FPL

December 5, 2014

L-2014-242
10 CFR 50.90

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

Subject: St. Lucie Nuclear Plant Units 1 and 2
Docket Nos.: 50-335 (Unit 1) and 50-389 (Unit 2)
APPLICATION TO ADOPT TSTF-505, REVISION 1, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES - RITSTF INITIATIVE 4B"

Dear Sir or Madam:

In accordance with the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), Florida Power and Light Company (FPL) is submitting a request for an amendment to the Technical Specifications (TS) for St. Lucie Nuclear Plant Units 1 and 2.

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

- Attachment 1 provides a description and assessment of the proposed change, the requested confirmation of applicability, and plant-specific verifications.
- Attachments 2 and 3 provide the marked up Unit 1 and Unit 2 TS pages, respectively, showing the proposed changes.
- Attachments 4 and 5 provide revised (clean) TS pages for Unit 1 and Unit 2, respectively.
- Attachments 6 and 7 provide marked up TS Bases pages showing the proposed changes (for information only).

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

FPL requests approval of the proposed license amendment by December 6, 2015, with the amendment being implemented within 180 days.

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

A001
NRR

This license amendment request has been reviewed by the St. Lucie Onsite Review Group.

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

Should you have any questions regarding this submittal, please contact Mr. Eric Katzman, Licensing Manager, at (772) 467-7734.

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

Executed on 5, December 2014

Sincerely,

Joseph Jensen
Site Vice President
St. Lucie Nuclear Plant

Attachments:

1. Description and Assessment
2. Proposed Technical Specifications Changes - Unit 1 (Mark-Ups)
3. Proposed Technical Specifications Changes - Unit 2 (Mark-Ups)
4. Revised Technical Specifications Pages - Unit 1
5. Revised Technical Specifications Pages - Unit 2
6. Proposed Technical Specifications Bases Changes - Unit 1 (Mark-Ups)
7. Proposed Technical Specifications Bases Changes - Unit 2 (Mark-Ups)

Enclosures:

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

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

ATTACHMENT 1

DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGE

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

1.0 DESCRIPTION

The proposed amendment would modify the Technical Specification (TS) requirements related to Completion Times (CTs) for Required Actions (RAs) to provide the option to calculate a longer, risk-informed CT (RICT). A new program, the Risk Informed Completion Time Program, is added to TS Section 6.0, "Administrative Controls."

The methodology for using the RICT Program is described in NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, which was approved by the NRC on May 17, 2007. Adherence to NEI 06-09 is required by the RICT Program. Florida Power & Light Company (FPL) is not proposing any deviations from NEI 06-09.

The proposed amendment is consistent with TSTF-505, Revision 1, “Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b”; however, only those Required Actions described in Enclosure 1 are proposed to be changed, which does not include all of the modified Required Actions in TSTF-505 and which includes some plant-specific Required Actions not included in TSTF-505.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

FPL has reviewed the model safety evaluation dated November 29, 2011 as part of the Federal Register Notice for Comment. This review included a review of the NRC staff's evaluation, as well as the supporting information provided to support TSTF-505 and the safety evaluation for NEI 06-09. As described in the subsequent paragraphs, FPL has concluded that the technical basis presented in the TSTF-505 proposal and the associated model safety evaluation prepared by the NRC staff are applicable to St. Lucie Nuclear Plant Units 1 and 2 and support incorporation of this amendment in the St. Lucie TS.

2.2 Verifications and Regulatory Commitments

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

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

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

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

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

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

5. Enclosure 5 provides the plant-specific baseline CDF and LERF to confirm that the potential risk increases allowed under the RICT Program are acceptable.

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

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

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

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

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

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

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

2.3 Optional Changes and Variations

Table 1 identifies each limiting condition of operation (LCO) and required action (RA) of TSTF-505, Revision 1 and the corresponding St. Lucie plant-specific LCO and RA where the RICT Program is proposed. Any differences between the plant-specific TS and TSTF-505 are identified and a justification is provided.

In general, the St. Lucie TS have the same content as the Standard TS (STS) for Combustion Engineering designed plants on which TSTF-505 was based; however, the St. Lucie TS were based on early versions of the Standard TS dating back to the initial licensing of the units, and therefore have different structure, formatting and numbering for most LCOs and actions. In order to simplify Table 1 and avoid confusion, such editorial differences are not specifically identified unless there is some unique effect which requires clarification. Editorial corrections and format changes are proposed that do not pertain to adoption of TSTF-505 and are conspicuously identified in TS mark-ups only. Lastly, the TS Bases do not contain the extent of information included in the Standard TS Bases regarding CTs. The St. Lucie TS Bases are revised only when and where appropriate.

TSTF-505 has different options available based on the plant-specific design considerations: for example, Section 3 TS include options for analog or digital instrumentation, and options based on whether a setpoint control program is used. The appropriate option applicable to the St. Lucie TS has been presented in Table 1 without presenting and explaining the unused options. TSTF-505 also includes TS with MODE 3 and MODE 4 applicability; St. Lucie is not adopting the RICT Program for these MODES at this time. These editorial differences are not considered optional changes or variations, and do not affect the applicability of TSTF-505 to the St. Lucie TS. St. Lucie TS do not have all of the LCOs and conditions in the scope of TSTF-505. Such cases are identified as variations in Table 1. In addition, some of the LCOs and conditions in the scope of TSTF-505 are not supported by the scope of the St. Lucie plant-specific PRA model; therefore, FPL is not adopting these LCOs into the proposed RICT Program. These cases are considered optional changes.

To accommodate implementation of TSTF-505 for the existing St. Lucie TS structure there are some plant-specific changes that are necessary, as are there some plant-specific TS requirements for which it is proposed to apply the RICT Program and some TSTF-505 conditions where FPL is not proposing to apply the RICT Program. Each instance is identified as a variation in Table 1.

FPL has identified two generic issues with TSTF-505 affecting Combustion Engineering plant TS that must be corrected for St. Lucie to properly implement the RICT Program. These are not considered variations since they are not plant-specific issues but rather issues with the TSTF-505 structure, and are identified as editorial changes in Table 1. (Note that TSTF-505 has spelling and grammar errors which have been corrected in the plant-specific TS mark-ups. Additionally, editorial corrections to existing St. Lucie TS and TS Bases have been conspicuously annotated in the TS and TS Bases mark-ups. These corrections are not identified as changes in Table 1.)

- TS 3.7.2 for main steam isolation valves (MSIVs) in TSTF-505 provides for unlimited operation in MODE 2 with one or more inoperable MSIVs provided the inoperable MSIVs are closed. The Condition applicable in MODE 1 for two or more inoperable MSIVs (Condition C) requires a shutdown to MODE 4 if the MSIVs are not restored to operable. This structure is not correct since once MODE 1 is exited, operation in MODE 2 is permitted allowing continued operation in MODE 2 with MSIVs inoperable but closed. In order to establish the correct structure for the St. Lucie TS, FPL offers that the proposed new action addressing two inoperable MSIVs include a MODE 2 shutdown requirement (rather than MODE 4), and to revise the existing action for one inoperable MSIV accordingly.

- TS 3.8.9 Condition D is a new Condition in TSTF-505 which specifically addresses loss of safety function due to multiple inoperable distribution subsystems. Conditions involving a loss of a safety function are prohibited from applying the RICT Program per the NEI guidance. Furthermore, the other Conditions (A, B, and C) are modified by TSTF-505 to address more than one inoperable distribution subsystem, so adopting TSTF-505 Condition D is considered unnecessary; therefore, FPL is not proposing to adopt this Condition.

There is one plant-specific condition for which FPL is proposing to apply the RICT Program. This condition is a variation as identified in Table 1 with additional detailed justification provided below:

- LCO/TS 3.3.1, Function 1, Action 1 addresses manual reactor trip channels. The St. Lucie TS have a 48-hour restoration action when one channel is inoperable. This differs from the STS, on which TSTF-505 is based, that require the affected reactor trip circuit breakers to be opened within one hour. Since no restoration time is identified before the action must be completed, the RICT Program is not applicable in TSTF-505. The St. Lucie TS are identical to the TSTF-505 TS for Westinghouse-designed plants for the manual reactor trip channels. TSTF-505 allows the RICT Program to be applied for one inoperable manual reactor trip channel for Westinghouse plants. Therefore, FPL is proposing to apply the RICT Program to its identical TS action.

Table 1

TSTF-505 LCO / RA	St. Lucie LCO / RA	NOTES
3.3.1 Reactor Protective System (RPS) Instrumentation (Analog) trip units, instrument and bypass removal channels	3.3.1.1 {Unit 1} 3.3.1 [Unit 2]	OPTIONAL CHANGE: FPL is not proposing to adopt the RICT Program for TS associated with LCO 3.3.1 in TSTF-505. The associated TS functions for RPS instrumentation are not modeled in detail in the plant-specific PRA.
3.3.3 A.1 RPS Logic and Trip Initiation (Analog) matrix logic channels	3.3.1.1 Function 12 Action 4 {Unit 1} 3.3.1 Function 11 Action 2 [Unit 2]	VARIATION: The St. Lucie TS do not include a restoration Action for matrix logic channel inoperability in Mode 1 or 2; therefore, the RICT Program is not applicable.
	3.3.1.1 Function 1 Action 1 {Unit 1} 3.3.1 Function 1 Action 1 [Unit 2] manual reactor trip channels	VARIATION: For inoperability of one manual reactor trip channel for Combustion Engineering (CE) plants, TSTF-505 does not apply the RICT Program because the required Action does not require restoration of the inoperable channel. The corresponding St. Lucie TS for one inoperable manual reactor trip channel (which is based on the CE standard TS in existence at the time the plants were licensed) provides a 48-hour restoration Action, which is the same as TSTF-505 for Westinghouse plants (TS 3.3.1 Condition B). TSTF-505 does apply the RICT Program for the Westinghouse TS Condition associated with one inoperable manual reactor trip channel. FPL is therefore proposing to apply the RICT Program when one manual reactor trip channel is inoperable, consistent with TSTF-505 for Westinghouse plants.
3.3.4 B.1 Engineered Safety Features Actuation System (ESFAS) Instrumentation (Analog)	3.3.2.1 Function 2.b Action 10c {Unit 1} 3.3.2 Function 2.b Action 18c [Unit 2] containment spray manual trip	VARIATION: FPL is proposing to maintain the existing restoration Action for two inoperable channels and to apply the RICT Program accordingly.

Table 1		
TSTF-505 LCO / RA	**St. Lucie LCO / RA**	**NOTES**
3.3.4 C.2.1 and C.2.2 ESFAS Instrumentation (Analog)	3.3.2.1 Function 5.b Action 13a {Unit 1} 3.3.2 Function 5.b Action 19 [Unit 2] containment sump recirculation	VARIATION: The St. Lucie TS differ from TSTF-505 in that restoration of the one inoperable trip unit (instrument channel) is not a TS required Action except for Function 5.b for containment sump recirculation (RAS). FPL is therefore only proposing to apply the RICT Program to the Action associated with Function 5.b.
3.3.4 D.2 ESFAS Instrumentation (Analog)	3.3.2.1 Functions 7.c, 8.a, 8.b Action 14c {Unit 1} 3.3.2 Functions 7.c, 8.a, 8.b Action 20c [Unit 2] AFAS / AFW isolation	VARIATION: The St. Lucie TS differ from TSTF-505 in that restoration of any one inoperable trip unit (instrument channel) is not a TS required Action except for Functions 7.c (auxiliary feedwater SG level low), 8.a (AFW isolation - SG differential pressure), and 8.b (AFW isolation - feedwater header differential pressure). FPL is therefore only proposing to apply the RICT Program to the Action associated with Functions 7.c, 8.a, and 8.b.
3.3.4 E ESFAS Instrumentation (Analog) automatic bypass removal	N/A	VARIATION: The St. Lucie TS do not have an Action associated with automatic bypass removal channel inoperability.
3.3.5 A.1 and B.1 ESFAS Logic and Manual Trip (Analog)	3.3.2.1 Functions 7.a, 7.b Actions 11a and 11b {Unit 1} 3.3.2 Functions 7.a, 7.b Actions 15a and 15b [Unit 2] AFAS manual, automatic initiation	
3.3.5 D.1 and E.1 ESFAS Logic and Manual Trip (Analog)	3.3.2.1 Functions 1.a, 2.a, 3.a, 4.a, 5.a Actions 8a, 8b {Unit 1} 3.3.2 Functions 1.a, 1.d, 2.a, 2.c, 3.a, 3.e, 4.d, 5.a, 5.c Action 12 [Unit 2] ESFAS manual trip channels	VARIATION: { Unit 1} - The St. Lucie TS address manual actuation capability but do not have an Action associated with inoperable automatic actuation logic.

Table 1

TSTF-505 LCO / RA	St. Lucie LCO / RA	NOTES
3.3.6 Diesel Generator (DG) - Loss of Voltage Start (LOVS) (Analog)	3.3.2.1 Function 6 {Unit 1} 3.3.2 Function 6 [Unit 2] loss of power	VARIATION: {Unit 1} - The St. Lucie TS do not include a restoration Action for loss of power instrumentation; therefore, the RICT Program is not applicable.
3.4.5 RCS Loops - Mode 3	3.4.1.2	OPTIONAL CHANGE: This LCO has Mode 3 applicability only. FPL is not proposing to apply the RICT Program in Mode 3.
3.4.9 Pressurizer	3.4.4 {Unit 1} 3.4.3 [Unit 2]	VARIATION: { Unit 1} - The St. Lucie TS do not include a restoration Action for inoperable pressurizer heaters; therefore, the RICT Program is not applicable. OPTIONAL CHANGE: [Unit 2] - FPL is not proposing to adopt the RICT Program for TS associated with LCO 3.4.9 in TSTF-505, as pressurizer heaters are not modeled in the plant-specific PRA.
3.4.10 A.1 Pressurizer Safety Valves	3.4.3 Action a {Unit 1} 3.4.2.2 Action a [Unit 2]	
3.4.11 B.3 AND E.3 Pressurizer Power Operated Relief Valves (PORVs)	N/A	VARIATION: The St. Lucie TS do not include an LCO for the PORVs in Mode 1 or 2.
3.4.11 C.2 and F.1 PORVs	3.4.12 {Unit 1} 3.4.4 Action a [Unit 2] PORV block valves	VARIATION: TSTF-505 provides two separate conditions for one inoperable PORV block valve (Condition C), or two PORV block valves inoperable (Condition F), and permits application of the RICT Program to each separate Required Action. The St. Lucie TS have a one-hour CT for restoration of one or more PORV block valves. Application of the RICT Program to this combined Condition is consistent with TSTF-505.

Table 1

TSTF-505 LCO / RA	St. Lucie LCO / RA	NOTES
3.4.14 RCS Pressure Isolation Valve (PIV) Leakage	N/A {Unit 1} 3.4.6.2 Action c [Unit 2] shutdown cooling auto-close interlock	VARIATION: [Unit 2] - The resolution of the plant-specific PRA model does not support calculation of a RICT for this condition.
3.5.1 A.1 Safety Injection Tanks (SITs)	3.5.1 Action a SIT boron concentration, water volume, nitrogen cover-pressure	
3.5.1 B.1 SITs	3.5.1 Action b SIT - reasons other than in Action a	
3.5.1 C.1 SITs	3.5.1 Action c multiple SITs	
3.5.2 A.1 Emergency Core Cooling System (ECCS) - Operating	3.5.2 Action a.1 ECCS subsystem - LPSI	
3.5.2 B.1 ECCS - Operating	3.5.2 Action a.2 ECCS subsystem - reasons other than in Action a.2	VARIATION: TSTF-505 Condition B addresses one or more inoperable trains, while the St. Lucie TS only address one inoperable ECCS subsystem. FPL is proposing to apply the RICT Program when one ECCS subsystem is inoperable for reasons other than LPSI inoperability.
3.5.2 C.1 ECCS - Operating	N/A	VARIATION: FPL is not proposing to include an Action regarding ECCS 100% flow due to design limitations.
3.5.3 A.1 ECCS - Shutdown	3.5.3	OPTIONAL CHANGE: This LCO has Mode 3 and 4 applicability only. FPL is not proposing to apply the RICT Program in Mode 3 or 4.
3.5.4 A.1 and B.1 Refueling Water Tank (RWT)	3.5.4 RWT borated water volume, temperature	EDITORIAL: The St. Lucie TS do not have separate Actions for the Refueling Water Tank when it is inoperable due to either boron concentration or temperature not within limits. TSTF-505 permits application of the RICT Program for any cause of inoperability, so the proposed change to the St. Lucie TS Action is consistent with TSTF-505.

Table 1

TSTF-505 LCO / RA	St. Lucie LCO / RA	NOTES
3.6.2 C.3 Containment Air Locks (Atmospheric and Dual)	3.6.1.3 Action b one or both air locks, inoperable door	EDITORIAL: TSTF-505 Condition C addresses one or more inoperable containment air locks due to causes other than an inoperable door (Condition A) or interlock mechanism (Condition B). The St. Lucie TS do not provide a separate Action for an inoperable interlock mechanism to permit continued operation indefinitely; however, FPL is proposing to adopt STS language for inoperability of both containment air locks due to an inoperable door, and to apply the RICT Program accordingly.
3.6.3 A.1, B.1, C.1, D.1, E.1, and F.1 Containment Isolation Valves (Atmospheric and Dual)	3.6.3.1 Actions a, b, c {Unit 1} 3.6.3 Actions a, b, c [Unit 2] containment isolation valves 3.6.1.7 Actions a, b, c [Unit 2] containment purge valves	EDITORIAL: TSTF-505 provides separate Conditions to address six different types of containment penetrations with different CTs. The RICT Program is applicable to the Required Actions to isolate the affected penetration(s) by different means. TSTF-505 also includes conforming changes to adjust the 31-day periodic verification to begin after the penetration has been isolated. The St. Lucie TS provide three Actions that require restoration of the inoperable isolation valve(s), or isolation of the affected containment penetrations by either of two means. These Actions are consistent with TSTF-505, but without distinguishing different types of containment penetrations and with a different grouping of the methods of isolation. Application of the RICT Program is consistent with TSTF-505. For Unit 2, the containment purge valves are in a separate TS (LCO 3.6.1.7); however, the required Actions to restore operability or isolate the affected penetration(s) are the same as in TSTF-505. The Unit 1 St. Lucie TS do not require periodic verification of isolation, so the TSTF-505 conforming changes are not required. For Unit 2, only TS 3.6.1.7 Action c has a periodic verification and so the TSTF-505 conforming change is required.

Table 1

TSTF-505 LCO / RA	St. Lucie LCO / RA	NOTES
3.6.6A A.1 Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)	3.6.2.1 Action 1.a one containment spray train	
3.6.6A C.1 Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)	3.6.2.1 Action 1.b one containment cooling train	
3.6.6A D.1 and D.2 Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System) one containment spray and one containment cooling train	N/A	VARIATION: The St. Lucie TS for simultaneous inoperability of one containment spray train and one containment cooling train require concurrent implementation of Actions 1.a and 1.b. TSTF-505 repeats the Required Actions to restore the containment spray or cooling train, but reduces the CT from 7 days to 72 hours. The St. Lucie Action for containment spray is already 72 hours, but the 7-day CT for a containment cooling train is not changed. This has no effect on the RICT Program, since if a RICT were entered for this condition, the RICT would become applicable at 72 hours which is the limiting CT, and is consistent with TSTF-505. Because the plant-specific TS Action simply makes reference back to exiting Actions for which the RICT Program is applicable, the TSTF-505 change is not required for this Action.
3.6.6A E.1 Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)	3.6.2.1 Action 1.d two containment cooling trains	

Table 1

TSTF-505 LCO / RA	St. Lucie LCO / RA	NOTES
3.6.6A F.1 Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)	3.6.2.1 Action 1.e two containment spray trains	
3.6.9 Hydrogen Mixing System (HMS) (Atmospheric and Dual)	N/A	VARIATION: The St. Lucie TS do not have an LCO corresponding to TSTF-505 LCO 3.6.9 for a hydrogen mixing system.
3.7.2 A.1 Main Steam Isolation Valves (MSIVs)	3.7.1.5 Mode 1 Action a one MSIV	EDITORIAL: The St. Lucie TS include the optional Action to close the inoperable MSIV to exit the Action in addition to restoring the inoperable MSIV to OPERABLE status. TSTF-505 only includes the restoration Action with the RICT Program applicable. This has no impact on RICT Program implementation for one inoperable open MSIV.
3.7.2 C.1 MSIVs	3.7.1.5 Mode 1 Action b two or more MSIVs	EDITORIAL: The default Condition in TSTF-505 for Condition C requires the plant to enter Mode 4; however, since Condition C is only applicable in Mode 1, the correct default Condition should only be to enter Mode 2 (TSTF-505 Condition B). FPL is proposing that the default shutdown for Action a be corrected to Mode 2. EDITORIAL: [Unit 2] - For Mode 1 Action a and Modes 2-4 Action, FPL is proposing to correct the end states to Modes that are just beyond the Mode(s) of applicability.
3.7.4 Atmospheric Dump Valves (ADVs)	N/A {Unit 1} 3.7.1.7 [Unit 2]	VARIATION: {Unit 1} - The St. Lucie TS do not have an LCO corresponding to TSTF-505 LCO 3.7.4 for atmospheric dump valves. VARIATION: [Unit 2] - FPL is not proposing to adopt TSTF-505 change for the corresponding plant-specific TS. The associated TS functions for are not modeled in detail in the plant-specific PRA.

Table 1		
TSTF-505 LCO / RA	**St. Lucie LCO / RA**	**NOTES**
3.7.5 A.1 AND B.1 Auxiliary Feedwater (AFW) System	3.7.1.2 Action a one AFW pump	EDITORIAL: The St. Lucie TS do not have separate Actions for one AFW train inoperable when it is inoperable due to a single steam supply being inoperable or for other causes. TSTF-505 permits application of the RICT Program for any cause of inoperability, so the proposed change to the St. Lucie TS Action is consistent with TSTF-505.
3.7.5 C.1 AFW System	3.7.1.2 Action b two AFW pumps	
3.7.6 A.2 Condensate Storage Tank (CST)	3.7.1.3 Action	
3.7.7 A.1 Component Cooling Water (CCW) System	3.7.3.1 Action a {Unit 1} 3.7.3 Action a [Unit 2] one CCW loop	EDITORIAL: The St. Lucie TS differ from TSTF-505 in nomenclature designating "loops" instead of "trains."
3.7.7 B.1 CCW System	3.7.3.1 Action b {Unit 1} 3.7.3 Action b [Unit 2] two CCW loops	EDITORIAL: The St. Lucie TS differ from TSTF-505 in nomenclature designating "loops" instead of "trains."
3.7.8 A.1 Service Water System (SWS)	3.7.4.1 Action a {Unit 1} 3.7.4 Action a [Unit 2] intake cooling water loop	EDITORIAL: The St. Lucie TS differ from TSTF-505 in nomenclature designating "intake cooling water loops" instead of "SWS trains."
3.7.8 B.1 SWS	3.7.4.1 Action b {Unit 1} 3.7.4 Action b [Unit 2] two intake cooling water loops	EDITORIAL: The St. Lucie TS differ from TSTF-505 in nomenclature designating "intake cooling water loops" instead of "SWS trains."
3.7.9 Ultimate Heat Sink (UHS)	3.7.5.1 ultimate heat sink	VARIATION: FPL does not propose to apply the RICT Program to the LCO associated with the ultimate heat sink. The plant-specific TS Actions either do not involve restoration of equipment, or provide an alternative Action which does not require a plant shutdown, so application of the RICT Program is not required.

Table 1

TSTF-505 LCO / RA	St. Lucie LCO / RA	NOTES
3.7.10 Essential Chilled Water (ECW)	N/A	VARIATION: The St. Lucie TS do not have an LCO corresponding to TSTF-505 LCO 3.7.10 for essential chilled water.
3.7.12 Control Room Emergency Air Temperature Control System (CREATCS)	3.7.7.1 {Unit 1} control room emergency ventilation 3.7.7 [Unit 2] control room emergency air cleanup	VARIATION: FPL is not proposing to include the control room cooling function in the scope of the RICT Program. The scope of the plant-specific LCO includes the radiological filtration function, and would therefore be outside the scope of the RICT Program.
3.8.1 A.3 AC Sources - Operating	3.8.1.1 Action a offsite circuits	
3.8.1 B.4 AC Sources - Operating	3.8.1.1 Action b diesel generator	
3.8.1 C.2 AC Sources - Operating	3.8.1.1 Action d required offsite A.C. circuits	
3.8.1 D.1 AC Sources - Operating	3.8.1.1 Action c offsite circuit and diesel generator	
3.8.1 E.1 AC Sources - Operating	3.8.1.1 Action e diesel generators	
3.8.1 F.1 AC Sources - Operating sequencers	N/A	VARIATION: The St. Lucie TS do not have a separate condition for sequencers.
3.8.1 G.1 AC Sources - Operating	3.8.1.1 Action f AC power source	

Table 1		
TSTF-505 LCO / RA	**St. Lucie LCO / RA**	**NOTES**
	3.8.1.1 Action g startup transformer	VARIATION: The St. Lucie TS include consideration of shared equipment between Units 1 and 2, and provide a separate Action for startup transformer inoperability under these conditions. The restoration Action for one inoperable offsite circuit (Action a) is repeated in this Action, so the RICT Program is applied for Action g for consistency with Action a.
3.8.4 A.1 and A.3 DC Sources - Operating	3.8.2.3 Action b {Unit 1} 3.8.2.1 Action b [Unit 2] battery chargers	VARIATION: The St. Lucie TS for an inoperable battery charger do not provide any restoration Actions; therefore, the RICT Program is not applicable.
3.8.4 B.1 and C.1 DC Sources - Operating	3.8.2.3 Action a {Unit 1} 3.8.2.1 Action a [Unit 2] battery(ies), one DC power source	EDITORIAL: The St. Lucie TS do not have separate Actions for the one DC electrical source inoperable when it is inoperable due to a battery being inoperable or for other causes. TSTF-505 permits application of the RICT Program for any cause of inoperability, so the proposed change to the St. Lucie TS Action is consistent with TSTF-505.
3.8.4 D.1 DC Sources - Operating	3.8.2.3 Action c {Unit 1} 3.8.2.1 Action c [Unit 2] two DC power sources	
3.8.7 A.1 and B.1 Inverters - Operating	N/A {Unit 1} 3.8.3.1 Actions b and d [Unit 2]	VARIATION: { Unit 1} - The St. Lucie TS do not have an LCO corresponding to TSTF-505 LCO 3.8.7 for inverters. EDITORIAL: [Unit 2] - Each St. Lucie Action provides two Actions, one to re-energize the associated instrument bus and one to restore the OPERABLE alignment through the inverter connected to its DC source. TSTF-505 has separate LCOs (3.8.7 and 3.8.9) for operability and for energizing the instrument busses, but the RICT Program is applied to both.

Table 1		
TSTF-505 LCO / RA	**St. Lucie LCO / RA**	**NOTES**
3.8.9 A.1, B.1, and C.1 Distribution Systems - Operating	3.8.2.1 Action {Unit 1} 3.8.3.1 Actions a, b, c, d [Unit 2] AC instrument bus, AC vital panel	
3.8.9 D.1 Distribution Systems - Operating loss of function	N/A	EDITORIAL: This new condition in TSTF-505 specifies a loss of a safety function, for which it is not permitted to apply a RICT; therefore, FPL does not propose to adopt this change.

3.0 REGULATORY SAFETY ANALYSIS

3.1 No Significant Hazards Consideration Determination

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

St. Lucie Nuclear Plant Units 1 and 2 request adoption of an approved change to the Standard Technical Specifications and plant-specific 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 TS Section 6.0, "Administrative Controls," under the appropriate subsection, entitled the "Risk Informed Completion Time Program."

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

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

   Response: No.

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

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

   Response: No.

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

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

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

   Response: No.

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

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

4.0 ENVIRONMENTAL CONSIDERATION

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

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

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATIONS CHANGES - UNIT 1 (MARK-UPS)

INSERT 1

or in accordance with the Risk Informed Completion Time Program,

INSERT 2

or in accordance with the Risk Informed Completion Time Program

INSERT 3 - Table 3.3-3, Action 8

NOTE
Action not applicable when second manual trip channel intentionally made inoperable.

b. With two channels inoperable, restore the inoperable channels to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 4 - Table 3.3-3, Action 10

NOTE
Actions 10c and 10d not applicable when two or more CSAS trip units or associated instruments intentionally made inoperable.

INSERT 5 - Table 3.3-3, Action 10

d. With the number of OPERABLE channels two or more less than the Minimum Channels OPERABLE, restore inoperable channels to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 6- Table 3.3-3, Action 11

a. With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

NOTE
Action not applicable when second AFAS manual trip or actuation logic channel intentionally made inoperable.

b. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE, restore the inoperable channels to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 7 - LCO 3.5.1, Safety Injection Tanks (SIT)

NOTE
Action not applicable when two or more SITs intentionally made inoperable.

c. With two or more SITs inoperable, restore SITs to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 8 - LCO 3.6.2.1, Containment Spray and Cooling Systems

NOTE
Action not applicable when second containment spray train or three or more containment spray or cooling trains intentionally made inoperable.

e. With two containment spray trains inoperable or any combination of three or more trains inoperable, restore containment spray trains and containment cooling trains to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

INSERT 9 - LCO 3.7.1.2, Auxiliary Feedwater System

NOTE
Action not applicable when second auxiliary feedwater pump intentionally made inoperable.

b. With two auxiliary feedwater pumps inoperable, restore at least one auxiliary feedwater pump to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 10 - LCO 3.7.1.5, Main Steam Isolation Valves

NOTE Action not applicable when both main steam isolation valves intentionally made inoperable.

b. With both MSIVs inoperable in MODE 1, restore MSIVs to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program; otherwise, be in MODE 2 within the next 6 hours.

INSERT 11 - LCO 3.7.3.1, Component Cooling Water System

NOTE Action not applicable when second component cooling water loop intentionally made inoperable.

b. With two component cooling water loops inoperable, restore at least one loop to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 12 - LCO 3.7.4.1, Intake Cooling Water System

NOTE Action not applicable when second intake cooling water loop intentionally made inoperable.

b. With two intake cooling water loops inoperable, restore at least one loop to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 13 - LCO 3.8.1.1, A.C. Sources - Operating

NOTE Action not applicable when three or more A.C. sources intentionally made inoperable.

f. With three or more A.C. sources inoperable, restore inoperable A.C. sources to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 14 - LCO 3.8.2.3, D.C. Distribution - Operating

NOTE
Action not applicable when second D.C. source intentionally made inoperable.

c. With two D.C. electrical sources inoperable, restore at least one D.C. electrical source to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 15 - Section 6.0, Administrative Controls (6.8.4)

o. 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:

(i) The RICT may not exceed 30 days;

(ii) A RICT may only be utilized in MODES 1 and 2;

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

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

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

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

(iv) 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.

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

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq$ 1% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq 10^{-4}$% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

(e) Deleted.

(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours, or be in HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

INSERT 1

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

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure – High	4	2	3	1, 2, 3	9#
c. Pressurizer Pressure – Low	4	2	3	1, 2, 3(a)	9#
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure – High-High	4	2(b)	3	1, 2, 3	10a#, 10b#, 10c , 10d
3. CONTAINMENT ISOLATION (CIS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure – High	4	2	3	1, 2, 3	9#
c. Containment Radiation – High	4	2	3	1, 2, 3, 4	9#
d. SIAS	---------- (See Functional Unit 1 above) ----------				
4. MAIN STEAM LINE ISOLATION (MSIS)					
a. Manual (Trip Buttons)	2/steam generator	1/steam generator	2/operating steam generator	1, 2, 3, 4	8
b. Steam Generator Pressure – Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	9#

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1725 psia; bypass shall be automatically removed when pressurizer pressure is $\geq$ 1725 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed in this MODE below 685 psig; bypass shall be automatically removed at or above 685 psig.

\# The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 8 - a. With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. INSERT 1

INSERT 3

ACTION 9 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the bypassed or tripped condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then place the inoperable channel in the tripped condition.

INSERT 4

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

c. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours. INSERT 1

INSERT 5

ACTION 11 - ~~With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.~~ INSERT 6

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION 13 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

INSERT 2

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If OPERABILITY ~~can not~~cannot be restored within 48 hours, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

ACTION 14 - With the number of channels OPERABLE one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If an inoperable SG level channel can not be restored to OPERABLE status within 48 hours, then AFAS-1 or AFAS-2 as applicable in the inoperable channel shall be placed in the bypassed condition. If an inoperable SG DP or FW Header DP channel can not be restored to OPERABLE status within 48 hours, then both AFAS-1 and AFAS-2 in the inoperable channel shall be placed in the bypassed condition. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

c. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 1

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of $\geq$ 2422.8 psig and $\leq$ 2560.3 psig.

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures > 281°F.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the next 6 hours.

INSERT 1

b. With two or more pressurizer code safety valves inoperable, be in HOT STANDBY within 6 hours and in HOT SHUTDOWN with all RCS cold leg temperatures $\leq$ 281°F within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 Verify each pressurizer code safety valves is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within +/- 1% of 2500 psia.

REACTOR COOLANT SYSTEM

PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.12 Each Power Operated Relief Valve (PORV) Block Valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

INSERT 2

With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.12 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

SAFETY INJECTION TANKS (SITs)

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. Between 1090 and 1170 cubic feet of borated water,

c. A minimum boron concentration of 1900 ppm, and

d. A nitrogen cover-pressure of between 230 and 280 psig.

format change

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

ACTION:

a. With one SIT inoperable due to boron concentration not within limits, or due to an inability to verify the required water volume or nitrogen cover-pressure, restore the inoperable SIT to OPERABLE status with 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. INSERT 2

b. With one SIT inoperable due to reasons other than those stated in ACTION-a, restore the inoperable SIT to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. INSERT 2

INSERT 7

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

2. Verifying that each safety injection tank isolation valve is open.

format change

~~* With pressurizer pressure ≥ 1750 psia.~~

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) pump,

b. One OPERABLE low-pressure safety injection pump,

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and

d. One OPERABLE charging pump*.

format change

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

ACTION:

a. 1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days INSERT 1 or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours INSERT 1 or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

format change

~~*~~

NOTE

One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e.

~~** With pressurizer pressure ≥ 1750 psia.~~

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained volume 477,360 gallons of borated water,

b. A minimum boron concentration of 1900 ppm,

c. A maximum water temperature of 100°F,

d. A minimum water temperature of 55°F when in MODES 1 and 2, and

e. A minimum water temperature of 40°F when in MODES 3 and 4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: INSERT 1

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable~~*~~:

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

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

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

one or both

in the affected air lock(s) and

4. The provisions of Specification 3.0.4 are not applicable.

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

INSERT 2

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

format change

~~*~~

NOTE

If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door. No more than one airlock door shall be open at any time.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: Containment Spray System: MODES 1, 2, and MODE 3 with Pressurizer Pressure ≥ 1750 psia.

Containment Cooling System: MODES 1, 2, and 3.

ACTION:

1. Modes 1, 2, and 3 with Pressurizer Pressure ≥ 1750 psia:

INSERT 2

a. With one containment spray train inoperable, restore the inoperable spray train to OPERABLE status within 72 hours ~~and within 10 days from initial discovery of failure to meet the LCO~~; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 54 hours.

INSERT 2

b. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 7 days ~~and within 10 days from initial discovery of failure to meet the LCO~~; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

c. With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered inoperable.

d. With two containment cooling trains inoperable, restore one cooling train to OPERABLE status within 72 hours; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

INSERT 2

INSERT 8

~~e. With two containment spray trains inoperable or any combination of three or more trains inoperable, enter LCO 3.0.3. immediately.~~

2. Mode 3 with Pressurizer Pressure < 1750 psia:

a. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours; otherwise be in MODE 4 within the next 6 hours.

b. With two containment cooling trains inoperable, enter LCO 3.0.3 immediately.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves shall be OPERABLE:

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours INSERT 2, or

b. Isolate each affected penetration within 4 hours INSERT 2 by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours INSERT 2 by use of at least one closed manual valve or blind flange; or

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

SURVEILLANCE REQUIREMENTS

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

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor driven feedwater pumps, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the ~~at least three~~ editorial auxiliary feedwater pump~~s (two motor driven pumps and one capable of being powered by an OPERABLE steam supply system)~~ to OPERABLE status within 72 hours INSERT 1 or be in HOT SHUTDOWN within the next 12 hours.

INSERT 9

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank shall be OPERABLE with a minimum contained volume of 153,400 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, restore the condensate storage tank to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the water level.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - a. With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours; otherwise, be in ~~HOT STANDBY~~ within the next 6 hours.

INSERT 10 INSERT 2 MODE 2

MODES 2 and 3 - With one or both main steam isolation valve(s) inoperable, subsequent operation in MODES 2 or 3 may proceed provided the isolation valve(s) is (are) maintained closed. Otherwise, be in at least HOT STANDBY within the next 6 hours and in ~~COLD~~ HOT SHUTDOWN within the following 24 hours.

The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve that is open shall be demonstrated OPERABLE by verifying full closure within 6.0 seconds when tested pursuant to the Inservice Testing Program.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

INSERT 11

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation Signal.

PLANT SYSTEMS

3/4.7.4 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent intake cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With only one intake cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

INSERT 12

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two intake cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation signal.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generator sets each with:

1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

2. A separate fuel storage system containing a minimum of 19,000 gallons of fuel, and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, except as provided in Action ~~g~~f. below, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the offsite circuit to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

INSERT 1

b. With one diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG~~*~~; restore the diesel generator to OPERABLE status within 14 days INSERT 1 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

format change

NOTE

If the absence of any common-cause failure cannot be confirmed, ~~this test~~SR 4.8.1.1.2.a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

ELECTRICAL POWER SYSTEMS

ACTION (continued)

c. With one offsite A.C. circuit and one diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG~~*~~. Restore at least one of the inoperable sources to OPERABLE status within 12 hours ~~or~~ INSERT 1 be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 ACTION Statement a or b, as appropriate, with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable A.C. power source. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

d. With two of the required offsite A.C. circuits inoperable, restore one of the inoperable offsite sources to OPERABLE status within 24 hours INSERT 1 or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow ACTION Statement a. with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable offsite A.C. circuit.

format change

NOTE
If the absence of any common-cause failure cannot be confirmed, ~~this test~~SR 4.8.1.1.2.a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

ELECTRICAL POWER SYSTEMS

ACTION (continued)

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in the at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator unit, follow ACTION Statement b. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator. INSERT 1

INSERT 13

g~~f~~ With one Unit 1 startup transformer (1A or 1B) inoperable and with a Unit 2 startup transformer (2A or 2B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 2 require the use of the startup transformer administratively available to both units, Unit 1 shall demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within 72 hours~~, or~~ be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability; and

b. Demonstrated OPERABLE at least once per 18 months by transferring (manually and automatically) unit power supply from the auxiliary transformer to the startup transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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

1. Verifying fuel level in the engine-mounted fuel tank~~,~~

2. Verifying the fuel level in the fuel storage tank~~,~~

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the engine-mounted tank~~,~~

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generator sets:

4160	volt Emergency Bus	1A3
4160	volt Emergency Bus	1B3
480	volt Emergency Bus	1A2
480	volt Emergency Bus	1B2
480	volt Emergency MCC Busses	1A5, 1A6, 1A7
480	volt Emergency MCC Busses	1B5, 1B6, 1B7
120	volt A.C. Instrument Bus	1MA
120	volt A.C. Instrument Bus	1MB
120	volt A.C. Instrument Bus	1MC
120	volt A.C. Instrument Bus	1MD

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying indicated power availability.

Amendment No.

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt D.C. bus No. 1A, 125-volt Battery bank No. 1A and a full capacity charger.

b. 125-volt D.C. bus No. 1B, 125-volt Battery bank No. 1B and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks or busses inoperable, restore the inoperable battery bank or bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.3.2.a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

INSERT 14

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying indicated power availability.

4.8.2.3.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

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

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

2. The total battery terminal voltage is greater than or equal to 129-volts on float charge.

m. Control Room Envelope Habitability Program (continued)

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

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 36 month assessment of the CRE boundary.

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 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

n. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(i) Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,
2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
3. A clear and bright appearance with proper color or a water and sediment content within limits;

(ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

(iii) Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

INSERT 15

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS CHANGES - UNIT 2 (MARK-UPS)

INSERT 1

or in accordance with the Risk Informed Completion Time Program,

INSERT 2

or in accordance with the Risk Informed Completion Time Program

INSERT 3 - Table 3.3-3, Action 12

NOTE
Action not applicable when second manual trip channel intentionally made inoperable.

b. With two channels inoperable, restore the inoperable channels to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 4 - Table 3.3-3, Action 15

NOTE
Action not applicable when second AFAS manual trip or actuation logic channel intentionally made inoperable.

b. With two channels inoperable, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 5 - Table 3.3-3, Action 18

NOTE
Actions 18c and 18d not applicable when two or more CSAS trip units or associated instruments intentionally made inoperable.

INSERT 6 - Table 3.3-3, Action 18

d. With the number of channels OPERABLE, two or more less than the Minimum Channels OPERABLE, restore inoperable channels to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 7 - LCO 3.5.1, Safety Injection Tanks (SIT)

NOTE
Action not applicable when two or more SITs intentionally made inoperable.

c. With two or more SITs inoperable, restore SITs to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 8 - LCO 3.6.2.1, Containment Spray and Cooling Systems

NOTE
Action not applicable when second containment spray train or three or more containment spray or cooling trains intentionally made inoperable.

e. With two containment spray trains inoperable or any combination of three or more trains inoperable, restore containment spray trains and containment cooling trains to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

INSERT 9 - LCO 3.7.1.2, Auxiliary Feedwater System

NOTE
Action not applicable when second auxiliary feedwater pump intentionally made inoperable.

b. With two auxiliary feedwater pumps inoperable, restore at least one auxiliary feedwater pump to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 10 - LCO 3.7.1.5, Main Steam Isolation Valves

NOTE
Action not applicable when both main steam isolation valves intentionally made inoperable.

b. With both main steam line isolation valves inoperable in MODE 1, restore main steam isolation valves to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program; otherwise, be in MODE 2 within the next 6 hours.

INSERT 11 - LCO 3.7.3, Component Cooling Water System

NOTE
Action not applicable when second component cooling water loop intentionally made inoperable.

b. With two component cooling water loops inoperable, restore at least one loop to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 12 - LCO 3.7.4, Intake Cooling Water System

NOTE
Action not applicable when second intake cooling water loop intentionally made inoperable.

b. With two intake cooling water loops inoperable, restore at least one loop to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 13 - LCO 3.8.1.1, A.C. Sources - Operating

NOTE
Action not applicable when three or more A.C. sources intentionally made inoperable.

f. With three or more A.C. sources inoperable, restore inoperable A.C. sources to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 14 - LCO 3.8.2.1, D.C. Sources - Operating

NOTE
Action not applicable when second D.C. source intentionally made inoperable.

c. With two D.C. electrical sources inoperable, restore at least one D.C. electrical source to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 15 - LCO 3.8.3.1, Onsite Power Distribution - Operating

NOTE
Action not applicable when more than one A.C. vital panel intentionally either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus.

d. With more than one A.C. vital panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital panels within one hour or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital panels from their associated inverters connected to their associated D.C. buses within one hour or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 16 - Section 6.0, Administrative Controls (6.8.4)

p. 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:

(i) The RICT may not exceed 30 days;

(ii) A RICT may only be utilized in MODES 1 and 2;

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

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

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

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

(iv) 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.

(v) 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.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a boric acid makeup pump through a charging pump to the Reactor Coolant System.

b. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a gravity feed valve through a charging pump to the Reactor Coolant System.

c. The flow path from the refueling water ~~storage~~ tank via a charging pump to the Reactor Coolant System.

editorial

OR

At least two of the following three boron injection flow paths shall be OPERABLE:

d. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both boric acid makeup pumps through a charging pump to the Reactor Coolant System.

e. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both gravity feed valves through a charging pump to the Reactor Coolant System.

f. The flow path from the refueling water ~~storage~~ tank, via a charging pump to the Reactor Coolant System.

editorial

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200 °F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

\# The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER in conjunction with (d) below; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is greater than or equal to 15% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq 10^{-4}$% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours, or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

INSERT 1

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Containment Pressure – High	4	2	3	1, 2, 3	13*, 14
c. Pressurizer Pressure – Low	4	2	3	1, 2, 3(a)	13*, 14
d. Automatic Actuation – Logic (delete hyphen)	2	1	2	1, 2, 3, 4	12
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Containment Pressure – High-High	4	2	3	1(b), 2(b), 3(b)	18a*, 18b*, 18c, 18d
c. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	12
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Safety Injection (SIAS)	See Functional Unit 1 for all Safety Injection Initiating Functions and Requirements				
c. Containment Pressure – High	4	2	3	1, 2, 3	13*, 14
d. Containment Radiation – High	4	2	3	1, 2, 3	13*, 14
e. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION (MSIS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3	16
b. Steam Generator Pressure – Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	13*, 14
c. Containment Pressure – High	4	2	3	1, 2, 3	13*, 14
d. Automatic Actuation Logic	2	1	2	1, 2, 3	12
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Refueling Water ~~Storage~~ Tank - Low editorial	4	2	3	1, 2, 3	19
c. Automatic Actuation Logic	2	1	2	1, 2, 3	12

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than 1836 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to 1836 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed in this MODE below 700 psia; bypass shall be automatically removed at or above 700 psia.

* The provisions of Specification 3.0.4 are not applicable.

ACTION OF STATEMENTS

ACTION 12 - a. With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 3

INSERT 1

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit	Functional Unit Bypassed
1. Containment Pressure -	Containment Pressure – High (SIAS, CIAS, CSAS) Containment Pressure – High (RPS)
2. Steam Generator Pressure -	Steam Generator Pressure – Low (MSIS) AFAS-1 and AFAS-2 (AFAS) Thermal Margin/Low Pressure (RPS) Steam Generator Pressure – Low (RPS)
3. Steam Generator Level -	Steam Generator Level – Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)
4. Pressurizer Pressure -	Pressurizer Pressure – High (RPS) Pressurizer Pressure – Low (SIAS) Thermal Margin/Low Pressure (RPS)

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below.

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Containment Pressure -	Containment Pressure – High (SIAS, CIAS, CSAS) Containment Pressure – High (RPS)
2. Steam Generator Pressure -	Steam Generator Pressure – Low (MSIS) AFAS-1 and AFAS-2 (AFAS) Thermal Margin/Low Pressure (RPS) Steam Generator Pressure – Low (RPS)
3. Steam Generator Level -	Steam Generator Level – Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)
4. Pressurizer Pressure -	Pressurizer Pressure – High (RPS) Pressurizer Pressure – Low (SIAS) Thermal Margin/Low Pressure (RPS)

ACTION 15 - a. With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 4

INSERT 1

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 17 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE ~~status within 48 hours~~ or place the inoperable channel in the tripped condition and verify that the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

INSERT 1

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then place the inoperable channel in the tripped condition.

b. With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.

INSERT 5

c. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours. INSERT 1

INSERT 6

ACTION 19 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. Within 1 hour the inoperable channel is placed in either the bypassed or tripped condition. If OPERABILITY ~~can not~~cannot be restored within 48 hours, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours. INSERT 2

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If an inoperable SG level channel can not be restored to OPERABLE status within 48 hours, then AFAS-1 or AFAS-2 as applicable in the inoperable channel shall be placed in the bypassed condition. If an inoperable SG DP or FW Header DP channel can not be restored to OPERABLE status within 48 hours, then both AFAS-1 and AFAS-2 in the inoperable channel shall be placed in the bypassed condition. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

b With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.

c. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 1

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
5. CONTAINMENT SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water ~~Storage~~ Tank – Low editorial	5.67 feet above tank bottom	4.62 feet to 6.24 feet above tank bottom
c. Automatic Actuation Logic	Not Applicable	Not Applicable
6. LOSS OF POWER		
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ 3120 volts	≥ 3120 volts
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	≥ 360 volts	≥ 360 volts
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	≥ 3848 volts with < 10-second time delay	≥ 3848 volts with < 10-second time delay
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	≥ 432 volts	≥ 432 volts
7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. SG 2A & 2B Level Low	≥ 19.0%	≥ 18.0 %
8. AUXILIARY FEEDWATER ISOLATION		
a. Steam Generator ΔP – High	≤ 275 psid	89.2 to 281 psid
b. Feedwater Header ΔP – High	≤ 150.0 psid	56.0 to 157.5 psid

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4	
b. Containment Pressure – High	S	R	M	1, 2, 3	
c. Pressurizer Pressure – Low	S	R	M	1, 2, 3	
d. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4	
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4	
b. Containment Pressure – High-High	S	R	M	1, 2, 3	
c. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4	
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4	
b. Safety Injection SIAS	N.A.	N.A.	R	1, 2, 3, 4	
c. Containment Pressure – High	S	R	M	1, 2, 3	
d. Containment Radiation – High	S	R	M	1, 2, 3	
e. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4	
4. MAIN STEAM LINE ISOLATION					
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3	
b. Steam Generator Pressure – Low	S	R	M	1, 2, 3	
c. Containment Pressure – High	S	R	M	1, 2, 3	
d. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4	
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	N.A.	N.A.	R	N.A.	
b. Refueling Water ~~Storage~~ Tank – Low	S	R	M	1, 2, 3	
d. Automatic Actuation Logic editorial	N.A.	N.A.	M(1), R(3)	1, 2, 3	

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Containment Pressure	2	1
2.	Reactor Coolant Outlet Temperature – T_{Hot} (Wide Range)	2	1
3.	Reactor Coolant Inlet Temperature – T_{Cold} (Wide Range)	2	1
4.	Reactor Coolant Pressure – Wide Range	2	1
5.	Pressurizer Water Level	2	1
6.	Steam Generator Pressure	2/steam generator	1/steam generator
7.	Steam Generator Water Level – Narrow Range	1/steam generator	1/steam generator
8.	Steam Generator Water Level – Wide Range	1/steam generator*	1/steam generator*
9.	Refueling Water ~~Storage~~ Tank Water Level editorial	2	1
10.	Auxiliary Feedwater Flow Rate (Each pump)	1/pump*	1/pump*
11.	Reactor Cooling System Subcooling Margin Monitor	2	1
12.	PORV Position/Flow Indicator	2/valve***	1/valve**
13.	PORV Block Valve Position Indicator	1/valve**	1/valve**
14.	Safety Valve Position/Flow Indicator	1/valve***	1/valve***
15.	Containment Sump Water Level (Narrow Range)	1****	1****
16.	Containment Water Level (Wide Range)	2	1
17.	Incore Thermocouples	4/core quadrant	2/core quadrant
18.	Reactor Vessel Level Monitoring System	2*****	1*****

* These corresponding instruments may be substituted for each other.

** Not required if the PORV block valve is shut and power is removed from the operator.

*** If not available, monitor the quench tank pressure, level and temperature, and each safety valve/PORV discharge piping temperature at least once every 12 hours.

**** The non-safety grade containment sump water level instrument may be substituted.

***** Definition of OPERABLE: A channel consists of eight (8) sensors in a probe of which four (4) sensors must be OPERABLE.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature – T_{Hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature – T_{Cold} (Wide Range)	M	R
4. Reactor Coolant Pressure – Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Water Level – Narrow Range	M	R
8. Steam Generator Water Level – Wide Range	M	R
9. Refueling Water ~~Storage~~ Tank Water Level editorial	M	R
10. Auxiliary Feedwater Flow Rate (Each pump)	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. PORV Position/Flow Indicator	M	R
13. PORV Block Valve Position Indicator	M	R
14. Safety Valve Position/Flow Indicator	M	R
15. Containment Sump Water Level (Narrow Range)	M	R
16. Containment Water Level (Wide Range)	M	R
17. Incore Thermocouples	M	R
18. Reactor Vessel Level Monitoring System	M	R

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of ≥ 2410.3 psig and ≤ 2560.3 psig.*

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures > 230°F.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the next 6 hours.
INSERT 1

b. With two or more pressurizer code safety valves inoperable, be in HOT STANDBY within 6 hours and in HOT SHUTDOWN with all RCS cold leg temperatures at ≤ 230°F within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 Verify each pressurizer code safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within +/- 1% of 2500 psia.

format change

~~*~~

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

REACTOR COOLANT SYSTEM

3/4.4.4 PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Each Power Operated Relief Valve (PORV) Block valve shall be OPERABLE. No more than one block valve shall be open at any one time.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

INSERT 2

a. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both block valves open, close one block valve within 1 hour, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of Action a. or b. above.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS (SITs)

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 1420 and 1556 cubic feet,

c. A boron concentration of between 1900 and 2200 ppm of boron, and

d. A nitrogen cover-pressure of between 500 and 650 psig.

format change

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

ACTION:

a. With one SIT inoperable due to boron concentration not within limits, or due to an inability to verify the required water volume or nitrogen cover-pressure, restore the inoperable SIT to OPERABLE status withjo 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. INSERT 2

b. With one SIT inoperable due to reasons other than those stated in ACTION-a, restore the inoperable SIT to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. INSERT 2

INSERT 7

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

2. Verifying that each safety injection tank isolation valve is open.

format change

in MODE 3 with

NOTE

~~With pressurizer pressure greater than or equal to 1750 psia.~~ When pressurizer pressure ~~is~~ less than 1750 psia, at least three safety injection tanks shall be OPERABLE, each with a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 1250 and 1556 cubic feet with a boron concentration of between 1900 and 2200 ppm of boron. With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 833 and 1556 cubic feet with a boron concentration of between 1900 and 2200 ppm of boron.

3/4.5.2 ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure safety injection pump,

b. One OPERABLE low pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and

format correction

d. One OPERABLE charging pump~~*~~.

format change

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

ACTION:

a. 1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 1

2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 1

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

NOTE

One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e.

~~* With pressurizer pressure greater than or equal to 1750 psia.~~

format change

3/4.5.4 REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained borated water volume 477,360 gallons,

b. A boron concentration of between 1900 and 2200 ppm of boron, and

c. A solution temperature of between 55°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is less then 55°F or greater than 100°F.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

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

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

ACTION:

a. With one containment air lock door inoperable~~*~~:

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

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

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

one or both

in the affected air lock(s) and

4. The provisions of Specification 3.0.4 are not applicable.

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

INSERT 2

format change

~~*~~

NOTE

If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door. No more than one airlock door shall be open at any time.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. Each 48-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. The 8-inch containment purge supply and exhaust isolation valves may be open for purging and/or venting as required for safety related purposes such as:

1. Maintaining containment pressure within the limits of Specification 3.6.1.4.

2. Reducing containment atmosphere airborne radioactivity and/or improving air quality to an acceptable level for containment access.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With a 48-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, close and/or seal close the open valve(s) or isolate the penetration(s) within 4 hours~~,~~; otherwise be in at least HOT STANDBY within the next 6 hours and ~~in~~ COLD SHUTDOWN within the following 30 hours. INSERT 2

b. With an 8-inch containment purge supply and/or exhaust isolation valve(s) open for reasons other than those stated in Specification 3.6.1.7.b, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours~~,~~; otherwise be in at least HOT STANDBY within the next 6 hours and ~~in~~ COLD SHUTDOWN within the following 30 hours. INSERT 2

INSERT 2

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Surveillance Requirements 4.6.1.7.3 and/or 4.6.1.7.4, within 24 hours either restore the inoperable valve(s) to OPERABLE status or isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve with resilient seals or blind flange, verify the affected penetration flowpath is isolated, and perform Surveillance Requirement 4.6.1.7.3 or 4.6.1.7.4 for resilient seated valves closed to isolate the penetration flowpath, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

1. Closed and de-activated automatic valve(s) with resilient seals used to isolate the penetration flowpath(s) shall be tested in accordance with either Surveillance Requirement 4.6.1.7.3 for 48-inch valves at least once per 6 months or Surveillance Requirement 4.6.1.7.4 for 8-inch valves at least once per 92 days.

insert "following isolation"

2. Verify the affected penetration flowpath is isolated once per 31 days for isolation devices outside containment and prior to entering MODE 4 from MODE 5 for isolation devices inside containment if not performed within the previous 92 days.

format change

NOTE
Verification of isolation devices by administrative means is acceptable when they are located in high radiation areas or they are locked, sealed, or otherwise secured by administrative means.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: Containment Spray System: MODES 1, 2, and MODE 3 with Pressurizer Pressure ≥ 1750 psia.

Containment Cooling System: MODES 1, 2, and 3.

ACTION:

1. Modes 1, 2, and 3 with Pressurizer Pressure ≥ 1750 psia:

   INSERT 2

   a. With one containment spray train inoperable, restore the inoperable spray train to OPERABLE status within 72 hours ~~and within 10 days from initial discovery of failure to meet the LCO~~; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 54 hours.

   b. With one containment cooling train inoperable, restore the INSERT 2 inoperable cooling train to OPERABLE status within 7 days ~~and within 10 days from initial discovery of failure to meet the LCO~~; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   c. With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered inoperable.

   d. With two containment cooling trains inoperable, restore one cooling train to OPERABLE status within 72 hours; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   INSERT 8 INSERT 2

   ~~e. With two containment spray trains inoperable or any combination of three or more trains inoperable, enter LCO 3.0.3. immediately.~~

2. Mode 3 with Pressurizer Pressure < 1750 psia:

   a. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours; otherwise be in MODE 4 within the next 6 hours.

   b. With two containment cooling trains inoperable, enter LCO 3.0.3 immediately

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

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

INSERT 2 INSERT 2

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

INSERT 2

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

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

SURVEILLANCE REQUIREMENTS

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

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

INSERT 1

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 9

~~b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.~~

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST #2) shall be OPERABLE with a contained volume of at least 307,000 gallons.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

INSERT 1

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

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

Amendment No.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

MODE 1 - a. With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least ~~HOT STANDBY~~ MODE 2 within the next 6 hours. INSERT 2

INSERT 10

MODES 2, 3, and 4 - With one or both main steam isolation valve(s) inoperable, subsequent operation in MODES 2, 3 or 4 may proceed provided the isolation valve(s) is (are) maintained closed. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 24 hours.

The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 6.75 seconds when tested pursuant to the Inservice Testing Program.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

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

ACTION:

a. With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 11

INSERT 1

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on an SIAS test signal.

format change

NOTE

When CCW pump 2C is being used to satisfy the requirements of this specification, the alignment of the discharge valves shall be verified to be consistent with the appropriate power supply at least once per 24 hours. Upon receipt of annunciation for improper alignment of the pump 2C motor power in relation to any of its motor-operated discharge valves positions, restore proper system alignment within 2 hours.

Amendment No.

PLANT SYSTEMS

3/4.7.4 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent intake cooling water loops shall be OPERABLE.

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

ACTION:

a. With only one intake cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 12

INSERT 1

SURVEILLANCE REQUIREMENTS

4.7.4 At least two intake cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a SIAS test signal.

format change

NOTE

When ICW pump 2C is being used to satisfy the requirements of this specification, the alignment of the discharge valves must be verified to be consistent with the appropriate power supply at least once per 24 hours.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. Two separate engine-mounted fuel tanks containing a minimum volume of 200 gallons of fuel each,

2. A separate fuel storage system containing a minimum volume of 42,500 gallons of fuel, and

3. A separate fuel transfer pump.

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

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, except as provided in Action g~~f~~. below, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the offsite circuit to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

INSERT 1

b. With one diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG~~*~~; restore the diesel generator to OPERABLE status within 14 days, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

INSERT 1

format change

NOTE

If the absence of any common-cause failure cannot be confirmed, ~~this test~~SR 4.8.1.1.2a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

ELECTRICAL POWER SYSTEMS

ACTION: (Continued)

c. With one offsite A.C. circuit and one diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG~~*~~. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 ACTION Statement a or b, as appropriate, with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable A.C. power source. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

INSERT 1

format change

NOTE

If the absence of any common-cause failure cannot be confirmed, ~~this test~~SR 4.8.1.1.2a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

ELECTRICAL POWER SYSTEMS

ACTION: (Continued)

INSERT 1

d. With two of the required offsite A.C. circuits inoperable, restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow ACTION Statement a. with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable offsite A.C. circuit.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours, or be in the at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator unit, follow ACTION Statement b. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator.

INSERT 1

INSERT 13

g~~f~~. With one Unit 2 startup transformer (2A or 2B) inoperable and with a Unit 1 startup transformer (1A or 1B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 1 require the use of the startup transformer administratively available to both units, Unit 2 shall demonstrate the operability of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability; and

b. Demonstrated OPERABLE at least once per 18 months by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery bank No. 2A and a full capacity charger.

b. 125-volt Battery bank No. 2B and a full capacity charger.

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

ACTION:

INSERT 1

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.1a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

INSERT 14

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

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

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

2. The total battery terminal voltage is greater than or equal to 129-volts on float charge.

Amendment No.

ELECTRICAL POWER SYSTEMS

ACTION:

a. With one of the required trains of A.C. Emergency busses not fully energized, re-energize the train within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

b. With one A.C. Instrument Bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus: (1) re-energize the A.C. Instrument Bus within 2 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) re-energize the A.C. Instrument Bus from its associated inverter connected to its associated D.C. Bus within 24 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

INSERT 1

c. With one D.C. Bus not energized from its associated Battery Bank, re-energize the D.C. Bus from its associated Battery Bank within 2 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT 1

INSERT 15

SURVEILLANCE REQUIREMENTS

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

n. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(i) Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. A clear and bright appearance with proper color or a water and sediment content within limits;

(ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

(iii) Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

o. Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14 , Revision 1, August 1975.

$\boxed{\text{INSERT 16}}$

ATTACHMENT 4

REVISED TECHNICAL SPECIFICATIONS PAGES - UNIT 1

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq$ 1% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq 10^{-4}$% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

(e) Deleted.

(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ACTION STATEMENTS

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

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

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure – High	4	2	3	1, 2, 3	9#
c. Pressurizer Pressure – Low	4	2	3	1, 2, 3(a)	9#
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure – High-High	4	2(b)	3	1, 2, 3	10a#, 10b#, 10c, 10d
3. CONTAINMENT ISOLATION (CIS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure – High	4	2	3	1, 2, 3	9#
c. Containment Radiation – High	4	2	3	1, 2, 3, 4	9#
d. SIAS	(See Functional Unit 1 above)				
4. MAIN STEAM LINE ISOLATION (MSIS)					
a. Manual (Trip Buttons)	2/steam generator	1/steam generator	2/operating steam generator	1, 2, 3, 4	8
b. Steam Generator Pressure – Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	9#

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1725 psia; bypass shall be automatically removed when pressurizer pressure is $\geq$ 1725 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed in this MODE below 685 psig; bypass shall be automatically removed at or above 685 psig.

\# The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 8 - a. With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when second manual trip channel intentionally made inoperable.

b. With two channels inoperable, restore the inoperable channels to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 9 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the bypassed or tripped condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then place the inoperable channel in the tripped condition.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

> **NOTE**
> Actions 10c and 10d not applicable when two ore more CSAS trip units or associated instruments intentionally made inoperable

c. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With the number of channels OPERABLE two or more less than the Minimum Channels OPERABLE, restore inoperable channels to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 11 - a. With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

> **NOTE**
> Action not applicable when second AFAS manual trip or actuation logic channel intentionally made inoperable.

b. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE, restore the inoperable channels to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION 13 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If OPERABILITY cannot be restored within 48 hours or in accordance with the Risk Informed Completion Time Program, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

ACTION 14 - With the number of channels OPERABLE one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If an inoperable SG level channel can not be restored to OPERABLE status within 48 hours, then AFAS-1 or AFAS-2 as applicable in the inoperable channel shall be placed in the bypassed condition. If an inoperable SG DP or FW Header DP channel can not be restored to OPERABLE status within 48 hours, then both AFAS-1 and AFAS-2 in the inoperable channel shall be placed in the bypassed condition. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

c. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of $\geq$ 2422.8 psig and $\leq$ 2560.3 psig.

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures > 281°F.

ACTION:

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

b. With two or more pressurizer code safety valves inoperable, be in HOT STANDBY within 6 hours and in HOT SHUTDOWN with all RCS cold leg temperatures $\leq$ 281°F within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 Verify each pressurizer code safety valves is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within +/- 1% of 2500 psia.

REACTOR COOLANT SYSTEM

PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.12 Each Power Operated Relief Valve (PORV) Block Valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one or more block valve(s) inoperable, within 1 hour or in accordance with the Risk Informed Completion Time Program either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.12 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

SAFETY INJECTION TANKS (SITs)

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. Between 1090 and 1170 cubic feet of borated water,

c. A minimum boron concentration of 1900 ppm, and

d. A nitrogen cover-pressure of between 230 and 280 psig.

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

ACTION:

a. With one SIT inoperable due to boron concentration not within limits, or due to an inability to verify the required water volume or nitrogen cover-pressure, restore the inoperable SIT to OPERABLE status with 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one SIT inoperable due to reasons other than those stated in ACTION-a, restore the inoperable SIT to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

NOTE
Action not applicable when two or more SITs intentionally made inoperable.

c. With two or more SITs inoperable, restore SITs to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

2. Verifying that each safety injection tank isolation valve is open.

ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) pump,

b. One OPERABLE low-pressure safety injection pump,

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and

NOTE
One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e.

d. One OPERABLE charging pump.

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

ACTION:

a. 1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained volume 477,360 gallons of borated water,

b. A minimum boron concentration of 1900 ppm,

c. A maximum water temperature of 100°F,

d. A minimum water temperature of 55°F when in MODES 1 and 2, and

e. A minimum water temperature of 40°F when in MODES 3 and 4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

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

APPLICABILITY: MODES 1, 2, 3 and 4.

NOTE
If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door. No more than one airlock door shall be open at any time.

ACTION:

a. With one containment air lock door inoperable:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: Containment Spray System: MODES 1, 2, and MODE 3 with Pressurizer Pressure $\geq$ 1750 psia.

Containment Cooling System: MODES 1, 2, and 3.

ACTION:

1. Modes 1, 2, and 3 with Pressurizer Pressure $\geq$ 1750 psia:
   a. With one containment spray train inoperable, restore the inoperable spray train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 54 hours.
   b. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.
   c. With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered inoperable.
   d. With two containment cooling trains inoperable, restore one cooling train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

NOTE
Action not applicable when second containment spray train or three or more containment spray or cooling trains intentionally made inoperable.

   e. With two containment spray trains inoperable or any combination of three or more trains inoperable, restore containment spray trains and containment cooling trains to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.
2. Mode 3 with Pressurizer Pressure < 1750 psia:
   a. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours; otherwise be in MODE 4 within the next 6 hours.
   b. With two containment cooling trains inoperable, enter LCO 3.0.3 immediately.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves shall be OPERABLE:

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, either:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor driven feedwater pumps, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the auxiliary feedwater pump to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT SHUTDOWN within the next 12 hours.

NOTE
Action not applicable when second auxiliary feedwater pump intentionally made inoperable.

b. With two auxiliary feedwater pumps inoperable, restore at least one auxiliary feedwater pump to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank shall be OPERABLE with a minimum contained volume of 153,400 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, restore the condensate storage tank to OPERABLE status within 4 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the water level.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - a. With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in MODE 2 within the next 6 hours.

> **NOTE**
> Action not applicable when both main steam isolation valves intentionally made inoperable.

b. With both MSIVs inoperable in MODE 1, restore MSIVs to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program; otherwise, be in MODE 2 within the next 6 hours.

MODES 2 and 3 - With one or both main steam isolation valve(s) inoperable, subsequent operation in MODES 2 or 3 may proceed provided the isolation valve(s) is (are) maintained closed. Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 24 hours.

The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve that is open shall be demonstrated OPERABLE by verifying full closure within 6.0 seconds when tested pursuant to the Inservice Testing Program.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when second component cooling water loop intentionally made inoperable.

b. With two component cooling water loops inoperable, restore at least one loop to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation Signal.

PLANT SYSTEMS

3/4.7.4 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent intake cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With only one intake cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

> **NOTE**
> Action not applicable when second intake cooling water loop intentionally made inoperable.

b. With two intake cooling water loops inoperable, restore at least one loop to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two intake cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation signal.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generator sets each with:

1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

2. A separate fuel storage system containing a minimum of 19,000 gallons of fuel, and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, except as provided in Action g. below, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the offsite circuit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

NOTE
If the absence of any common-cause failure cannot be confirmed, SR 4.8.1.1.2.a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

b. With one diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG; restore the diesel generator to OPERABLE status within 14 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

ELECTRICAL POWER SYSTEMS

ACTION (continued)

> **NOTE**
> If the absence of any common-cause failure cannot be confirmed, SR 4.8.1.1.2.a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

c. With one offsite A.C. circuit and one diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 ACTION Statement a or b, as appropriate, with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable A.C. power source. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

d. With two of the required offsite A.C. circuits inoperable, restore one of the inoperable offsite sources to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow ACTION Statement a. with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable offsite A.C. circuit.

ACTION (continued)

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in the at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator unit, follow ACTION Statement b. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator.

NOTE
Action not applicable when three or more A.C. sources intentionally made inoperable.

f. With three or more A.C. sources inoperable, restore inoperable A.C. sources to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

g. With one Unit 1 startup transformer (1A or 1B) inoperable and with a Unit 2 startup transformer (2A or 2B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 2 require the use of the startup transformer administratively available to both units, Unit 1 shall demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability; and

b. Demonstrated OPERABLE at least once per 18 months by transferring (manually and automatically) unit power supply from the auxiliary transformer to the startup transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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

1. Verifying fuel level in the engine-mounted fuel tank
2. Verifying the fuel level in the fuel storage tank
3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the engine-mounted tank

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generator sets:

4160	volt Emergency Bus	1A3
4160	volt Emergency Bus	1B3
480	volt Emergency Bus	1A2
480	volt Emergency Bus	1B2
480	volt Emergency MCC Busses	1A5, 1A6, 1A7
480	volt Emergency MCC Busses	1B5, 1B6, 1B7
120	volt A.C. Instrument Bus	1MA
120	volt A.C. Instrument Bus	1MB
120	volt A.C. Instrument Bus	1MC
120	volt A.C. Instrument Bus	1MD

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying indicated power availability.

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt D.C. bus No. 1A, 125-volt Battery bank No. 1A and a full capacity charger.

b. 125-volt D.C. bus No. 1B, 125-volt Battery bank No. 1B and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks or busses inoperable, restore the inoperable battery bank or bus to OPERABLE status within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.3.2.a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

NOTE
Action not applicable when second D.C. source intentionally made inoperable.

c. With two D.C. electrical sources inoperable, restore at least one D.C. electrical source to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying indicated power availability.

4.8.2.3.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

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

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

2. The total battery terminal voltage is greater than or equal to 129-volts on float charge.

o. 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:

(i) The RICT may not exceed 30 days;

(ii) A RICT may only be utilized in MODES 1 and 2;

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

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

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

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

(iv) 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.

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

ATTACHMENT 5

REVISED TECHNICAL SPECIFICATIONS PAGES - UNIT 2

REACTIVITY CONTROL SYSTEMS

FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a boric acid makeup pump through a charging pump to the Reactor Coolant System.

b. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a gravity feed valve through a charging pump to the Reactor Coolant System.

c. The flow path from the refueling water tank via a charging pump to the Reactor Coolant System.

OR

At least two of the following three boron injection flow paths shall be OPERABLE:

d. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both boric acid makeup pumps through a charging pump to the Reactor Coolant System.

e. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both gravity feed valves through a charging pump to the Reactor Coolant System.

f. The flow path from the refueling water tank, via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200 °F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER in conjunction with (d) below; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is greater than or equal to 15% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq 10^{-4}$% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Containment Pressure – High	4	2	3	1, 2, 3	13*, 14
c. Pressurizer Pressure – Low	4	2	3	1, 2, 3(a)	13*, 14
d. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	12
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Containment Pressure – High-High	4	2	3	1(b), 2(b), 3(b)	18a*, 18b*, 18c, 18d
c. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	12
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Safety Injection (SIAS)	See Functional Unit 1 for all Safety Injection Initiating Functions and Requirements				
c. Containment Pressure – High	4	2	3	1, 2, 3	13*, 14
d. Containment Radiation – High	4	2	3	1, 2, 3	13*, 14
e. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION (MSIS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3	16
b. Steam Generator Pressure – Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	13*, 14
c. Containment Pressure – High	4	2	3	1, 2, 3	13*, 14
d. Automatic Actuation Logic	2	1	2	1, 2, 3	12
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	19
c. Automatic Actuation Logic	2	1	2	1, 2, 3	12

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than 1836 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to 1836 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed in this MODE below 700 psia; bypass shall be automatically removed at or above 700 psia.

* The provisions of Specification 3.0.4 are not applicable.

ACTION OF STATEMENTS

ACTION 12 - a. With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when second manual trip channel intentionally made inoperable.

b. When two channels inoperable, restore the inoperable channels to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit	Functional Unit Bypassed
1. Containment Pressure -	Containment Pressure – High (SIAS, CIAS, CSAS) Containment Pressure – High (RPS)
2. Steam Generator Pressure -	Steam Generator Pressure – Low (MSIS) AFAS-1 and AFAS-2 (AFAS) Thermal Margin/Low Pressure (RPS) Steam Generator Pressure – Low (RPS)
3. Steam Generator Level -	Steam Generator Level – Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)
4. Pressurizer Pressure -	Pressurizer Pressure – High (RPS) Pressurizer Pressure – Low (SIAS) Thermal Margin/Low Pressure (RPS)

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below.

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Containment Pressure -	Containment Pressure – High (SIAS, CIAS, CSAS) Containment Pressure – High (RPS)
2. Steam Generator Pressure -	Steam Generator Pressure – Low (MSIS) AFAS-1 and AFAS-2 (AFAS) Thermal Margin/Low Pressure (RPS) Steam Generator Pressure – Low (RPS)
3. Steam Generator Level -	Steam Generator Level – Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)
4. Pressurizer Pressure -	Pressurizer Pressure – High (RPS) Pressurizer Pressure – Low (SIAS) Thermal Margin/Low Pressure (RPS)

ACTION 15 - a. With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

NOTE
Action not applicable when second AFAS manual trip or actuation logic channel intentionally made inoperable.

b. With two channels inoperable, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 17 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or place the inoperable channel in the tripped condition and verify that the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then place the inoperable channel in the tripped condition.

b. With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.

NOTE
Actions 18c and 18d not applicable when two or more CSAS trip units or associated instruments intentionally made inoperable.

c. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With the number of channels OPERABLE two or more less than the Minimum Channels OPERABLE, restore inoperable channels to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 19 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. Within 1 hour the inoperable channel is placed in either the bypassed or tripped condition. If OPERABILITY cannot be restored within 48 hours or in accordance with the Risk Informed Completion Time Program, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If an inoperable SG level channel can not be restored to OPERABLE status within 48 hours, then AFAS-1 or AFAS-2 as applicable in the inoperable channel shall be placed in the bypassed condition. If an inoperable SG DP or FW Header DP channel can not be restored to OPERABLE status within 48 hours, then both AFAS-1 and AFAS-2 in the inoperable channel shall be placed in the bypassed condition. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

b With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.

c. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
5. CONTAINMENT SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank – Low	5.67 feet above tank bottom	4.62 feet to 6.24 feet above tank bottom
c. Automatic Actuation Logic	Not Applicable	Not Applicable
6. LOSS OF POWER		
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	$\geq$ 3120 volts	$\geq$ 3120 volts
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	$\geq$ 360 volts	$\geq$ 360 volts
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	$\geq$ 3848 volts with < 10-second time delay	$\geq$ 3848 volts with < 10-second time delay
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	$\geq$ 432 volts	$\geq$ 432 volts
7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. SG 2A & 2B Level Low	$\geq$ 19.0%	$\geq$ 18.0 %
8. AUXILIARY FEEDWATER ISOLATION		
a. Steam Generator ΔP – High	$\leq$ 275 psid	89.2 to 281 psid
b. Feedwater Header ΔP – High	$\leq$ 150.0 psid	56.0 to 157.5 psid

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Pressure – High	S	R	M	1, 2, 3
c. Pressurizer Pressure – Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Pressure – High-High	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Safety Injection SIAS	N.A.	N.A.	R	1, 2, 3, 4
c. Containment Pressure – High	S	R	M	1, 2, 3
d. Containment Radiation – High	S	R	M	1, 2, 3
e. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4
4. MAIN STEAM LINE ISOLATION				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. Steam Generator Pressure – Low	S	R	M	1, 2, 3
c. Containment Pressure – High	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4
5. CONTAINMENT SUMP RECIRCULATION (RAS)				
a. Manual RAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Refueling Water Tank – Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Containment Pressure	2	1
2.	Reactor Coolant Outlet Temperature – T_{Hot} (Wide Range)	2	1
3.	Reactor Coolant Inlet Temperature – T_{Cold} (Wide Range)	2	1
4.	Reactor Coolant Pressure – Wide Range	2	1
5.	Pressurizer Water Level	2	1
6.	Steam Generator Pressure	2/steam generator	1/steam generator
7.	Steam Generator Water Level – Narrow Range	1/steam generator	1/steam generator
8.	Steam Generator Water Level – Wide Range	1/steam generator*	1/steam generator*
9.	Refueling Water Tank Water Level	2	1
10.	Auxiliary Feedwater Flow Rate (Each pump)	1/pump*	1/pump*
11.	Reactor Cooling System Subcooling Margin Monitor	2	1
12.	PORV Position/Flow Indicator	2/valve***	1/valve**
13.	PORV Block Valve Position Indicator	1/valve**	1/valve**
14.	Safety Valve Position/Flow Indicator	1/valve***	1/valve***
15.	Containment Sump Water Level (Narrow Range)	1****	1****
16.	Containment Water Level (Wide Range)	2	1
17.	Incore Thermocouples	4/core quadrant	2/core quadrant
18.	Reactor Vessel Level Monitoring System	2*****	1*****

* These corresponding instruments may be substituted for each other.

** Not required if the PORV block valve is shut and power is removed from the operator.

*** If not available, monitor the quench tank pressure, level and temperature, and each safety valve/PORV discharge piping temperature at least once every 12 hours.

**** The non-safety grade containment sump water level instrument may be substituted.

***** Definition of OPERABLE: A channel consists of eight (8) sensors in a probe of which four (4) sensors must be OPERABLE.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature – T_{Hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature – T_{Cold} (Wide Range)	M	R
4. Reactor Coolant Pressure – Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Water Level – Narrow Range	M	R
8. Steam Generator Water Level – Wide Range	M	R
9. Refueling Water Tank Water Level	M	R
10. Auxiliary Feedwater Flow Rate (Each pump)	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. PORV Position/Flow Indicator	M	R
13. PORV Block Valve Position Indicator	M	R
14. Safety Valve Position/Flow Indicator	M	R
15. Containment Sump Water Level (Narrow Range)	M	R
16. Containment Water Level (Wide Range)	M	R
17. Incore Thermocouples	M	R
18. Reactor Vessel Level Monitoring System	M	R

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

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

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of $\geq$ 2410.3 psig and $\leq$ 2560.3 psig.

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures > 230°F.

ACTION:

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

b. With two or more pressurizer code safety valves inoperable, be in HOT STANDBY within 6 hours and in HOT SHUTDOWN with all RCS cold leg temperatures at $\leq$ 230°F within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 Verify each pressurizer code safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within +/- 1% of 2500 psia.

REACTOR COOLANT SYSTEM

3/4.4.4 PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Each Power Operated Relief Valve (PORV) Block valve shall be OPERABLE. No more than one block valve shall be open at any one time.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more block valve(s) inoperable, within 1 hour or in accordance with the Risk Informed Completion Time Program either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both block valves open, close one block valve within 1 hour, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of Action a. or b. above.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS (SITs)

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 1420 and 1556 cubic feet,

c. A boron concentration of between 1900 and 2200 ppm of boron, and

d. A nitrogen cover-pressure of between 500 and 650 psig.

NOTE
When in MODE 3 with pressurizer pressure less than 1750 psia, at least three safety injection tanks shall be OPERABLE, each with a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 1250 and 1556 cubic feet with a boron concentration of between 1900 and 2200 ppm of boron. With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 833 and 1556 cubic feet with a boron concentration of between 1900 and 2200 ppm of boron.

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

ACTION:

a. With one SIT inoperable due to boron concentration not within limits, or due to an inability to verify the required water volume or nitrogen cover-pressure, restore the inoperable SIT to OPERABLE status with 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one SIT inoperable due to reasons other than those stated in ACTION-a, restore the inoperable SIT to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

NOTE
Action not applicable when two or more SITs intentionally made inoperable.

c. With two or more SITs inoperable, restore SITs to OPERABLE status within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the borated water volume and nitrogen cover-pressure in the tanks are within their limits, and
2. Verifying that each safety injection tank isolation valve is open.

3/4.5.2 ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure safety injection pump,

b. One OPERABLE low pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and

> **NOTE**
> One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e.

d. One OPERABLE charging pump.

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

ACTION:

a. 1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours..

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained borated water volume 477,360 gallons,

b. A boron concentration of between 1900 and 2200 ppm of boron, and

c. A solution temperature of between 55°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is less then 55°F or greater than 100°F.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

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

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

ACTION:

> **NOTE**
> If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door. No more than one airlock door shall be open at any time.

a. With one containment air lock door inoperable:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

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

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. Each 48-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. The 8-inch containment purge supply and exhaust isolation valves may be open for purging and/or venting as required for safety related purposes such as:

1. Maintaining containment pressure within the limits of Specification 3.6.1.4.

2. Reducing containment atmosphere airborne radioactivity and/or improving air quality to an acceptable level for containment access.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With a 48-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, close and/or seal close the open valve(s) or isolate the penetration(s) within 4 hours or in accordance with the Risk Informed Completion Time Program, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With an 8-inch containment purge supply and/or exhaust isolation valve(s) open for reasons other than those stated in Specification 3.6.1.7.b, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Surveillance Requirements 4.6.1.7.3 and/or 4.6.1.7.4, within 24 hours or in accordance with the Risk Informed Completion Time Program either restore the inoperable valve(s) to OPERABLE status or isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve with resilient seals or blind flange, verify the affected penetration flowpath is isolated, and perform Surveillance Requirement 4.6.1.7.3 or 4.6.1.7.4 for resilient seated valves closed to isolate the penetration flowpath, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

1. Closed and de-activated automatic valve(s) with resilient seals used to isolate the penetration flowpath(s) shall be tested in accordance with either Surveillance Requirement 4.6.1.7.3 for 48-inch valves at least once per 6 months or Surveillance Requirement 4.6.1.7.4 for 8-inch valves at least once per 92 days.

NOTE
Verification of isolation devices by administrative means is acceptable when they are located in high radiation areas or they are locked, sealed, or otherwise secured by administrative means.

2. Verify the affected penetration flowpath is isolated once per 31 days following isolation for isolation devices outside containment and prior to entering MODE 4 from MODE 5 for isolation devices inside containment if not performed within the previous 92 days.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: Containment Spray System: MODES 1, 2, and MODE 3 with Pressurizer Pressure ≥ 1750 psia.

Containment Cooling System: MODES 1, 2, and 3.

ACTION:

1. Modes 1, 2, and 3 with Pressurizer Pressure ≥ 1750 psia:

   a. With one containment spray train inoperable, restore the inoperable spray train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 54 hours.

   b. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   c. With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered inoperable.

   d. With two containment cooling trains inoperable, restore one cooling train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

NOTE
Action not applicable when second containment spray train or three or more containment spray or cooling trains intentionally made inoperable.

   e. With two containment spray trains inoperable or any combination of three or more trains inoperable, restore containment spray trains and containment cooling trains to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

2. Mode 3 with Pressurizer Pressure < 1750 psia:

   a. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours; otherwise be in MODE 4 within the next 6 hours.

   b. With two containment cooling trains inoperable, enter LCO 3.0.3 immediately.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

NOTE
Action not applicable when second auxiliary feedwater pump intentionally made inoperable.

b. With two auxiliary feedwater pumps inoperable, restore at least one auxiliary feedwater pump to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST #2) shall be OPERABLE with a contained volume of at least 307,000 gallons.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

MODE 1 - a. With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least MODE 2 within the next 6 hours.

NOTE
Action not applicable when both main steam isolation valves intentionally made inoperable.

b. With both main steam line isolation valves inoperable in MODE 1, restore main steam isolation valves to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program; otherwise, be in MODE 2 within the next 6 hours.

MODES 2, 3 and 4 - With one or both main steam isolation valve(s) inoperable, subsequent operation in MODES 2, 3 or 4 may proceed provided the isolation valve(s) is (are) maintained closed. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 24 hours.

The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 6.75 seconds when tested pursuant to the Inservice Testing Program.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

NOTE
When CCW pump 2C is being used to satisfy the requirements of this specification, the alignment of the discharge valves shall be verified to be consistent with the appropriate power supply at least once per 24 hours. Upon receipt of annunciation for improper alignment of the pump 2C motor power in relation to any of its motor-operated discharge valves positions, restore proper system alignment within 2 hours.

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

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

ACTION:

a. With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when second component cooling water loop intentionally made inoperable.

b. With two component cooling water loops inoperable, restore at least one loop to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on an SIAS test signal.

PLANT SYSTEMS

3/4.7.4 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

> **NOTE**
> When ICW pump 2C is being used to satisfy the requirements of this specification, the alignment of the discharge valves must be verified to be consistent with the appropriate power supply at least once per 24 hours.

3.7.4 At least two independent intake cooling water loops shall be OPERABLE.

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

ACTION:

a. With only one intake cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

> **NOTE**
> Action not applicable when second intake cooling water loop intentionally made inoperable.

b. With two intake cooling water loops inoperable, restore at least one loop to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two intake cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a SIAS test signal.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. Two separate engine-mounted fuel tanks containing a minimum volume of 200 gallons of fuel each,

2. A separate fuel storage system containing a minimum volume of 42,500 gallons of fuel, and

3. A separate fuel transfer pump.

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

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, except as provided in Action g. below, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the offsite circuit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

NOTE
If the absence of any common-cause failure cannot be confirmed, SR 4.8.1.1.2a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

b. With one diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG; restore the diesel generator to OPERABLE status within 14 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

ELECTRICAL POWER SYSTEMS

ACTION: (Continued)

NOTE
If the absence of any common-cause failure cannot be confirmed, SR 4.8.1.1.2a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

c. With one offsite A.C. circuit and one diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 ACTION Statement a or b, as appropriate, with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable A.C. power source. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

ELECTRICAL POWER SYSTEMS

ACTION: (Continued)

d. With two of the required offsite A.C. circuits inoperable, restore one of the inoperable offsite sources to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow ACTION Statement a. with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable offsite A.C. circuit.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in the at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator unit, follow ACTION Statement b. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator.

NOTE
Action not applicable when three or more A.C. sources intentionally made inoperable.

f. With three or more A.C. sources inoperable, restore inoperable A.C. sources to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

g. With one Unit 2 startup transformer (2A or 2B) inoperable and with a Unit 1 startup transformer (1A or 1B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 1 require the use of the startup transformer administratively available to both units, Unit 2 shall demonstrate the operability of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability; and

b. Demonstrated OPERABLE at least once per 18 months by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery bank No. 2A and a full capacity charger.

b. 125-volt Battery bank No. 2B and a full capacity charger.

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

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.1a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

NOTE
Action not applicable when second D.C. source intentionally made inoperable.

c. With two D.C. electrical sources inoperable, restore at least one D.C. electrical source to OPERABLE status within 1 hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

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

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

2. The total battery terminal voltage is greater than or equal to 129-volts on float charge.

ELECTRICAL POWER SYSTEMS

ACTION:

a. With one of the required trains of A.C. Emergency busses not fully energized, re-energize the train within 8 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. Instrument Bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus: (1) re-energize the A.C. Instrument Bus within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) re-energize the A.C. Instrument Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. Bus not energized from its associated Battery Bank, re-energize the D.C. Bus from its associated Battery Bank within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when more than one A.C. vital panel intentionally either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus.

d. With more than one A.C. vital panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital panels within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital panels from their associated inverters connected to their associated D.C. buses within one hour or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

n. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(i) Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,
2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
3. A clear and bright appearance with proper color or a water and sediment content within limits;

(ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

(iii) Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

o. Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14 , Revision 1, August 1975.

p. 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:

(i) The RICT may not exceed 30 days;

(ii) A RICT may only be utilized in MODES 1 and 2;

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

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.
2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.
3. Revising the RICT is not required If the plant configuration change would lower plant risk and would result in a longer RICT.

(iv) 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.

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

ATTACHMENT 6

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS BASES - UNIT 1 (MARK-UPS)

FOR INFORMATION ONLY

INSERT 1

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

INSERT 2

or in accordance with the Risk Informed Completion Time Program

INSERT 3 - B 3.5.1, Safety Injection Tanks

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

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

INSERT 4 - B 3.6.2.1, Containment Spray and Cooling Systems

With two containment spray trains or any combination of three or more Containment Spray System and Containment Cooling System trains inoperable, sufficient containment spray trains and/or containment cooling trains must be restored to OPERABLE status so that no more than one containment spray train or two containment cooling trains are inoperable within one hour or in accordance with the Risk Informed Completion Time Program. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of sufficient trains. In MODE 3 with Pressurizer Pressure < 1750 psia, containment spray is not required.

Action 1.e is modified by a Note stating it is not applicable when two containment spray trains or any combination of three or more containment spray and cooling trains are intentionally made inoperable. This Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one containment spray train or a combination of two containment spray and cooling trains are inoperable for any reason and a second containment spray train or additional containment spray or cooling trains are found to be inoperable, or if two containment spray trains or any combination of three or more containment spray and cooling trains are found to be inoperable at the same time.

INSERT 5 - B 3.7.1.5, Main Steam Line Isolation Valves

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

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

INSERT 6 - B 3.7.1.2, Auxiliary Feedwater Pumps

With two of three AFW pumps inoperable in MODE 1, 2, or 3, the Required Action is to restore at least one of the inoperable AFW pumps 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 AFW pump. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Action is modified by a Note stating it is not applicable when the second AFW pump 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 AFW pump is inoperable for any reason and a second AFW pump is found to be inoperable, or if two AFW pumps are found to be inoperable at the same time.

INSERT 7 - B 3.7.3.1, Component Cooling Water System

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

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

INSERT 8 - B 3.7.4.1, Intake Cooling Water System

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

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

INSERT 9 - B 3.8.1.1, A.C. Sources - Operating

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

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

INSERT 10 - B 3.8.2.3, D.C. Distribution - Operating

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

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

SECTION NO.: 3/4.5	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 7 OF ADM-25.04 EMERGENCY CORE COOLING SYSTEMS (ECCS) ST. LUCIE UNIT 1	PAGE: 3 of 6
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BASES FOR SECTION 3/4.5

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

(SITs)

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

INSERT 1

The limit of 72 hours for operation with an SIT that is inoperable due to boron concentration not within limits, or due to the inability to verify liquid volume or cover-pressure, considers that the volume of the SIT is still available for injection in the event of a LOCA. If one SIT is inoperable for other reasons, the SIT may be unable to perform its safety function and, based on probability risk assessment, operation in this condition is limited to 24 hours. INSERT 2

INSERT 3

SECTION NO.: 3/4.5	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 7 OF ADM-25.04 EMERGENCY CORE COOLING SYSTEMS (ECCS) ST. LUCIE UNIT 1	PAGE: 4 of 6
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3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) (continued)

BASES (continued)

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

TS 3.5.2.c and 3.5.3.a require that ECCS subsystem(s) have an independent OPERABLE flow path capable of automatically transferring suction to the containment sump on a Recirculation Actuation Signal. The containment sump is defined as the area of containment below the minimum flood level in the vicinity of the containment sump strainers. Therefore, the LCOs are satisfied when an independent OPERABLE flow path to the containment sump strainer is available.

TS 3.5.2.d requires that an ECCS subsystem(s) have OPERABLE charging pump and associated flow path from the BAMT(s). Reference to TS 3.1.2.2 requires that the Train A charging pump flowpath is from the BAMT(s) through the boric acid makeup pump(s). The Train B charging pump flowpath is from the BAMT(s) through the gravity feed valve(s). #3

TS 3.5.2, ACTION a.1. provides an allowed outage/action completion time (AOT) of up to 7 days from initial discovery of failure to meet the LCO provided the affected ECCS subsystem is inoperable only because its associated LPSI train is inoperable. INSERT 1 ~~This 7 day AOT is based on the findings of a deterministic and probabilistic safety analysis and is referred to as a "risk informed" AOT extension. Entry into this ACTION requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP) which is described in the Administrative Procedure (ADM 17.08) that implements the Maintenance Rule pursuant to 10 CFR 50.65.~~

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained.

SECTION NO.: 3/4.6	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 8 OF ADM-25.04 CONTAINMENT SYSTEMS ST. LUCIE UNIT 1	PAGE: 5 of 10
REVISION NO.: ~~8~~		

3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS

The OPERABILITY of the containment spray and cooling systems ensures that depressurization and cooling capability will be available to limit post-accident pressure and temperature in the containment to acceptable values. During a Design Basis Accident (DBA), at least one containment cooling train and one containment spray train are capable of maintaining the peak pressure and temperature within design limits. One containment spray train has the capability, in conjunction with the Spray Additive System, to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analyses. To ensure that these conditions can be met considering single-failure criteria, two spray trains and two cooling trains must be OPERABLE.

The 72 hour action interval specified in ACTION 1.a and ACTION 1.d, and the 7 day action interval specified in ACTION 1.b take into account the redundant heat removal capability and the iodine removal capability of the remaining operable systems, and the low probability of a DBA occurring during this period. INSERT 1 ~~The 10 day constraint for ACTIONS 1.a and 1.b is based on coincident entry into two ACTION conditions (specified in ACTION 1.c) coupled with the low probability of an accident occurring during this time.~~ If the system(s) cannot be restored to OPERABLE status within the specified completion time, alternate actions are designed to bring the unit to a mode for which the LCO does not apply. The extended interval (54 hours) specified in ACTION 1.a to be in MODE 4 includes 48 hours of additional time for restoration of the inoperable CS train, and takes into consideration the reduced driving force for a release of radioactive material from the RCS when in MODE 3. ~~With two containment spray trains or any combination of three or more containment spray and containment cooling trains inoperable in MODES 1, 2, or Mode 3 with Pressurizer Pressure ≥ 1750 psia, the unit is in a condition outside the accident analyses and LCO 3.0.3 must be entered immediately. In MODE 3 with Pressurizer Pressure < 1750 psia, containment spray is not required.~~

INSERT 4

The specifications and bases for LCO 3.6.2.1 are consistent with NUREG-1432, Revision 0 (9/28/92), Specification 3.6.6A (Containment Spray and Cooling Systems; Credit taken for iodine removal by the Containment Spray System), and the plant safety analyses.

SECTION NO.: 3/4.7	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 9 OF ADM-25.04 PLANT SYSTEMS ST. LUCIE UNIT 1	PAGE: 5 of 13
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3/4.7 PLANT SYSTEMS (continued)

<u>BASES</u> (continued)

3/4.7.1 TURBINE CYCLE (continued)

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. The dose calculations for an assumed steam line rupture include the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves (MSIVs) ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

INSERT 5

3/4.7.1.6 SECONDARY WATER CHEMISTRY

This section left blank intentionally.

SECTION NO.: 3/4.7	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 9 OF ADM-25.04 PLANT SYSTEMS ST. LUCIE UNIT 1	PAGE: 4 of 13
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3/4.7 PLANT SYSTEMS (continued)

BASES (continued)

3/4.7.1 TURBINE CYCLE (continued)

106.5 = Power Level-High Trip Setpoint for two loop operation

X = Total relieving capacity of all safety valves per steam line in lbs/hour (6.192×10^6 lbs/hr.)

Y = Maximum relieving capacity of any one safety valve in lbs/hour (7.74×10^5 lbs/hr.)

Surveillance Requirement 4.7.1.1 verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The MSSV setpoints are 1000 psia +/-3% (4 valves each header) and 1040 psia +2/-3% (4 valves each header) for OPERABILITY; however, the valves are reset to 1000 psia +/-1% and 1040 psia +/- 1%, respectively, during the Surveillance to allow for drift. The LCO is expressed in units of psig for consistency with implementing procedures. #4

The provisions of Specification 3.0.4 do not apply. This allows entry into and operation in MODE 3 prior to performing the Surveillance Requirements so that the MSSVs may be tested under hot conditions.

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS (AFW)

The OPERABILITY of the auxiliary feedwater pumps ensures that the Reactor Coolant System can be cooled down to less than 325°F from normal operating conditions in the event of a total loss of off-site power.

Any two of the three auxiliary feedwater pumps have the required capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS temperature to 325°F where the shutdown cooling system may be placed into operation for continued cooldown.

INSERT 6

3/4.7.1.3 CONDENSATE STORAGE TANKS

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain HOT STANDBY for one hour and then cooldown of the Reactor Coolant System to less than 325°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere. The minimum usable volume to satisfy the criteria stated above is 130,500 gallons, which is ensured by the LCO for the CST volume of 153,400 gallons. #4

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3/4.7 PLANT SYSTEMS (continued)

BASES (continued)

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200-psig are based on a steam generator RT_{NDT} of 50°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

(CCW)

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident

INSERT 7

3/4.7.4 INTAKE COOLING WATER SYSTEM

(ICW)

The OPERABILITY of the intake cooling water system ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

INSERT 8

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

The limitation on minimum water level is based on providing an adequate cooling water supply to safety related equipment until cooling water can be supplied from Big Mud Creek.

Cooling capacity calculations are based on an ultimate heat sink temperature of 95°F. It has been demonstrated by a temperature survey conducted from March 1976 to May 1981 that the Atlantic Ocean has never risen higher than 86°F. Based on this conservatism, no ultimate heat sink temperature limitation is specified.

SECTION NO.: 3/4.8	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 10 OF ADM-25.04 ELECTRICAL POWER SYSTEMS ST. LUCIE UNIT 1	PAGE: 4 of 7
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3/4.8 ELECTRICAL POWER SYSTEMS (continued)

BASES (continued)

TS 3.8.1.1, ACTION "b" provides an allowed outage/action completion time (AOT) of up to 14 days tote restore a single inoperable diesel generator to operable status. INSERT 1 ~~This AOT is based on the findings of a deterministic and probabilistic safety analysis and is referred to as a "risk informed" AOT. Entry into this action requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP), which is described in the Administrative Procedure that implements the Maintenance Rule pursuant to 10 CFR 50.65.~~

All EDG inoperabilities must be investigated for common-cause failures regardless of how long the EDG inoperability persists. When one diesel generator is inoperable, required ACTIONS 3.8.1.1.b and 3.8.1.1.c provide an allowance to avoid unnecessary testing of EDGs. If it can be determined that the cause of the inoperable EDG does not exist on the remaining OPERABLE EDG, then SR 4.8.1.1.2.a.4 does not have to be performed. Eight (8) hours is reasonable to confirm that the OPERABLE EDG is not affected by the same problem as the inoperable EDG. If it cannot otherwise be determined that the cause of the initial inoperable EDG does not exist on the remaining EDG, then satisfactory performance of SR 4.8.1.1.2.a.4 suffices to provide assurance of continued OPERABILITY of that EDG. If the cause of the initial inoperability exists on the remaining OPERABLE EDG, that EDG would also be declared inoperable upon discovery, and ACTION 3.8.1.1.e would be entered. Once the failure is repaired (on either EDG), the common-cause failure no longer exists.

Ambient conditions are the normal standby conditions for the diesel engines. Any normally running warmup systems should be in service and operating, and manufacturer's recommendations for engine oil and water temperatures and other parameters should be followed.

INSERT 9

INSERT 10

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

decrease margin

The Surveillance Requirements for demonstrating the OPERABILTY of the DC system battery cell interconnection resistances are based on criteria recommended by the manufacturer. The table contained in TSSR 4.8.2.3.2.c.3 is provided to define the maximum individual and maximum average allowable values for battery cell interconnection resistances.

3/4.8 ELECTRICAL POWER SYSTEMS (continued)

BASES (continued)

The Surveillance Requirements for demonstrating the OPERABILTY of the DC system battery cell interconnection resistances are based on criteria recommended by the manufacturer. The table contained in TSSR 4.8.2.3.2.c.3 is provided to define the maximum individual and maximum average allowable values for battery cell interconnection resistances.

decrease margin

The maximum individual battery cell interconnection resistance values are based on the negligible impact of voltage drop and connection heating, during peak DC system load conditions. A maximum individual battery interconnection resistance value of ≤ 150 x 10-6 ohms is used for connections, which use inter-cell (bus-bar type) connections and for the battery set output terminal connections. The maximum individual battery interconnection resistance value of ≤ 200 x 10-6 ohms is used for the inter-tier and inter-rack connections, which are subject to additional resistance of the cables used to extend between the different level tiers of each battery rack and of the adjacent battery rack.

The maximum average battery cell interconnection resistance value of ≤ 50 x 10-6 ohms is the average of the interconnection resistance limit for all inter-cell, inter-tier, inter-rack and output terminals in the series-connected battery bank string. The ≤ 50 x 10-6 ohms criteria was selected in order to ensure that the battery cell interconnection voltage drop does not exceed the vendor criteria limit of less than 33.66 mV (average) for each battery cell interconnection, during the maximum design current load profile. The battery manufacturer has rated the battery bank set for full rated output, given adherence to limiting the average interconnection resistance to less than 33.66 mV drop between cells. For battery cell interconnections, which are monitored via multiple measurement points between two adjacent cells, these measurements must first be averaged for the connection between the affected adjacent cells, before averaging the values for all cells used in the full battery bank set.

ATTACHMENT 7

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS BASES - UNIT 2 (MARK-UPS)

FOR INFORMATION ONLY

INSERT 1

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

INSERT 2

or in accordance with the Risk Informed Completion Time Program

INSERT 3 - B 3.3.2, Engineered Safety Features Actuation System Instrumentation

If two auxiliary feedwater actuation system (AFAS) manual trip or actuation logic channels are inoperable, the Action is to restore at least one channel to OPERABLE status within 1 hour. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of at least one channel. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

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

With three or more containment spray actuation system (CSAS) trip units or associated instruments inoperable (i.e., two or more channels less than the Minimum Channels OPERABLE requirement) the Action is to restore sufficient trip units or associated instruments to OPERABLE status within 1 hour to restore the containment spray actuation system initiation function. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration of sufficient channels to restore initiation function. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

Actions 18c (two CSAS trip units inoperable) and 18d (three or more CSAS trip units inoperable) are modified by a Note stating the Action is not applicable when two or more CSAS trip units or associated instruments are intentionally made inoperable. These Actions are not intended for voluntary removal of redundant systems or components from service. The Actions are only applicable if one CSAS trip unit or associated instrument is inoperable for any reason and additional CSAS trip units or associated instruments are found to be inoperable, or if two or more CSAS trip units or associated instruments are found to be inoperable at the same time.

INSERT 4 - B 3.5.1, Safety Injection Tanks

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

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

INSERT 5 - B 3.6.2.1, Containment Spray and Cooling Systems

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

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

INSERT 6 - B 3.7.1.2, Auxiliary Feedwater

With two of three AFW pumps inoperable in MODE 1, 2, or 3, the Required Action is to restore at least one of the inoperable AFW pumps 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 AFW pump. Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.
The Action is modified by a Note stating it is not applicable when the second AFW pump 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 AFW pump is inoperable for any reason and a second AFW pump is found to be inoperable, or if two AFW pumps are found to be inoperable at the same time.

INSERT 7 - B 3.7.1.5, Main Steam Line Isolation Valves

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

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

INSERT 8 - B 3.7.3, Component Cooling Water System

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

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

INSERT 9 - B 3.7.4, Intake Cooling Water System

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

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

INSERT 10 - B 3.8.1.1, A.C. Sources - Operating

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

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

INSERT 11 - B 3.8.2.1, D.C. Sources

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

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

INSERT 12 - B 3.8.3.1 Onsite Power Distribution - Operating

Onsite Power Distribution

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

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

SECTION NO.: 3/4.3	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 5 OF ADM-25.04 INSTRUMENTATION ST. LUCIE UNIT 2	PAGE: 3 of 6
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BASES FOR SECTION 3/4.3

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensure that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

INSERT 3

The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. For the Steam Generator Water Level Low Functional Unit, the trip setpoint and methodology used to determine the trip setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in the UFSAR. The two table notations are consistent with the recommended notes provided in NRC's letter to NEI Technical Specifications Methods Task Force for Setpoint Allowances dated September 5, 2005.

CE Owners Group topical report CEN-403, Revision 1-A, March 1996, provides the basis to allow ESFAS subgroup relay testing on a STAGGERED TEST BASIS. Such testing requires each subgroup relay to be tested at least once per 18 months (refueling cycle), with approximately equal numbers of relays being tested at 6 month subintervals. Subgroup relays which cannot be tested with the unit at power should be scheduled for testing during plant shutdowns. If two or more ESFAS subgroup relays fail in a 12-month period, the design, maintenance, and testing of all ESFAS subgroup relays should be considered to evaluate the adequacy of the surveillance interval. If it is determined that the surveillance interval is inadequate for detecting a single relay failure, the surveillance interval should be decreased such that an ESFAS subgroup relay failure prior to occurrence of a second failure can be detected.

SECTION NO.: 3/4.5	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 7 OF ADM-25.04 EMERGENCY CORE COOLING SYSTEMS (ECCS) ST. LUCIE UNIT 2	PAGE: 3 of 7
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BASES FOR SECTION 3/4.5

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

(SITs)

The OPERABILITY of each of the Reactor Coolant System (RCS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the assumptions used for safety injection tank injection in the safety analysis are met.

The safety injection tank power-operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limit of 72 hours for operation with an SIT that is inoperable due to boron concentration not within limits, or due to the inability to verify liquid volume or cover-pressure, considers that the volume of the SIT is still available for injection in the event of a LOCA. ~~If one SIT is inoperable for other reasons, the SIT may be unable to perform its safety function and, based on probability risk assessment, operation in this condition is limited to 24 hours.~~

INSERT 1

INSERT 2

INSERT 4

The practice of calibrating and testing the SIT isolation valve interlock function below 515 psia (the current plant practice is to set and test the interlock function at 500 psia) meets the requirements of Technical Specification Surveillance 4.5.1.1.d.1. The staff accepted that testing the SIT isolation interlock at a more conservative setpoint demonstrates operability at and above the setpoint (NRC letter from William C. Gleaves to J.A. Stall dated November 2, 1999, subject "St. Lucie Unit 2 – Amendment Request Regarding Safety Injection Tank and Shutdown Cooling System Isolation Interlock Surveillances (TAC No. MA5619)."

SECTION NO.: 3/4.5	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 7 OF ADM-25.04 EMERGENCY CORE COOLING SYSTEMS (ECCS) ST. LUCIE UNIT 2	PAGE: 4 of 7
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3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) (continued)

BASES (continued)

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS hot leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

TS 3.5.2.c and 3.5.3 require that ECCS subsystem(s) have an independent OPERABLE flow path capable of automatically transferring suction to the containment on a Recirculation Actuation Signal. The containment sump is defined as the area of containment below the minimum flood level in the vicinity of the containment sump strainers. Therefore, the LCOs are satisfied when an independent OPERABLE flow path to the containment sump strainer is available. R2

TS 3.5.2.d requires that an ECCS subsystem(s) have an OPERABLE charging pump and associated flow path from the BAMT(s). Reference to TS 3.1.2.2 requires that the one charging pump flow path is from the BAMT(s) through the boric acid makeup pump(s). The second charging pump flowpath is from the BAMT(s) through the gravity feed valves. R2

TS 3.5.2, ACTION a.1. provides an allowed outage/action completion time (AOT) of up to 7 days from initial discovery of failure to meet the LCO provided the affected ECCS subsystem is inoperable only because its associated LPSI train is inoperable. INSERT 1 ~~This 7 day AOT is based on the findings of a deterministic and probabilistic safety analysis and is referred to as a "risk informed" AOT extension. Entry into this ACTION requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP) which is described in the Administrative Procedure (ADM-17.08) that implements the Maintenance Rule pursuant to 10 CFR 50.65.~~

In Mode 3 with RCS pressure < 1750 psia and in Mode 4, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS

The OPERABILITY of the containment spray and cooling systems ensures that depressurization and cooling capability will be available to limit post-accident pressure and temperature in the containment to acceptable values. During a Design Basis Accident (DBA), at least one containment cooling train and one containment spray train are capable of maintaining the peak pressure and temperature within design limits. One containment spray train has the capability, in conjunction with the Iodine Removal System, to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analyses. To ensure that these conditions can be met considering single-failure criteria, two spray trains and two cooling trains must be OPERABLE.

The 72 hour action interval specified in ACTION 1.a and ACTION 1.d, and the 7 day action interval specified in ACTION 1.b take into account the redundant heat removal capability and the iodine removal capability of the remaining operable systems, and the low probability of a DBA occurring during this period. INSERT 1 ~~The 10 day constraint for ACTIONS 1.a and 1.b is based on coincident entry into two ACTION conditions (specified in ACTION 1.c) coupled with the low probability of an accident occurring during this time.~~ If the system(s) cannot be restored to OPERABLE status within the specified completion time, alternate actions are designed to bring the unit to a mode for which the LCO does not apply. The extended interval (54 hours) specified in ACTION 1.a to be in MODE 4 includes 48 hours of additional time for restoration of the inoperable CS train, and takes into consideration the reduced driving force for a release of radioactive material from the RCS when in MODE 3. ~~With two containment spray trains or any combination of three or more containment spray and containment cooling trains inoperable in MODES 1, 2, or Mode 3 with Pressurizer Pressure ≥ 1750 psia, the unit is in a condition outside the accident analyses and LCO 3.0.3 must be entered immediately. In MODE 3 with Pressurizer Pressure < 1750 psia, containment spray is not required.~~

INSERT 5

The specifications and bases for LCO 3.6.2.1 are consistent with NUREG-1432, Revision 0 (9/28/92), Specification 3.6.6A (Containment Spray and Cooling Systems; Credit taken from iodine removal by the Containment Spray System), and the plant safety analyses.

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3/4.7 PLANT SYSTEMS (continued)

BASES (continued)

3/4.7.1 TURBINE CYCLE (continued)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM (continued)

The steam turbine-driven AFW pump receives steam from either main steam header upstream of the main steam isolation valve. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump. The turbine driven AFW pump supplies a common header capable of feeding both steam generators, with DC powered control valves actuated to the appropriate steam generator by the Auxiliary Feedwater Actuation System (AFAS).

The AFW System supplies feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The AFW System mitigates the consequences of any event with a loss of normal feedwater. The limiting Design Basis Accidents and transients for the AFW System are as follows:

1. Feedwater Line Break, and
2. Loss of normal feedwater.

INSERT 6

Surveillance Requirement (SR) 4.7.1.2.d verifies that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by the ASME Code. Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. this test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component Operability, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, discussed in the ASME Code, at 3 month intervals satisfies this requirement. This SR is modified to defer performance until suitable test conditions are established for the steam turbine-driven AFW pump within 24 hours after entering Mode 3 and prior to entering Mode 2.

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3/4.7 PLANT SYSTEMS (continued)

BASES (continued)

3/4.7.1 TURBINE CYCLE (continued)

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will comply with the dose criterion provided in 10 CFR 50.67 in the event of a steam line rupture. The dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves (MSIVs) ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. ~~The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements is consistent with the assumptions used in the safety analyses.~~

INSERT 7

The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements is consistent with the assumptions used in the safety analyses. The specified 6.75 second full closure time represents the addition of the maximum allowable instrument response time of 1.15 seconds and the maximum allowable valve stroke time of 5.6 seconds. These maximum allowable values should not be exceeded because they represent the design basis values for the plant.

3/4.7 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.7.1 TURBINE CYCLE (continued)

3/4.7.1.7 ATMOSPHERIC DUMP VALVES

The limitation on maintaining the atmospheric dump valves in the manual mode of operation is to ensure the atmospheric dump valves will be closed in the event of a steam line break. For the steam line break with atmospheric dump valve control failure event, the failure of the atmospheric dump valves to close would be a valid concern were the system to be in the automatic mode during power operations.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations to 100°F and 200 psig are based on a steam generator RT_{NDT} of 20°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

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

INSERT 8

3/4.7.4 INTAKE COOLING WATER SYSTEM

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

INSERT 9

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3/4.8 ELECTRICAL POWER SYSTEMS (continued)

BASES (continued)

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (continued)

The four hour completion time upon discovery that an opposite train required feature is inoperable is to provide assurance that a loss of offsite power, during the period that a EDG is inoperable, does not result in a complete loss of safety function of critical redundant required features. The four hour completion time allows the operator time to evaluate and repair any discovered inoperabilities. This completion time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The four hour completion time only begins on discovery that both an inoperable EDG exists and a required feature on the other train is inoperable.

TS 3.8.1.1, ACTION "b" provides an allowed outage/action completion time (AOT) of up to 14 days to restore a single inoperable diesel generator to operable status. INSERT 1 ~~This AOT is based on the findings of a deterministic and probabilistic safety analysis and is referred to as a "risk informed" AOT. Entry into this action requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP), which is described in the Administrative Procedure that implements the Maintenance Rule pursuant to 10 CFR 50.65.~~

All EDG inoperabilities must be investigated for common-cause failures regardless of how long the EDG inoperability persists. When one diesel generator is inoperable, required ACTIONS 3.8.1.1.b and 3.8.1.1.c provide an allowance to avoid unnecessary testing of EDGs. If it can be determined that the cause of the inoperable EDG does not exist on the remaining OPERABLE EDG, then SR 4.8.1.1.2.a.4 does not have to be performed. Eight (8) hours is reasonable to confirm that the OPERABLE EDG is not affected by the same problem as the inoperable EDG. If it cannot otherwise be determined that the cause of the initial inoperable EDG does not exist on the remaining EDG, then satisfactory performance of SR 4.8.1.1.2.a.4 suffices to provide assurance of continued OPERABILITY of that EDG. If the cause of the initial inoperability exists on the remaining OPERABLE EDG, that EDG would also be declared inoperable upon discovery, and ACTION 3.8.1.1.e would be entered. Once the failure is repaired (on either EDG), the common-cause failure no longer exists.

INSERT 10

INSERT 11

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3/4.8 ELECTRICAL POWER SYSTEMS (continued)

BASES (continued)

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (continued)

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

The Surveillance Requirements for demonstrating the OPERABILTY of the DC system battery cell interconnection resistances are based on criteria recommended by the manufacturer. The table contained in TSSR 4.8.2.3.2.c.3 is provided to define the maximum individual and maximum average allowable values for battery cell interconnection resistances.

The maximum individual battery cell interconnection resistance values are based on the negligible impact of voltage drop and connection heating, during peak DC system load conditions. A maximum individual battery interconnection resistance value of ≤ 150 x 10-6 ohms is used for connections, which use inter-cell (bus-bar type) connections and for the battery set output terminal connections. The maximum individual battery interconnection resistance value of ≤ 200 x 10-6 ohms is used for the inter-tier and inter-rack connections, which are subject to additional resistance of the cables used to extend between the different level tiers of each battery rack and of the adjacent battery rack.

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The maximum average battery cell interconnection resistance value of ≤ 50 x 10-6 ohms is the average of the interconnection resistance limit for all inter-cell, inter-tier, inter-rack and output terminals in the series-connected battery bank string. The ≤ 50 x 10-6 ohms criteria was selected in order to ensure that the battery cell interconnection voltage drop does not exceed the vendor criteria limit of less than 33.66 mV (average) for each battery cell interconnection, during the maximum design current load profile. The battery manufacturer has rated the battery bank set for full rated output, given adherence to limiting the average interconnection resistance to less than 33.66 mV drop between cells. For battery cell interconnections, which are monitored via multiple measurement points between two adjacent cells, these measurements must first be averaged for the connection between the affected adjacent cells, before averaging the values for all cells used in the full battery bank set.

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3/4.8 ELECTRICAL POWER SYSTEMS (continued)

BASES (continued)

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (continued)

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

INSERT 12

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

The OPERABILITY of the motor operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

St. Lucie Nuclear Plant Units 1 and 2

Enclosure 1

LIST OF REVISED REQUIRED ACTIONS TO CORRESPONDING PRA FUNCTIONS

1.0 INTRODUCTION

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

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

This enclosure provides confirmation that the St. Lucie (PSL) PRA models include the necessary scope of SSCs and their functions to address each proposed application of the Risk-Informed Completion Time (RICT) Program to the proposed scope TS LCO Conditions, and provides the information requested for Item 2 of the NRC safety evaluation. The scope of the comparison includes each of the TS LCO conditions and associated required actions within the scope of the RICT Program, as identified in Table 1 of the LAR.

This document lists each TS LCO/Condition to which the RICT Program may be applied and, for each Required Action, describes the corresponding SSC and associated function modeled in the PRA. This is to include the applicable success criteria used in the PRA model compared to the licensing basis criteria when calculating RICTs. The calculated RICT is provided for the condition to which the RICT applies.

2.0 SCOPE

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

- Column "TS LCO/Condition": Lists all of the LCOs and condition statements within the scope of the 4B implementation.
- Column "SSCs Covered by TS LCO/Condition": The SSCs addressed by each action requirement.
- Column "SSCs Modeled in PRA": Indicates whether the SSCs addressed by the TS LCO/Condition are included in the PRA.

- Column "Function Covered by TS LCO/Condition": A summary of the required function(s) from the design basis analyses.
- Column "Design Success Criteria": A summary of the success criteria from the design basis analyses.
- Column "PRA Success Criteria": The function success criteria modeled in the PRA.
- Column "Disposition": Justification or resolution to address any inconsistencies between the TS and PRA functions, regarding the scope of SSCs and the success criteria. Where the PRA scope of SSCs is not consistent with the TS, additional information is provided to describe how the LCO condition can be evaluated using appropriate surrogate events. Differences in the success criteria for TS functions are addressed to demonstrate the PRA criteria provide a realistic estimate of the risk of the TS condition as required by NEI 06-09.

The corresponding SSCs for each TS LCO and the associated TS functions are identified and compared to the PRA. This description also includes the design success criteria and the applicable PRA success criteria. Any differences between the scope or success criteria are described in the table. Scope differences are justified by identifying appropriate surrogate events which permit a risk evaluation to be completed using the CRMP tool for the RICT program. Differences in success criteria typically arise due to the requirement in the PRA standard (for example, SC-B1) to make PRAs realistic rather than bounding, whereas design basis criteria are necessarily conservative and bounding. The use of realistic success criteria is necessary to conform to capability category II of the PRA standard as required by NEI 06-09.

The estimated RICT calculations are provided in Tables E1-2 and E1-3 for each individual condition to which the RICT applies (assuming no other SSCs modeled in the PRA are unavailable simultaneously). Actual RICT values will be calculated based on the actual plant configuration using a current revision of the PRA model which represents the as-built, as-operated condition of the plant, as required by NEI 06-09 and the NRC safety evaluation, and may differ from the RICTs presented.

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

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in PRA	Function Covered by TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
{3.3.1.1 (Unit 1)} [3.3.1 (Unit 2)] Reactor Protective Instrumentation Function 1 – Manual Reactor Trip	2 channels	No	(1) Manually trip reactor on demand	(1) 1 of 2 channels	(1) Not modeled - see Disposition	The operator action for failure to actuate a manual reactor trip will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
{3.3.2.1 (Unit 1)} [3.3.2 (Unit 2)] Engineered Safety Features Actuation System (ESFAS) Instrumentation			(1) Associated ESF action will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint			Only a limited subset of functions associated with LCO 3.3.1 are in the proposed scope of the amendment request, as discussed below for each function.
Function 1a – Safety Injection (SIAS) – Manual (Trip Buttons)	2 channels	No		(1) 1 of 2 channels	(1) Not modeled - see Disposition	The operator action for failure to actuate a manual SIAS will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Function 1d – Safety Injection (SIAS) – Automatic Actuation Logic [Unit 2 only]	2 channels	No		(1) 1 of 2 channels	(1) Not modeled - see Disposition	The PRA model includes a basic event which addresses failure of SIAS logic. This event will be used to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.

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

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in PRA	Function Covered by TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
Function 2a – Containment Spray (CSAS) – Manual (Trip Buttons)	2 channels	No		(1) 1 of 2 channels	(1) Not modeled - see Disposition	The operator action for failure to actuate a manual CSAS will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Function 2b – CSAS – Containment Pressure – High-High	4 channels	Yes		(1) 2 of 4 channels	(1) SAME	SSCs for CSAS on containment pressure – high-high are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
Function 2c – CSAS – Automatic Actuation Logic [Unit 2 only]	2 channels	No		(1) 1 of 2 channels	(1) Not modeled - see Disposition	Automatic CSAS actuation logic is not credited in the PRA. The operator action for failure to actuate a manual CSAS will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Function 3a – Containment Isolation Actuation Logic (CIAS) – Manual (Trip Buttons)	2 channels	No		(1) 1 of 2 channels	(1) Not modeled - see Disposition	The operator action for failure to manually close isolation valves after failure of the CIS will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.

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

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in PRA	Function Covered by TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
Function 3e – CIAS – Automatic Actuation Logic [Unit 2 only]	2 channels	No		(1) 1 of 2 channels	(1) Not modeled - see Disposition	The PRA model includes a basic event which addresses failure of CIAS logic. This event will be used to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Function 4a – Main Steam Line Isolation (MSIS) – Manual (Trip Buttons) {Unit 1 only}	2 channels per SG	No		(1) 1 of 2 channels per SG	(1) Not modeled - see Disposition	The main steam isolation valves failure to close will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Function 4d – MSIS – Automatic Actuation Logic [Unit 2 only]	2 channels	No		(1) 1 of 2 channels	(1) Not modeled - see Disposition	The main steam isolation valves failure to close will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Function 5a – Containment Sump Recirculation (RAS) – Manual RAS (Trip Buttons)	2 channels	No		(1) 1 of 2 channels	(1) Not modeled - see Disposition	The operator action for failure to actuate manual RAS will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Function 5b – RAS - Refueling Water Tank - Low	4 channels	Yes		(1) 2 of 4 channels	(1) SAME	SSCs for RAS on refueling water tank – low are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
TS LCO/Condition	**SSCs Covered by TS LCO/ Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO/ Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Disposition**
Function 5c – RAS - Automatic Actuation Logic [Unit 2 only]	2 channels	Yes		(1) 1 of 2 channels	(1) SAME	SSCs for RAS on refueling water tank – low are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
Function 7a – Auxiliary Feedwater (AFAS) – Manual (Trip Buttons)	4 channels per SG	No		(1) 2 of 4 channels per SG	(1) Not modeled - see Disposition	The operator action for failure to actuate AFW manually will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Function 7b – AFAS – Automatic Actuation Logic	4 channels per SG	No		(1) 2 of 4 channels per SG	(1) Not modeled - see Disposition	The PRA model includes a basic event which addresses failure of AFAS automatic actuation logic. This event will be used to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Function 7c – AFAS – SG Level - Low	4 channels per SG	Yes		(1) 2 of 4 channels per SG	(1) SAME	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.

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

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in PRA	Function Covered by TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
{3.4.3 (Unit 1)} [3.4.2.2 (Unit 2)] Code Safety Valves	3 code safety valves	Yes	(1) Prevent RCS pressure from exceeding safety limit	(1) 3 of 3 code safety valves (for limiting transient)	(1) 2 of 3 code safety valves (ATWS event) (in conjunction with PORVs)	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The design basis event is a loss of load with immediate reactor trip not credited (subsequent trip on high reactor pressure is credited). Non-ATWS events with partial RPS failure are not probabilistically significant, so they are not considered in the PRA. The success criteria in the PRA for the limiting ATWS events use realistic analyses for RCS pressure control crediting operation of the PORVs. This is consistent with the PRA standards for capability category II.
{3.4.12 (Unit 1)} [3.4.4 (Unit 2)] Power Operated Relief Valve (PORV) Block Valves	2 PORV block valves	Yes	(1) Isolate open or leaking PORV	(1) Associated block valve manually closed	(1) SAME	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.

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

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in PRA	Function Covered by TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.5.1 Safety Injection Tanks (SITs)	4 SITs	Yes	(1) Initial cooling mechanism during large RCS pipe ruptures	(1) 3 of 4 SITs to intact cold legs (cold leg break LOCA) or 4 of 4 SITs (hot leg break LOCA)	(1) SAME for cold leg LOCA, no separate criteria for hot leg LOCA	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria for cold leg LOCAs, and are based on realistic criteria for hot leg LOCAs. Success criteria in PRA are based on plant-specific realistic analyses consistent with the PRA standards for capability category II.

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

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in PRA	Function Covered by TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.5.2 Emergency Core Cooling System (ECCS) Subsystems – Operating	2 high-pressure safety injection (HPSI) pumps 2 low-pressure safety injection (LPSI) pumps 2 charging pumps Flowpaths from refueling water tank and containment sump	Yes (partial)	(1) Sufficient emergency core cooling in the event of a LOCA (2) Long term core cooling in the recirculation mode (3) Long term reactivity control	(1) 1 of 2 HPSI and 1 of 2 LPSI pumps with credit for only 75% of the total flow to intact cold legs (2) 1 of 2 HPSI with suction from containment sump to cold legs [and (unit 2 only) hot legs]; decay heat removal from Containment Cooling Systems (3) 1 of 2 charging pumps and associated boration flowpath	(1) 1 of 2 HPSI pumps (small and medium LOCA only) and 1 of 2 LPSI (large LOCA only) to any intact cold leg (2) 1 of 2 HPSI pumps to any intact cold leg; [1 of 2 HPSI pumps (unit 2) to any hot leg] for medium and large LOCA; decay heat removal from Containment Cooling Systems (3) Not modeled - see Disposition	SSCs are modeled consistent with the TS scope (except for the charging pumps) and so can be directly evaluated using the CRMP. The function of the charging pumps is addressed by TS 3.1.2.4 in MODES 1-4, but this LCO not in the scope of TSTF-505. The PRA success criteria differ from the design basis in 1) injection into any intact cold leg, 2) not requiring HPSI for large LOCAs, 3) not requiring recirculation for small LOCAs if secondary cooling and shutdown cooling are available, 4) crediting configurations of hot and cold leg flowpaths during recirculation based on realistic analyses for specific break locations, and 5) requiring a hot leg injection path for unit 1 for medium or large LOCAs using non-ECCS flowpaths. Success criteria in PRA are based on plant-specific realistic analyses consistent with the PRA standards for capability category II.

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

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in PRA	Function Covered by TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.5.4 Refueling Water Tank	1 Refueling Water Tank per unit	Yes	(1) Sufficient water for ECCS injection for a LOCA to permit recirculation (2) Reactor will remain subcritical following a LOCA	Refueling Water Tank boron concentration, temperature, and level within limits	SAME	The PRA does not explicitly model the impact of out of limit boron or temperature, but these can be conservatively addressed for the RICT Program by assuming the RWST is unavailable. Therefore, this LCO condition can be evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.6.1.3 Containment Air Locks	2 air locks	No	(1) Meet restrictions on containment integrity and containment leak rate		Not modeled - see Disposition	SSCs for the containment air locks can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event which involves a large, pre-existing containment leak; this would be bounding for risk associated with an inoperable air lock door with at least one door closed, and can be used as a bounding surrogate.
3.6.1.7 [Unit 2 only] Containment Ventilation System	Purge Supply and Exhaust Isolation Valves	No	(1) 48" valves sealed closed (2) 8" valves open only for safety-related purposes	(1) Each valve sealed (2) Each valve closed unless open for safety-related purposes with leakrate in limits	Not modeled - see Disposition	SSCs for the containment purge supply and exhaust isolation valves can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event which involves a large, pre-existing containment leak; this would be bounding for risk associated with an inoperable air lock door with at least one door closed, and can be used as a bounding surrogate.

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

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in PRA	Function Covered by TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.6.2.1 Containment Spray (CS) and Containment Cooling Systems	2 CS trains 2 containment cooling trains	Yes	(1) Limit containment post-accident pressure and temperature (2) Maintain iodine concentrations below those assumed in the safety analyses	(1) 1 of 2 CS trains and 1 of 2 containment cooling trains (2) 1 of 2 CS trains	(1) 1 of 2 CS trains or 2 of 4 containment fan coolers (2) Not modeled	The SSCs in the TS scope are modeled in the PRA. The iodine removal function of the CS trains is not required for mitigation of severe accidents and is not modeled. The success criteria in the PRA are based on plant-specific realistic analyses to support long-term decay heat removal consistent with the PRA standards for capability category II.
{3.6.3.1 (Unit 1)} [3.6.3 (Unit 2)] Containment Isolation Valves	2 active or passive isolation devices on each fluid penetration line	Yes (in part)	(1) Each containment penetration isolated within the time limits assumed in the safety analyses	(1) 1 of 2 isolation devices per penetration isolate within required stroke time.	(1) SAME for PRA modeled penetrations. All other penetrations evaluated as not significant sources of fission product leakage and are screened out.	SSCs for containment isolation valves not in the PRA model can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event which involves a large, pre-existing containment leak; this would be bounding on risk on an inoperable isolation valve and can be used as a bounding surrogate. The PRA does not explicitly model the impact of excessive stroke time. This condition can be addressed for the RICT Program by conservatively assuming the inoperable containment isolation valve is unclosable if it is open. Otherwise, the success criteria in the PRA are consistent with the design basis criteria.

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

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in PRA	Function Covered by TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.7.1.2 Auxiliary Feedwater (AFW) System	2 motor-driven pumps and 1 turbine-driven pump	Yes	(1) Supply feedwater to SGs to remove RCS decay heat and reduce RCS temperature to 325°F	(1) 2 of 3 pumps	(1) SAME for most limiting event (ATWS); 1 of 3 pumps for non-ATWS events	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are based on realistic analyses for removal of decay heat from the reactor, consistent with the PRA standards for capability category II.
3.7.1.3 Condensate Storage Tank (CST)	CST	Yes	(1) Source of water to SGs for removing heat from RCS	(1) CST aligned with minimum water volume	(1) SAME	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.7.1.5 Main Steam Isolation Valves (MSIVs)	2 MSIVs	Yes	(1) Ensure no more than one SG blows down in the event of a steam line rupture	(1) MSIV on affected steamline closes	(1) SAME	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions						
TS LCO/Condition	**SSCs Covered by TS LCO/ Condition**	**SSCs Modeled in PRA**	**Function Covered by TS LCO/ Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Disposition**
{3.7.3.1 (Unit 1)} [3.7.3 (Unit 2)] Component Cooling Water (CCW) System	2 trains	Yes	(1) Cooling of vital components and ESF equipment during normal and accident conditions	(1) 1 of 2 trains	(1) SAME	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
{3.7.4.1 (Unit 1)} [3.7.4 (Unit 2)] Intake Cooling Water System	2 trains	Yes	(1) Cooling of vital components and ESF equipment during normal and accident conditions	(1) 1 of 2 trains	(1) SAME	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.8.1.1 AC Sources - Operating	2 offsite circuits 2 diesel generators (DG)	Yes	(1) Sufficient power for safe shutdown and mitigation and control of accident conditions	(1) Automatically power associated safety-related busses	(1) SAME	SSCs for offsite circuits not in the PRA model can be evaluated by a bounding assessment as permitted by NEI 06-09. The PRA model includes an event for Start-Up Transformers; this would be bounding on risk on an inoperable offsite circuit and can be used as a bounding surrogate. The success criteria in the PRA are consistent with the design basis criteria.

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

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in PRA	Function Covered by TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
{3.8.2.1 (Unit 1)} [3.8.3.1 (Unit 2)] AC Power Distribution System	4160 V busses 1[2]A3 1[2]B3 480 V busses 1[2]A2, 1[2]B2, 1[2]A5, 1[2]A6, 1[2]A7, 1[2]B5, 1[2]B6, 1[2]B7 120 V instrument busses 1[2]MA, 1[2]MB, 1[2]MC, 1[2]MD	Yes	(1) Sufficient power for safe shutdown and mitigation and control of accident conditions	(1) Align to provide power to busses	(1) SAME	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
{3.8.2.3 (Unit 1)} [3.8.2.1 (Unit 2)] DC Power Distribution System	125 V DC bus 1[2]A and 1[2]B and associated battery bank and charger	Yes	(1) Sufficient power for safe shutdown and mitigation and control of accident conditions	(1) Aligned to provide power to associated busses from battery and associated charger	(1) SAME	SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.

3.0 RESULTS

To calculate the estimated RICT for each TS LCO, the St. Lucie Internal Events PRA Model (Reference 3) was used, as well as the resulting cutsets of the Internal Flood PRA (Reference 4) and Fire PRA from NFPA 805 (Reference 5). The change in core damage and large early release frequency (ΔCDF/ΔLERF) was calculated for each hazard separately, then combined in a spreadsheet to calculate the total ΔCDF and ΔLERF. PRAQuant was used to quantify each condition in the Internal Events model, and SYSIMP was used to calculate the increase in risk using Internal Flood and NFPA 805 pre-generated cutsets This could then be used to calculate the RICT values (time to reach a ΔCDP of 1.0E-05/yr. or ΔLERP of 1.0E-06/yr.). The RICT estimates for Unit 1 and Unit 2 are provided in Tables E1-2 and E1-3 below.

Table E1-2: Unit 1 In Scope TS/LCO Conditions RICT Estimate

TS LCO/Condition	RICT Estimate[1] (days)
3.3.1.1 Reactor Protective Instrumentation Function 1, Action 1 One of two manual reactor trip channels inoperable	>30
3.3.2.1 ESFAS Instrumentation Function 1a, Action 8 One of two manual SI channels inoperable	>30
3.3.2.1 ESFAS Instrumentation Function 2a, Action 8 One of two manual CSAS channels inoperable	>30
3.3.2.1 ESFAS Instrumentation Function 2b, Action 10c Two of four Containment Pressure High-High CSAS channels inoperable	>30
3.3.2.1 ESFAS Instrumentation Function 3a, Action 8 One of two manual CIS channels inoperable	>30
3.3.2.1 ESFAS Instrumentation Function 4a, Action 8 One of two manual MSIS channels inoperable	>30
3.3.2.1 ESFAS Instrumentation Function 5a, Action 8 One of two manual RAS channels inoperable	>30
3.3.2.1 ESFAS Instrumentation Function 5b, Action 13 One of four Refueling Water Tank - Low channels inoperable	2
3.3.2.1 ESFAS Instrumentation Function 7a, Action 11 One of four manual AFAS channels Logic channels inoperable	>30

Table E1-2: Unit 1 In Scope TS/LCO Conditions RICT Estimate	
TS LCO/Condition	**RICT Estimate[1] (days)**
3.3.2.1 ESFAS Instrumentation Function 7b, Action 11 One of four AFAS Automatic Actuation Logic channels inoperable	>30
3.3.2.1 ESFAS Instrumentation Function 7c, Action 14 One of four SG Level - Low channels on one SG inoperable	>30
3.4.3 Safety Valves - Operating Action a One of three code safety valves inoperable	>30
3.4.12 PORV Block Valves Action (undesignated) One PORV block valve inoperable	< 1
3.5.1 SITs Actions a and b One SIT inoperable	>30
3.5.2 ECCS Subsystems - Operating Action a.1 One ECCS subsystem LPSI pump inoperable	>30
3.5.2 ECCS Subsystems - Operating Action a.2 One ECCS subsystem inoperable	7
3.5.4 Refueling Water Tank [2] Action (undesignated) Refueling Water Tank inoperable	< 1
3.6.1.3 Containment Air Locks [3] Action a.1 One Containment Air Lock door inoperable Action b One Containment Air Lock inoperable	6
3.6.2.1 CS and Containment Cooling Systems Action 1.a One CS train inoperable	>30
3.6.2.1 CS and Containment Cooling Systems Action 1.b One Containment Cooling train inoperable	>30
3.6.2.1 CS and Containment Cooling Systems Action 1.c One CS train and one Containment Cooling train inoperable	>30
3.6.2.1 CS and Containment Cooling Systems Action 1.d Two Containment Cooling trains inoperable	>30
3.6.3.1 Containment Isolation Valves [3] Actions a, b, or c One containment isolation valve inoperable	6

Table E1-2: Unit 1 In Scope TS/LCO Conditions RICT Estimate	
TS LCO/Condition	RICT Estimate[1] (days)
3.7.1.2 AFW Action a One AFW pump inoperable	3
3.7.1.3 Condensate Storage Tank Action (undesignated) Condensate Storage Tank inoperable	< 1
3.7.1.5 Main Steam Isolation Valves Action (undesignated) One of two MSIVs inoperable	>30
3.7.3.1 CCW System Action (undesignated) One of two independent CCW trains inoperable	6
3.7.4.1 ICW System Action (undesignated) One of two independent ICW trains inoperable	>30
3.8.1.1 AC Sources - Operating Action a One of two offsite circuits inoperable	3
3.8.1.1 AC Sources - Operating Action b One of two diesel generator sets inoperable	22
3.8.1.1 AC Sources - Operating Action c One of two offsite circuits and one of two diesel generator sets inoperable	3
3.8.1.1 AC Sources - Operating Action d Two of two offsite circuits inoperable	3
3.8.1.1 AC Sources - Operating Action e Two of two diesel generator sets inoperable	< 1
3.8.1.1 AC Sources - Operating Action f Unit startup transformer inoperable with other unit startup transformer connected to same offsite circuit and administratively available to both units, and other unit requires use of startup transformer	3
3.8.2.1 AC Distribution - Operating Action (undesignated) One AC electrical bus inoperable	< 1
3.8.2.3 DC Distribution - Operating Action a One battery bank or bus inoperable	< 1

Table E1-3: Unit 2 In Scope TS/LCO Conditions RICT Estimate

TS LCO/Condition	RICT Estimate[1] (days)
3.3.1 Reactor Protective Instrumentation Function 1, Action 1 One of two manual reactor trip channels inoperable	>30
3.3.2 ESFAS Instrumentation Functions 1a/d, Action 12 One of two manual/automatic SI channels inoperable	>30
3.3.2 ESFAS Instrumentation Functions 2a/c, Action 12 One of two manual/automatic CSAS channels inoperable	>30
3.3.2 ESFAS Instrumentation Function 2b, Action 18c Two of four Containment Pressure High-High CSAS channels inoperable	>30
3.3.2 ESFAS Instrumentation Functions 3a/e, Action 12 One of two manual/automatic CIS channels inoperable	>30
3.3.2 ESFAS Instrumentation Functions 4a/d, Action 12 One of two manual/automatic MSIS channels inoperable	>30
3.3.2 ESFAS Instrumentation Functions 5a/c, Action 12 One of two manual/automatic RAS channels inoperable	>30
3.3.2 ESFAS Instrumentation Function 5b, Action 19a One of four Refueling Water Tank - Low channels inoperable	6
3.3.2 ESFAS Instrumentation Functions 7a/b, Action 15 One of four manual AFAS channels or Automatic Actuation Logic channels inoperable	13
3.3.2 ESFAS Instrumentation	>30

Table E1-3: Unit 2 In Scope TS/LCO Conditions RICT Estimate	
TS LCO/Condition	**RICT Estimate[1] (days)**
Function 7c, Action 20 One of four SG Level - Low channels on one SG inoperable	
3.4.2.2 Safety Valves - Operating Action a One of three code safety valves inoperable	>30
3.4.4 PORV Block Valves Action a One PORV block valve inoperable	20
3.4.4 PORV Block Valves Action b Two PORV block valve inoperable	< 1
3.5.1 SITs Actions a and b One SIT inoperable	>30
3.5.2 ECCS Subsystems - Operating Action a.1 One ECCS subsystem LPSI pump inoperable	>30
3.5.2 ECCS Subsystems - Operating Action a.2 One ECCS subsystem inoperable	3
3.5.4 Refueling Water Tank [2] Action (undesignated) Refueling Water Tank inoperable	< 1
3.6.1.3 Containment Air Locks [3] Action a.1 One Containment Air Lock door inoperable Action b One Containment Air Lock inoperable	5
3.6.1.7 Containment Ventilation System [3] Action c Purge Supply and/or Exhaust valve inoperable	5
3.6.2.1 CS and Containment Cooling Systems Action 1.a One CS train inoperable	11
3.6.2.1 CS and Containment Cooling Systems Action 1.b One Containment Cooling train inoperable	>30
3.6.2.1 CS and Containment Cooling Systems Action 1.c One CS train and one Containment Cooling train inoperable	11

Table E1-3: Unit 2 In Scope TS/LCO Conditions RICT Estimate

TS LCO/Condition	RICT Estimate[1] (days)
3.6.2.1 CS and Containment Cooling Systems Action 1.d Two Containment Cooling trains inoperable	>30
3.6.3 Containment Isolation Valves [3] Actions a, b, or c One containment isolation valve inoperable	5
3.7.1.2 AFW Action a One AFW pump inoperable	3
3.7.1.3 Condensate Storage Tank Action (undesignated) Condensate Storage Tank inoperable	< 1
3.7.1.5 Main Steam Isolation Valves Action (undesignated) One of two MSIVs inoperable	>30
3.7.3 CCW System Action (undesignated) One of two independent CCW trains inoperable	3
3.7.4 ICW System Action (undesignated) One of two independent ICW trains inoperable	>30
3.8.1.1 AC Sources - Operating Action a One of two offsite circuits inoperable	4
3.8.1.1 AC Sources - Operating Action b One of two diesel generator sets inoperable	14.00
3.8.1.1 AC Sources - Operating Action c One of two offsite circuits and one of two diesel generator sets inoperable	2
3.8.1.1 AC Sources - Operating Action d Two of two offsite circuits inoperable	2
3.8.1.1 AC Sources - Operating Action e Two of two diesel generator sets inoperable	3
3.8.1.1 AC Sources - Operating Action f Unit startup transformer inoperable with other unit startup transformer connected to same offsite circuit and administratively available to both units, and other unit requires use of startup transformer	4

Table E1-3: Unit 2 In Scope TS/LCO Conditions RICT Estimate	
TS LCO/Condition	**RICT Estimate[1] (days)**
3.8.2.1 DC Distribution - Operating Action a One battery bank inoperable	< 1
3.8.3.1 Onsite Power Distribution - Operating Action a One AC electrical bus inoperable	< 1
3.8.3.1 Onsite Power Distribution - Operating Action b One AC instrument bus not energized from its inverter connected to its DC bus	>30
3.8.3.1 Onsite Power Distribution - Operating Action c One DC bus not energized from its associated battery bank	< 1

1 RICT estimates are based on the most recent PSL Internal Events, Internal Flood, and NFPA 805 models of record. Any changes made to these models will impact the calculated RICT values.

2 RICT evaluated for limiting condition of the associated tank being empty.

3 RICT evaluated for limiting condition of loss of containment function for a large containment penetration.

References

1. ML071200238, *Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines (TAC No. MD4995)*, Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.

2. NEI 06-09, *Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document*, Nuclear Energy Institute, Revision 0, November 2006.

3. PSL-BFJR-12-001 Rev. 0, *St. Lucie EPU PSA Model Update for Units 1 & 2*, FPL, May 11, 2012.

4. PSL-BFJR-11-005 Rev. 0, *Internal Flood Analysis for St. Lucie Units 1 & 2,* FPL, July 2, 2013.

5. Report 0493060006.105 Rev. 4, *St. Lucie Nuclear Plant Fire PRA Summary Report NUREG/CR-6850 Task 16*, ERIN Engineering, March 2013.

St. Lucie Nuclear Plant Units 1 and 2

Enclosure 2

INFORMATION SUPPORTING CONSISTENCY WITH REGULATORY GUIDE 1.200, REVISION 2

Table of Contents

E2-1.0 INTRODUCTION

NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 1) Section 2.3.4 identifies that the probabilistic risk assessment (PRA) shall be reviewed using the guidance of Regulatory Guide (RG) 1.200 (Reference 2) for a PRA which meets Capability Category II for the supporting requirements (SRs) of the internal events at power PRA standard (Reference 3), and that deviations shall be justified and documented. Section 4.0, Item 3 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 4) for NEI 06-09 requires the license amendment request (LAR) to include a discussion of the results of peer reviews and self-assessments conducted for the plant-specific PRA models which support the RMTS Program, including the resolution or disposition of any identified deficiencies (i.e., findings and observations from peer reviews). The scope of this information includes the internal events PRA model, and other models for which additional standards have been endorsed by a revision to RG 1.200.

This enclosure provides information on the technical adequacy of the St. Lucie PRA internal events, internal flood, and internal fire models which support the Risk-Informed Completion Time (RICT) Program, in support of the LAR to revise Technical Specifications (TS) to implement NEI 06-09. This information is consistent with the requirements of Item 3 of Reference 3, and addresses each PRA model for which a RG 1.200 endorsed standard exists. The information is provided in Attachments A, B, and C to this document.

Note that other external hazards including seismic hazards are not addressed by PRA models, and are further discussed in Enclosure 4. Shutdown modes of operation are not in the scope of the RICT Program, and so low-power and shutdown PRA models are not addressed. No other PRA standards are endorsed by RG 1.200.

No changes have been made to the internal event, internal flood, or fire PRA models since the peer reviews that would constitute an upgrade as defined by ASME/ANS RA-Sa-2009, and therefore no additional focused-scope peer reviews are required to support implementation of the RICT Program. Future changes to the St. Lucie PRA models will be performed consistent with station procedures for design changes, procedure changes, and equipment performance monitoring. This will also include updates to implemented risk informed applications as applicable and appropriate.

E2-2.0 BACKGROUND

In response to NRC Generic Letter 88-20, the St. Lucie PRA Level 1 and Level 2 models (collectively known as internal events analysis) were originally developed and submitted in 1993. The document called St. Lucie Individual Plant Examination (IPE) (Reference 7). The PRA models addressed risk assessment of at-power operation. Later, in 1996, the IPE was supplemented with the Individual Plant Examination for External Events (IPEEE) document submittal (Reference 8) to address external events such as internal fire and flood, among other events. This PRA was then subjected to a number of reviews, internal and external, during its preparation as well as extensive reviews by the NRC through national labs following its publication.

In July 2002, the Combustion Engineering Owners Group (CEOG) performed a peer review of the St. Lucie Units 1 and 2 Level 1 and Level 2 2002-PSA models update. The review followed a process that was adopted by industry reference NEI-00-02, Rev. A3 (Probabilistic Risk Assessment (PRA) Peer Review Process Guidance, Nuclear Energy Institute, March 2000). The resulting Findings and Observations (F&Os) of the CEOG peer review were published in the February 2003 Westinghouse publication WCAP-16034, Rev 0, St. Lucie Unit 1 and 2: Probabilistic Risk Assessment Peer Review Report, CEOG Task 1037 (Reference 9). The assessment covered all aspects of the PRA model and documentation. The result of the assessment ranked the findings on a scale of "A" to "D", with "A" being the most significant. Each of the findings was presented with observations and comments.

In December 2005, MARACOR Software & Engineering, Inc. performed an independent review of St. Lucie Unit 1 & 2 PSA models update. The review process was based on conformance to Category II of the ASME PRA Standard (ASME-RA-S-2002), Addenda A (RA-Sa-2003). The resulting recommendations were published in the MARACOR report, "An Independent Review of the Port St. Lucie PRA against the Requirements of the ASME PRA Standard" dated December 13, 2005.

Following the issuance of the ASME PRA Standard and Regulatory Guide 1.200 (RG 1.200), in October 2007, St. Lucie PRA Group performed a self-assessment of the 2002 peer review and 2005 assessment to identify gaps and actions needed to conform to the requirements delineated by RG-1.200 Rev 1. The current PSL gap analysis uses the RA-Sa-2009 version of the standard as endorsed by RG 1.200, Revision 2.

To supplement the original peer review and internal gap analysis, and to further improve the quality of the updated internal events models used in the Fire PRA, subsequent focused scope peer reviews for St. Lucie were conducted. A LERF Focused Peer Review (Reference 10) was conducted in July 2009, which identified no findings. A self-assessment was completed in March 2014, indicated below, identified no gap in LERF Focused Peer Review when applied ASME/ANS RA-SA-2009 as endorsed by RG 1.200, Revision 2.

A focused peer review of Common-Cause Failure (CCF) methodology and data (Reference 11) was completed in August 2009. In April 2011, a focused peer review (Reference 12) was performed by the PWROG that included Human Reliability Analysis (HRA), Internal Flooding Analysis (IF), and Data Analysis (DA) for compliance against the most current combined PRA standard, ASME/ANS RA-Sa–2009, as endorsed by Regulatory Guide 1.200, Rev. 2. The internal flooding analysis focused peer review was performed because the latest internal flooding analysis was much more comprehensive than the original screening analysis that was performed for the IPEEE. Although the basic methods used for the HRA had not changed substantially, the HRA focused peer review was performed because of the enhanced HRA dependency analysis and the use of the HRA Calculator software in the latest model, and the fact that HRA plays a significant role in the determination of the dominant sequences and overall risk profile.

In December 2013, a focused peer review was conducted for Interfacing System LOCA (ISLOCA) initiating events (Reference 13) in accordance with RG 1.200 Revision 2. The ISLOCA analysis utilized a completely different approach from the previous analysis.

A self-assessment has been developed in March 2014 (Reference 14) to include an integrated document identifying potential gaps between peer reviews conducted using earlier PRA Standard revisions that were endorsed by earlier revisions of RG 1.200 and the current PRA ASME Standard as endorsed by RG 1.200 Revision 2.

Significant findings from the peer reviews are listed in Attachments A & B, along with their resolutions.

E2-3.0 CONFORMANCE WITH PRA ASME STANDARD

The following sections describe the conformance and capability of the St. Lucie PRA against the major parts of PRA ASME Standard.

E2-3.1 ASME Part 2 - Internal Events

The internal events portion of the St. Lucie PRA has been updated a number of times since the original IPE submittal.

As described in Section 2.2, there have been one global peer review (full scope) and several focused peer reviews to include various ASME elements such as HR, DA, and LE as well as other PRA areas with cross-connection among ASME elements (e.g., CCF, and ISLOCA). The following peer reviews have been conducted against internal events supporting requirements:

- In 2002, a review of all technical elements was performed by CEOG using the industry PSA Certification process, the precursor to the PRA Standard. All of the findings and observations have been addressed in the model updates following this peer review.
- In 2005, a self-assessment was performed by MARACOR against ASME-RA-Sa-2003.
- In 2009, a focused peer review on Large Early Release Frequency (LERF) was performed (HLR-LE). This review was conducted by PWROG using ASME RA-Sb-2005 as endorsed by RG 1.200 Revision 1 and resulted in zero findings and observations.
- In 2011, a focused peer review was performed by PWROG for the elements DA, and HR. This assessment replaced the 2002 peer review for those elements that were in scope. This review was done using the current PRA Standard (ASME/ANS RA-Sa-2009) as endorsed by RG 1.200 Revision 2. All of the findings and suggestions have been resolved, and, where changes were necessary, addressed in a model update

In addition to these peer reviews; there have been 2 subsequent focused peer reviews for specific PRA areas associated with the St. Lucie PRA models; mainly common-cause failure (CCF) methodology and Interfacing Systems LOCA (ISLOCA) modeling. Each of these PRA areas included review of applicable cross-cut of multiple ASME elements.

- In 2009, a focused peer review of CCF methodology and respective data was performed. This review covered all SRs in the ASME Standard (ASME/ANS RA-Sa-2009) as endorsed by RG 1.200 Revision 2 which have a relationship to CCF. All of the findings and observations have been addressed in the model updates following this peer review.
- In 2013, a focused peer review of ISLOCA methodology and respective data was performed. This review covered all SRs in the ASME Standard (ASME/ANS RA-

Sa-2009) as endorsed by RG 1.200 Revision 2 which have a relationship to ISLOCA. The respective findings and observations are currently being reviewed and responses are being provided. Each finding will be addressed in a model update (if applicable) and documented accordingly.

A self-assessment was completed in March 2014 as an integrated document identifying potential gaps between peer reviews and self-assessments conducted using earlier PRA Standard revisions that were endorsed by earlier revisions of RG 1.200 and the current PRA ASME Standard as endorsed by RG 1.200 Revision 2. No gaps identified to impact the current model results except those associated with ISLOCA focused peer review that are currently being resolved.

Conclusion

The current open items do not represent significant deficiency in the analyses necessary to support the 4b application. The current St. Lucie PRA meets all Part 2 (internal event) CC II requirements of the PRA Standard.

E2-3.2 ASME Part 3 - Internal Flooding

In 2011, a focused peer review was performed by PWROG for Internal Flooding element (IF). This review was done using the current PRA Standard (ASME/ANS RA-Sa-2009) as endorsed by RG 1.200 Revision 2. All of the findings and suggestions have been resolved, and, where changes were necessary, addressed in a model update

Conclusion

There are no open items which represent significant deficiency in the analyses necessary to support the 4b application. The current St. Lucie PRA meets all Part 3 (internal flood) CC II requirements of the PRA Standard.

E2-3.3 ASME Part 4 - Internal Fire

A fire PRA was performed for St. Lucie as part of the 1994 IPEEE submittal. Since it was done for the IPEEE, it was more of a screening analysis to discover any fire vulnerabilities than an attempt to determine a realistic estimate of core damage risk due to fire. It has not been updated since the original submittal.

St. Lucie is an NFPA-805 plant, and therefore has a fire PRA to support the NFPA-805 effort. The fire PRA uses the latest internal events PRA model as a basis. The St. Lucie NFPA-805 fire PRA uses NUREG/CR-6850 guidance as required by NFPA-805, and thus produces a conservative estimate of core damage risk due to fire.

A peer review of the St. Lucie (PSL) Fire PRA was performed at PSL using the NEI 07-12 Fire PRA peer review process, and the PRA standard (ASME/ANS RA-Sa–2009) as endorsed by RG 1.200, Revision 2. The purpose of this review was to provide a method for establishing the technical quality and adequacy of the Fire PRA for the spectrum of potential risk-informed plant licensing applications for which the Fire PRA may be used. The PSL Fire PRA Peer Review was a full-scope review of all the technical elements of Part 4 of the ASME/ANS standard.

The Fire PRA update addressed the Supporting-Requirement-assessed deficiencies (i.e., Not Met or CCI). Completion of recommendations related to Supporting Requirement assessments and 'Finding' F&Os results in a Capability Category II assessment for the majority of the Supporting Requirements.

Conclusion

Based on the completion of peer review recommendations and the assessment of deferred items, the St. Lucie Fire PRA is adequate to support this application. The fire PRA is a conservative representation of the fire risk from operation of St. Lucie Nuclear Plant.

E2-4.0 CONCLUSION

The St. Lucie PRA model of record fully meets all the requirements of Part 2 (Internal Events) and Part 3 (Internal Flood) of the current ASME/ANS PRA Standard as endorsed by RG 1.200 Revision 2. All significant findings from peer reviews or other technical reviews have been (or are currently being) addressed and closed. This is considered to meet the guidance addressed in NEI 06-09.

Based on the completion of peer review recommendations and the assessment of deferred items, the St. Lucie Fire PRA is adequate to support this application, with the caveat that the PRA is a conservative representation of the fire risk from operation of St. Lucie Nuclear Plant. The Fire PRA model will be exercised to obtain quantitative fire risk insights, but refinements may need to be made on a case-by-case basis.

Seismic risk at St. Lucie is minimal and will not be a significant factor in the 4b application. This is further discussed in Enclosure 4.

E2-5.0 REFERENCES

1. NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b Risk-Managed Technical Specifications (RMTS) Guidelines", Nuclear Energy Institute, November 2006.

2. NRC Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Office of Nuclear Regulatory Research, March 2009.

3. ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", American Society of Mechanical Engineers and American Nuclear Society, February 2009.

4. ML071200238, Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines (TAC No. MD4995)," Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.

5. PSL-BFJR-12-001, Revision 0, St. Lucie EPU PSA Model Update for Units 1&2, Florida Power and Light Co., May 2012.

6. PSL-BFJR-11-005, Revision 0, Internal Flood Analysis for St. Lucie Units 1 & 2, Florida Power and Light Co., June 2013.

7. PSL-IPE, Revision 0, St. Lucie Units 1 & 2 Individual Plant Examination, Florida Power and Light Co., December 1993.

8. PSL-IPEEE, Revision 0, Individual Plant Examination of External Events for St. Lucie Units 1 and 2, Florida Power and Light Co., December 1994.

9. WCAP-16034, Revision 00, St. Lucie Units 1 and 2: Probabilistic Risk Assessment Peer Review Report, CEOG Task 1037, Westinghouse, February 2003.

10. LTR-RAM-II-09-038, Revision 0, Focused Scope Peer Review of the St. Lucie Units 1 and 2 Large Early Release Frequency PRA Against the ASME PRA Standard Requirements, Westinghouse, July 2009.

11. Focused Peer Review of St. Lucie PSA Common Cause Failure (CCF) Methodology, Revision 0, Michael Lloyd, August 2009.

12. LTR-RAM-II-11-054, Revision 0, RG 1.200 PRA Focused Peer Review Against the ASME PRA Standard Requirements for Plant St. Lucie (PSL) Probabilistic Risk Assessment, Westinghouse, July 2011.

13. RSC 13-49, Revision 0, St. Lucie Nuclear Power Plants Focused Scope PRA Peer Review: Interfacing Systems LOCA, Reliability and Safety Consulting Engineers, Inc., December 2013.

14. PSL-BFJR-13-020, Revision 0, NRC RG-1.200 Rev. 2 Self-Assessment For St. Lucie Units 1 & 2, Florida Power and Light Co., March 2014.

15. Florida Power & Light Company St. Lucie Nuclear Power Plant Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition, Transition Report, March 2013.

E2-Attachment A - Internal Events and Internal Flooding Peer Review Findings

Table E2-A1 summarizes facts and observations with significance ranking "A", "B", or "Finding" from the previously referenced peer reviews for Internal Events and Internal Flooding.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
AS-01	AS-A5 AS-A7 SY-A1	Cutset %ZZSU1*CMM1AVCCCF appears overly conservative. Each CCW header provides approximately 8000 gpm. The largest accident loads are the shutdown cooling heat exchangers (4500 gpm) and the fan coolers (1200 gpm each). The N-loads are the SPF HXs (2900 gpm), let down heat exchanger (less than 1400 gpm), the RCP cooling (250 gpm each), and the boric acid concentrators (775 gpm). During a small LOCA, the heat load on the containment fan coolers is significantly lower than a design basis accident. The load on the SDC HXs does not exist until re-circulation. Eventually, the LOCA will lead to the failure of the RCPs even if the operators do not trip the pumps. When the RCPs are not running the heat load is further reduced. The SPF will act to moderate temperature changes due to the large volume of water. Not only will the peak containment temperature and pressures will be much lower during the small break LOCA, but also the decay heat removal will not solely be provided through the break. Secondary side heat removal is quite effective during the small LOCA break sizes. This issues combine to form a reasonable basis for not requiring N-header isolation during a small break LOCA. Considering the initial flow rates through the TCCW HXs and the Open Blowdown HXs, it would be a more difficult argument to make to not require the closure of ICW MOVs 21-2 and 21-3. Considering that most of the heat removal can still be provided by the S/Gs, it is probably reasonable to removal this closure as well. This being said the failure both ICW isolation and N-header isolation should probably be considered failure unless a more detailed calculation is available.	A	Add basis for excluding N-header isolation following a small break LOCA.	This finding has been resolved and closed by an update to the model/documentation. Success Criteria for N-Header and ICW-to-TCW HXs isolation valves have been revised as well as CCF data analysis during previous model maintenance and update.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
AS-02	AS-A5 AS-A7 ASA10 AS-B1 SC-A6 QU-A1	Considering the significance of aligning OTCC and the high human action failure probability. It would be prudent to credit to develop multiple human action failure probabilities depending on the type of trip. Breaking out the trips based on S/G water level would be a good start (low, normally, and high).	A	Add three flavors of OTCC mitigation.	This finding has been resolved and closed by an update to the model/documentation. Multiple operator actions were added to the model to account for different available times to initiate OTC (based on SG level and trip and whether AFW operated for some time after trip). The current model included Human Failure Events (HFEs) to initiate OTC following normal or low level trips with short term loss of FW, loss of FW after operating for at least 4 hrs, and loss of FW after CST depletion.
AS-03	AS-A2 AS-A4 AS-A3 AS-A5 AS-A7 AS-A10 AS-B1 HR-E1 HR-F1 HR-F2 SC-A3 SC-A6 SY-A2 SY-A22 QU-A1	MFW is not credited post trip. This leads to quite a few high level cutsets that are overly conservative. If post S/G level control is automatic, then only the control system hardware need be modeled. If not, then the human action to control S/G water level need be modeled. The availability of the TBVs post quick open prevents the need for hot well make-up. Crediting the ADVs for use with MFW would require the modeling of hotwell make-up.	A	Credit MFW post trip following IEs where MFW is available.	This finding has been resolved and closed by an update to the model/documentation. MFW modeling was enhanced and credited post trip where applicable.
AS-04	AS-A5 AS-A7 AS-A10 AS-B1 SC-A6 QU-A1	RWT rupture is assumed to fail shutdown cooling. This seems overly conservative. Without make-up the level in the RCS would drop, but there is more than enough fluid in the Boric Acid Tanks and the VCT to restore this level. The level does not need to be fully restored to allow shutdown cooling. The level need only be above the hot leg. Estimated Level Drop 2250 psia at 600 F (0.0217 ft^3/lbm) to 100 psia at 300F (0.01766 ft^3/lbm). Given RCS liquid volume of 10,400 ft^3, this means approximately 18,500 gallons are required to restore the PRZ level. Each Boric Acid Tank contains 9700 gallons the VCT contains 4000 gallons. Fully PRZ level is not required full shutdown cooling when core damage is the alternative.	B	Require either (RWT) or (2 BASTs and a VCT) for shutdown cooling	This finding was reviewed and closed with no further action. This scenario was reviewed to be contrary to current plant practices and EOPs. Use of RWT and BAMTs is required when RCS needs makeup due to shrinkage following Rx trip. The reviewer assumes that Ops will continue to SDC, even in case if RWT rupture were to occur during makeup to the point where RCS cooling continues at a rate of 100F/hr, and RCS level drops down to about midloop level. EOP-02, step 4.5 prevent against this behavior by requiring that OP ensure RCS inventory control is maintained, and PZR level is restored between 30% to 35%. SDC will not continue until PZR level is restored. The concerned scenario is perceived as not credible.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
AS-06	AS-A3 AS-A4 AS-A5 AS-A7 AS-A10 AS-B1 HR-E1 HR-F1 HR-F2 SC-A3 SC-A6 SY-A2 SY-A22 QU-A1.	Consider adding low pressure feed (using Condensate pumps) to the model for accident sequences involving loss of all MFW/AFW. Using condensate pumps to feed the SG's is in both EOP 6 'Total Loss of Feed' and EOP-15 'Functional Recovery Procedure'. Operations is directed to use low pressure feed in 1-EOP-06 (Step 8.B.3.1). Crediting low-pressure feed will eliminate those core damage sequences where the MFW pumps are lost, but the condensate pumps are available. If the TBVs are not available, then the hot well make-up control system (or an operation action) must be modeled to incorporate this alternative. Adding LPF could reduce dependency on Once Through Cooling for a number of accident sequences. (See F&O AS-03 also)	B	Consider including Low Pressure Feed from the condensate pumps in accident sequences that include TLOF. (Current model remains conservative).	This finding has been resolved and closed by an update to the model/documentation. LOMFW IE events were combined in data update into a single IE and recovery events were developed based on the type of events that have occurred. Low pressure feed would be included if applicable based on IE data review.
AS-08	AS-A5 AS-A7 AS-A10 AS-B1 QU-A1 SC-A6 SY-A1	Check Valves 09294 and 09252 are common for both AFW, MFW, and Low Pressure Feed. These CKVs currently appear only in the AFW system. The may be some events (e.g. LOL) where the turbine trips and steam generator pressure rises enough to cause the closure of these check valves. Under these scenarios, the failure of both of these checks would fail all secondary side heat removal. Currently, these CKVs are modeled under FMM1SGCVLV. This event has a failure probability far lower than several three element CKV groups in the AFW system. There does not appear to be a basis for this difference. The failure likelihoods (independent and common cause) of the check valves in the AFW system should be consistent or the basis for the difference is documented. Further, as the random failure of these CKVs could cause a LOFW trip and eliminate all secondary side feed to a single S/G, this is worthy of consideration as an initiating event.	A	Document basis for CKV failure rates and ensure CKVs appear in the MFW and low pressure feed portions of the tree.	This finding has been resolved and closed by an update to the model/documentation. The models were revised to ensure failure of the referenced check valves have the expected impact on loss of feedwater to applicable SG. Data Analysis was revised to use the latest industry as well as plant-specific data.
AS-11	N/A	Documentation used to provide the basis of event tree structure is not adequately traceable to the underlying analysis.	B	Provide references in accident sequence analysis documents to the supporting thermal-hydraulic analyses. Describe how analyses are made applicable to the PSA, e.g., justify why licensing and design basis analyses, which make various non-PSA related assumptions and are often very conservative, can be used for defining accident sequences and timings.	This finding has been resolved and closed by an update to the model/documentation. The current revision of the stand-alone Accident Sequence Analysis has revised and corrected many editorial issues existed in its predecessor analysis document.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
AS-12	AS-A5 AS-A7 AS-A10 AS-B1 QU-A1 SC-A6	Currently, shutdown cooling is credited as a long-term cooling method to eliminate the re-circulation requirement on certain ranges of LOCA breaks. A certain amount of water must be above the bottom of the hot leg to avoid drawing vapor into the shutdown cooling system. Some calculation must be done to ensure that the RCS will be above this critical point. This calculation could be quite simple: determine the RCS water level at the point of shutdown cooling entry conditions, determine the leakage rate at the point, verify the RCS level will be adequate for the remaining part of the 24 hr mission without re-circulation or RCS make-up. If this is not true, then addition make-up must be modeled through the emergency sump or CVCS.	B	Do simple calculation. Take appropriate action.	This finding was reviewed and closed with no further action. To enter SDC during the course of mitigating a LOCA, an RCS level of 30% in the Pressurizer is required by EOP-3. Further, once SDC has been entered ONP-01-03 requires OTC to be re-established if RCS level falls below 29 feet 9.5 inches (Top of Hot Leg). Operation of SDC in the above referenced condition is procedurally not allowed and physically not possible. Therefore the question is highly hypothetical and not applicable at PSL.
AS-13	AS-A2 AS-A3 AS-A5 AS-A7 AS-A10 AS-B1 HR-C1 HR-C2 HR-C3 QU-A1 QU-B10 SC-A3 SC-A6 SY-A1 SY-A7 SY-A8 SY-A9 SY-A11 SY-A12 SY-C1 SY-C2	The PORVs are only assumed to lift given total loss of secondary side heat removal or a loss of load with no anticipatory trip. This appears non-conservative. The only loss of load trips considered are TT and loss of off-site power trips. This is based on an informal calculation that shows the RCS pressure exceeds 2300 psia, but stays below the PORV open set point of 2400 psia. This does not consider variations in the time delay between the turbine trip and the reactor trip nor does it consider variations in the PRZ pressure set point. Consideration of these variables may lead the analyst to conclude that the likelihood of a PORV lift during this condition is much larger than analyzed. Further, the portion of the tree (under Gate U1QT99) that models the circuitry associated with the anticipatory trip only contains a single basic event. No other support system dependencies appear. For example, does the status of pressurizer spray affect this calculation? Are there support system failures that could cause a loss of load and disable or degrade the anticipatory trip function?	B	Model support systems for anticipatory trip. Consider a likelihood that the PORVs open on LOL with anticipatory trip. Consider a higher PORV challenge likelihood during other types of trip where the delay between the turbine and reactor trip is larger (e.g. spurious MSIV closure)	This finding has been resolved and closed by an update to the model/documentation. Input on trips likely to challenge PORVs were received from fuels group and incorporated in models update - see logic under gate U1QT03.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
AS-14	AS-A5 AS-A7 AS-A10 AS-B1 QU-A1 SC-A6	The top Unit 1 CD cutset is %ZZSU1*GMM1MRMOV. This cutset appear to be overly conservative. The base failure rate of a hand valve to transfer closed without a demand is in the 2E-7 range. The likelihood of a HV transferring closed should be 5 to 10 times lower. Further, the mission time for these MOVs is based on a 3-month test interval. As these MOVs are common to CS, LPSI, and HPSI (6 total pumps), it is highly doubtful that the MOVs will go more than a few weeks without passing flow. Considering this, the likelihood of this event is between a factor of 20 to 50 lower than currently estimated.	A	Use a more realistic failure likelihood for hand valves (or electrically isolated MOVs) spurious transferring closed. Adjust the mission time considering that 6 pumps with quarterly surveillance testing use the same flow path.	This finding has been resolved and closed by an update to the model/documentation. Changed exposure time for recirc valves transferring closed during standby to 2.5 months (see discussion below) (a) Changed TC rate for manual valve TC to 1.2E-07/hr based on latest generic data calc. (b) Per discussion with the pump and valve test engineer, the ECCS pumps (HPSI, LPSI, CS), and thus recirc flow paths, are tested within a week or so of each other. The 3-month exposure could be reduced at most by a couple of weeks. #-month test assumption is valid. No further change required.
DA-A1-01	DA-A1 DA-E1 DA-E2	Identifiers (i.e., type codes) are provided for the various types of components included in the PRA models for PSL Units 1 and 2. However, no evidence was provided to illustrate how the type codes are linked to the basic events in the PRA models or how to verify that the type codes are properly implemented. The system notebooks identified the basic events for which probabilities are required. This included basic events for independent and common cause failure of equipment to start and run, unavailability due to test/maintenance, and recovery of a function. Common cause failure basic events are listed in Tables 2-5 of PSL-BFJR-06-008, Rev. 2, and basic events due to unavailability are listed in Tables 13 - 16 of PSL-BFJR-11-08, Rev. 0. The identifiers for types of components are included in Tables 21 and 22 of PSL-BFJR-11-08, Rev. 0. However, a link between the type codes and basic events could not be identified.	Finding	The documentation should be enhanced to demonstrate the connection between the various types codes in Tables 21 and 22 of PSL-BFJR-11-08 and the basic events that are included in the PRA model. The enhancement can include a description of the naming scheme used for the basic events.	This finding has been resolved and closed by an update to the model/documentation. Naming scheme, Type Codes, and how Type Codes are linked to basic events in CAFTA models are all described in the IPE document which has not changed. PRA maintenance and update documents only describe departure or addition to specific modeling logic since IPE. However, the latest revision of PSL Data Analysis document considered all elements of the peer review recommendations were applicable.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
DA-C13-01	DA-C13 DA-E2	It was not clear how the assumptions obtained from knowledge plant personnel that was used to establish out-of-service hours. Table 13 identifies OOS hours and only modes 1-4 (on-line) were considered, as reported in Section 5.3. It is noted in the self assessment that the comment column in Table 13 contains "assumptions obtained from interviews with knowledgeable plant personnel" however this is not thoroughly documented. Look at examples in Section 5.3.3. - Check service water unavailability. (installed spare was OOS for a year).	Finding	Information obtained from knowledge plant personnel should be documented and include how the information is being used as part of the data analysis.	This finding has been resolved and closed by an update to the model/documentation. Intake Cooling Water (ICW) system has 3 pumps that are rotated for service on equal basis to allow only two pumps operating and deliver flow to 2 trains during at power operation. The identified comment was not found in the Data Analysis Document. The SR states "INTERVIEW knowledgeable plant personnel (e.g., engineering, plant operations, etc.) to estimate ranges in the unavailable time per maintenance act for components, trains, or systems for which the unavailabilities are significant basic events." There is no specific requirement to document such interview listed in this SR. All inputs to PSL Data Analysis are provided either by system engineers or ex-shift manager used to hold SRO license at PSL and currently working as PRA group member in support of PSL PRA development whose review and signature meets, if not exceeds, the requirements of this SR. The Revised PSL Data Analysis document was further enhanced to add a paragraph to clarify how input is received or communicated from knowledgeable plant personnel during completion of PSL Data Analysis. This F&O is considered resolved.
DA-C14-01	DA-C14 DA-E1 DA-E2	It appears that the treatment of coincident unavailability for inter-systems was considered. However, there was no clear documentation to demonstrate such treatment. Coincident unavailability due to maintenance for different trains of the same system (intra-system) is not allowed by established plant procedures. Therefore, the calculation of coincident unavailabilities for intra-systems as a result of planned and repetitive activities was not calculated. There was no clear documentation on the treatment of coincident unavailability for inter-systems. Discussion with the utility PRA staff indicated that review of the plant operating history was performed to identify potential coincident unavailablity for inter-systems. No such unavailabilities were identified. The PRA staff also demonstrated that the PRA model accounts for coincident unavailability for inter-systems by the use of appropriate mutually exclusive logic.	Finding	The documentation for data analysis should be enhanced to clearly address the treatment of coincident unavailability for inter-systems. This can include a documented review of plant operating history to demonstrate whether planned repetitive activities are performed that allowed the unavailability due to maintenance between systems.	This finding was reviewed and closed with no further action. This SR requires "EXAMINING" coincident unavailability due to maintenance for redundant equipment (both intrasystem and intersystem) that is a result of a planned, repetitive activity based on actual plant experience. The key words here are "PLANNED" and "REPETITIVE". At PSL, there is no coincident unavailability of "PLANNED" and "REPETITIVE" maintenance for redundant equipment (both intrasystem and intersystem) to be allowed, per plant T/S, procedure, guidelines, and instructions.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
DA-D4-01	DA-D4 DA-E1 DA-E2	No clear evidence was provided to demonstrate that the posterior distributions resulting from Bayesian updates were checked for reasonableness based on the plant-specific evidence that was used. These items have been considered and checked however no further discussion or examples are given (see Section 5.4.6). It appears this has been done, however more information should be provided to verify this (or that all the numbers should have been Bayesian updated).	Finding	The documentation should be updated and enhanced to included criteria to be used in establishing the appropriateness of posterior distributions resulting from Bayesian updates. NUREG/CR-6823 describes an approach for performing a consistency check of prior distributions used in Bayesian updates.	This finding has been resolved and closed by an update to the model/documentation. The Data Analysis calculation document discussed the criteria that were used to meet the intent of this SR. Revision 1 of the same document is enhanced to specifically address this F&O and added discussion for how consistency of data and Prior was performed by examining the difference between Bayesian Updated Mean value and Prior Mean value for each Bayesian updated analysis and showed that such difference is significantly less than 9E-03. NUREG/CR-6823 considered a difference of 0.05 as small. This concludes that use of Prior data in each Bayesian Update analysis was reasonable as Priors and Posteriors are comparable and close to each other with small difference.
DE-01	DA-C1 DA-D5 DA-D6 SY-B1 SY-B3 SY-B4	The common cause analysis has very few electrical components (AC and DC) considered for common cause grouping. The EDG's, batteries, and reactor trip breakers appear to be the only electrical components in the CC analysis. An evaluation or analysis to justify the exclusion of other electrical components (breakers, relays, inverters, MCC's, etc...) could not be found in the references.	B	Expand common cause analysis to justify limited electrical equipment inclusion. Or Provide clear justification for exclusion of certain AC & DC components.	This finding has been resolved and closed by an update to the model/documentation. CCF modeling and data was expanded to include data and components consistent with component types provided in INEL 94-0064 and latest related industry documents.
HR-D1-01	HR-D1	Some inconsistencies have been identified between the documentation, the HRA calculator file and the CAFTA model; one example is AHFL1CSTIV, which is indicated as 2.7E-5 in the summary table 3.0 while appears to have the floor value from ASEP in the HRA calculator file (i.e., 1E-5) (See F&O HR-D1-01). A more conservative value has ben entered in the model, with respect of what the HRA calculator provides.	Finding	Ensure consistency between the HRA calculator results and the summary table 3.0 (also explicitly include the conversion from median to mean for better traceability).	This finding has been resolved and closed by an update to the model/documentation. The bases for pre-initiators HFEs were documented in Calculation "St. Lucie Pre-Initiator Human Interaction Analysis" that was developed by ERIN Engineering and Research, Inc. in July 2009, and was later revised by a subsequent repetitive model updates. The referenced event was confusing for peer reviewer due to use of earlier revision of HRA calculator that maintained Median values which were converted to Mean values outside the calculator and before use in CAFTA. All values used in CAFTA are Mean values.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
HR-G6-01	HR-G6	This SR requires a check of the post-initiator HEP quantifications, and a review of the final HEPs to check their reasonableness given the scenario context, plant history, procedures, etc. Although a cutset review was performed as part of the quantification to verify the correct application of HEPs, no detailed discussion of the final HEPs to check their reasonableness with respect to each other was found. In particular, several single HEPs were identified that had higher probabilities of success than combination events that credited the same HEPs. (e.g. event JHFPSDCR vs Combination138), but discussion of how/why that was reasonable was provided. The combinations identified do not appear to be complete. For example, the combination of FHFP1RECMFW-N and GHFPOTCTGT42 shows up in the cutsets, but is not addressed in the combination analysis. Additionally, the dependency analysis that was performed did not have a reasonableness check of the total combined human failure provided. Several total combined failures were significantly lower than 1E-10, with several lower that 1E-16. The HRA notebook does not have a lower bound for total combined human failures. A failure probability of 1E-16 is equivalent to the operation failing for all actions in the given sequence to be 1 out of 10,000,000,000,000,000 times. Based on feedback from other reviews and regulator opinion, this is a very optimistic view of the potential for operations to recover a sequence. Several total combined human failure probabilities applied in the model maybe over optimistic and exceed the best practice for the lower limit.	Finding	Provide a discussion of the reasonableness of the final HEP values taking into consideration the scenario context, plant history, procedures, etc. In particular discuss the single events that have higher probabilities of success versus their associated combination events since this is not typically seen. Apply a lower bound for total combined human failures. A typical best practice for a lower limit is 1E-06. If no lower bound is deemed warranted, discuss the validity of the combination events that have extremely low HEPs.	(a) The recovery rules in the recovery rule file are applied in order of ascending probability. In some cases, the probability of a single HFE can be less than a combination event in which the HFE is a constituent. This can occur when the HFE is not the first chronological HFE in the combination. As an example, let's say the probability of HFE A is lower than the combination of HFE A and HFE B. This can occur when the probability of HFE A is appreciably higher than HFE B, and HFE A is dependent on HFE B in the combination event AB. It can be argued that combination event probability should not be higher than any of the constituent HFE probabilities, and that is how the PSL recovery rules are applied. No matter what the interpretation, the effect on the risk calculation is minimal. (b) This is due to the fact that the list of combination events was generated for the HRA well before the PSL models had been finalized. The list of combination events was generated from the latest draft model updates that were available at the time. The models have changed considerably since that time, so it is not surprising that some unanalyzed combination events are showing up. Those cutsets that have a combination of HFEs that do not appear in the HRA dependency analysis will at least have credit for one of the HFEs applied in the cutsets. At worst, the cutsets frequency will be somewhat conservative. (c) To check the reasonableness of the HFE combinations and dependency analysis, a sensitivity study was performed using HFE combination floor values of 1E-5 and 1E-6. This analysis was documented and combination events with a probability lower than these floor values were reviewed for timing, cues, etc. to check the dependency with other operator actions in the cutset.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
HR-I2-01	HR-I2	This SR is associated with the documentation of the process used to identify, characterize, and quantify the pre-initiator, post-initiator, and recovery actions considered in the PRA including the inputs, methods and results. Although the overall HRA analysis looks very good, the current documentation for the dependency analysis and treatment of post-initiator HRAs is incomplete. The current documentation only states that all post-initiator HEPs are set to 1.0 and then fed into the HRA calculator to determine the dependency between the HEP events. There is no discussion of how the rest of the process is performed, including how the HRA Recovery File is used to "reset" combination events to the appropriate values based on the dependency analysis, no discussion on why the HEP values in the BE file are set to 0.5, no discussion of how the HRA calculators dependency analysis was validated, etc. Additionally, there is no assurance that all HEP combinations have been identified and evaluated - see F&O HR-G6-01 for more detail.	Finding	Provide a complete discussion of the dependency analysis process followed, including the format/structure of the recovery rules file and the basis/use of nominal values for HEPs in the BE file, the check/validation of the HRA Calculator output file for dependencies, etc.	This finding has been resolved and closed by an update to the model/documentation. The latest revision of HRA analysis included use of revised dependency analysis methodology that ensured generation of HEP combinations consistent with dependencies between HFEs that are considered in HRA Calculator. The HRA Analysis document included detailed steps taken to generate the revised dependency analysis.
HR-I3-01	HR-13	There is no discussion on model related uncertainties for pre-initiator HRA calculations. For post-initiator HFE, the EF indicated in Table 9 are then not propagated in the CAFTA file. It is therefore not clear how the uncertainty parameters are treated in the model. A complete uncertainty assessment involves both stochastic uncertainties (included in the HRA calculator) and epistemic (model) uncertainties. A discussion on the assumptions made in the analysis and their potential for impact on the HEP calculations is required to meet the SR. The inconsistency between the EF discussed in the post-initiator HRA notebook (table 9 in Section 3.3) and the actual CAFTA file does not allow a correct uncertainty analysis. Table 9 states that generic Error Factors are used, but there are no error factors in the BE file, so it is unclear how the error factors are propagated. Also, the Combination events, and the renamed post-initiator single events are not included in the BE file so it is unclear how their error factors are included in the analysis - or if they are even considered.	Finding	Provide a discussion on the model uncertainties associated with pre-initiator and post-initiator HFEs. Ensure that the EFs discussed in the documentation are used in the CAFTA model. Provide a discussion of how the EFs are modified/impacted by the dependency assessment.	This finding has been resolved and closed by an update to the model/documentation. Pre-initiator and post-initiator HFE EFs were added to CAFTA RR-file so UNCERT can use them in the uncertainty calculations. Discussion of HRA EF uncertainties is provided in the PSL HRA analysis document.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IE-01	AS-B1 IE-A1 IE-A2 IE-A5 IE-A6 IE-A9	LOSP Initiating Event was extracted from generic industry data going back 20 years or more (Calculation No.PSL-BF-JR-01-005). No data trending was applied to establish a downward trend in LOSP annual frequency. The latest biannual EPRI report on LOSP frequency concluded that: LOSP frequency has trended downward and has stabilized over the last few years. The non-trended derived PSL LOSP frequency (Total value approximately 5.3E-02 Table 6.1) is very conservative; and the probability of non-recovery of offsite power (shown in the Log-normal cumulative figures) is also too high. This high degree of conservatism in PSL LOSP frequency and associated non-recovery probabilities may lead a PRA practitioner to determine unnecessarily high risk for some applications that would otherwise be acceptable.	B	Perform PSL LOSP data trending or use the trended and analyzed data provided in the latest biannual EPRI report on Loss of Offsite Power at Nuclear Power Plants through 1999 (contact Frank Rahn at EPRI). The up-to-date generic data would be applicable to PSL and would be sufficiently conservative since PSL1 and PSL2 have not experienced any LOSP during the last dozen years. Once the EPRI data is used, the existing detailed report on LOSP frequency (PSL-BFJR-01-005) may be archived.	This finding has been resolved and closed by an update to the model/documentation. The latest Off-Site Power Non-Recovery probability calc document included consideration of LOSP industrial events occurred during 1997-2008 as published by EPRI documents. Further events from 2008 on will be considered during the cyclical maintenance and update of this calculation document. The downward trend of LOSP annual frequency should have improved effects on the model and thus the current levels are considered conservative.
IE-04	AS-B1 IE-A2 IE-A5 IE-A6	St. Lucie includes loss of individual 120VAC instrument buses as initiators, but does not address multiple bus failures as initiators. The 120 VAC buses power the RPS/ESFAS. Failure of 2 buses could result in spurious actuation of multiple safety systems given the 2 of 4 actuation logic. This type of initiator has not been addressed for the St. Lucie PRA. Multiple actuations could have unanticipated effects such as actuation of the Feed Only Good logic for both steam generators at the same time that AFAS was actuated. This would result in no auxiliary feedwater being supplied to the steam generators. Note: panels are co-located in pairs. Construction activity noted in area. Construction materials/ debris could block cooling intakes and cause failure. This is one example of potential common cause mechanism.	B	FP&L needs to specifically evaluate the potential impact of failure of multiple 120VAC buses in section 2 of calculation PSL-BFJR-02-001, REVISION 0. If these potential initiators are found to have a significant impact on system response, they should be incorporated in the model.	This finding was reviewed and closed with no further action. Multiple instrument bus failures are judged to be a low probability. Impact of loss of two instrument busses is judged to be covered by the LODC IE which impacts two instrument channels.
IE-05	AS-B1 IE-B1 IE-B2 IE-B3 IE-B3 IE-B4 IE-C13	PSL PRA-2.P presents the ISLOCA calculation. It has not been updated since 1992. This calculation does not address the RCP seal cooler heat exchanger tube leak ISL path nor does it discuss treatment of common cause failure of valves for the other ISL paths.	B	The ISLOCA analysis needs to be updated to include the RCP seal cooler heat exchanger tube leak ISL path. The issue of common cause failure of valves also needs to be discussed. in many cases common cause failure of valves in various pathways can be discounted based on different operating conditions, but this issue needs to be discussed explicitly.	This finding has been resolved and closed by an update to the model/documentation. Latest update of ISLOCA analysis was completely revised and issued on 4/18/2011. The revised analysis considered all previously identified issues related to ISL.
IE-07	DA-C16 IE-A8 IE-A9 IE-C1 IE-C11 IE-C2 IE-C3 IE-C4	The IE data documentation is scattered in different reports and in different revisions of the same report. Sometimes, inconsistent values are provided (see F&O IE-05).	B	Revise IE data reports in a consistent manner	This finding has been resolved and closed by an update to the model/documentation. Updated IE data are revised and documented in the latest stand-alone PSL Data Analysis document.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IE-08	IE-D1 IE-D2	All of the St. Lucie PRA documentation are calculations covered by the FP&L engineering calculation procedure, ENG-QI-1.5. This procedure requires independent review and signoff of all calculations performed per this procedure. The latest St. Lucie PRA update was not fully completed at the time of the peer review so most documents had not been independently reviewed at the time of the peer review.	A	Do Review	This finding has been resolved and closed by an update to the model/documentation. The peer review team reviewed draft calculation document dated 2002. All calculation documents generated in support of current PRA model update were independently reviewed/signed-off and approved consistent with the current Quality Instructions and PRA standards requirements.
IE-C5-01	IE-C5	The approach for generating the initiating event frequency utilizes an adjustment factor that ratios the exposure time. The exposure time is not the mission time for reliability and the approach does not produce an appropriate value for frequency of occurrence. Utilize the idea of initiating event as the first valve failure based on having to remain isolated for the period of one year. Then consider the unavailability of the other valves in the line based on exposure time. Systematically address each valve as if it is the holding valve.	Finding	The adjustment factor is not appropriate for unavailability.	Resolution in-progress. See E2-Table E2-C1
IE-C6-01	IE-C6	A screening approach is utilized for some lines based on low frequency but this is not quantified. The SR indicates a frequency expectation for screening. Define the estimate for the lines screened on low frequency and show that the calculated frequency supports screening.	Finding	There is not quantification of excluded ISLOCA scenarios.	Resolution in-progress. See E2-Table E2-C1
IE-C9-01	IE-C9 IE-C10 SC-A5	The fault tree model used for the ISLOCA paths assumes that the status of all valves is known when the plant is brought online and the corresponding exposure time is the refueling interval. However, based on discussions with knowledgeable staff, there is no positive means to know that more than one isolation valve is actually holding. Use of status lights is not definitive since there is a +/-5% margin between light changing and valve seating. The exposure time should be based on a positive flow test which may not occur on a refueling basis but based on other studies could be as much as the life of the plant.	Finding	Model should use one year exposure time for the first failure and then do the CCDP following the failure. F&O finding level produced.	Resolution in-progress. See E2-Table E2-C1
IFEV-A1-01	IFEV-A1 IFSN-A5	The identification of scenario-induced failures of components was not complete. A check of the flood scenario discussion and supporting information provided in Excel spreadsheet, PSL_FrequencyCalculations_AndTag_Database_3-24-2011.xlsx, indicated that corresponding plant initiating event group for each flood scenario was identified. However, identification of scenario induced failures of components was not complete, as discussed in F&O IFSN-A5-01.	Finding	Refer to F&O IFSN-A5-01.	This finding has been resolved and closed by an update to the model/documentation. Equipment locations and vulnerabilities are discussed in the Internal Flooding Calculation document which is further revised.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IFEV-A7-01	IFEV-A7	The consideration of human-induced floods was not included in the internal flooding evaluation. The consideration of human-induced floods was not included in the internal flooding evaluation. EPRI report 1013141 that provided generic data for flood initiating event frequencies stated that "Human induced causes of flooding that do not involve piping system pressure boundary failure such as overfilling tanks and inappropriate valve operations that release fluid from the system are not included."	Finding	Human-induced floods should be evaluated and included in the internal flooding analysis.	This finding was reviewed and closed with no further action. As noted in the main Flooding Analysis document, no condition reports that would reflect such problems, associated with the possibility that plant design and operating practices might affect the likelihood of flooding, were to be found. This possibility was reviewed with experienced plant staff after the peer review that identified one issue "the periodic transfer of waste water from Unit 2 to Unit 1." The document was further revised to address the issue.
IFEV-B3-01	IFEV-B3	Sources of model uncertainty and related assumptions were not documented. A review of the notebook sections that discussed the flood scenarios did not provide any evidence that uncertainties associated with the internal flooding initiating event frequencies was addressed and documented. For example, a source of uncertainty may include the range of the initiating event frequencies for each flood scenario.	Finding	The sources of model uncertainty and related assumptions should be documented to meet this requirement.	This finding has been resolved and closed by an update to the model/documentation. The main internal flood analysis document as well as Internal Flood Quantification document were further revised to addresses the analysis uncertainty.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IFPP-B1-01	IFPP-A5 IFPP-B1 IFPP-B2 IFSN-A12 IFSN-A13 IFSN-A15	The documentation associated with the plant partitioning is scattered between the initial portion of the document and the walkdown report in Attachment B, which in reality is a discussion of the screening of main structures such as major structures. The walkdown report does not include any explanatory picture and mixes the definition of the area and their screening, without spelling out the generic criteria used for the screening of specific structures. This organization of the information is prone to confusion; moreover, since the area identification and the screening are mixed, some overlook have been noticed. For example, the walkdown notes explicitly mention which bldg has been walked down and the DG BLDG is not listed among those, still, the screening of the DG BLDG is only discussed in the walkdown report and they are all screened out on the basis that there is no service water (DG are air cooled); there is nevertheless no mention of the potential spray effects of Fire Protection system on a single DG (FP lines have been noticed during the walkdown that may have the potential to spray on the DG cabinet). While the screening of the DG BLD may still be possible (FP lines may be dry since there are large FP valves immediately outside of the DG building that may be deluge valve, or the DG AOT may be sufficient to recover from a spray event on the DG cabinet), the presence of a flood source that has the potential for impacting PRA equipment needs to be addressed. The screening out of the Turbine Building is another example of screening process inconsistent with the screening criteria provided in the standard. While it is true that the TB BDLG is open, a rupture in the condenser expansion joints will induce an initiating event and for this reason the area cannot be screened out for flood considerations. The flood scenario generated by a rupture of the condenser expansion joint may be screened for other reasons (e.g., it may be folded into an already existing IE category with identical plant effect but higher IEF), still a discussion of the reasoning and of the screening criteria needs to be provided. Finally, section 4.1.1 points to the walkdown notes but incorrectly indicating Attachment C rather than Attachment B.	Finding	Clearly spell out in the text the screening criteria used for not including in the analysis some of the major structure and assure that the walkdowns notes support the screening. Specify in the walkdown report if/how the information collected through existing plant database/documentation has been checked for accuracy to ensure that the plant partitioning reflects the as built, as operated plant. For screening of areas, it is suggested that a summary table is provided with the areas that are screened and the associated rationale.	This finding was reviewed and closed with no further action. No flood zones or flood sources located within the reactor auxiliary building were screened out. The flooding analysis document was further revised to address flooding originating in the other buildings or areas even if this is not truly "internal" flooding. These do not result in additional scenarios that need be quantified as no previously unaddressed reactor scram need ensue after such an event. This SR explicitly requires discussing other than spray/submergence failure modes.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IFPP-B3-01	IFPP-B3	There is no discussion of sources of uncertainties associated with the plant partitioning phase. The SR specifically requires a discussion of the uncertainties associated with the plant partitioning. The self-assessment table (Table 3.5.1.1) points to the quantification notebook, which is not addressing uncertainties associated with the plant partitioning but only stochastic uncertainties propagation in the final results.	Finding	Explicitly discuss sources of uncertainties associated with the plant partitioning phase. Plant partitioning is for example highly dependent on location and normal position of doors (i.e., a normally open rather than a normally closed door can change the definition of a flood area).	This finding has been resolved and closed by an update to the model/documentation. The internal flood analysis documents were revised to addresses the analysis uncertainty. The principles governing the plant partitioning into flood zones are discussed in the analysis document. In general, flood zones are individual plant areas that could reasonably contain or delay propagation of water or in which water levels might be significantly different to those in adjoining areas. Walls, curbs and doors were used to identify flood boundaries. Individual adjoining rooms were combined into single flood zones if there is no impediment to the propagation of flood water between them. Conservative assumptions governing the quantification of the model – all equipment vulnerable to spray damage assumed to fail at the onset of flooding in a flood zone, no credit for recovery of flooded components, no credit for drains that would limit the height of the flood - would more than compensate for any uncertainties in the partitioning scheme. The analysis document was further revised to state that the possibility of a normally closed door being open was considered in calculating flood heights but was found to mitigate the consequences in scenarios of concern.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IFQU-A10-01	IFQU-A10	Some inconsistencies in the mapping between the flood events and the basic events associated with impacted equipment has been identified. According to table 20, the top cutsets have to do with ATWS induced by a spray event on the reactor trip switchgear. Spray on the reactor trip switchgear would result in loss of power, which would result in the trip itself. Therefore, even though the reactor trip switchgears are actually impacted by the spray event, the flood initiator needs not to be mapped with the basic event associated with the switchgear because their failure is in the direction of the success. Another example of suspect inconsistency in the mapping is observed from the review of the main CDF contributors. A spray from room 1RAB43-59/58 is not expected to impact both trains since the originating room only hosts 1 train of batteries.	Finding	Review mapping between impacted components and associated basic events to ensure that the flood induce failure is consistent with the failure mode modeled in the actual BE.	This finding was reviewed and closed with no further action. Mapping between impacted components and associated basic events was reviewed to ensure that the flood induce failure is consistent with the failure mode modeled in the PRA model. Changes to the mapping tables were implemented to address the concerns of this F&O. The one inconsistency related to the spray event affecting the trip switchgear has been corrected. The scenario referenced in the second part of the review comment does not involve a spray in rooms 1RAB43-58 and -59 but rather a flood emanating from the battery rooms to the neighboring switchgear rooms through the connecting doors and submerging various electrical components inside. Since the analysis does not credit isolation of the break, the flooding will persist over the number of hours. The mapping reflects the equipment disabled by the accumulating water not only in the battery rooms but also in the neighboring electrical rooms, affecting both electrical trains.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IFQU-A1-02	IFQU-A1	Some inconsistencies in the mapping between the flood events and the basic events associated with impacted equipment has been identified. According to table 20, the top cutsets have to do with ATWS induced by a spray event on the reactor trip switchgear. Spray on the reactor trip switchgear would result in loss of power, which would result in the trip itself. Therefore, even though the reactor trip switchgears are actually impacted by the spray event, the flood initiator needs not to be mapped with the basic event associated with the switchgear because their failure is in the direction of the success. Another example of suspect inconsistency in the mapping is observed from the review of the main CDF contributors. A spray from room 1RAB43-59/58 is not expected to impact both trains since the originating room only hosts 1 train of batteries.	Finding	Review mapping between impacted components and associated basic events to ensure that the flood induce failure is consistent with the failure mode modeled in the actual BE.	This finding was reviewed and closed with no further action. Mapping between impacted components and associated basic events was reviewed to ensure that the flood induce failure is consistent with the failure mode modeled in the PRA model. Changes to the mapping tables were implemented to address the concerns of this F&O. The one inconsistency related to the spray event affecting the trip switchgear has been corrected. The scenario referenced in the second part of the review comment does not involve a spray in rooms 1RAB43-58 and -59 but rather a flood emanating from the battery rooms to the neighboring switchgear rooms through the connecting doors and submerging various electrical components inside. Since the analysis does not credit isolation of the break, the flooding will persist over the number of hours. The mapping reflects the equipment disabled by the accumulating water not only in the battery rooms but also in the neighboring electrical rooms, affecting both electrical trains.
IFQU-A5-01	IFQU-A5	Flood-specific actions are credited as successful without a supporting HRA. There are examples of flood specific HRA (e.g., isolation of CC header) that are credited as successful without an HRA being performed.	Finding	Perform HRA on the action that are credited.	This finding has been resolved and closed by an update to the model/documentation. The latest revision of Internal Flooding Quantification included revised Flooding-HRA analysis consistent with the EPRI guideline.
IFQU-A6-01	IFQU-A6	Flood impact on HRA is not documented. Section 4.5 on HRA only lists the changes made to the HEP (or not made) without any explanation of the reason why. There is no discussion on how the flood specific PSF are addressed and of which flood scenario requires modification of existing HEPs or dependency values.	Finding	Document the HRA for flood-induced PSF.	This finding has been resolved and closed by an update to the model/documentation. The latest revision of Internal Flooding Quantification included revised Flooding-HRA analysis consistent with the EPRI guideline.
IFQU-B2-01	IFQU-B2	The flooding quantification notebook only provides a list of FRANX files and a summary of the results. The documentation of the PRA modeling and quantification of the internal flooding can only be inferred by the FRANX tables.	Finding	Explicitly provide a discussion on how the information from the Internal Flooding analysis notebook and associated attachments is translated in the PRA modeling through the FRANX software.	The latest revision of Internal Flooding Quantification document included discussions and listing of mapped tables and data used in FRANX.

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ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IFQU-B3-01	IFQU-B3	There is no discussion on how the EF are calculated for the stochastic uncertainties in the Internal Flooding analysis notebook (section 4.2 or 4.3 provide the IEF calculations for the various cases but not the EF, which do not also appear in the "calculation" tab of the Excel file PSL Frequency Calculations and TAG Database (3-24-2011).xlsx. An uncertainty analysis is presented in the Flooding quantification notebook but the EF associated with the Flooding initiators are not discussed.	Finding	Discuss the EF associated with the flooding initiators.	This finding has been resolved and closed by an update to the model/documentation. The latest revision of the internal flooding quantification document provided definition of Error Factors and discussion of how the Error Factors were used in the quantification.
IFSN-A5-01	IFEV-A1 IFSN-A1 IFSN-A2 IFSN-A5 IFSN-A6 IFSN-A10	While the equipment located in each area is indicated, its vulnerability is not always specified. For example: Appendix A for room 1RAB43-56 lists (under the "equipment located within area") equipment without any consideration on its vulnerability (i.e., elevation and spray vulnerability columns are empty). This appears in numerous other flood areas. This challenges the reliability of the selection process for equipment impacted for each scenario since there is no clear way to defend how the equipment impacted from each scenario is selected.	Finding	Ensure the information associated with the "equipment located within area" is entered. Specify if the equipment listed in that section is PRA equipment or other equipment that may be able to induce an IE.	This finding has been resolved and closed by an update to the model/documentation. Equipment locations and vulnerabilities are presented and discussed in the Internal Flooding Analysis document.
IFSN-A6-01	IFSN-A6	There is no discussion on other than spray/submergence failure modes SR explicitly requires to discuss other than spray-submergence failure modes.	Finding	If other-than spray and submergence failure modes are not addressed, explicitly say so.	This finding was reviewed and closed with no further action. Failure modes associated with pipe whip, jet impingement and the other consequences of High Energy Line Breaks were explicitly considered in the St. Lucie flooding analysis. However, as noted in the analysis document, the main steam and feedwater lines at PSL run outside the reactor auxiliary buildings into containment; their rupture is therefore not an internal flooding event. The concern is therefore limited to the rupture of the steam generator blowdown system in the piping penetration room and the steam generator blowdown room and the chemical and volume control system in the piping penetration room, the letdown heat exchanger room and the valve gallery. These flooding scenarios are fully addressed in multiple sections in the PSL Internal Flooding Analysis document.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IFSN-A7-01	IFSN-A7	There is no discussion on the basis for the assessing the vulnerability of equipment located in the area. There is no clear traceability of the actual equipment impacted for each scenarios. For example, the equipment listed in room 2RAB-10-16B includes equipment such as HCV-3625 and HCV-3627. For the first valve there is an indication of the elevation from the ground, but not for the second one; for both valves there is no indication of spray vulnerabilities. Both these valves are not indicated as impacted in the scenario 4.3.1.71 (according to the Excel file "PSL Frequency Calculations and TAG Database (3-24-2011).xlsx" under the calculation tab or under the "Unit_2_Tagged_Sections(QA)" tab. It is not possible to asses why they are not included and if this is appropriate.	Finding	Identify for each equipment the basis for their vulnerability such that the impacted equipment selection can be assessed. Examples of information that can be provided to facilitate the analysis are: Are they protected/sealed? Do they fail as they are and in the required position to respond to an accident? Are they manual or check valves not vulnerable to spray or submergence? Are they PRA equipment or not (see also F&O IFSN-A5-01).	This finding was reviewed and closed with no further action. As listed in selected scenario descriptions in the internal flood analysis document, damage to the individual components can be subsumed into the damage to pumps and the loss of equipment trains. Accordingly, there is no need to address the failure of these individual components. The document was further revised to highlight this position.
IFSO-A3-01	IFSO-A3	The screening of potential flood sources for each of the flood areas could not be determined. It does not appear the screening of any potential flood sources was performed. The flood areas for St. Lucie units 1 and 2 are listed in Table 4.1.1.1 and shown graphically in Figures 4.1.1.2-4.1.1.7 of PSL-BFJR-11-005, Rev. 0 for St. Lucie Unit 1. Likewise, the flood areas for St. Lucie Unit 2 are shown in Figures 4.1.1.8-4.1.1.13 of PSL-BFJR-11-005, Rev. 0. There is no listing of flood areas that were screened out from further evaluation and it could not be determine if screening of flood areas was performed. For example, no potential flood source is listed in Attachment A for the Control Room for Unit 2 (i.e., flood area 2RAB62-42i). This is a flood area that may be screened out because of the lack of a potential flood source. Attachment A noted that the emergency diesel generator flood areas were screened out, see page B-2 of PSL-BFJR-11-005, Rev. 0, from further evaluation. The screening of the emergency diesel generator flood areas is not allowed because it contains fire protection piping as a potential flood source and PRA-related component.	Finding	The screening of flood areas should be performed and documented. This should included the documentation of criteria used and their application in performing the screening and an assessment of fire protection piping that requires pre-actuation. [More refinement later!]	This finding was reviewed and closed with no further action. No flood zones or flood sources located within the reactor auxiliary building were screened out. The flooding analysis document was augmented to address flooding originating in the other buildings or areas even if this is not truly "internal" flooding. These do not result in additional scenarios that need be quantified as no previously unaddressed reactor scram need ensue after such an event. This SR explicitly requires discussing other than spray/submergence failure modes.
IFSO-A4-01	IFSO-A4	No evidence was provided to indicate that human-induced mechanisms were considered to determine their impact as potential sources of flooding. The flooding notebook indicated that the EPRI guideline, as documented in report 1019194, was used in performing the flooding analysis. The EPRI Guidance identified the flooding mechanism that would result in a release. No evidence could be found on the treatment of human-induced flooding. It appears that only pipe failures were considered as flooding mechanism.	Finding	All flood-induced mechanisms should be identified, including human-induced mechanisms that occur during maintenance activities and the failure modes of piping and other components.	This finding was reviewed and closed with no further action. As noted in the main Flooding Analysis document, no condition reports that would reflect such problems, associated with the possibility that plant design and operation practices might affect the likelihood of flooding, were to be found. This possibility was reviewed with experienced plant staff after the peer review who identified one issue "the periodic transfer of waste water from Unit 2 to Unit 1." The document was further revised to address the issue.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IFSO-A4-02	IFSO-A4	Flooding mechanisms including the failure modes of pipes, tanks, etc. are required to be identified for each potential flooding source. No evidence can be found in the documentation to indicate that human-induced flooding mechanisms were addressed. The flooding notebook indicated that the EPRI guideline, as documented in report 1019194, was used in performing the flooding analysis. The EPRI Guidance identified the flooding mechanism that would result in a release. No evidence could be found on the treatment of human-induced flooding. It appears that only pipe failures were considered as flooding mechanism.	Finding	All flooding mechanisms identified in the supporting requirement should be addressed in the identification of plant-specific flood sources.	This finding was reviewed and closed with no further action. As noted in the main Flooding Analysis document, no condition reports that would reflect such problems, associated with the possibility that plant design and operation practices might affect the likelihood of flooding, were to be found. This possibility was reviewed with experienced plant staff after the peer review who identified one issue "the periodic transfer of waste water from Unit 2 to Unit 1." The document was further revised to address the issue.
IFSO-A5-01	IFSN-A1 IFSN-A3 IFSN-A16 IFSO-A5	Capacity and temperature/pressure of flood sources are not clearly defined. While the flow rate is explicitly discussed for each source that is indicated in Appendix A and then discussed in the various scenario definition sections, there is no explicit discussion of the overall capacity of each source. In the SR explicitly requires identifying the capacity. For example, for scenario 4.2.1.1, a human action is credited to isolate the CC header, which has impact on the overall source capacity, which in turn impacts how far along the potential propagation path the scenario may have potential impact (i.e., if the capacity is higher because the human action is not successful, some water can propagate in the switchgear room at the lower elevation).	Finding	Clearly define for each scenario what is the overall flood capacity is associated to each source. If the capacity is modified by an operator action, clearly identify the operator action and discuss the operator action in the analysis, addressing whether different set of equipment (due for example to a different or more extensive propagation) is impacted in case of a successful or unsuccessful operator action.	This finding has been resolved and closed by an update to the model/documentation. Temperature and pressure data have been added to the release scenario spreadsheet created in response to the finding related to SR IFSN-B1.
IFSO-A5-02	IFSO-A5	The capacity of the flood sources and operating conditions (i.e., pressure and temperature) were not included in the characterization of the release for each flood. The characteristics of release for each flood source were identified in terms of the type of breach, range of flow rates, and capacity of the source. Based on information provided in Attachment D, a breach that results in a spray event is characterized by a flow rate of approximately 100 gpm. It appears that any breach that results in a flow rate greater than 100 gpm is characterized as a "flood" rupture. A check of the information related to Flooding Scenarios provided in Sections 4.2 and 4.3 revealed that in general the flood sources for each of the flood areas are characterized the a range of flow rates, and in certain cases the capacity of the flood source is also included. However, no clear evidence was provided to indicate the operating characteristics (i.e., pressure and temperature) of the flood source).	Finding	The operating conditions and capacity of each flood source should be included in the characterization of the release.	This finding has been resolved and closed by an update to the model/documentation. Temperature and pressure data have been added to the release scenario spreadsheet created in response to the finding related to SR IFSN-B1.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IFSO-A6-01	IFQU-A11 IFSN-A17 IFSO-A6	Confirmatory walkdown to assess the accuracy of the information associated with the source identification and scenario definition were not performed. One walkdown was performed before the identification of the flood source began but flood sources have not been confirmed during a dedicated confirmatory walkdown. Some potential inconsistencies between the isometric drawings used for the identification of the flood sources and actual configuration has been observed during the peer review walkdown. For example Appendix A indicates more than 138' of CC piping in the U2 Battery room A (2RAB43-35) but no CC piping has been observed in the room during the walkdown. On the other hand, demin water lines to the emergency eyewash have been observed during the peer review walkdown in the battery room, which are not listed in the Appendix A datasheet. 2RAB43-36 also does not show DW lines although it is expected that eyewash station are also present and they are indeed shown in the architectural drawing). In 2RAB43-36, the batteries are mentioned to be potentially vulnerable to spray from fire protection but no fire protection is listed as potential source in the room. Appendix A shows multiple examples of datasheet being incomplete even for critical rooms such as the ECCS rooms (see for example 2RAB-10-16B) that would challenge the selection of impacted equipment.	Finding	Perform confirmatory walkdown for flood source identification, and vulnerable equipment and confirm the information collected from other sources and listed in Appendix A.	This finding has been resolved and closed by an update to the model/documentation. Confirmatory partial-walkdowns were performed after development of this F&O and pipe isometric drawings were re-reviewed for accuracy. The walkdown revealed that the CCW piping segments inside the vital battery rooms at el. 43' are hidden within a pipe chase near the ceiling and therefore not visible. As a result, the analysis has been corrected by deleting the CCW piping from the list of potential flood sources in the battery rooms. Any water from postulated breaks inside the chase was assumed to divert from the battery rooms to the adjoining rooms. However, the water supply pipe to the shower station was added as a potential flood source in each battery room. The spreadsheet calculating rupture frequencies was updated with the above corrections which also corrected the input to FRANX. Finally, the datasheets were reviewed and updated as well.
IFSO-B2-01	IFSO-B2	A list of flood sources that require further evaluation was not developed. The criteria that were used for screening flood sources were not clearly developed and documented. Flood sources were identified and included in the documentation. The screening criteria used to eliminate flood areas and flood sources from further evaluation was not documented. It appears that some level of screening was performed for flood areas and flood sources. However, a listing of the flood sources that require further examination was not provided.	Finding	Screening criteria used to eliminate flood sources from further evaluation should be developed and documented. The results obtained from applying the screening criteria should also be included in the documentation.	This finding was reviewed and closed with no further action. As stated in Section 4.1.3 of the PSL Internal Flooding Analysis, no screening of flood sources (within the reactor auxiliary building) was performed. The flood sources in all flood zones within the auxiliary and fuel handling buildings are listed in analysis document. All listed flood sources are explicitly considered in the analysis. Flood sources in other buildings were not considered unless their rupture might precipitate or ensuing damage causes a reactor scram. The screening of these other buildings is addressed in the response to the finding pertaining to IFEV-A6.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
IFSO-B3-01	IFSO-B3	No evidence was provided that discusses modeling uncertainties and related assumptions associated with flood sources. A review of the notebook sections that discussed the potential plant floods and flood scenarios did not provide any evidence that modeling uncertainties and related assumptions associated with flood sources was documented.	Finding	A discussion on modeling uncertainties and related assumptions associated with flood sources should be included as part of the documentation.	This finding has been resolved and closed by an update to the model/documentation. The main internal flood analysis document as well as Internal Flood Quantification document were further revised to addresses the analysis uncertainty.
MU-02	N/A	PSL developed no criteria upon which to base the need for a model update. Impacts written against the model may remain pending for a long time. Incorporating into the model a pending impact is based only on judgment call. In addition, a fixed periodic PRA model update schedule should be established. The update periodicity should be consistent with the principle of a living PRA.	A	Like most other plants, PSL should establish a time cutoff (say within 30 days) for implementing model impacts that has about 15% or more change to CDF / LERF values. The 15% change could be up or down from the existing CDF / LERF values. Prior living PRA-based decisions should then be re-examined for continued validity. PRA customers should be advised of the change in CDF / LERF as soon as the implementation of the change is completed. A fixed periodic PRA model update schedule should be established. The update periodicity should be consistent with the principle of a living PRA.	This finding was reviewed and closed with no further action. The PRA models maintenance and update are developed in accordance with NextEra fleet procedure/standard "PRA CONFIGURATION CONTROL AND MODEL MAINTENANCE". The frequency of model update is based on priority setting of the proposed changes to be developed by pertinent model custodian/staff engineer.
QU-02	AS-C3 DA-E3 HR-I3 IE-D3 QU-E4 SC-C3 SY-C3	A lot of results sections in the quantification report are blank with a "later" in place of the table or results.	B	Finish quantification and fill appropriate tables.	This finding has been resolved and closed by an update to the model/documentation. The current PRA Update documents included final results and completed analysis.
QU-04	AS-B5 AS-B6 AS-C3 DA-E3 HR-I3 IE-D3 QU-B6 QU-E1 QU-E2 QU-E4 QU-F2 QU-F4 SC-C3 SY-C3	No uncertainty analysis has been performed on the results from Unit 1 or Unit 2 quantification results.	B	Complete uncertainty analysis on results.	This finding has been resolved and closed by an update to the model/documentation. Completed uncertainty/sensitivity analysis and related evaluations are included in the current flood Quantification document.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
SL-CCF-02	IE-A6	CCF of Turbine Building supply fans, i.e., IMFFCCFTBSWGRF$ (CCF FACTOR - (2/2) TURB SWGR RM FANS FAIL TO RUN), is modeled to result in the loss of 6.9KV buses 1A1 and 1B1 and the loss 4KV buses 1A2 and 1B2, but this event is not modeled as a contributor to loss of these bus initiators. A similar comment also applies to the Unit 2 model. It should be noted that these HVAC-related CCF contributors could not be incorporated into the current St. Lucie PRA model, because this model does not include the loss of HVAC as an initiating event, per a statement in Section 3.2 of the St. Lucie Initiating Events Notebook (Reference 22). However, other than to say that only two plants model a Loss of HVAC initiating event, this notebook provides little basis for its exclusion. Due to its potential risk significance, such basis is appropriate. Basis for Significance: This F&O was assigned a Significance of B, because the absence of this CCF initiator in the initiating event fault trees is judged to not meet SR IE-A6 for any CC level on CCF and this contributor is judged to be risk significant. Additional Discussion: This finding applies to both St. Lucie Unit 1 and 2 PRA models.	B	Add the subject CCFs to their appropriate initiating event fault trees or document basis for their exclusion.	This finding was reviewed and closed with no further action This CCF is credited under the system part of the fault tree and if it were to be credited under IE fault tree, it would be double counted. Nevertheless, adding the CCF under the IE fault tree will have no impact on the results and conclusion of overall risk insights as the CCF probability is of the order of 2E-5 while the IE fault tree is dominated by annual breaker failure of the order of 1E-3. Thus, the finding is considered of no significance.
SL-CCF-06	SY-B8	No documented evidence was available that a process was applied to identify new common cause failures due to spatial and environmental hazards. This is required by SR SY-B8 for CCF. Basis for Significance: This F&O was assigned a significance of B (rather than A), because it is judged unlikely that the results of a plant walkdown or alternative investigation intending to identify new common cause failures due to spatial or environmental hazards would discover any CCFs that are not already modeled. Additional Discussion: This finding applies to both St. Lucie Unit 1 and 2 PRA models.	B	Perform plant walkdowns seeking common cause failures due to spatial and environmental hazards. These may be due to radiation, heat, humidity, vibration, etc. Special attention should be given to like components, with similar functions, in the same system, and which are in close proximity of each other. Document locations inspected. Note that an operator or system engineer may be good company on such walkdowns, because they will be familiar with plant equipment. Document findings, walkdown dates, locations visited, personnel participating, etc. Note that these walkdowns could be incorporated into walkdowns conducted for fire risk model development or flooding model updates. However, the CCF inspection is not necessarily compatible with the objectives and focus of these other inspections.	This finding has been resolved and closed by an update to the model/documentation. The System Notebooks were revised to include walkdowns worksheets where applicable.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
SL-CCF-10	DA-D6	Plant-specific data was not reviewed for CCF events as required by SR DA-D6. Basis for Significance: This F&O was assigned a Significance of B, because the plant-specific review is required for CC II, but the "discovery" of plant-specific CCFs that are not included in the generic CCF database is unlikely given that the generic databases were developed via extensive and exhaustive research efforts. Additional Discussion: This finding applies to both St. Lucie Unit 1 and 2 PRA models.	B	Perform a plant-specific data review for CCF events. Compare findings with data in the NRC generic database. If any new CCF events are identified, request INEL to consider them for possible inclusion into the NRC database.	This finding was reviewed and closed with no further action Limited scope review of plant-specific data found no event to be added to those being considered in NRC CCF database.
SL-CCF-12	IE-A6	The CCF of ICW traveling screen plugging and the CCF of ICW strainers plugging as contributors to the loss of ICW initiator fault tree are missing from the model and no explanation for their absence is provided. Common cause contributors to the loss of ICW are judged to be both credible and potentially risk-significant. This judgment is based on the failure of a intake screen reported in LER 84-09, 1011/84 (Unit 1), the fact that these issues are addressed in the plant Off-Nominal Operating Procedure 064030, and that data is available for both of these failures in the NRC CCF database. Given that common cause is likely to be a dominant contributor to the loss of ICW and that the nominal loss of ICW frequency is judged to be very low (~1E-5/rx-yr), the modeling of the loss of ICW initiating event is judged to not meet SR IE-A6 for any CC level on CCF. Basis for Significance: This F&O was assigned a Significance of A due to its potential risk significance. Additional Discussion: This finding applies to both St. Lucie Unit 1 and 2 PRA models.	A	Evaluate the appropriateness of including common cause contributors to loss of ICW and, if appropriate, include them in the model, otherwise, document basis for their exclusion. Note that that it may be more appropriate to use a plant-specific estimate of the loss of ICW due to environmental effects in lieu of using generic CCF component CCF failure data (such as traveling screen plugging or service water pump strainer plugging) to estimate the loss of ICW frequency.	This finding has been resolved and closed by an update to the model/documentation. The PSL models were revised to include CCF of ICW traveling screen plugging and the CCF of ICW strainers plugging as contributors to the loss of ICW system. They were not considered under IE fault tree to eliminate double-counting.
ST-01	SC-A6	FP&L does not directly address reactor vessel capability. In section 2.3.5 of PSL-BFJR-02-001, FP&L dismisses reactor vessel rupture as being of low risk significance because of a low generic failure probability and also dismissed PTS as being of low risk significance based on generic analyses. Therefore, reactor vessel failure is not included in the model at all.	B	Reactor vessel rupture should be included in the model and quantified using the generic failure frequency. This will not have any significant impact on CDF but will provide a placeholder to address any future issues. This is consistent with industry practices and NEI subtier evaluation criteria.	This finding was reviewed and closed with no further action Vessel rupture Initiating Event is considered in the PSL models. The considered frequency was adopted from CEOG position paper "Evaluation of the Initiating Event Frequency for Reactor Vessel Rupture".

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
ST-02	LE-D1 LE-D1 LE-D2 LE-D2 SC-A6	The containment capability analysis included in the IPE submittal is a simplified analysis based on the generic approach in NUREG/CR-2442 and NUREG/CR-3653 using St Lucie specific information in the simplified equation. This analysis provided an estimate of containment ultimate pressure value that was used to generate a containment fragility curve based on containment fragility curves for other similar containment designs shifted so that the median was at the St. Lucie ultimate containment pressure. The analysis did not address temperature effects and only included a single failure mode, liner tear at the spring line. The analysis did not address other containment failure points such as liner tear at the containment hatches or penetrations. The level 2 analyses did consider release pathways including containment bypass, containment isolation and containment failure. Only the single failure mode of unspecified location was used for containment.	B	If the level 2 analyses are to be updated, a more detailed containment capability analysis should be performed to include temperature impacts on material properties and to evaluate other potential failure locations and sizes. If the level 2 analyses are not updated, FP&L may want to switch to the NRC simplified LERF model where details of the containment capability evaluation are not as important.	This finding has been resolved and closed by an update to the model/documentation. A simplified approach for Level 2 analysis was developed and issued by Westinghouse in 2009. The analysis has considered all aspects of containment capability features. The analysis was peer reviewed and there with no findings.
SY-01	N/A	There are no references to engineering calculations or analyses to support the system analysis success criteria, either in the system analysis documents or the accident sequence analysis. The basis for success criteria should be included in the system analysis documentation in order to facilitate review, update, and application of the model. For example, for AFW, the success criteria section of the AFW system analysis document does not give a basis for the success criterion that is described (flow to 1 SG). The required flow rate to remove decay heat should be compared to the capacity of a single pump, including the effects of potential flow diversion through the recirc line (since failure of the recirc line is assumed to be subsumed in the injection failure) and blowdown (since isolation of blowdown is assumed not to be needed). This could be done using engineering analysis or thermal-hydraulic analysis, but the basis should be described in the AFW system analysis document. Another example is the basis for the AFW success criterion for ATWS (flow to both SGs).	B	Include in success criteria section of system analysis documents references to engineering calculations and thermal-hydraulic analyses that provide the basis for the success criteria.	This finding has been resolved and closed by an update to the model/documentation. Stand-alone Success Criteria documents were developed and issued in 2009 for pre-EPU and post-EPU, respectively.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
SY-08	AS-A10 AS-A5 AS-A7 AS-B1 QU-A1 SC-A6 SY-A1	It appears that in general key control systems in the St. Lucie Plant are not modeled. AFW flow control is not modeled. The AFAS system appear to control the based on the NR SG water level (opens at 19% decreasing) closes at 29% NR. There is no calculation available to determine the number of cycles required for automatic flow control. Additionally the consequence of steam generator overfill is not modeled. It should be noted that increased cycles affect a wide array of components: the relays in the control circuitry, the check valves cycled as flow is interrupted to the SG, etc. This affects not only the independent failure rates, but the common cause failure likelihood as well.	A	Model the components within the control systems down to the relay level. Provide a basis for number of cycles. Model all of the consequences control system failures (e.g. overfill/underfill) or provide a defendable discussion on why the consequences are not modeled.	This finding was reviewed and closed with no further action Per discussions with operations personnel, AFAS would start pumps and open flow valves to provide AFW flow to SGs. Small adjustments to valve position over time would be performed by the operator to maintain desired SG level. There would not be a series of valve open and close cycles. It is judged that the assumed 3 valve cycles would be adequate to capture or bound the total valve failure prob.
SY-12	AS-A10 AS-A5 AS-A7 AS-B1 QU-A1 SC-A6 SY-A1	It appears that in general key control systems in the St. Lucie Plant are not modeled. In the fault tree the AFW flow control system is demanded 3 times, but the basis for using 3 demands is unclear. No analysis has been done to determine the number of cycle the AFW system will undergo. Further, the common cause MOV demand failure rate does only considers a single demand. The model does not differentiate between an overfill and underfill. Overfills in general could lead to the failure of the turbine driven AFW pumps. Note: If the MOVs are demanded twice, it is doubtful that the failure likelihood would double. But it is also clear the failure likelihood will increase. Given the importance of the AFW MOVs, any increase to the failure rates can be quite significant.	A	Model the components within the control systems down to the relay level. Provide a basis for number of cycles. Model all of the consequences control system failures (e.g. overfill/underfill) or provide a defendable discussion on why the consequences are not modeled.	This finding was reviewed and closed with no further action Per discussions with operations personnel, AFAS would start pumps and open flow valves to provide AFW flow to SGs. Small adjustments to valve position over time would be performed by the operator to maintain desired SG level. There would not be a series of valve open and close cycles. It is judged that the assumed 3 valve cycles would be adequate to capture or bound the total valve failure prob.
SY-14	AS-A10 AS-A5 AS-A7 AS-B1 QU-A1 SC-A6 SY-A1	There is not a single event model that represents debris clogging the sump (i.e. both headers blocked).	B	Add a single sump clogging basic event to both sump headers.	This finding has been resolved and closed by an update to the model/documentation. The current models included CCF event for sump plugging.

Table E2-A1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Peer Review Recommendation	Resolution
SY-15	DA-A4 DA-C1 DA-C2 DA-C3 DA-C6 DA-C7 DA-C8 DA-C9 SY-A11 SY-A12 SY-A19 SY-A6 SY-A7	The implementation of the Alpha Parameter methodology for common cause analysis has resulted in conditions that appear to be an over estimation of the contribution from common cause and results that do not make obvious sense (i.e. cutsets in which the common cause failure of three check valves [three AFW pump discharge check valves] is more likely than the common cause failure of two check valves [two MFW check valves to the steam generators I-V09294 and I-V09252]). The implementation of the methodology includes an assumption in the development of the parameters of staggered testing. This assumption may be non-conservative. The common cause failure of the check valves in the pump recirculation lines was not considered and justification provided for not including them was not included. Some of the issues may be the result of the use of component specific and generic alpha parameter data.	B	A clearer description of the implementation of the alpha parameter methodology would clarify part of the problem. Common cause events that are important contributors to the results should be developed consistently with component specific alpha parameter development.	This finding has been resolved and closed by an update to the model/documentation. The latest revision of CCF analysis document clearly describes the application of Alpha-Factor method and applicable data using 2009 INL/NRC database.
SY-A2-01	SY-A2 QU-D2	The current ISLOCA model includes a failure of the SDC isolation valve to fail to close as one manner by which a loss of isolation may occur combined under an "OR" gate with a pre-initiator error involving the operator failing to correctly close the valve. If the valve mechanically failed during startup the operators would not enter into power operation so the failure mode is not valid. It could be postulated that if it failed the operators could fail to take appropriate actions which would be a pre-initiator action, but this would require the two events to be "ANDed" which would substantially decrease the likelihood of occurrence. Closing the valve at power is not plausible due to the high RCS pressure so the closure would not be valid with regard to isolation.	Finding	The current assessment for a failure to return to power with the valves in the correct position does not appear to take credit for self annunciated faults. The assessment should include failures of the operators to ensure restoration similar to post-maintenance operations.	Resolution in-progress. See E2-Table E2-C1
TH-02	AS-A9 SC-A2 SC-A6 SC-B1 SC-B2 SC-B3 SC-B4	FP&L uses a combination of FSAR and best estimate analyses to support success criteria. There is a calculation, PSL-1FJF-93-063, which documents MAAP runs supporting success criteria evaluation. However, the accident sequence analysis report, PSL-BFJR-02-001, does not have any direct references to the cases within the MAAP analyses report linking specific success criteria assumptions to specific MAAP runs.	B	The accident sequence analysis report should directly reference any MAAP of other transient analyses used to establish specific success criteria or timing.	This finding has been resolved and closed by an update to the model/documentation. Stand-alone Success Criteria documents were developed and issued in 2009 for pre-EPU and post-EPU, respectively.

E2-Attachment B - Internal Fire PRA Peer Review Findings

Table E2-B1 summarizes facts and observations the previously referenced peer reviews for Internal Fire.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
CS	4kV power and 125VDC control cables required to support the operation of the Containment Spray Pump were not identified. Fire PRA Plant Response model and other Fire PRA support tasks are adversely affected. Perform a comparison of the components identified on the MSO (multiple spurious operation) list against the Fire PRA components for which new cable selection was performed (i.e., components not previously identified on the Appendix R safe shutdown equipment list). Verify that the cable selection for the common components supports all credited operations.	A3	CS-A3, CS-A4	01	Finding	Fire PRA Plant Response model and other Fire PRA support tasks are adversely affected.	Perform a comparison of the components identified on the MSO (multiple spurious operation) list against the Fire PRA components for which new cable selection was performed (i.e., components not previously identified on the Appendix R safe shutdown equipment list). Verify that the cable selection for the common components supports all credited operations.	Reviewed component failure modes to ensure that components for which operation is credited include required power cables.
CS	Include all load cables and applicable control circuit cables as required cables for credited switchgear, since concurrent faults on the load cables and control circuit could prevent proper tripping of the breaker and result in loss of the switchgear. Also review faults on CT cables for their potential impact on breaker operability. These recommendations apply to all credited switchgear.	A6		01	Finding	An analysis has not been completed and needs to be completed to assure this issue evaluated.	Assess all the load power cables and the applicable portions of the associated control circuits in the Fire PRA for their potential impact on the Fire PRA. Concurrent damage to the power cable(s) and control circuit could affect the automatic over-current trip capability of the affected breaker, which in turn could adversely affect the ability of the switchgear to remain energized. This should be assessed for all switchgear credited in the Fire PRA.	Breakers with -CNTL and -PWR have been added to the analysis and to the fault tree. CNTL/PWR cable failures cause failure of the bus.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
CS	The documentation for new cable selection and cable routing is highly fragmented. In the documents that were reviewed, there are no references to the plant source documents and document revisions to provide traceability.	C2		01	Finding	The documentation for cable selection did not include a reference to plant source documents.	Provide a consistent document that shows Fire PRA components, functions, cable associated, fire zone location with a reference to plant source documents.	Documentation updates have been implemented to consolidate the cable selection and cable routing data and associated methodologies.
CS	There is no documented methodology for cable location to fire areas.	C1		01	Finding	The documentation did not exist.	Development a documented methodology for locating cable to fire areas.	Documentation updates have been implemented to consolidate the cable selection and cable routing data and associated methodologies.
CS	No evaluation was performed to verify that the new components and cables associated with the Fire PRA is bounded by the existing overcurrent coordination analysis.	B1	CS-C4	01	Finding	The evaluation was not completed at this time.	Evaluate the new cables and components and verify that they are bounded by the current overcurrent coordination analysis.	A detailed review of the coordination analysis was performed including those power supplies associated with Fire PRA components.
CS	There were cable location assumptions that were made and documented in the scenarios task. PSL Fire PRA Scenario Report, Rev 1, Attachment A, has two scenarios that made assumptions (1_47 and 1_26) that cables designated as Y3 were not in the fire area. The justification was a statement that the cables were "Judged not to have cables in this zone due to location of component". No other justification was provided to determine that the cable was not in the area. More justification is needed to document the assumption on cable routing.	A11	CS-C3	01	Finding	There is no justification for the assumed cable routing. SR CS-A11 and CS-C3 cannot be verified without the justification and documentation to validate the assumption on cable routing for components that had no cable selection or routing.	Provide supporting justification and documentation for assumed cable routing.	All exclusions of component/cable fire impacts are based on developed component/cable fire routing data. Eliminated exclusions based on assumptions of routing.
FQ	No identification of significant contributors was available. Appendix C of the Fire PRA Summary report stated that this will come later.	E1	FQ-E1	01	Finding		Perform the analysis of significant contributors in accordance with FQ-E1	Added Importance measures from appended cutsets to Summary Report.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
PP	Draft Report NISYS-1251-0001 was reviewed and provides a validation of the FHA and documents the plant specific walkdowns performed for each fire zone boundary. Finding written to finalize this report and incorporate by reference into the plant partitioning report.	B7 and C3	B7 and C3	01	Finding	Document needs to be finalized and incorporated into project documents to provide the technical basis.	Provide evidence of walkdowns to confirm partitioning.	Incorporated reference to report in PP/FIF report. Added Reference 9 to the report.
PP	Need list of excluded areas with basis. Work must have been done to decide what was excluded, but was not presented. Criteria is clearly presented but use of the criteria is not. Necessary to support definition of Global Boundary and whether all appropriate compartments were included.	C2	A1	01	Finding	SR unable to be reviewed. List is necessary to perform review and to ensure technical adequacy.	Include list in report; include for each item the justification for exclusion from further analysis.	Added Note 3 to Table 2-1 regarding basis for exclusion of buildings which do not contain equipment or cables which impact the Fire PRA.
PP	Evidence was presented to the reviewer that raceways supporting PRA equipment exists in the "no man's land" area between unit 1 and unit 2. This area is not currently included as part of an analyzed compartment, however no analysis exists as to why it meets the criteria for exclusion presented in Section 2.1.1. of the report.	A1		01	Finding	Additional Analysis required to ensure PRA addresses fire failures appropriately in this area.	Document a basis for exclusion from the analysis, or add compartments to the fire PRA analysis and quantify the fire failures.	Added Note 3 to Table 2-1 regarding basis for exclusion of buildings which do not contain equipment or cables which impact the Fire PRA.
CF	The basis for the conditional failure probability used in the Altered Events table was not documented.	B1		01	Finding	Documentation/ reference supporting the credited conditional failure probabilities provides the technical basis for applicability of these treatments.	Provide basis for the conditional failure probabilities used in the Altered Events table.	Provided additional detail in altered events table with reference to 6850 basis for value used.
ES	No information was identified in the Component and Cable Selection Report (Report 0493060006.101, Revision 1) or the HRA Evaluation Report (Report 0493060006.102, Revision 0) that characterized instrument availability or spurious operability for individual fires.	C2	ES-C2	01	Finding		A review of control room instrumentation should be performed to identify, on a fire-zone basis, those instruments in which unavailable or spurious indications could mislead the operator into performing undesirable actions.	Provided clarification in HRA report, Section 3.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
ES	Tables 4.2-1, 4.2-2 (to be completed for Unit 2), B-1 and B-2 provide information on instrumentation associated with PRA basic events and SSEL mapping and disposition. The HRA Evaluation Report (Report 0493060006.102, Revision 0), Tables A-1 - A-4 and Appendix C provide information on the instrumentation associated with important control room actions. Appendix R instrumentation is specifically identified by bold formatting. However, no information was provided that would allow the impact of a specific fire on the instrumentation set to be identified. For essential instrumentation this information is available in the Response to Fire procedures. The reduced set of instrumentation associated with a fire zone should be used to support estimation of the human failure probabilities associated with a fire scenario.	C1	ES-C1	01	Finding		Expand the Component and Cable Selection Report to address the impact of a fire in each fire zone (or area) on instrumentation addressed in the HRA Evaluation Report.	One set of SSD instrumentation will remains available to meet SSD systems for an area wide fire. The correlation between SSD instrumentation and operator actions provided in the HRA report confirms that for each HFE Appendix R instrumentation is available to support the cue for the action. Guidance provided in SSD procedures will identify the instruments available post fire and focus operator cues on these instruments. Since the instrumentation availability is defined on a fire area wide fire basis it will provide a conservative basis for instrumentation available for an individual scenario within the fire area. Incorporated additional discussion in HRA report, Section 3.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
ES	PI-03-003 provides instruction for circuit analysis to include review of interlocks, instrumentation, and support system dependencies. Cable routing database was reviewed and confirmed that interlocks, instrumentation, and support system cables were included in equipment effects. However, demonstration of a review of power supplies, etc. was not readily apparent in the Component Selection report. The development of the Fire PRA equipment list inherently considers the entire component and its supporting equipment; however, it is important to document this information to support peer reviews and applications. It is suggested that document the review to show the interlocks, power supplies, etc. are included (or referenced) in the development of the Component Selection section. The equipment selection report states that SSEL equipment required to place the plant in hot standby, the PRA end state, are included in the analysis while equipment only associated with taking the plant to cold shutdown were excluded from analysis. No information is provided to facilitate the assignment of individual SSEL instrumentation to specific plant states, which complicates review against this SR. Expand Component and Cable Selection tables to allow SSEL components to be associated with specific plant states. Components are linked to fault tree Basic Events, but suggest document all potential fire induced sequences are confirmed to be associated with a reactor trip initiating event in the fault tree. Improve component selection report to address items identified in this F&O.	D1	ES-A2, ES-A3, ES-B4, ES-D1	01	Finding		Improve component selection report, address items identified in this F&O.	SSD and FPRA documentation revised to provide enhanced documentation of component selection and cable selection.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
IGN	Bayesian updates to generic fire frequencies were performed on a reactor-year basis, consistent with the Standard. The analysis does not include consideration of plant availability as required.	A5	IGN-A5	01	Finding		Revise updated frequencies to include consideration of plant availability.	Attachment K provides the basis for reactor years used, incorporating capacity factor via removal of outage durations.
IGN	An analysis supporting the estimation of plant-specific reactor-years is not described (the number of reactor-years is specified). Add a description of the process of estimating the number of plant-specific reactor years to the fire frequency report.	B4	IGN-B4	01	Suggestion		Add a description of the process of estimating the number of plant-specific reactor years to the fire frequency report.	Attachment K added to provide the basis for the reactor years used.
PRM	Overall PRM documentation is sparse and doesn't provide the information addressed in the SRs associated with the HLRs described in the Category I, II and III criteria of PRM-C1. In addition, the development of changes made in Tables D1 and D3 are not described (PRM-B9).	C1	PRM-B9	01	Finding		Recommend a separate PRM report that documents in a structured and consistent way the requirements described in the PRM SRs.	Added discussion in Component/Cable report Section 5.0.
FQ	Fire-related SSD actions are currently modeled only through the AlteredEvents file in FRANC, which bypasses the dependency analysis.	C1		01	Finding		Any fire-related SSD actions modeled in the final Fire PRA should be evaluated for potential dependencies with other actions.	Incorporated multipliers applied to cutsets with multiple screening HEPs. See Section 4.1 and Appendix B of HFE Report.
FQ	Documentation of the CDF and LERF analysis to the extent required in the FQ-F1 supporting requirement has not been developed. CDF and LERF values are provided on a scenario bases, but these are not ranked. Basic event correlations have not been addressed nor have uncertainty analyses been performed.	F1		01	Finding	FQ-F1	Document the CDF and LERF analysis to the extent required in the FQ-F1 supporting requirement should be completed as the analysis proceeds.	Additional documentation, including parametric uncertainty have been performed and incorporated into the Fire PRA documentation.
SF	Section 3.13 of the St. Lucie Fire PRA Summary report discusses the seismic/Fire interaction issue, 0493060006.105, Rev 1., concludes, with no supporting evidence that there is no issue and pointed to a set of references as providing the requisite supporting information. A review of these references indicated that they pertained to the seismic issues associated with A-46 resolution and GL-88-20. They did not contain any discussion of seismic/fire issues such as the potential for unique fire initiators, the potential for spurious operation or failure of fire detection and suppression systems, the potential for common cause failure of multiple suppression systems or the impact on fire brigade response.	A1	SF-A2, SF-A3, SF-A4, SF-A5, SF-B1	01	Finding		The five SRs associated with HLR-SR-A specify five specific aspects to evaluate qualitatively to ensure that the insights from the original IPEEE evaluations remain valid in light of knowledge gained from the new Fire PRA. FP&L needs to upgrade the write-up in Section 3.13 of the St. Lucie Fire PRA Summary report to specifically discuss the items in each of the SRs.	The scope of seismic analyses performed for the IPEEE is considered to be sufficient given the low seismic event frequency and magnitudes expected at the PSL site.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
HRA	Section 4.1 of H0493060006.102, Rev. 0, briefly discusses reviewing fire failures to identify operator recovery actions for these failures. It was indicated that these recovery actions were included with a screening value of 0.01. No additional information on these recovery actions was provided in the HRA report. A review of the FRANC AlteredEvents File indicated that these "recovery actions were incorporated into the model by altering the failure probability of a related equipment failure basic event to the screening value for the recovery action. The sole documentation was the comment field for the AlteredEvent. The AlteredEvent file also had some additional events that were clearly identified as operator actions. Again, there was no related information in the HRA report. Discussions with St. Lucie personnel revealed that these were actions added to the model logic for several MSOs and set to 1.0. These events were listed in the BE mapping table in the Scenario Report, but were not discussed in the HRA report. The conclusion is that St. Lucie did identify these actions, but the documentation of these actions was severely limited to the point that it was extremely difficult to locate this information	A2	HRA-B2, HRA-E1, PRM-B11	01	Finding		The HRA report should be modified to provide additional information for the fire-specific actions. As a minimum, a table should be added to list the AlteredEvent elements added to cover a recovery action. The table should define the operator action and provide a summary description of the action and associated equipment, identify the event being altered to account for the action, the assigned probability and the basis for the assigned probability. For each recovery action retained, this basic information should be supplemented with the standard information needed to define and quantify a human action (e.g., timing, cues, etc.) For the MSO-related operator actions, as a minimum, have a reference to the BE mapping table with an explanation of what the actions represent. Any that are retained, must be fully documented.	Additional discussion and process applied for screening HEPs is added to HFE Report section 4.1.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
HRA	A number of fire-specific HFEs were identified. Some of these were incorporated into the model via the AlteredEvents table with the definition of the HFE limited to a brief statement in the comment field for the altered event. Other events were added to the model to support the MSO logic with the values set to 1.0. The intent is to determine which HFEs to retain and which HFEs to delete. However, at this point they are in the model with limited documentation and no characterization. As such, the definition of these HFEs is not complete and provides no scenario specific information beyond the fire scenario ID in the AlteredEvents file.	B3		01	Finding			Additional discussion and process applied for screening HEPs is added to HFE Report section 4.1.
FSS	PSL reviewed their cable types and modeled targets as non-IEEE-383 qualified with damage thresholds of thermoplastic cable. No references or description of the cable review was provided. The Fire Scenario Report simply states that 'Most of the targets are cable trays containing non-IEEE-383 qualified cables.' Recommend providing a description of how that determination was made, possibly including references to cable purchase orders, procurement documents, etc.	H2		01	Finding	No basis for target damage thresholds were provided as required by the SR.	Recommend providing a description of how that determination was made, possibly including references to cable purchase orders, procurement documents, etc.	For PSL Unit 1 documentation is not needed to substantiate the use of thermoplastic cable damage criteria. Had thermoset and/or IEEE-383 cable damage criteria or flame spread characteristics been credited, additional documentation would be needed. For Unit 2 cables are thermoset but the use of Kerite-FR cables requires that the thermoplastic damage criteria be used. Thermoset cable flame spread criteria is applicable to U2.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
FSS	PSL did not postulate hydrogen (H2) fires other than the turbine generator H2 fires. PSL used the basis that their H2 piping contains excess flow check valves. However, this will not prevent H2 fires. It's likely that plants experiencing H2 fires that contributed to the "potentially challenging" fire frequency also had excess flow check valves. Recommend either postulating H2 fires or developing a stronger technical justification for their exclusion. PSL did not appear consider all pump lube oil fire scenarios (e.g., AFW pumps, Charging Pumps, HPSI pumps, LPSI pumps, MFW pumps, etc.). These scenarios often involve significant quantities of oil causing widespread damage in the fire compartment. They can also contribute to multi-compartment fire risk. Note that some lube oil scenarios appear to have been considered by PSL. Specifically, MFW and turbine lube oil fires were postulated. In speaking with the analysts, they indicated that other pumps tend not to have large quantities of lube oil and that source-target data for oil scenarios was often collected during walkdowns. However, there was little documentation of this, and very few oil scenarios were quantified in FRANC.	A1		01	Finding	PSL did not postulate H2 fires and oil fires as specified by NUREG/CR-6850, and minimal basis for this deviation was provided. These fires can be risk significant due to the potential for widespread damage in the fire compartment.	Either postulate H2 and oil fires or develop a stronger technical justification for their exclusion.	Hydrogen for VCT tank isolated from other equipment components. AFW steam driven pump oil fire addressed in AFW C pump fire. Located in outdoor area thus limiting impact of this fire.
FSS	1_55E Scenario F09 (IMUX-4 Cabinet) was quantified with no targets (i.e., UNL-only). However, during the peer review walkdowns, a stack of five cable trays (C31, C30, M30, M31, and L30). However, these trays were not postulated to fail in the FRANC quantification. Failure of these trays represents a potential 1.0 CCDP (similar to adjacent heat trace panels) and CDF 1E-7.	A4		01	Finding	Risk-significant targets (CCDP of 1.0) were not modeled as damaged when they would indeed be damaged.	Re-quantify scenario with affected targets failed.	Revised/Corrected.
FSS	A 0.1 CCDP was modeled for main control room fires in which operators rely on the alternate shutdown panel (i.e, abandonment). There could be scenarios where the damage caused by the fire cannot be mitigated from the alternate shutdown panel. For example, if a particular scenario requires the HPSI pumps to function, and those pumps are not controllable from the alternate shutdown panel, then the 0.1 CCDP may not be appropriate.	A6		01	Finding	In certain scenarios, the current Fire PRA model may credit the alternate shutdown panel when it is not sufficient to mitigate the scenario.	Review the scenarios in which alternate shutdown is modeled. Perform an assessment as to whether the alternate shutdown panel can mitigate the fire-induced failures and adjust the CCDP appropriately.	Specific CCDPs are calculated for each C/R abandonment/non-abandonment scenario. Calculated CCDPs are increased to account for potential impact of abandonment for the CR abandonment cases.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
FSS	This Suggestion F&O is at PSL's request to provide an F&O for all SRs meeting CC-I, including a suggestion on how to achieve CC-II. Time-dependent Heat Release Rate (HRR) profiles are required to be implemented to meet CC-II. This is most related to calculating non-suppression probabilities, and would require a fair amount of additional analysis (specific to each source) than the generic NSPs currently modeled. CC-I met. This is just a suggestion for how to meet CC-II. Model time-dependent HRR profiles for risk-significant scenarios. Calculate NSPs specific to the timing associated with the HRR profile and geometric configuration of each risk significant ignition source.	C2	FSS-C2	01	Suggestion	CC-I met. This is just a suggestion for how to meet CC-II.	Model time-dependent HRR profiles for risk-significant scenarios. Calculate NSPs specific to the timing associated with the HRR profile and geometric configuration of each risk significant ignition source.	Incorporated time dependent HRR profiles and associated NSPs.
FSS	This Suggestion F&O is at PSL's request to provide an F&O for all SRs meeting CC-I, including a suggestion on how to achieve CC-II. PSL used generic, generally bounding severity factors. In order to achieve CC-II, severity factors can be developed based on the specific geometry and fire characteristics of each scenario. For each risk significant ignition source, this would require measuring data such as distance to the nearest target and applying fire modeling equations to calculate the fraction of fires that are non-damaging versus damaging. CC-I met. This is just a suggestion for how to meet CC-II. Develop severity factors specific to each risk significant ignition source based on the specific fire characteristics and geometry of each source.	C4	FSS-C4	01	Suggestion	CC-I met. This is just a suggestion for how to meet CC-II.	Develop severity factors specific to each risk significant ignition source based on the specific fire characteristics and geometry of each source.	Incorporated scenario specific configuration and severity factors.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
FSS	This Suggestion F&O is at PSL's request to provide an F&O for all SRs meeting CC-I, including a suggestion on how to achieve CC-II. PSL developed and applied generic non-suppression probabilities by reviewing the EPRI Fire Events Database. Note F&O FSS-H1-01 to document a strong technical basis for this approach. In order to meet CC-II, PSL should review plant-specific data to ensure no outlier behavior from the generic estimates. CC-I met. This is just a suggestion for how to meet CC-II. In order to meet CC-II, PSL should review plant-specific data to ensure no outlier behavior from the generic estimates.	D7	FSS-D7	01	Suggestion	CC-I met. This is just a suggestion for how to meet CC-II.	In order to meet CC-II, PSL should review plant-specific data to ensure no outlier behavior from the generic estimates.	Confirmed no outlier behavior for suppression and detection system availability.
FSS	This Suggestion F&O is at PSL's request to provide an F&O for all SRs meeting CC-I, including a suggestion on how to achieve CC-II. PSL did not postulate failures due to smoke damage. This is sufficient for CC-I. In order to meet CC-II, PSL should evaluate fire risk associated with failures caused by smoke, and not just temperature / thermal radiation. CC-I met. This is just a suggestion for how to meet CC-II. In order to meet CC-II, PSL should evaluate fire risk associated with failures caused by smoke, and not just temperature / thermal radiation.	D9	FSS-D9	01	Suggestion	CC-I met. This is just a suggestion for how to meet CC-II.	In order to meet CC-II, PSL should evaluate fire risk associated with failures caused by smoke, and not just temperature / thermal radiation.	Qualitative analysis provided which documents that the thermal damage criteria envelopes the smoke and sensitive electronics damage criteria.
FSS	PSL's multi-compartment evaluation consisted of a two-stage screening approach. During the first stage, a 0.0074 barrier failure probability (which corresponds to a solid wall) was inappropriately applied. This resulted in several scenarios being inappropriately screened at the first stage.	G1		01	Finding	Inappropriate application of the 0.0074 multiplier may result in screening scenarios that are potentially significant.	Simply don't apply the 0.0074 screening criteria at the first stage of the screening process.	HGL/MCA evaluation has been revised to consider adjacent zones with fixed openings where the 0.0074 criteria is not applicable.
FSS	Documentation of PSLs multi-compartment analysis, as well as most of the FSS-related tasks, was light. These analyses seemed technically adequate, however it took a fair amount of verbal explanation to understand. Recommend improving documentation of this analysis.	H8		01	Finding	The methodology could not be understood without significant verbal explanation.	Document the methodology, inputs, outputs, and conclusions in a manner that can allow a Fire PRA engineer to understand the analysis without significant explanation.	Revised HGL/MCA analysis. Methods associated with panel factors and lower transient HRR have been eliminated from the analysis.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
FSS	Attachment A of the Fire Scenario Report documents cases where certain failures/BEs were excluded from the mapping based on an either assumed cable routing. These cases were spot-checked and no problems were noted. However, no discussion of the uncertainties associated with this assumed routing was provided, as required by the SR. Note that failures/BEs appear only to have been excluded when there was a high confidence in the assumed cable routing. For example, there is a high confidence that main feedwater is not affected in containment.	E4		01		No discussion of the uncertainties associated with this assumed routing was provided, as required by the SR.	Simply provide of uncertainties associated with assumed cable routing.	Y3 component exclusions are now based on cable routing only.
FSS	In several cases, PSL implemented methods beyond those available in beyond industry accepted guidance documents (e.g., NUREG/CR-6850 and its supplements). For example, PSL created their own multipliers / severity factors for fires that cause damage beyond the ignition source by reviewing the EPRI Fire Events Database. A second example is that PSL modeled transient fires using the motor fire heat release rate distribution, which is much smaller than the transient fire distribution. A third example is not applying the "Location Factor" to account for wall/corner effects on flame height and plume temperature distribution. While these methods seem appropriate, documentation of the technical bases for these methods was generally lacking. Methods beyond industry accepted guidance (e.g., NUREG/CR-6850 and its supplements) should have documented technical bases of similar quality and magnitude to those provided in NUREG/CR-6850. Also, PSL should be aware that methods beyond industry accepted guidance documents may be viewed critically by the NRC.	H1		01	Finding	While these methods seem appropriate, the level of documentation provided did not allow detailed review by the peer reviewers. In addition, methods beyond industry accepted guidance (e.g., NUREG/CR-6850 and its supplements) should have documented technical bases of similar quality and magnitude to those provided in NUREG/CR-6850.	Simply provide stronger documentation of the technical bases where methods beyond industry guidance were implemented. For example, when severity factors were developed based on a Fire Events Database Review, documentation might include an explicit listing and written disposition of each event.	Beyond 6850 methods, panel factor approach, has been eliminated from the PSL Fire PRA. The use of the 69 kW HRR for transient fires has been limited to those fire zones in which "zero transients" are allowed in order to account for the potential violation of the administrative controls.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
HRA	The definitions of the HFEs for existing actions used the existing internal events definitions, which were defined in the EPRI HRA Calculator. Modifications were made to account for general categories of time available, accessibility, and complexity. This appears to be adequate for Cat 1 where a task analysis is not needed. For Cat 2, a more detailed analysis of HFEs for specific fires needs to be performed, along with a corresponding task analysis.	B3	HRA-D2	03	Finding		For existing internal events actions included in the Fire PRA, provide a more complete definition to support the quantification. Note, the detail of the definition can be scaled to the significance of the action (see HRA-C1, Cat 2).	Use of HRA multipliers provides a bounding assessment of the impact of the fire on HEPs defined by the internal events model. HRA Calculator is used to define the new values for combination event recoveries given these revised base HEP values.
UNC	The referenced SRs (e.g., QU-E3) requires an estimation of the uncertainty distribution for fire-induced CDF, which is not included in the Fire PRA.	A1		03	Finding		Provide an estimate of the uncertainty of fire-initiated CDF (or propagate CDF uncertainty).	Uncertainty evaluation performed and incorporated into the summary report.
UNC	The uncertainty analysis documented in Appendix D of the Fire PRA Summary Report covers the major sources of uncertainty, except for those associated specifically with LERF.	A1		01	Finding	See requirements of UNC-A1, specifically reference SRs LE-F2 and LE-F3	Add a LERF-specific section to the uncertainty analysis and document the unique impacts of Fire PRA on the LERF analysis and results.	Added sensitivity and uncertainty analysis for LERF for both PSL units.
HRA	Screening HEP quantification was used to adjust the existing internal event PRA to account for fire impacts. This included feasibility factors (cues availability, accessibility of local action) and adjustment factors based on time available and complexity. This approach is appropriate for the stage of the Fire PRA.	C1		01	Finding		To satisfy Cat 2 requirements, perform detailed human reliability analyses for the significant HFEs in the context of specific fire scenarios.	Use of HRA multipliers provides a bounding assessment of the impact of the fire on HEPs defined by the internal events model. HRA Calculator is used to define the new values for combination event recoveries given these revised base HEP values.
HRA	For new fire-related actions, there is no evidence of any definition of the HFE beyond the title in the AlteredEvents table.	B3		02	Finding		Once the fire response procedures are finalized, the HFE definitions should be completed for operator actions modeled sufficient to support the quantification. Note, the detail of the definition can be scaled to the significance of the action (see HRA-C1, Cat 2).	The use of the screening approach for adjusting FPIE model HEPs and the use of screening HEPs is sufficient to support this application.

Table E2-B1: Fire PRA Peer Review Results Summary

Element	Discussion	Supporting Reqt	Related SRs	Observation No	Level of Significance	Basis for Significance	Possible Resolution	Disposition
HRA	A review of modeled actions is planned to be performed once draft procedures are generated from the Fire PRA. However, at present no such review has been performed except for a limited board walkthrough documented in Appendix C of the Human Failure Evaluation report.	A4		01	Finding		Once fire response procedures are finalized, perform talk-throughs with plant operations and training personnel, at least for risk-significant actions, to support the HRA for these actions.	The use of the screening approach for adjusting FPIE model HEPs and the use of screening HEPs is sufficient to support this application. A review against the draft post fire procedure revision is identified as an implementation item in LAR Table S-2, Item 11.

E2-Attachment C - Currently Open F&Os and Impact on 4b Application

Table E2-C1: Currently Open F&Os and Impact on 4b Application

F&O	Affected SRs	Description	Comment	Impact on 4b Application
IE-C5-01	IE-C5	The approach for generating the initiating event frequency utilizes an adjustment factor that ratios the exposure time. The exposure time is not the mission time for reliability and the approach does not produce an appropriate value for frequency of occurrence. Utilize the idea of initiating event as the first valve failure based on having to remain isolated for the period of one year. Then consider the unavailability of the other valves in the line based on exposure time. Systematically address each valve as if it is the holding valve.	The adjustment factor is not appropriate for unavailability.	No Impact on 4b. ISLOCA F&Os will be resolved prior to application implementation
IE-C6-01	IE-C6	A screening approach is utilized for some lines based on low frequency but this is not quantified. The SR indicates a frequency expectation for screening. Define the estimate for the lines screened on low frequency and show that the calculated frequency supports screening.	There is not quantification of excluded ISLOCA scenarios.	No Impact on 4b. ISLOCA F&Os will be resolved prior to application implementation
IE-C9-01	IE-C9 IE-C10 SC-A5	The fault tree model used for the ISLOCA paths assumes that the status of all valves is known when the plant is brought online and the corresponding exposure time is the refueling interval. However, based on discussions with knowledgeable staff, there is no positive means to know that more than one isolation valve is actually holding. Use of status lights is not definitive since there is a +/-5% margin between light changing and valve seating. The exposure time should be based on a positive flow test which may not occur on a refueling basis but based on other studies could be as much as the life of the plant.	Model should use one year exposure time for the first failure and then do the CCDP following the failure. F&O finding level produced.	No Impact on 4b. ISLOCA F&Os will be resolved prior to application implementation

Table E2-C1: Currently Open F&Os and Impact on 4b Application

F&O	Affected SRs	Description	Comment	Impact on 4b Application
SY-A2-01	SY-A2 QU-D2	The current ISLOCA model includes a failure of the SDC isolation valve to fail to close as one manner by which a loss of isolation may occur combined under an "OR" gate with a pre-initiator error involving the operator failing to correctly close the valve. If the valve mechanically failed during startup the operators would not enter into power operation so the failure mode is not valid. It could be postulated that if it failed the operators could fail to take appropriate actions which would be a pre-initiator action, but this would require the two events to be "ANDed" which would substantially decrease the likelihood of occurrence. Closing the valve at power is not plausible due to the high RCS pressure so the closure would not be valid with regard to isolation.	The current assessment for a failure to return to power with the valves in the correct position does not appear to take credit for self-annunciated faults. The assessment should include failures of the operators to ensure restoration similar to post-maintenance operations.	No Impact on 4b. ISLOCA F&Os will be resolved prior to application implementation

St. Lucie Nuclear Plant Units 1 and 2

Enclosure 3

Information Supporting Technical Adequacy of Probabilistic Risk Assessment (PRA) Models without PRA Standards Endorsed by Regulatory Guide 1.200, Revision 2

This enclosure is not applicable to the St. Lucie Nuclear Plant submittal. Florida Power & Light is not proposing to use any PRA models in its Risk-Informed Completion Time Program for which a PRA standard, endorsed by the NRC in RG 1.200, Revision 2, does not exist.

St. Lucie Nuclear Plant Units 1 and 2

Enclosure 4

INFORMATION SUPPORTING JUSTIFICATION OF EXCLUDING SOURCES OF RISK NOT ADDRESSED BY THE PRA MODELS

E4-1.0 Introduction

Section 4.0, item 5 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines" (Reference 2) requires that the License Amendment Request (LAR) provide a justification for excluding any risk sources determined to be insignificant to the calculation of configuration-specific risk, and will provide a discussion of any conservative or bounding analyses to be applied to the calculation of risk-informed completion times (RICTs) for sources of risk not addressed by the PRA models.

This attachment provides information supporting justification of excluding sources of risk not addressed by the St. Lucie (PSL) PRA.

E4-2.0 SCOPE

NEI 06-09 and the associated Pressurized Water Reactor (PWR) Owners Group (PWROG) guidance (Reference 3) do not provide a specific list of hazards to be considered in an RICT program. However, NUREG-1855 (Reference 4) provides regulatory guidance on risk-informed decision-making relative to hazards that are not considered in the PRA model. Specifically, Section 6 of NUREG-1855 provides the following list of external hazards that should be addressed either via a bounding analysis or included in a PRA calculation:

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

E4-3.0 TECHNICAL APPROACH

The guidance contained in NEI 06-09 states that all hazards that contribute significantly to incremental risk of a configuration must be quantitatively addressed in the implementation of RMTS. Consistent with NUREG-1855, the process includes the ability to address external hazards by

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

The ASME/ANS PRA Standard (Reference 5) has endorsed the following set of five external hazard screening criteria:

(1) The hazard would result in equal or lesser damage than the events for which the plant has been designed. This requires an evaluation of plant design bases to estimate the resistance of plant structures and systems to a particular external hazard.

(2) The hazard has a significantly lower mean frequency of occurrence than another event (taking into account the uncertainties in the estimates of both frequencies), and the hazard could not result in worse consequences than the other event.

(3) The hazard cannot occur close enough to the plant to affect it. Application of this criterion needs to take into account the range of magnitudes of the hazard for the recurrence frequencies of interest.

(4) The hazard is included in the definition of another event.

(5) The hazard is slow in developing, and it can be demonstrated that sufficient time exists to eliminate the source of the threat or to provide an adequate response.

The review of external hazards considers two aspects of the contribution to risk. The first is the contribution from the occurrence of beyond design basis conditions (i.e., winds greater than design). These beyond design basis conditions challenge the functionality of the systems, structures, and components (SSCs) to support safe shutdown of the plant. The second aspect addressed are the challenges caused by external conditions that are within the design basis, but still require some plant response to assure safe shutdown (i.e., high winds causing loss of offsite power). While the plant design basis assures that the safety related equipment necessary to respond to these challenges are protected, the occurrence of these conditions nevertheless cause a demand on these systems and can impact configuration risk.

Note that when the effect of a particular hazard is not mitigatable using the plant SSCs, then there is no impact on the changes in risk calculated to support the RICT Program, and so these hazards can be screened as well. Only events which create a demand for mitigation equipment are potentially relevant to the RICT Program.

The review and disposition of each external hazard is addressed in Table E4-1. Unless otherwise specified, all information is based on the Individual Plant Examination of External Events (IPEEE) (Reference 6).

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
Aircraft Impacts	There are no low level military training airways within 10 miles, or low level federal airways within 2 miles. There are no major airports within 10 miles of PSL, the closest being Palm Beach International (48 miles); there are three non-commercial airports in the vicinity of the plant, the closest being St. Lucie County (12 miles). The two more distant airports were screened from consideration based on the Standard Review Plan (SRP) criteria (Reference 7) methods. The frequency of an aircraft impact at PSL due to air traffic associated with St. Lucie County airport was calculated as 8E-7/year (SRP method), with a more realistic method yielding 3.4E-9/year (Sandia method). Available FAA data from Airport Master Record for the 12 month period ending 1/24/2011 shows approximately a 16% increase in aircraft operations at St. Lucie County Airport since the IPEEE calculations were performed, so that the conservative SRP methodology still demonstrates an impact frequency below 1E-6/year. Similar or smaller increases were noted for the other two airports which do not change their screened status.	Projected air traffic from the small airports and airways do not pose a significant safety impact to PSL based on the design of the facility and the low frequency of aircraft impacts. The type of air traffic from the nearby airports is mostly smaller planes which would not be expected to be capable of causing significant damage to safety-related structures. It is therefore concluded that no unique PRA model for aircraft impacts is required in order to assess configuration risk for the RICT Program.
External Flooding	The external flooding hazard includes flooding from a maximum probable hurricane, storm surge, waves, erosion, and probable maximum precipitation. Other potential hazards, including tsunami, dam failure, and flooding from streams and rivers, are not applicable to PSL or were screened. Analysis demonstrates that either potential flood waters do not enter a structure containing safety-related equipment, or that the affected equipment is above the resulting flood level. A review of these hazards concluded that PSL conforms to the SRP criteria and	External flooding scenarios do not pose a significant safety impact to PSL based on the design of the facility and conformance to the SRP and Regulatory Guide 1.59. It is therefore concluded that no unique PRA model for external flooding scenarios is required in order to assess configuration risk for the RICT Program.

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
	Regulatory Guide 1.59 (Reference 8); therefore, there are no vulnerabilities. Additional evaluation of higher rainfall intensities than previously evaluated over shorter periods also demonstrated no adverse impact to plant structures and systems.	
Extreme Winds and Tornados (including generated missiles)	The category I structures at PSL are designed to withstand design basis tornado wind speeds (360 mph), and non-category I structures are similarly designed for this wind speed. External missile generation will not result in a loss of safe shutdown capability or increased severity of a LOCA by design or protection of SSCs to withstand missile impact or separation of redundant components to preclude simultaneous failure due to a single missile impact. The site design basis for high winds and tornadoes was reviewed, and conform to the appropriate SRP criteria for high winds and tornadoes for PSL unit 2. For unit 1, the design was reviewed and conforms to the SRP criteria for most SSCs; the hazard frequency was evaluated for the Diesel Oil tank, Component Cooling Water and Intake Cooling Water piping and found to be acceptably low (<1E-6/year failure frequency due to external missile impacts).	Extreme winds and tornadoes do not pose a significant safety impact to PSL based on the design of the structures, low frequency of occurrence of the events, and conformance to the SRP. There are no significant failure modes of important SSCs due to high winds or missile impacts, and it is therefore concluded that no unique PRA model for extreme winds and tornadoes is required in order to assess configuration risk for the RICT Program.
External Fires	Forest fires in the plant vicinity were evaluated as having a minimal potential impact on the plant, and are bounded by the effects of a loss of offsite power.	The impact of an external fire is bounded by the existing loss of offsite power initiating event. It is therefore concluded that no unique PRA model for external fires is required in order to assess configuration risk for the RICT Program.
Accidents From Nearby Facilities	There are no military bases, missile installations, chemical plants, hazardous material storage areas, or drilling operations within 10 miles of PSL. A review of nearby facilities was conducted and	Nearby facility accidents do not pose a significant safety impact to PSL based on conformance to the SRP. It is therefore concluded that no unique PRA model for facility accidents is required in order to assess configuration risk

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
	concluded that PSL conformed to the appropriate SRP criteria. Based on a review of satellite images, there are no new facilities in the vicinity of PSL, and so the conclusions of the IPEEE remain valid.	for the RICT Program.
Pipeline Accidents (e.g., natural gas)	There are no pipelines with 5 miles of PSL, which conforms to the SRP criteria. A review of available current information shows no new pipelines have been installed in the vicinity of PSL, and so the conclusions of the IPEEE remain valid.	There are no pipelines in sufficient proximity to the plant site to cause a significant hazard. It is therefore concluded that no unique PRA model for pipeline accidents is required in order to assess configuration risk for the RICT Program.
Release of Chemicals Stored at the Site	The accidental release of toxic chemicals may affect control room habitability. The initiating event would be a chemical spill or tank rupture caused, for example, by a handling accident, container failure, or some other accident. After the material is released, to contribute significantly to risk, it must be carried by some mechanism to the control room air intake. Evaluations of ammonia, chlorine, and carbon dioxide demonstrate that even with conservative assumptions on atmospheric conditions the frequency of toxic levels in the control room is well below SRP guidance. Other chemicals stored on site are screened as not able to cause a challenge to control room habitability.	There are no chemicals on site which can cause a significant challenge to control room habitability, and there is no impact to other plant mitigating equipment. It is therefore concluded that no unique PRA model for chemical releases is required in order to assess configuration risk for the RICT Program.
Seismic Events	Due to low seismicity region where St. Lucie is located, plant risk due to seismic initiating event is the lowest in USA. Therefore, NRC allowed PSL to perform a screening analysis, including walkdown, in lieu of standard A-46 analysis that would require either Seismic Margin Analysis (SMA) or Seismic PRA (SPRA). Using the most recent Ground Motion Response Spectra (GMRS) performed by EPRI and published by March 31, 2014, GMRS for PSL was not shown to scale higher than the site's Safe Shutdown Earthquake (SSE) level.	Seismic events are not a significant hazard due to the plant location in a low seismicity region. Therefore, no unique PRA model for seismic events is required in order to assess configuration risk for the RICT Program.

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
Transportation Accidents	The nearest major highways are U. S. 1 4.8 miles west-southwest, which is sufficient to preclude adverse effects on the plant, and State Route A1A 0.2 miles east. The possible effects of liquid propane releases from an accident on A1A adversely impacting the plant were evaluated as less than 1E-7/year. The nearest railroad is 2 miles west of the plant, which is sufficient to preclude adverse effects on the plant from explosions. Chemical releases were evaluated as adversely affecting the plant less than 1E-6/year. The nearest shipping routes in the Atlantic Ocean are 10 miles east which is sufficient to preclude adverse effects on the plant. The Intracoastal Waterway is 1.2 miles west. FSAR analysis of the explosion effects of the most hazardous materials and quantities likely to be shipped on the Intracoastal Waterway (gasoline in 16,000 barrel quantities) demonstrates a minimum safe distance of 797 feet, while the plant is approximately 6000 feet from the waterway. The SRP criteria for transportation accidents are therefore satisfied. Based on a review of satellite images, there are no new transportation routes in the vicinity of PSL, and so the conclusions of the IPEEE remain valid.	Transportation accidents cannot cause damage to the plant or are shown to have a bounding CDF of less than 1E-6 per year, consistent with the SRP criteria. It is therefore concluded that no unique PRA model for transportation accidents is required in order to assess configuration risk for the RICT Program.
Turbine-Generated Missiles	Turbine missiles and their potential plant impacts are discussed in UFSAR Chapter 3.5. Modern manufacturing and quality control procedures have eliminated the credibility of turbine rotor failures. However, such failures are hypothesized and the plant designed to protect SSCs required for safe shutdown and accident mitigation. For unit 1, a missile probability analysis was performed which demonstrates the frequency of missile generation	The frequency of turbine missile generation is very small compared to other initiating events. The consequences of the event are limited to a single failure of redundant safe shutdown equipment. The failure likelihood due to turbine missiles is much less than the failure probability from other causes. The likelihood of an initiating event involving a turbine missile, with the exact orientation necessary to cause failure of one redundant train of exposed equipment, and a random failure of the other redundant train, is

Table E4-1
Evaluation of Risks from External Hazards

External Hazard	Evaluation	Disposition for RICT Program
	is 1.88E-6/year, which was found acceptable to the NRC (Reference 9). For unit 2, turbine missile generation cannot cause a LOCA or penetrate the control room. Exposed outdoor equipment required for safe shutdown is protected from turbine missile damage by spatial separation of redundant components.	bounded by other scenarios modeled in the PRA, such as a plant trip with common cause failure of the equipment. It is therefore concluded that no unique PRA model for turbine missile accidents is required in order to assess configuration risk for the RICT Program.

4.0 REFERENCES

1. ML071200238, *Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines (TAC No. MD4995),"* Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.

2. NEI 06-09, *Risk-Informed Technical Specifications Initiative 4B: Risk-Managed Technical Specifications (RMTS) Guidelines*, Nuclear Energy Institute, Revision 0-A, November 2006.

3. WCAP-16952-NP, *Supplemental Implementation Guidance for the Calculation of Risk Informed Completion Time and Risk Managed Action Time for RITSTF Initiative 4B*, August 2010.

4. NUREG-1855, *Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making*, Volume 1, March 2009.

5. American Society of Mechanical Engineers and American Nuclear Society, *Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME/ANS RA-Sa-2009, New York (NY), February 2009.

6. FP&L Letter L-94-318, *NRC Generic Letter 88-20 Supplement 4, Individual Plant Examination of External Events for Severe Accident Vulnerabilities Report*, December 15, 1994.

7. NUREG-75/087, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*, 1975.

8. U. S. NRC, Regulatory Guide 1.59, *Design Basis Floods for Nuclear Power Plants*, Revision 2.

9. TP-04124-NP-A, *Missile Probability Analysis for the Siemens 13.9 m2 Retrofit Design of Low-Pressure Turbine by Siemens AG*, June 7, 2004 (Includes NRC Final Safety Evaluation letter dated March 30, 2004).

St. Lucie Nuclear Plant Units 1 and 2

Enclosure 5

BASELINE CDF AND LERF

E5-1.0 INTRODUCTION

Section 4.0, Item 6 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide the plant-specific total CDF and LERF to confirm applicability of the limits of Regulatory Guide (RG) 1.174, Revision 1 (Reference 3). (Note that RG 1.174, Revision 2 (Reference 4), issued by the NRC in May 2011, did not revise these limits.)

This attachment demonstrates that the total CDF and total LERF are below the guidance of RG 1.174, specifically, 1E-4/year CDF and 1E-5/year LERF, such that the risk metrics of NEI 06-09 may be applied to the St. Lucie Risk-Informed Completion Time (RICT) Program.

Table E5-1 provides the CDF and LERF values that resulted from a quantification of the baseline average annual models (Reference 5), which include contributions from internal events, internal flooding, fire, and seismic hazards. Other external hazards are below accepted screening criteria and therefore do not contribute significantly to the totals.

Table E5-1: Total Baseline Average Annual CDF/ LERF

Hazard	Unit 1 CDF (per rx-yr)	Unit 1 LERF (per rx-yr)	Unit 2 CDF (per rx-yr)	Unit 2 LERF (per rx-yr)
Internal Events	5.34E-06	7.79E-07	6.77E-06	2.32E-07
Internal Flood	8.58E-07	1.73E-07	8.98E-08	2.56E-09
Internal Fire	5.16E-05	6.96E-06	6.96E-05	7.96E-06
Seismic	3.49E-06	3.49E-07	3.49E-06	3.49E-07
Other Hazards*	4.01E-06	4.01E-07	4.01E-06	4.01E-07
Total	**6.53E-05**	**8.66E-06**	**8.40E-05**	**8.94E-06**

*** Includes all External Initiating events identified and listed in the IPEEE document, excluding Internal Flood, Internal Fire, and Seismic Events listed above.**

Notes:

- Seismic CDF is evaluated using the most updated site mean hazard curve developed by EPRI and published in 2014. LERF is conservatively considered as 10% of CDF value.
- Listed values reflect the anticipated configuration of the plant upon full implementation of NFPA 805 and related plant modifications to resolve fire protection issues. At the time of implementation of the RICT Program, the PRA model used will reflect the existing configuration of the plant.)

As demonstrated in the table, the total CDF and total LERF are within the guidance of RG 1.174 to permit small changes in risk which may occur during RICT Program implementation of extended Completion Times. Therefore, the St. Lucie RICT Program is consistent with NEI 06-09 guidance.

E5-2.0 REFERENCES

1. ML071200238, Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines (TAC No. MD4995)," Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.
2. NEI 06-09, Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Revision 0, November 2006.
3. Regulatory Guide 1.174, An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 1, November 2002.
4. Regulatory Guide 1.174, An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 2, May 2011.
5. PSL-BFJR-12-001, Revision 0, St. Lucie EPU Model Update for Units 1&2, May 2012.
6. PSL-BFJR-11-006, Revision 0, Quantification of Internal Flood Analysis for St. Lucie Units 1 & 2, November 2013.
7. PSL-IPEEE, Revision 0, Individual Plant Examination of External Events for St. Lucie Units 1 and 2, December 1994.
8. LTR-RAM-II-09-048, Revision 0, Transmittal PSL-1 EPU Report (Assessment of Post-EPU Risk from Fire, Flood, Other External Events and Shutdown Operation for St. Lucie Unit 1) Rev. 0, November 2009.
9. LTR-RAM-II-10-014, Revision C, Transmittal PSL-2 EPU Report (Assessment of Post-EPU Risk from Fire, Flood, Other External Events and Shutdown Operation for St. Lucie Unit 2), May 2010.
10. LTR-RAM-I-14-015 Rev 0, "Seismic Core Damage Frequency Estimates for St. Lucie, Westinghouse Electric Company, March 31, 2014.

St. Lucie Nuclear Plant Units 1 and 2

Enclosure 6

JUSTIFICATION OF APPLICATION OF AT-POWER PRA MODELS TO SHUTDOWN MODES

This attachment is not applicable to the St. Lucie Nuclear Plant submittal. FPL is proposing to apply the Risk-Informed Completion Time Program only in Modes 1 and 2 and not in the shutdown Modes.

St. Lucie Nuclear Plant Units 1 and 2

Enclosure 7

PRA MODEL UPDATE PROCESS

E7-1.0 INTRODUCTION

Section 4.0, Item 8 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide a discussion of the licensee's programs and procedures which assure the PRA models which support the RMTS are maintained consistent with the as-built/as-operated plant.

This attachment describes the administrative controls and procedural processes applicable to the configuration control of PRA models used to support the Risk-Informed Completion Time (RICT) Program, which will be in place to ensure that these models reflect the as-built/as-operated plant. Plant changes, including physical modifications and procedure revisions, will be identified and reviewed prior to implementation to determine if they could impact the PRA models per EN-AA-105, *Probabilistic Risk Assessment (PRA) Program* (Reference 3), and EN-AA-105-1000, *PRA Configuration Control and Model Maintenance* (Reference 4). The configuration control program will ensure these plant changes are incorporated into the PRA models as appropriate. The process will include discovered conditions associated with the PRA models, which will be addressed by the applicable site Corrective Action Program.

Should a plant change or a discovered condition be identified that has a significant impact to the RICT Program calculations as defined by the Configuration Control Program, an interim update of the PRA model will be implemented. Otherwise, the PRA model change is incorporated into a subsequent periodic model update. Such pending changes are considered when evaluating other changes until they are fully implemented into the PRA models. Periodic updates are performed no less frequently than every five years.

E7-2.0 PRA MODEL UPDATE PROCESS

E7-2.1 Internal Event, Internal Flood, Fire, and Seismic Event PRA Maintenance and Update

The Fleet risk management process ensures that the applicable PRA model used for the RICT Program reflects the as-built/as-operated plant for each of the NextEra/FPL units. The PRA configuration control process delineates the responsibilities and guidelines for updating the full power internal event, internal flood, fire, and seismic PRA models, and includes both periodic and interim PRA model updates. The process includes provisions for monitoring potential impact areas affecting the technical elements of the PRA models (e.g., due to plant changes, plant/industry operational experience, or errors or limitations identified in the model), assessing the individual and cumulative risk impact of unincorporated changes, and controlling the model and necessary computer files, including those associated with the configuration risk management program (CRMP) model.

E7-2.2 Review of Plant Changes for Incorporation into the PRA Model

(1) Plant changes or discovered conditions, as defined in the PRA Configuration Control Program, are reviewed for potential impact to the PRA models and including the CRMP model and the subsequent risk calculations which support the RICT Program (NEI 06-09 Section 2.3.4, Items 7.2 and 7.3, and Section 2.3.5, Items 9.2 and 9.3).

(2) Plant changes that meet the criteria defined in the PRA configuration control program (including consideration of the cumulative impact of other pending changes) will be immediately incorporated in the applicable PRA model(s), consistent with the NEI 06-09 guidance. Otherwise, the change is assigned a priority and is incorporated at a subsequent periodic update consistent with procedural requirements. (NEI 06-09 Section 2.3.5, Item 9.2)

(3) PRA updates for plant changes are performed at least once every two refueling cycles, consistent with the guidance of NEI 06-09 (NEI 06-09 Section 2.3.4, Item 7.1, and Section 2.3.5, Item 9.1).

(4) If a PRA model change is required for the CRMP model, but cannot be immediately implemented for a significant plant change or discovered condition, either:

A. Alternative analyses to conservatively bound the expected risk impact of the change will be performed. In such a case, these alternative analyses become part of the RICT Program calculation process until the plant changes are incorporated into the PRA model during the next update. The use of such bounding analyses is consistent with the guidance of NEI 06-09.

B. Appropriate administrative restrictions on the use of the RICT Program for extended Completion Times are put in place until the model changes are completed, consistent with the guidance of NEI 06-09.

These actions satisfy NEI 06-09 Section 2.3.5, Item 9.3.

E7-3.0 REFERENCES

1. ML071200238, Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines (TAC No. MD4995)," Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.
2. NEI 06-09, Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Revision 0, November 2006.
3. EN-AA-105, *Probabilistic Risk Assessment (PRA) Program.*
4. EN-AA-105-1000, *PRA Configuration Control and Model Maintenance.*

St. Lucie Nuclear Plant Units 1 and 2

Enclosure 8

ATTRIBUTES OF THE CRMP MODEL

E8-1.0 Introduction

Section 4.0, Item 9 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide a description of PRA models and tools, including identification of how the baseline probabilistic risk assessment (PRA) model is modified for use in the CRMP tools, quality requirements applied to the PRA models and CRMP tools, consistency of calculated results from the PRA model and the CRMP tools, and training and qualification programs applicable to personnel responsible for development and use of the CRMP tools. The scope of structures, systems, and components (SSCs) within the CRMP will be provided. This item should also confirm that the CRMP tools can be readily applied for each Technical Specification (TS) limiting condition for operation (LCO) within the scope of the plant-specific submittal.

This attachment describes the necessary changes to the peer-reviewed baseline PRA models for use in the CRMP software to support the Risk-Informed Completion Time (RICT) Program. The process employed to adapt the baseline models for CRMP use is demonstrated:

(a) to preserve the core damage frequency (CDF) and large early release frequency (LERF) quantitative results;
(b) to maintain the quality of the peer-reviewed PRA models; and
(c) to correctly accommodate changes in risk due to configuration-specific considerations.

Quality controls and training programs applicable for the CRMP are also discussed in this enclosure. Additional considerations regarding the fire PRA model to address implementation of National Fire Protection Association (NFPA)-805 as the licensing basis for the fire protection program is also discussed at the end of this attachment.

E8-2.0 Translation of Baseline PRA Model for Use in CRMP

The baseline PRA models for internal events, including the internal flood models, are the peer-reviewed models, updated when necessary to incorporate plant changes to reflect the as-built/as-operated plant. The internal fire models currently supporting NFPA-805 which will be used in the RICT Program are also in this same category. Prior to implementation of the RICT Program, Internal events model will be integrated with internal flood model and internal fire model, for each unit, to develop a one-top integrated baseline model. This baseline model will be translated to develop the CRMP model to be used for the RICT Program by removing mutual exclusive maintenance practice usually prohibited by plant procedures or guidelines, and allowing user-specified train alignments.

It is intended to use EPRI's EOOS Software to facilitate all configuration-specific risk calculations, and in particular, to support the RICT Program implementation. The integrated models may include additional changes that are currently logged in the database for periodic model maintenance and update that are considered pending for the upcoming cycles of model update in accordance with plant procedures.

There are two model changes planned that are specifically required for RICT Program implementation other than those already been identified as pending per the periodic model maintenance and update process.

(a) Adding appropriate flags to allow users of CRMP model to select respective quadrant of the refueling cycle which allows adjustment of the probability associated with Unfavorable Moderator Temperature Coefficient (UMTC) needed for Anticipated Transient Without Scram (ATWS).

(b) Revise logic associated with Initiating Event Fault Tree to replace existing events associated with fixed Mean-Time-To-Repair (MTTR) of 72 hours with basic events with annual failure.

Development of the integrated models, and the changes that might be required thereof are controlled using plant procedures and calculations, which include all necessary quality controls and reviews.

E8-3.0 Scope of Systems, Structures, and Components within the CRMP

In addition to the SSCs modeled for each TS LCO in the scope of the RICT Program (described in Enclosure 1), the additional SSCs and/or corresponding functions which are in the PRA models but not in plant TS are listed below.

- Instrument Air System and Turbine Cooling Water System
- Main Feedwater and Condensate systems, pumps, and valves.
- Treatment Water Storage System as makeup source for AFW CST.

E8-4.0 Quality Requirements and Consistency of PRA Model and CRMP Tools

The approach for establishing and maintaining the quality of the PRA models, including the CRMP model, includes both a PRA maintenance and update process (described in Enclosure 7), and the use of self-assessments and independent peer reviews (described in Enclosure 2).

The information provided in Enclosure 2 demonstrates that the site's internal event, internal flood, and internal fire PRA models reasonably conform to the associated industry standards endorsed by Regulatory Guide 1.200. This information provides a robust basis for concluding that the PRA models are of sufficient quality for use in risk-informed licensing actions.

For maintenance of an existing CRMP model, changes made to the baseline PRA model in translation to the CRMP model will be controlled and documented. An acceptance test is performed after every CRMP model update to verify proper translation of the baseline PRA models and acceptance of all changes made to the baseline PRA models pursuant to translation to the CRMP model. This testing also verifies correct mapping of plant components to the basic events in the CRMP model.

E8-5.0 Training and Qualification

PRA staff is responsible for development and maintenance of the CRMP model. The PRA staff is trained in accordance with the site's Engineering personnel training program. Operations and Work Control staffs will use the CRMP tool under the RICT Program and staffs are trained in accordance with a program using National Academy for Nuclear Training (ACAD) documents, which is also accredited by INPO.

E8-6.0 Application of the CRMP Tool to the RICT Program Scope

The plant will use the EPRI software program EOOS, whose future revisions is currently known as *Phoenix*, as its CRMP platform. This program is specifically designed by EPRI to support implementation of RMTS, and is currently used at site. EOOS will permit the user to evaluate all configurations within the scope of the RICT Program using appropriate mapping of equipment to PRA model elements.

E8-7.0 Additional Considerations for NFPA-805 Modifications

The existing fire PRA model includes credit for committed plant modifications to be implemented as part of the transition of the fire protection licensing basis to NFPA-805 (as described in commitment 3 contained in Enclosure 5 of Reference 5). At the expected time of implementation of the RICT Program, not all of these committed modifications will be implemented. FPL proposes to use the risk insights from the post-transition fire PRA model and commits as part of the RICT Program implementation to maintain compensatory measures in place until the associated plant modifications are implemented, as follows:

- At any time when a RICT is in effect, a continuous fire watch will be established in the Cable Spreading rooms until incipient detection and hot shutdown panel modifications, described in Reference 5, are implemented.
- At any time when a RICT is in effect, welding and cutting activities will be prohibited in impacted fire zones/areas as identified by, and in accordance with, the CRMP procedure.

E8-8.0 REFERENCES

1. ML071200238, *Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines (TAC No. MD4995),*" Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.

2. NEI 06-09, *Risk-Informed Technical Specifications Initiative 4B: Risk-Managed Technical Specifications (RMTS) Guidelines*, Nuclear Energy Institute, Revision 0-A, November 2006.

3. Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Revision 2, March 2009.

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

5. FPL Letter L-2013-099, "*License Amendment Request for Transition to 10 CFR 50.48(c) – NFPA 805 Performance-Based for Fire Protection for Light Water Reactor Generating Plants (2001) Edition),*" March 22, 2013.

St. Lucie Nuclear Plant Units 1 and 2

Enclosure 9

KEY ASSUMPTIONS AND SOURCES OF UNCERTAINTY

E9-1.0 Introduction

Section 4.0, item 10 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines" (Reference 2) requires that the License Amendment Request (LAR) provide a discussion of how the key assumptions and sources of uncertainty were identified, and how their impact was assessed and dispositioned.

This enclosure provides a discussion of how the key assumptions and sources of uncertainty were identified, and how their impact on the Risk-Informed Completion Time (RICT) Program was assessed and dispositioned.

E9-2.0 Process for Identification of Key Assumptions and Sources of Uncertainty

Sources of model uncertainty and related assumptions, defined consistent with Regulatory Guide (RG) 1.200 Revision 2 (Reference 3) and the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Probabilistic Risk Assessment (PRA) Standard (Reference 4), have been identified for the baseline PRA models using the guidance of NUREG-1855 (Reference 5) and EPRI TR-1016737 *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessment* (Reference 6).

The detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 and Section 3.1.1 of EPRI TR-1016737. The process in these references was mostly developed to evaluate the uncertainties associated with the internal events PRA model; however, the approach can be applied to other types of hazard groups.

E9-3.0 Disposition of Key Assumptions and Sources of Uncertainty

The list of assumptions and sources of uncertainty from Reference (7) were reviewed to identify those which would be significant for the evaluation of configuration-specific changes in risk. If the model uses a non-conservative treatment, or methods which are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine the impact on RICT Program calculations. Only those assumptions or sources of uncertainty which could significantly impact the configuration risk calculations were considered key for this application.

The internal events PRA models are used to support the fire and seismic PRA, and so the assumptions and uncertainties evaluated would apply to these PRA models as well.

Key assumptions and sources of uncertainty for the RICT Program application are identified and dispositioned in Table (E9-1).

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
GENERIC IE-A-4 Human-induced initiating events. Support system initiating events do not explicitly consider human-induced events.	While not explicitly considered, the generic and plant-specific data used to develop the frequencies of these initiators would include events whose cause was human error.	For those systems in the scope of the RICT Program which are also in the scope of initiating events, risk management actions (RMAs) will include consideration of actions to enhance protection of the remaining available train or busses.
GENERIC IE-A-9 Common cause failures (CCF). CCFs are not included in initiating event models for electrical busses and other system initiators. (Similar item IE-B-3)	CCFs for electrical buses and panels are considered to be very rare events, and Combustion Engineering Owners' Group (CEOG) guidelines do not include recommendations for CCF groupings for buses. CCF of components in systems whose failure was considered an initiating event, (i.e., loss of ICW or CCW) are included in the system mitigation fault tree logic.	For RICT Program delta risk calculations, CCF is not significant since the failure probability is based on the remaining available train or component. NEI 06-09 guidelines require the use of RMAs to address potential CCFs when emergent failures occur. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC IE-B-1 Subsumed events. System failures which result in a reactor trip may be grouped with another bounding initiating event. An exception is failure of Risk-Significant HVAC which is considered subsumed by the reactor trip (T1) event.	A room heatup study did not suggest an imminent reactor trip following loss of HVAC support to components since the temperature increase is very slow to develop, and current plant compensatory measures suffice to control temperature rise at affected locations.	The treatment of loss of HVAC in considered realistic, and the potential non-conservatism in subsuming this event as a reactor trip event does not introduce any significant uncertainty. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Reactor Coolant System (RCS) pressure challenge. Pressurizer power-operated relief valves (PORVs) are assumed challenged due to reactor trip on high pressurizer pressure if the anticipatory reactor trip does not function. Pressurizer Code Safety Valves are assumed challenged if the turbine trip results in a high pressurizer pressure trip and at least one PORV does not open.	The assumption is used as the basis for considering turbine trip separately from a reactor trip due to the potential for additional challenges to relief valves associated with a turbine (loss of load) trip. The current modeling approach is judged not to impact the overall results in a significant manner.	The assumption of a pressure challenge is conservatively modeled, and would not adversely affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Excessive LOCA mitigation. Core damage is assumed for interfacing systems LOCA (ISLOCA) and reactor vessel rupture initiating events - no mitigation is credited.	The current modeling approach is consistent with current industry practice but introduces a very slight conservatism that should not impact the overall results in a significant manner.	The assumption has a potentially non-conservative impact on the calculated RICT for ECCS systems, since they may actually provide some mitigation capability for these events which would not be reflected in the calculations. Since the assumption is consistent with current industry practice and is judged to be a small impact, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Loss of HVAC. An extended loss of the Non-Risk Significant HVAC system may have a long term operational impact on the PSL units, but it is assumed that sufficient time would be available to make repairs to the system before shutdown would be required.	The assumption is used as the basis for screening the system failure as a potential initiating event. Room heat-up analysis indicated sufficient time available for controlled shutdown if HVAC is completely lost and compensatory measures are not adequate to control rising temperature for key safety components.	The assumption to screen Non-Risk Significant HVAC as an initiator is justified by analysis, and no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Relief valve capacity. The unit 2 shutdown cooling suction line inside-containment relief valves are assumed to have sufficient capacity to prevent overpressurization of downstream piping.	This assumption is judged to be reasonable given the design features of the suction line and the relief valves.	This assumption does not affect RICT calculations, since only the frequency of ISLOCA initiating events is impacted, and these events are assumed not to be mitigated. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Pre-initiator screening. All pre-initiator human failure events (HFEs) in the ISLOCA models are evaluated using a screening probability.	This assumption adds a small amount of conservatism to the ISLOCA results.	This assumption does not affect RICT calculations, since only the frequency of ISLOCA initiating events is impacted, and these events are assumed not to be mitigated. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Passive CCF. Only CCFs of active components are considered in the ISLOCA models.	This assumption adds a small amount of non-conservatism to the ISLOCA results. The assumption is consistent with CEOG guidance for treatment of passive components.	This assumption does not affect RICT calculations, since only the frequency of ISLOCA initiating events is impacted, and these events are assumed not to be mitigated. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Low pressure piping structural capacity. Low pressure piping is assumed to fail upon over-pressurization due to an ISLOCA event.	The failure of piping given an overpressure is probably not a certainty, and this assumption is conservative for the ISLOCA analysis.	This assumption does not affect RICT calculations, since only the frequency of ISLOCA initiating events is impacted, and these events are assumed not to be mitigated. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC ISLOCA break size. Breaks in lines of less than 1" diameter are assumed to be non-risk significant for the ISLOCA analysis.	This assumption is based on judgment that sufficient time should be available to depressurize the RCS prior to depletion of the refueling water storage tank (RWST).	This assumption does not affect RICT calculations, since only the frequency of ISLOCA initiating events is impacted, and these events are assumed not to be mitigated. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
AS-A-3 Thermal-hydraulic codes. The code MAAP 4 is used and although this is a consensus approach, the NRC has recently posed potential issues regarding the code's appropriate application. (Similar item SC-B-6)	MAAP-4 is a consensus approach for PRA analysis at this time; pending resolution of the NRC issues, this may change and would be re-visited as a PRA update or upgrade as part of the normal PRA model maintenance practices.	As a consensus approach, the use of MAAP-4 is acceptable for PRA. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
AS-B-1 Room cooling calculations. Room heatup calculations have not been performed for all important equipment areas. Therefore, it is possible that some HVAC-related dependencies may not be fully addressed in the current model.	This assumption may introduce some non-conservatism in the PRA results, as HVAC-related failures may contribute to CDF or LERF. However, the overall magnitude of this non-conservatism is judged to be small.	No HVAC systems are included in the scope of the RICT Program. The slight conservatism in not modeling HVAC failures as a contributor to supported system failure is not significant due to the expected failure timing (long term) and the possible compensatory measures to provide alternative cooling. No additional considerations are required to address this source of uncertainty in the RICT Program.
AS-B-2 Temperature-dependent failure criteria. As room heatup has not been fully evaluated for PSL (see AS-B-1), all relevant temperature-dependent failure criteria may not be fully addressed.	This assumption may introduce some non-conservatism in the PRA results.	See disposition of AS-B-1.
PLANT-SPECIFIC Credit for secondary steam relief. Failure of secondary steam relief is neglected for secondary heat removal for transients.	Since adequate steam relief is provided by the passive relief valves, the assumption that secondary steam relief always succeeds is judged to only be a small non-conservatism.	Secondary relief valves are not in the scope of the RICT Program This slight non-conservatism would have a negligible impact on RICT calculations, and therefore no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Credit for primary makeup. For main steam line break scenarios, primary makeup is included to consider the initial overcooling and reduced inventory in the RCS.	This assumption to require operation of primary makeup may be slightly conservative, but is judged to have no significant effect on overall results.	This slight conservatism would have a negligible impact on RICT calculations, and therefore no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Hot leg recirculation. All medium LOCAs require hot leg recirculation cooling.	Since the requirement for hot leg recirculation is dependent upon break size and location, not all medium LOCA scenarios require hot leg recirculation, and so this assumption is slightly conservative.	This assumption would have a small conservative impact on RICT calculations for ECCS equipment. Since the impact is conservative, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Hot leg recirculation. Large LOCAs do not require hot leg recirculation cooling.	This is a non-conservative assumption, but large LOCAs are not a significant contributor to risk so the impact is judged to be small.	This assumption is judged to have a small impact on RICT calculations for ECCS equipment. Based on the small impact of these assumptions, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Steam generator isolation. For steam generator tube rupture (SGTR) events, failure to isolate any of the following constitutes and isolation failure for the affected SG: main steam isolation valve (MSIV), MSIV bypass valve, SG blowdown lines, and auxiliary feedwater (AFW) steam supply. Main Feedwater isolation and bypass isolation valves close on the safety injection signal, and are not evaluated for failure to remain closed.	These assumptions regarding isolation of a ruptured SG are judged to have a very small impact on the analysis results.	Based on the small impact of these assumptions, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Passive piping failure. For most system models, ruptures and other passive piping failures (other than the tanks and heat exchangers) were not modeled.	This assumption is slightly conservative, but the likelihood of a piping passive failure is judged to be small relative to other active system failures.	Based on the small impact of this assumption, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC DC power mission time. Failure of DC control power over the full mission time is assumed to prevent system components from operating.	This assumption is conservative since DC power is only required for a short period at the time that components are actuated.	This conservative assumption will result in conservative calculations of RICTs. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Room cooling. Room cooling is not required for chemical and volume control system pumps during accident conditions.	This non conservative assumption would not impact short term operation of the pumps as room heatup effects would only slowly occur over time.	Although non-conservative, requiring room cooling and including appropriate recovery actions for alternate cooling is judged to have a negligible impact on the results. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Time-dependent model. For ECCS operation, simplifications were made in the time-dependent modeling of injection and recirculation phase failures.	The overall impact of these assumptions is judged to be very small.	Given the expected very small impact on results, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Unit 2 HPSI valves. For unit 2, closure of the HPSI header isolation valves (V3654 and V3656) during injection is assumed to fail the associated HPSI header and failure of these valves to close when initiating hot leg recirculation is assumed to fail the associated hot leg injection line.	This is a conservative assumption since it is possible that some flow may still be available even with these failures occurring.	This conservative assumption will result in conservative calculations of RICTs. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Ventilation maintenance. Maintenance is not performed on either the main standby supply or exhaust fans for the Reactor Auxiliary Building (RAB) Ventilation system during power operation.	This is a slightly non-conservative assumption.	This assumption has no impact on RICT calculations, since the actual plant configuration is assessed and not assumed. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Instrument Air (IA) valves. Failure of the IA compressor full load solenoid valves was conservatively assumed to fail the compressor.	This assumption may add a small conservatism to the model, as the affected compressor can still continue to operate at reduced load. However, the overall effect of this assumption on the PRA results is judged insignificant due to the relative risk insignificance of the IA system.	This slight conservatism in IA modeling is judged not to have any significant impact on RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Reactor Protection System (RPS) CCF. CCF of some RPS components are not in the model.	This assumption represents a small non-conservatism in the RPS model. RPS system failure is dominated by CCF due to its high level of redundancy.	For RICT calculations involving RPS components in the scope of the program, CCFs within the RPS would not contribute to the delta risk calculation for the RICT, since the calculation would be based on the remaining functional capability of the remaining operable RPS train, and CCF contributions would cancel out. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Statistical distribution assumptions. A lognormal distribution with an error factor of 10 is assumed for all test/maintenance events in the model.	This assumption potentially impacts uncertainty bands of calculations, but does not impact point estimate calculations.	The RICT Program uses a no-maintenance model with the test/maintenance basic events set to zero, so this assumption has no impact on RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Unavailability. One hour of out-of-service time has been assumed for various components that have not been removed from service during power operation over the data collection period.	This approach conservatively includes the failure experience of these components that have not required any maintenance unavailability. The overall amount of conservatism added to the PRA results is judged to be very small.	The RICT Program uses a no-maintenance model with the test/maintenance basic events set to zero, so this assumption has no impact on RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC AFW CCF. AFW CCF events include some conservative data and assumptions.	The net impact of these assumptions probably results in a small conservatism in the PRA results.	This conservative assumption will not impact calculations of RICTs, since CCF does not contribute to configuration-specific risk change (CCF contributions appear in both the configuration-specific and baseline, and cancel out.) Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Staggered testing. Non-staggered testing is assumed when the testing scheme cannot be established.	This assumption is slightly conservative, and the overall impact on the results is judged to be small.	This conservative assumption will result in conservative calculations of RICTs. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC CCF error factors. The error factor calculations assume that the alpha factor uncertainty parameters can be transformed to EFs without the introduction of additional uncertainty.	This assumption impacts only parametric uncertainty calculations. The impact is judged to be small.	Parametric uncertainty calculations are not used in the RICT Program. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Staggered testing impact on uncertainty. When calculating the alpha factor uncertainty it was assumed that applying a staggered or non-staggered testing scheme will not impact the uncertainty because it is a ratio of the 95th percentile to the mean.	This assumption impacts only parametric uncertainty calculations. The impact is judged to be small.	Parametric uncertainty calculations are not used in the RICT Program. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC CCF error factors The CCF alpha factors are assumed to have an upper bound error factor of 15.	This assumption impacts only parametric uncertainty calculations. The impact is judged to be small.	Parametric uncertainty calculations are not used in the RICT Program. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC CCF treatment. CCF between similar components is not included when failure of any component fails the required function.	This assumption has no impact on results, since it is assumed that any single independent failure of one component in a CCF group fails the function. Adding in a CCF contribution would be offset by reducing the independent failure by the same amount, so there would be no net impact on results.	There is no effect on RICT Calculations due to this assumption. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC CCF treatment. For situations in which more than four components were contained within a CCF group, simplification of the CCF modeling was performed, as recommended in NUREG-1855.	This assumption may add a small amount of non-conservatism to the PRA results; however, the impact of this assumption is judged to be insignificant.	This conservative assumption will result in conservative calculations of RICTs. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
IF-A-1 Flood area identification. Flood areas are defined by fire zones, and may not always ensure spray effects across zones is proscribed. (Similar assumption IF-C-5)	This assumption is non-conservative, but the potential impact is judged to be small since it is not likely that the equipment damage for a significant spray across fire zones would be more limiting than a major flood in the initiating zone.	It is judged that this non-conservative assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
IF-B-1 Floor drain impacts. No credit is taken for drain flow unless the flow would exacerbate the flood scenario. (Similar assumption IF-C-5)	This assumption is likely to be conservative, but could introduce uncertainty in flood scenario timing. No detailed evaluation of the potential for such impacts is documented.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
IF-C-2 Subsumed events. All flooding events are analyzed as resulting from the largest pipe break in the flood area with smaller breaks subsumed by the larger breaks. (Similar assumptions IF-C-11, IF-C-12, IF-D-2)	This assumption is expected to conservatively impact internal flooding calculations. However, flood mitigation features may have different relative importance based on the flood size.	It is judged that this conservative assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
IF-C-4 Flood mitigation. Specific actions and procedures that will be followed to mitigate a flood are not clearly addressed in the internal flooding PRA model. (Similar assumptions IF-C-7, IF-C-15, IF-C-16)	Actions taken to mitigate a flood can be affected by mitigation features and the flood rate assumed can affect the accident progression and impact of mitigation features.	The model is conservative since it does not credit any human actions to recover or isolate flood scenarios. If credited, the impact would be to reduce the frequency of flood initiators, and increase the duration of any impacted RICT. Based on a conservative treatment, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
IF-C-6 Spray protection. Instruments and electrical panels which are environmentally qualified for exposure to steam are able to withstand the effects of spray or splashing. Equipment further than 30 feet from a spray source would not be damaged. (Similar assumption IF-C-13)	This assumption is based on engineering judgment.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
IF-C-10 Propagation pathways. Assumptions related to the propagation of flooding through failed barriers and ventilation ducts are considered a source of uncertainty. (Similar assumption IF-C-18)	Failure or unavailability of barriers is not considered in the analysis. Propagation through ducts is only considered if the duct is submerged.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
IF-C-17 Effectiveness in adverse environments. Air and hydraulic valves are not subject to failure when exposed to spray or steam and that equipment that is qualified for a steam environment will survive spray.	This assumption is based on engineering judgment.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
IF-C-19 Barrier failure/unavailability. The failure or maintenance unavailability of flood barriers is not considered. (Similar assumption IF-C-21)	This is a non-conservative assumption, but the impact is expected to be small due to the passive nature of the barriers and the limited maintenance unavailability typically incurred.	It is judged that this assumption would not significantly affect RICT calculations. However, the unavailability of internal flood barriers will be addressed through consideration of RMAs.
IF-D-1 Flood frequency data. Data sources are combined and may result in double-counting of failures due to overlap. Older data may not be directly applicable to current operational practices. Data does not include valve mis-positioning as a flood initiator.	The data represents a source of uncertainty in the analysis, which has conservative and non-conservative impacts.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Internal flood reactor trip. The plant is assumed to be tripped if a flood could require a plant shutdown.	This assumption may be slightly conservative since a controlled plant shutdown should be less significant than a reactor trip.	It is judged that this conservative assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Internal flood mitigation. The operators will not intervene to terminate a flood until 24 hours has elapsed.	This is a conservative assumption.	It is judged that this conservative assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Internal flood door failure. Doors fail at a flood height of 3-feet.	This assumption could result in changes to flood scenarios and therefore result in conservative or non-conservative changes in results. If the flood involves a large source, then the assumption would not be relevant, since if the door would eventually fail as the flood level continued to increase.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Internal flood damage modes. In specific flood scenarios involving high energy line breaks, no pipe whip or jet impingement induced damage of components occurs and that these failures will not impact isolation.	While non-conservative, this assumption is judged to have a small impact since flood scenarios involving high energy line breaks have low frequencies. The FSAR which does not identify these failure modes for high energy line breaks in the affected areas.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Internal CCW flood. The operator will isolate the ruptured CCW header.	Failure of the operator to isolate the header could result in a loss of all CCW, so this assumption is non-conservative. However, there are distinct cues and procedures to direct these actions, so it is unlikely that the header would not be isolated, so the assumption is reasonable.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
PLANT-SPECIFIC Internal flood in control room. Damage to equipment in the control room occurs when the flood level reaches one inch.	A different damage height could result in different equipment damage timing. However, since the control room is continuously occupied, it is unlikely that this level will be reached and therefore this assumption is judged to have little impact.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Internal flood in battery room. Rupture of piping in the battery room does not result in a loss of DC initiating event.	Water released due to pipe rupture in battery room will not stay in the room, but will flow to adjacent areas and other locations at lower elevations below the room through the room door gap to downstairs HVAC equipment area, even when the room drains are plugged/clogged. Therefore, the impact is expected to be a plant transient due to momentary loss of DC power,	It is judged that this assumption is reasonable and would not impact RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Internal flood in letdown line. For two scenarios involving letdown line ruptures, the leak is isolated before heat and humidity cause damage to equipment.	Consideration of these additional failure modes might increase flooding risks; however, these effects are expected to be slowly developing, and would not impact environmentally qualified equipment. Therefore, the impact is judged to be small.	It is judged that this assumption would not significantly affect RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
PLANT-SPECIFIC Internal flood door gaps. A one-quarter inch door gap is assumed unless specific measurements are made.	This assumption is used to estimate flow of water through room doors that are assumed to have a minimum of 1/4" gap unless a specific bigger/smaller gap is measured as indicated.	It is judged that this assumption is reasonable and would not impact RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
GENERIC Grid Stability. Recently the stability of at least some local areas of the electric power grid has been questioned. The potential duration and complexities of recovery from such events may not be reflected in the offsite power recovery analysis.	Loss of offsite power frequency and recovery are based on industry-wide data and plant-specific battery capabilities to permit alignment of breakers for recovery.	There is no statistical basis to assess loss of offsite power events differently at this time. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Support system initiating events. The use of plant-specific models has led to inconsistencies in the treatment of CCF and equipment recovery.	Recovery of support system initiators is not credited. A mean time to repair of a failed train is assessed to determine the appropriate mission time for the second train, where applicable for a support system initiator. CCF is treated in the mitigating system tree.	Not crediting recovery for support system initiators is conservative for estimating the frequency of these initiating events, which conservatively impacts RICT calculations. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Core cooling success following containment failure. Loss of containment heat removal leading to long-term containment over-pressurization and failure can be a significant contributor in some PRAs. Consideration of the containment failure mode might result in additional mechanical failures of credited systems. Containment venting through "soft" ducts or containment failure can result in loss of core cooling due to environmental impacts on equipment in the reactor building, loss of NPSH on ECCS pumps, steam binding of ECCS pumps, or damage to injection piping or valves.	Success of containment heat removal is required to support success of ECCS recirculation.	This source of uncertainty is addressed by requiring containment heat removal to support ECCS recirculation. No additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
GENERIC Containment sump/strainer performance. All PWRs are improving ECCS sump management practices, including installation of new sump strainers at most plants. There is not a consistent method for the treatment of ECCS sump performance.	Common cause sump plugging events included in the PRA model.	Since the failure mode is addressed in the PRA, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Impact of failure of RCS pressure relief. Certain scenarios can lead to RCS/RPV pressure transients requiring pressure relief. Usually, there is sufficient capacity to accommodate the pressure transient. However, in some scenarios, failure of adequate pressure relief can be a consideration. Various assumptions can be taken on the impact of inadequate pressure relief.	Generic success criteria based on CEOG guidance for pressure relief are used. Failure of pressure relief is assumed to proceed to core damage.	Since the failure mode is addressed in the PRA consistent with applicable owner group guidance, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Operability of equipment in beyond design basis environments. Due to the scope of PRAs, scenarios may arise where equipment is exposed to beyond design basis environments (w/o room cooling, w/o component cooling, w/ deadheading, in the presence of an un-isolated LOCA in the area, etc.)	It is assumed that equipment will fail to operate in conditions beyond its environmental qualifications. This assumption may add a small amount of conservatism to the base PRA results, if equipment could continue to operate under these conditions. However, the impact of this assumption on the PRA results is believed to be small.	This assumption conservatively treats the source of uncertainty. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption/Uncertainty	Discussion	Disposition for RICT Program
GENERIC Credit for Emergency Response Organization (ERO). Most PRAs do not give much, if any credit, for initiation of the ERO, including actions included in plant-specific severe accident management guidelines (SAMGs) and the new B5b mitigation strategies. The additional resources and capabilities brought to bear via the ERO can be substantial, especially for long-term events.	Credit for ERO is not taken. Credit for some direction from the ERO for longer-term actions would be a realistic assumption, and not crediting these actions is a slight conservative treatment.	This assumption conservatively treats the source of uncertainty. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Core melt arrest in-vessel. Typically, the treatment of core melt arrest in-vessel has been limited. However, recent NRC work has indicated that there may be more potential than previously credited.	Arresting an in-vessel core melt event is only included for loss of offsite power sequences through recovery of offsite power. All SBO sequences that do not arrest core melt progression are assumed to have no CHR capability.	This assumption conservatively treats the source of uncertainty. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
GENERIC Ex-vessel cooling of lower head. The lower vessel head of some plants may be submerged in water prior to the relocation of core debris to the lower head. This presents the potential for the core debris to be retained in-vessel by ex-vessel cooling. This is a complex analysis impacted by insulation, vessel design and degree of submergence.	Ex-vessel cooling of the lower head is not considered due to uncertainties in the behavior of the lower head penetrations and the presence of insulation surrounding the lower head. This is considered a realistic treatment.	This assumption realistically treats the source of uncertainty. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-1 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption/Uncertainty	Discussion	Disposition for RICT Program
GENERIC Core debris contact with containment. In some plants, core debris can come in contact with the containment shell (e.g., some BWR Mark Is, some PWRs including free-standing steel containments). Molten core debris can challenge the integrity of the containment boundary. Some analyses have demonstrated that core debris can be cooled by overlying water pools.	This is not considered as an early failure mechanism because there is no direct path for core debris to contact the containment shell.	This assumption realistically treats the source of uncertainty with regards to early containment failure, and therefore there is no impact on LERF calculations for a RICT. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

E9-4.0 REFERENCES

1. ML071200238, *Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines (TAC No. MD4995)*," Letter from Jennifer M. Golder (NRR) to Biff Bradley (NEI), May 17, 2007.

2. NEI 06-09, *Risk-Informed Technical Specifications Initiative 4B: Risk-Managed Technical Specifications (RMTS) Guidelines*, Nuclear Energy Institute, Revision 0-A, November 2006.

3. Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Revision 2, March 2009.

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

5. NUREG-1855, Volume 1, *Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making.*

6. EPRI TR-1016737, *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments*, December 2008.

7. PSL-SNBK-UNCERT, *Uncertainty Notebook for St. Lucie Units 1 & 2*, August 2013.

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Enclosure 10

PROGRAM IMPLEMENTATION

Introduction

This enclosure provides a description of the implementing programs and procedures regarding the plant staff responsibilities for the Risk-Informed Completion Time (RICT) Program, including training of plant personnel, and specifically discusses the decision process for risk management action implementation during extended Completion Times (CT).

RICT Program and Procedures

FPL will develop a program description and implementing procedures for the RICT Program. The program description will establish the management responsibilities and general requirements for risk management, training, implementation, and monitoring of the RICT program. More detailed procedures will provide specific responsibilities, limitations, and instructions for implementing the RICT program. The program description and implementing procedures will incorporate the programmatic requirements for Risk Managed Technical Specifications included in NEI 06-09.

The Operations Department (licensed operators) is responsible for compliance with the Technical Specifications (TS) and will be responsible for implementation of RICTs and risk management actions (RMA). Entry into the RICT program will require management approval prior to pre-planned activities and as soon as practicable following emergent conditions.

The procedures for the RICT program will address the following attributes consistent with NEI 06-09:

- Plant management positions with authority to approve entry into the RICT Program.
- Important definitions related to the RICT Program.
- Departmental and position responsibilities for activities in the RICT Program.
- Plant conditions for which the RICT Program is applicable.
- Limitations on implementing RICTs under voluntary and emergent conditions.
- Implementation of the RICT Program 30-day back stop limit.
- Use of the Configuration Risk Management Program (CRMP) tool.
- Guidance on recalculating RICT and risk management action time within 12 hours or within the most limiting front-stop CT after a plant configuration change.
- Requirements to identify and implement RMAs when the RMAT is exceeded or is anticipated to be exceeded.
- Guidance on the use of RMAs including the conditions under which they may be credited in RICT calculations.
- Guidance on crediting PRA functionality.
- Conditions for exiting a RICT.
- Requirements for training on the RICT Program.
- Documentation requirements related to individual RICT evaluations, implementation of extended CTs, and accumulated annual risk.

RICT Program Training

The scope of training for the RICT Program will include rules for the new TS program, CRMP software, TS Actions included in the program, and procedures. This training will be conducted for the following NextEra personnel:

Site Personnel

- Operations Director
- Operations Personnel (Licensed and Non- Licensed)
- Operations Training
- Outage Manager
- On-line Manager
- Planning and Scheduling Personnel
- Work Week Managers
- Licensing Personnel
- Selected Maintenance Personnel
- Engineering
- Risk Engineering
- Other Selected Management

Corporate Personnel

- Operations Corporate Functional Area Manager
- Fleet Outages Corporate Functional Area Manager
- Licensing Management and Personnel
- Risk Engineering Management and Personnel
- Training Management and Personnel
- Other Selected Management

Training will be carried out in accordance with NextEra training procedures and processes. These procedures were written based on the Institute of Nuclear Power Operations (INPO) Accreditation (ACAD) requirements, as developed and maintained by the National Academy for Nuclear Training. NextEra has planned three levels of training for implementation of the RICT Program. They are described below:

Level 1 Training

This is the most detailed training. It is intended for the individuals who will be directly involved in the implementation of the RICT Program. This level of training includes the following attributes:

- Specific training on the revised TS
- Record keeping requirements
- Case studies
- Hands-on experience with the CRMP tool for calculating RMAT and RICT
- Identifying appropriate RMAs
- Determining PRA functionality

- Common cause failure considerations
- Other detailed aspects of the RICT Program

Level 2 Training

This training is applicable to supervisors, managers, and other personnel who need a broad understanding of the RICT Program. It is significantly more detailed than level 3 training (described below), but it is different from level 1 training in that hands-on time with the CRMP tool and case studies are not included. The concepts of the RICT Program will be taught, but this group of personnel will not be qualified to perform the tasks for actual implementation of the RICT Program.

Level 3 Training

This training is intended for the remaining personnel who require an awareness of the RICT Program. These employees need basic knowledge of RICT Program requirements and procedures. This training will cover RICT Program concepts that are important to disseminate throughout the organization.

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Enclosure 11

MONITORING PROGRAM

Introduction

This enclosure provides a description of the process applied to monitor the cumulative risk impact of implementation of the Risk-Informed Completion Time (RICT) Program, specifically the calculation of cumulative risk of extended Completion Times (CTs). Calculation of the cumulative risk for the RICT Program is discussed in Step 14 of Section 2.3.1 and Step 7.1 of Section 2.3.2 of NEI 06-09, *Risk Informed Technical Specifications Initiative 4b*. General requirements for a Performance Monitoring Program for risk-informed applications are discussed in Regulatory Guide (RG) 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis*, Element 3.

Description of Monitoring Program

The RICT Program will require calculation of cumulative risk impact at least every refueling cycle, not to exceed 24 months, consistent with the guidance in NEI 06-09, Revision 0. For the assessment period under evaluation, data will be collected for the risk increase associated with each application of an extended CT for both core damage frequency (CDF) and large early release frequency (LERF), and the total risk will be calculated by summing all risk associated with each RICT application. This summation is the change in CDF or LERF above the zero maintenance baseline levels during the period of operation in the extended CT (i.e., beyond the front-stop CT). The change in risk will be converted to average annual values.

The total average annual change in risk for extended CTs will be compared to the guidance of RG 1.174, Figures 4 and 5, acceptance guidelines for CDF and LERF, respectively. If the actual annual risk increase is acceptable (i.e., not in Region I of Figures 4 and 5 of RG 1.174), then RICT Program implementation is acceptable for the assessment period. Otherwise, further assessment of the cause of exceeding the acceptance guidelines of RG 1.174 and implementation of any necessary corrective actions to ensure future plant operation is within the guidelines will be conducted under the corrective action program.

The evaluation of cumulative risk will also identify areas for consideration, such as:

- RICT applications that dominated the risk increase
- Risk contributions from planned vs. emergent RICT applications
- Risk management actions (RMA) implemented but not credited in the risk calculations
- Risk impact from applying RICT to avoid multiple shorter duration outages
- Any specific RICT application that incurred a large proportion of the risk

Based on a review of the considerations above, corrective actions will be developed and implemented as appropriate. These actions may include:

- Administrative restrictions on the use of RICTs for specific high-risk configurations
- Additional RMAs for specific configurations
- Rescheduling planned maintenance activities
- Deferring planned maintenance to shutdown conditions
- Use of temporary equipment to replace out-of-service systems, structures or components (SSC)
- Plant modifications to reduce risk impact of future planned maintenance configurations

In addition to impacting cumulative risk, implementation of the RICT Program may potentially impact the unavailability of SSCs. The existing Maintenance Rule (MR) monitoring programs under 10 CFR 50.65(a)(1) and (a)(2) provide for evaluation and disposition of unavailability impacts which may be incurred from implementation of the RICT Program. The SSCs in the scope of the RICT Program are also in the scope of the MR, which allows the use of the MR Program. RG 1.177, *An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications,* Section 3.2, Maintenance Rule Control, discusses that the scope of evaluations required under the Maintenance Rule should include prior related TS changes, such as extension of CTs.

The monitoring program for the MR, along with the specific assessment of cumulative risk impact described above, serve as the Implementation and Monitoring Program for the RICT Program as described in Element 3 of RG 1.174 and NEI 06-09.

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Enclosure 12

RISK MANAGEMENT ACTION EXAMPLES

Introduction

This attachment describes the process for identification and implementation of Risk Management Actions (RMA) applicable during extended Completion Times (CT) and provides examples of RMAs. RMAs will be governed by plant procedures for planning and scheduling maintenance activities. The procedures will provide guidance for the determination and implementation of RMAs when entering the Risk-Informed Completion Time (RICT) Program consistent with the guidance provided in NEI 06-09, Revision 0.

Responsibilities

For planned entries into the RICT Program, the department responsible for performing the maintenance or other activity is responsible for developing the RMAs with assistance from Operations and Reliability and Risk Assessment Group (PRAG). Operations is responsible for approval and implementation of RMAs. For emergent entry into extended CTs, Operations is also responsible for developing the RMAs.

Procedural Guidance

For planned maintenance activities, implementation of RMAs will be required if it is anticipated that the risk management action time (RMAT) will be exceeded. The RMAs will be implemented at the earliest possible time, without waiting for the actual RMAT to be exceeded. For emergent activities, RMAs must be implemented if the RMAT is reached. Also, if an emergent event occurs requiring recalculation of a RMAT already in place, the procedure will require a re-evaluation of the existing RMAs for the new plant configuration to determine if new RMAs are appropriate. These requirements of the RICT Program are consistent with the guidance of NEI 06-09.

RMAs are implemented no later than the time at which an incremental core damage probability (ICDP) of 1E-6 is reached, or no later than the time when an incremental large early release probability (ILERP) of 1E-7 is reached. If, as the result of an emergent condition, the instantaneous core damage frequency (ICDF) or the instantaneous large early release frequency (ILERF) exceeds 1E-3 per year or 1E-4 per year, respectively, RMAs are also required to be implemented. These requirements are consistent with the guidelines of NEI 06-09.

By determining which structures, systems, or components (SSCs) are most important from a CDF or LERF perspective for a specific plant configuration, RMAs may be created to protect these SSCs. Similarly, knowledge of the initiating event or sequence contribution to the configuration-specific CDF or LERF allows development of RMAs that enhance the capability to mitigate such events. If the planned activity or emergent condition includes a SSC that is identified to impact Fire PRA, as identified in the current Configuration Risk Management

Program (CRMP), Fire PRA specific RMAs (10 CFR 50.65 (a)(4) Fire) associated with that SSC shall be implemented per the current plant procedure.

It is possible to credit RMAs in RICT calculations; however, such quantification of RMAs is neither required nor expected by NEI 06-09. Nonetheless, if RMAs will be credited to determine realistic RICTs, the procedure instructions will be consistent with the guidance in NEI 06-09.

NEI 06-09 classifies RMAs into the three categories described below:

1) Actions to increase awareness and control.

- Shift brief
- Pre-job brief
- Training
- Presence of system engineer or other expertise related to the activity
- Special purpose procedure to identify risk sources and contingency plans

2) Actions to reduce the duration of maintenance activities.

- Pre-staging materials
- Conducting training on mock-ups
- Performing the activity around the clock
- Performing walk-downs on the actual system(s) to be worked on prior to beginning work

3) Actions to minimize the magnitude of the risk increase.

- Suspend or minimize activities on redundant systems
- Suspend or minimize activities on other systems that adversely affect the CDF or LERF
- Suspend or minimize activities on systems that may cause a trip or transient to minimize the likelihood of an initiating event that the out-of-service component is meant to mitigate
- Use temporary equipment to provide backup power, ventilation, etc.
- Reschedule other risk significant activities

<u>Examples</u>

Example RMAs that may be considered during a RICT Program entry for a diesel generator (DG) or a battery to reduce the risk impact and ensure adequate defense-in-depth are:

A. Diesel Generator:

(1) Evaluate the condition of the offsite power supply, switchyard, and the grid prior to entering a RICT, and implement the RMAs below during times of high grid stress conditions, such as during high demand conditions.

(2) Defer switchyard activities, such as of discretionary maintenance on the main, auxiliary, or startup transformers associated with the unit.

(3) Defer maintenance that affects the reliability of the trains associated with the operable DGs.

(4) Defer planned maintenance activities on station blackout mitigating systems, and treat those systems as protected equipment.

(5) Contact the dispatcher on a periodic basis to provide information on DG status and the power needs of the facility, and to obtain grid status.

(6) Implement 10 CFR 50.65(a)(4) fire-specific RMAs associated with the impacted DG.

B. Battery:

(1) Limit the immediate discharge of the affected battery, if possible.

(2) Recharge the affected battery to float voltage conditions using a spare battery charger, if possible.

(3) Evaluate the capacity of the remaining battery and protect its ability to perform its safety function.

(4) Periodically verify battery float voltage is equal to or greater than the minimum required float voltage for remaining batteries.