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10 CFR 50.90

RS-20-009

January 31, 2020

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

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Application to Revise LaSalle County Station Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC), requests an amendment to Renewed Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2.

The proposed amendment would modify Technical Specifications (TS) requirements to permit the use of Risk Informed Completion Times in accordance with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b," (ADAMS Accession No. ML18183A493). A model safety evaluation was provided by the NRC to the TSTF on November 21, 2018 (ADAMS Accession No. ML18267A259).

- Attachment 1 provides a description and assessment of the proposed changes, the requested confirmation of applicability, and plant-specific verifications.
- Attachment 2 provides the existing TS pages marked up to show the proposed changes.
- Attachment 3 provides the existing TS Bases pages marked up to show the proposed changes and is provided for information only.
- Attachment 4 provides a cross-reference between the Technical Specifications included in TSTF-505, Rev. 2 and the LSCS plant-specific TS.
- Attachment 5 provides a PRA implementation item that must be completed prior to implementing the Risk Informed Completion Time (RICT) Program.
- Attachment 6 provides proposed License Conditions that require completion of the item listed in Attachment 5 (see above) prior to implementation of the RICT Program.

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- Attachment 7 provides information supporting the redundant means available to mitigate accidents for instrumentation governed by the TS proposed to be included as part of the RICT program in this submittal.

EGC proposes an amendment to Renewed Facility Operating License Nos. NPF-11 and NPF-18 for LSCS, Units 1 and 2. The proposed amendment would modify the licensing basis by the addition of a license condition for each unit to address the item described in Attachment 5 of this letter prior to implementation of the RICT Program. The proposed License Conditions are specified in Attachment 6.

These proposed changes have been reviewed and approved by the LSCS Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed amendment by January 31, 2021. The amendment shall be implemented within 180 days following NRC approval, or following completion of the License Conditions specified in Attachment 6, on a per unit basis, whichever is later.

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

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

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this letter, please contact Ryan Sprengel at (630) 657-2814.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 31st day of January 2020.

Respectfully,

Dwi Murray
Sr. Manager - Licensing
Exelon Generation Company, LLC

Attachments:

1. Description and Assessment
2. Proposed Technical Specification Changes – Mark-Ups
3. Proposed Technical Specification Bases Changes – Mark-Ups (For Information Only)
4. Cross-Reference of TSTF-505 and LaSalle County Station Technical Specifications
5. LaSalle County Station RICT Program PRA Implementation Item
6. Proposed Renewed Facility Operating License (RFOL) Changes – Mark-Ups
7. Evaluation of Instrumentation and Control Systems

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 Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)
6. Justification of Application of At-Power PRA Models to Shutdown Modes
7. PRA Model Update Process
8. Attributes of the Real Time Risk Model
9. Key Assumptions and Sources of Uncertainty
10. Program Implementation
11. Monitoring Program
12. Risk Management Action Examples

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – LaSalle County Station
Illinois Emergency Management Agency – Division of Nuclear Safety
NRR Project Manager, LaSalle County Station

ATTACHMENT 1

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

Description and Assessment

ATTACHMENT 1

DESCRIPTION AND ASSESSMENT

1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC), requests an amendment to Renewed Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2.

The proposed amendment would modify the Technical Specifications (TS) requirements related to Completion Times (CTs) for Required Actions (Required Action allowed outage times for LSCS) to provide the option to calculate a longer, risk-informed CT. A new program, the Risk-Informed Completion Time (RICT) Program, is added to TS Section 5.0, "Administrative Controls."

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

The proposed amendment is consistent with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b." However, only those Required Actions described in Attachment 4 and Enclosure 1, as reflected in the proposed TS mark-ups provided in Attachment 2, are proposed to be changed, because some of the modified Required Actions in TSTF-505 are not applicable to LSCS, and there are some plant-specific Required Actions not included in TSTF-505 that are included in this proposed amendment.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

EGC has reviewed TSTF-505, Revision 2, and the model safety evaluation dated November 21, 2018 (ADAMS Accession No. ML18267A259). This review included the information provided to support TSTF-505 and the safety evaluation for NEI 06-09-A. As described in the subsequent paragraphs, EGC has concluded that the technical basis is applicable to LSCS, Units 1 and 2, and support incorporation of this amendment in the LSCS TS.

2.2 Verifications and Regulatory Commitments

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

1. Enclosure 1 identifies each of the TS Required Actions to which the RICT Program will apply, with a comparison of the TS functions to the functions modeled in the probabilistic risk assessment (PRA) of the structures, systems and components (SSCs) subject to those actions.

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

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

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

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

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

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

8. Enclosure 8 provides a description of how the baseline PRA model, which calculates average annual risk, is evaluated and modified for use in the Real Time Risk tool to assess real time configuration risk, and describes the scope of, and quality controls applied to the Real Time Risk tool.

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

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

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

EGC is proposing the following variations from the TS changes described in TSTF-505, Revision 2, or the applicable parts of the NRC's model safety evaluation dated November 21, 2018. These options were recognized as acceptable variations in TSTF-505 and the NRC's model safety evaluation.

In a few instances, the LSCS TS use different numbering and titles than the Standard Technical Specifications (STS) on which TSTF-505 was based. These differences are administrative and do not affect the applicability of TSTF-505 to the LSCS TS. Attachment 4 provides specific information.

Attachment 4 is a cross-reference that provides a comparison between the NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," Required Actions included in TSTF-505 and the LSCS Required Actions included in this license amendment request. LSCS Units 1 and 2 are a BWR/5 design, LSCS TS Required Actions that align to NUREG-1434, "Standard Technical Specifications, General Electric Plants, BWR/6," Required Actions are identified in Attachment 4. The attachment includes a summary description of the referenced Required Actions, which is provided for information purposes only and is not intended to be a verbatim description of the Required Actions. The cross-reference in Attachment 4 identifies the following:

1. LSCS Required Actions that have identical numbers to the corresponding NUREG-1433/NUREG-1434 Required Actions are not deviations from TSTF-505, except for administrative deviations (if any) such as formatting. These deviations are administrative with no impact on the NRC's model safety evaluation dated November 21, 2018.

2. LSCS Required Actions that have different numbering than the NUREG-1433/ NUREG-1434 Required Actions are an administrative deviation from TSTF-505 with no impact on the NRC's model safety evaluation dated November 21, 2018.

3. For NUREG-1433/NUREG-1434 Required Actions that are not contained in the LSCS TS, the corresponding TSTF-505 mark-ups for the Required Actions are not applicable to LSCS. This is an administrative deviation from TSTF-505 with no impact on the NRC's model safety evaluation dated November 21, 2018.

4. The model application provided in TSTF-505 includes an attachment for typed, camera-ready (revised) TS pages reflecting the proposed changes. LSCS is not including such an attachment due to the number of TS pages included in this submittal that have the potential to be affected by other unrelated license amendment requests and the straightforward nature of the proposed changes. Providing only mark-ups of the proposed TS changes satisfies the requirements of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," in that the mark-ups fully describe the changes desired. This is an administrative deviation from TSTF-505 with no impact on the NRC's model safety evaluation dated November 21, 2018. Because of this deviation, the contents and numbering of the attachments for this amendment request differ from the attachments specified in the model application in TSTF-505.

5. The model application provided in TSTF-505 includes mark-ups to Completion Times for NUREG-1433/NUREG-1434 in a format using an "OR" Logical Connector followed by "In accordance with the Risk Informed Completion Time Program." Several existing Required Actions have two Completion Times connected by the Logical Connector "AND" in the current LSCS TS. LSCS TS Section 1.2, "Logical Connectors," specifies that Completion Times only use first level logic. Therefore, the proposed markups have been modified for these Required Actions to embed "or in accordance with the Risk Informed Completion Time Program" into the existing Completion Times. This follows LSCS TS Section 1.2 and does not create a second level logic for the Completion Times. This is an administrative deviation from TSTF-505 with no impact on the NRC's model safety evaluation dated November 21, 2018.

6. There are several plant-specific LCOs and associated Required Actions for which LSCS is proposing to apply the RICT Program that are variations from TSTF-505 as identified in Attachment 4. Attachment 4 was created using the BWR/4 standard from NUREG-1433, with exceptions annotated on Attachment 4 and summarized below, including annotations for use of the BWR/6 standard from NUREG-1434. Additional details are contained in Attachment 4 for TS Conditions and Required Actions.

   - TS 3.3.4.2 – Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT) Instrumentation. The LSCS Instrumentation is closest to the BWR/6 standard from NUREG-1434.
   - TS 3.3.5.1 – Emergency Core Cooling System (ECCS) Instrumentation. The LSCS Instrumentation is closest to the BWR/6 standard from NUREG-1434. LSCS TS 3.3.5.1.D is a plant-specific condition with a restoration action and allowed outage time of 24 hours. This submittal does not propose a change to the instrumentation tables.
   - TS 3.5.1 – ECCS-Operating. The LSCS TS 3.5.1 is closest to the BWR/6 standard from NUREG-1434.
   - TS 3.5.3 – RCIC System. The LSCS TS 3.5.3 is closest to the BWR/6 standard from NUREG-1434.
   - TS 3.8.1 – AC Sources-Operating. LSCS TS 3.8.1.C is a plant-specific condition with a restoration action and allowed outage time of 72 hours.
   - TS 3.8.4 – DC Sources-Operating. LSCS TS 3.8.4.E is a plant-specific condition with a restoration action and allowed outage time of 7 days.
   - TS 3.8.7 – Distribution Systems-Operating. LSCS TS 3.8.7.D is a plant-specific condition with a restoration action and allowed outage time of 7 days.

   EGC has determined that the application of a RICT for these LSCS plant-specific LCOs is consistent with TSTF-505 and with the NRC's model safety evaluation dated November 21, 2018. Application of a RICT for plant-specific LCOs will be controlled under the RICT Program. The RICT Program provides the necessary administrative controls to permit extension of Completion Times and thereby delay reactor shutdown or remedial actions, if risk is assessed and managed within specified limits and programmatic requirements. The specified safety function or performance levels of TS required structures, systems or components (SSCs) are unchanged, and the remedial actions, including the requirement to shut down the reactor, are also unchanged; only the Required Action allowed outage times are extended by the RICT Program.

   Application of a RICT will be evaluated using the methodology and probabilistic risk guidelines contained in NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, which was approved by the NRC on May 17, 2007 (ADAMS Accession No. ML071200238). The NEI 06-09-A, Revision 0 methodology includes a requirement to perform a quantitative assessment of the potential impact of the application of a RICT on risk, to reassess risk due to plant configuration changes, and to implement compensatory measures and risk management actions (RMAs) to maintain the risk below acceptable regulatory risk thresholds. In addition, the NEI 06-09-A, Revision 0 methodology satisfies the five key safety principles specified in Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," dated August 1998 (ADAMS

Accession No. ML003740176), relative to the risk impact due to the application of a RICT.

Therefore, the proposed application of RICT in the LSCS plant-specific Required Actions is consistent with TSTF-505, Revision 2, and with the NRC's model safety evaluation dated November 21, 2018.

EGC has reviewed these changes and determined that they do not affect the applicability of TSTF-505, Revision 2, to the LSCS TS.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Determination

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

LaSalle County Station, Units 1 and 2, requests adoption of an approved change to the standard technical specifications (STS) 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 Chapter 5.0, "Administrative Controls," entitled the "Risk-Informed Completion Time Program."

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

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

   Response: No.

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

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

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

   Response: No.

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

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

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

Response: No.

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

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

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

3.2 Conclusions

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

4.0 ENVIRONMENTAL CONSIDERATION

EGC has reviewed the environmental evaluation included in the model safety evaluation published on November 21, 2018 (ADAMS Accession No. ML18267A259) as part of the Notice of Availabilty. EGC has concluded that the NRC staff findings presented in the evaluation are applicable to LaSalle County Station, Units 1 and 2.

The proposed changes would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed changes.

ATTACHMENT 2

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

<u>Proposed Technical Specification Changes – LaSalle County Station Mark-Ups</u>

1.3 Completion Times

EXAMPLES	EXAMPLE 1.3-7 (continued) is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.
INSERT 1	
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	In accordance with the Surveillance Frequency Control Program

(continued)

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

OR

In accordance with the Risk Informed Completion Time Program

ACTIONS

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

1. Separate Condition entry is allowed for each channel.

2. When Functions 2.b and 2.c channels are inoperable due to the APRM indication not within limits, entry into associated Conditions and Required Actions may be delayed for up to 2 hours if the APRM is indicating a lower power value than the calculated power, and for up to 12 hours if the APRM is indicating a higher power value than the calculated power.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. <u>OR</u> A.2 Place associated trip system in trip.	12 hours 12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. <u>OR</u> B.2 Place one trip system in trip.	6 hours 6 hours

OR

In accordance with the Risk Informed Completion Time Program

(continued)

3.3 INSTRUMENTATION

3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Four channels of feedwater system and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more feedwater system and main turbine high water level trip channels inoperable.	A.1 Place channel in trip.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Feedwater system and main turbine high water level trip capability not maintained.	B.1 Restore feedwater system and main turbine high water level trip capability.	2 hours

(continued)

3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

LCO 3.3.4.2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level—Low Low, Level 2; and

b. Reactor Steam Dome Pressure—High.

OR

In accordance with the Risk Informed Completion Time Program

APPLICABILITY: MODE 1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable.	A.1	Restore channel to OPERABLE status.	14 days
	OR		
	A.2	--------NOTE-------- Not applicable if inoperable channel is the result of an inoperable breaker. -------------------- Place channel in trip.	14 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1 --------NOTE-------- Only applicable for Functions 1.a, 1.b, 2.a and 2.b. --------------------- Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>	
	B.2 --------NOTE-------- Only applicable for Functions 3.a and 3.b. --------------------- Declare High Pressure Core Spray (HPCS) System inoperable.	1 hour from discovery of loss of HPCS initiation capability
	<u>AND</u>	
	B.3 Place channel in trip.	24 hours <u>OR</u> --------NOTE-------- Not applicable when a loss of function occurs. ----------------------- In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	--------NOTE-------- Only applicable for Functions 1.c and 2.c. --------------------- Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	AND		
	C.2	Restore channel to OPERABLE status.	24 hours OR --------NOTE-------- Not applicable when a loss of function occurs. ------------------------ In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (con	D.3	--------NOTE--------- Only applicable for Functions 1.g and 2.f. --------------------- Restore channel to OPERABLE status.	24 hours
	AND		
	D.4	Restore channel to OPERABLE status.	7 days
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1	Declare Automatic Depressurization System (ADS) valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	AND		
	E.2	Place channel in trip.	96 hours from discovery of inoperable channel concurrent with HPCS or reactor core isolation cooling (RCIC) inoperable AND 8 days

OR

--------NOTE--------
Not applicable when a loss of function occurs.

In accordance with the Risk Informed Completion Time Program

or in accordance with the Risk Informed Completion Time Program

--------NOTE--------
The Risk Informed Completion Time Program is not applicable when a loss of function occurs.

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	F.1	--------NOTE--------- Only applicable for Functions 4.c, 4.e, 4.f, 4.g, 5.c, 5.e, and 5.f. ---------------------	
		Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	<u>AND</u>		
	F.2	Restore channel to OPERABLE status. or in accordance with the Risk Informed Completion Time Program	96 hours from discovery of inoperable channel concurrent with HPCS or RCIC inoperable <u>AND</u> 8 days
G. Required Action and associated Completion Time of Condition B, C, D, E, or F not met.	G.1	Declare associated supported feature(s) inoperable.	Immediately

--------NOTE--------
The Risk Informed Completion Time Program is not applicable when a loss of function occurs.

3.3 INSTRUMENTATION

3.3.5.3 Reactor Core Isolation Cooling (RCIC) System Instrumentation

LCO 3.3.5.3 The RCIC System instrumentation for each Function in Table 3.3.5.3-1 shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.3-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	B.1 Declare RCIC System inoperable.	1 hour from discovery of loss of RCIC initiation capability
	AND	
	B.2 Place channel in trip.	24 hours OR --------NOTE-------- Not applicable when a loss of function occurs. ------------------------ In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	C.1	Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	D.1	--------NOTE-------- Only applicable if RCIC pump suction is not aligned to the suppression pool. -------------------- Declare RCIC System inoperable.	1 hour from discovery of loss of RCIC initiation capability
	AND		
	D.2.1	Place channel in trip.	24 hours OR --------NOTE-------- Not applicable when a loss of function occurs. ------------------------ In accordance with the Risk Informed Completion Time Program
	OR		
	D.2.2	Align RCIC pump suction to the suppression pool.	24 hours
E. Required Action and associated Completion Time of Condition B, C, or D not met.	E.1	Declare RCIC System inoperable.	Immediately

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

1. Separate Condition entry is allowed for each channel.

2. For Function 1.e, when automatic isolation capability is inoperable for required Reactor Building Ventilation System corrective maintenance, filter changes, damper cycling, or required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 4 hours.

3. For Function 1.e, when automatic isolation capability is inoperable due to loss of reactor building ventilation or for performance of SR 3.6.4.1.3 or SR 3.6.4.1.4, entry into associated Conditions and Required Action may be delayed for up to 12 hours.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place channel in trip. or in accordance with the Risk Informed Completion Time Program	12 hours for Functions 2.b, 2.f, and 5.a AND 24 hours for Functions other than Functions 2.b, 2.f, and 5.a

(continued)

3.3 INSTRUMENTATION

3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When the associated diesel generator (DG) is required to be OPERABLE by LCO 3.8.2, "AC Sources—Shutdown."

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place channel in trip.	1 hour OR --------NOTE-------- Not applicable when a loss of function occurs. ----------------------- In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Declare associated DG inoperable.	Immediately

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS—Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC is required to be OPERABLE. AND B.2 Restore HPCS System to OPERABLE status.	Immediately 14 days OR In accordance with the Risk Informed Completion Time Program
C. Two low pressure ECCS injection/spray subsystems inoperable.	C.1 Restore one low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	12 hours
E. One required ADS valve inoperable.	E.1 Restore required ADS valve to OPERABLE status.	14 days OR In accordance with the Risk Informed Completion Time Program
F. Required Action and associated Completion Time of Condition E not met.	F.1 Be in MODE 3.	12 hours

(continued)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE. AND A.2 Restore RCIC System to OPERABLE status.	Immediately 14 days OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.3 Restore air lock to OPERABLE status.	24 hours OR In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.2.1	-------------------NOTES------------------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.1. -- Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.2.2	Verify only one door in the primary containment air lock can be opened at a time.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

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

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------NOTE--------- Only applicable to penetration flow paths with two or more PCIVs. --------------------- One or more penetration flow paths with one PCIV inoperable for reasons other than Condition D.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours or in accordance with the Risk Informed Completion Time Program, except for main steam line AND 8 hours or in accordance with the Risk Informed Completion Time Program for main steam line
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 --------NOTES-------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	Once per 31 days following isolation for isolation devices outside primary containment <u>AND</u> Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 -------NOTES-------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. --------------------- Verify the affected penetration flow path is isolated.	Once per 31 days following isolation
D. One or more penetration flow paths with MSIV leakage rate or hydrostatically tested line leakage rate not within limit.	D.1 Restore leakage rate to within limit.	4 hours for hydrostatically tested line leakage not on a closed system <u>AND</u> 8 hours for MSIV leakage <u>AND</u> 72 hours for hydrostatically tested line leakage on a closed system

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.1.6 Suppression Chamber-to-Drywell Vacuum Breakers

LCO 3.6.1.6 Each suppression chamber-to-drywell vacuum breaker shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1 Restore the vacuum breaker to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
C. One suppression chamber-to-drywell vacuum breaker not closed.	C.1 Close both manual isolation valves in the affected line. AND C.2 Restore the vacuum breaker to OPERABLE status.	4 hours 72 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1	Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time of Condition A not met.	B.	Be in MODE 3.	12 hours
C. Two RHR suppression pool cooling subsystems inoperable.	C.1	Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 AND D.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

LCO 3.6.2.4 Two RHR suppression pool spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool spray subsystem inoperable.	A.1 Restore RHR suppression pool spray subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Two RHR suppression pool spray subsystems inoperable.	B.1 Restore one RHR suppression pool spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System

LCO 3.7.1 Two RHRSW subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHRSW subsystem inoperable.	A.1	--------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown," for RHR shutdown cooling subsystem made inoperable by RHRSW System. --------------------- Restore RHRSW subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
	A.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one division concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	A.3	Restore required offsite circuit to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One required Division 1, or 2 DG inoperable. <u>OR</u> Required opposite unit Division 2 DG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
	B.2	Declare required feature(s), supported by the inoperable DG(s), inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	B.3.1	Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	<u>OR</u>		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	<u>AND</u>		
	B.4	Restore required DG(s) to OPERABLE status.	14 days <u>OR</u> In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. --------NOTE--------- 1. Not applicable to the Division 2 DG and the opposite unit Division 2 DG during installation of Division 2 CSCS isolation valves during a single Unit 1 Refueling Outage completed prior to July 1, 2024, and during a single Unit 2 Refueling Outage completed prior to July 1, 2023, while the outage unit is in MODE 4,5, or defueled. --------------------- Required Division 3 DG inoperable. <u>OR</u> One required Division 1, 2, or 3 DG inoperable and the required opposite unit Division 2 DG inoperable.	C.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>	
	C.2 Declare required feature(s), supported by the inoperable DG(s), inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	<u>AND</u>	
	C.3.1 Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	<u>OR</u>	
	C.3.2 Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	<u>AND</u>	
	C.4 Restore required DG(s) to OPERABLE status.	72 hours <u>OR</u> --------NOTE-------- Not applicable when a loss of function occurs. ------------------------ In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two required offsite circuits inoperable.	D.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable. AND D.2 Restore one required offsite circuit to OPERABLE status.	12 hours from discovery of Condition D concurrent with inoperability of redundant required feature(s) 24 hours
E. One required offsite circuit inoperable. AND One required Division 1, 2, or 3 DG inoperable.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems—Operating," when Condition E is entered with no AC power source to any required division. ---------------------------- E.1 Restore required offsite circuit to OPERABLE status. OR E.2 Restore required DG to OPERABLE status.	OR In accordance with the Risk Informed Completion Time Program 12 hours 12 hours

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources—Operating

LCO 3.8.4 The Division 1 125 VDC and 250 VDC, Division 2 125 VDC, Division 3 125 VDC, and the opposite unit Division 2 125 VDC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required Division 1, 2, or 3 125 VDC battery charger on one division inoperable. <u>OR</u> One required Division 2 or opposite unit Division 2 battery charger on one division inoperable. <u>OR</u> One required Division 1 250 VDC battery charger inoperable.	A.1	Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	2 hours
		<u>AND</u>	
	A.2	Verify battery float current ≤ 2 amps.	Once per 12 hours
		<u>AND</u>	<u>OR</u> In accordance with the Risk Informed Completion Time Program
	A.3	Restore required battery charger(s) to OPERABLE status.	7 days
B. Division 1 or 2 125 VDC electrical power subsystem inoperable for reasons other than Condition A.	B.1	Restore Division 1 and 2 125 VDC electrical power subsystems to OPERABLE status.	2 hours

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met for the Division 3 DC electrical power subsystem. OR Division 3 DC electrical power subsystem inoperable for reasons other than Condition A.	C.1	Declare High Pressure Core Spray System inoperable.	Immediately
D. Required Action and associated Completion Time of Condition A not met for the Division 1 250 VDC electrical power subsystem. OR Division 1 250 VDC electrical power subsystem inoperable for reasons other than Condition A.	D.1	Declare associated supported features inoperable.	Immediately
E. Required Action and associated Completion Time of Condition A not met for the opposite unit Division 2 DC electrical power subsystem. OR Opposite unit Division 2 DC electrical power subsystem inoperable for reasons other than Condition A.	E.1	Restore opposite unit Division 2 DC electrical power subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Distribution Systems—Operating

LCO 3.8.7 The following electrical power distribution subsystems shall be OPERABLE:

a. Division 1 and Division 2 AC and 125 V DC distribution subsystems;

b. Division 3 AC and 125 V DC distribution subsystems;

c. Division 1 250 V DC distribution subsystem; and

d. The portions of the opposite unit's Division 2 AC and 125 V DC electrical power distribution subsystems capable of supporting the equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.7.4, "Control Room Area Filtration (CRAF) System," LCO 3.7.5, "Control Room Area Ventilation Air Conditioning (AC) System," and LCO 3.8.1, "AC Sources-Operating."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or both Division 1 and 2 AC electrical power distribution subsystems inoperable.	A.1	Restore Division 1 and 2 AC electrical power distribution subsystems to OPERABLE status.	8 hours OR --------NOTE-------- Not applicable when a loss of function occurs. ------------------------ In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or both Division 1 and 2 125 V DC electrical power distribution subsystems inoperable.	B.1 Restore Division 1 and 2 125 V DC electrical power distribution subsystem(s) to OPERABLE status.	2 hours OR --------NOTE-------- Not applicable when a loss of function occurs. ------------------------ In accordance with the Risk Informed Completion Time Program
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	12 hours
D. One or more required opposite unit Division 2 AC or DC electrical power distribution subsystems inoperable.	-------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.1 when Condition C results in the inoperability of a required offsite circuit. ---------------------------- D.1 Restore required opposite unit Division 2 AC and DC electrical power distribution subsystem(s).	 7 days OR In accordance with the Risk Informed Completion Time Program
E. Required Action and associated Completion Time of Condition D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours

(continued)

5.5 Programs and Manuals

5.5.16 Surveillance Frequency Control Program (continued)

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

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

INSERT 2

INSERT 1 (EXAMPLE 1.3-8)

EXAMPLE 1.3-8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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

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

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

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

INSERT 2 (New TS 5.5.17)

5.5.17 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-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

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

c. When a RICT is being used, any change to the plant configuration, as defined in NEI 06-09-A, Appendix A, must be considered for the effect on the RICT.

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

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

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

d. For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

   1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

   2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

ATTACHMENT 3

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

Proposed Technical Specification Bases Changes – LaSalle County Station Mark-Ups

(For Information Only)

APPLICABILITY (continued)

ensure the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE during these conditions, when only a single control rod can be withdrawn.

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CFR 50.67 (Ref. 4) limits following a LOCA involving significant fission product releases. The SLC System is designed to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water (Ref. 3).

or in accordance with the Risk Informed Completion Time Program

ACTIONS

A.1

If one SLC System subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the shutdown function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability and inability to meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the unit shutdown function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive System to shut down the reactor.

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable, given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

(continued)

BASES

ACTIONS (continued)

(i.e., the GAF is low (conservative)). The GAF for any channel is defined as the power value determined by the heat balance divided by the APRM reading for that channel. Upon completion of the gain adjustment, or expiration of the allowed time, the channel must be returned to OPERABLE status or the applicable Condition entered and the Required Actions taken. This Note is based on the time required to perform gain adjustments on multiple channels and additional time is allowed when the GAF is out of limits but conservative.

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

<u>A.1 and A.2</u>

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 10) to permit restoration of any inoperable required channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases.) If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a scram or recirculation pump trip (RPT)), Condition D must be entered and its Required Action taken.

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

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Required Actions B.1 and B.2 limit the time the RPS scram logic for any Function would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in Reference 10 for the 12 hour Completion Time. Within the 6 hour allowance, the associated Function will have all required channels either OPERABLE or in trip (or in any combination) in one trip system.

Completing one of these Required Actions restores RPS to an equivalent reliability level as that evaluated in Reference 10, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels, if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision as to which trip system is in the more degraded state should be based on prudent judgment and current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or RPT, it is permissible to place the other trip system or its inoperable channels in trip.

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

(continued)

BASES

ACTIONS (continued)

of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable Feedwater System and Main Turbine High Water Level Trip Instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable Feedwater System and Main Turbine High Water Level Trip Instrumentation channel.

A.1

With one or more channels inoperable and trip capability maintained, the remaining OPERABLE channels can provide the required trip signal. However, overall instrumentation reliability is reduced because a single failure in one of the remaining channels concurrent with feedwater controller failure, maximum demand event, or a variable leg failure may result in the instrumentation not being able to perform its intended function. Therefore, continued operation is only allowed for a limited time. If the inoperable channel cannot be restored to OPERABLE status within the Completion Time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in a feedwater turbine, motor-driven feedwater pump, or main turbine trip), Condition C must be entered and its Required Action taken.

The Completion Time of 7 days is based on the low probability of the event occurring coincident with a single failure in a remaining OPERABLE channel. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

B.1

With the feedwater system and main turbine high water level trip capability not maintained, the feedwater system and main turbine high water level trip instrumentation cannot perform its design function. Therefore, continued operation

(continued)

ACTIONS

A Note has been provided to modify the ACTIONS related to ATWS-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ATWS-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable ATWS-RPT instrumentation channel.

A.1 and A.2

With one or more channels inoperable, but with ATWS-RPT trip capability for each Function maintained (refer to Required Action B.1 and C.1 Bases), the ATWS-RPT System is capable of performing the intended function. However, the reliability and redundancy of the ATWS-RPT instrumentation is reduced, such that a single failure in the remaining trip system could result in the inability of the ATWS-RPT System to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of ATWS-RPT, 14 days is provided to restore the inoperable channel (Required Action A.1). Alternately, the inoperable channel may be placed in trip (Required Action A.2), since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not

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

(continued)

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.3. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition G must be entered and its Required Action taken.

C.1 and C.2

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the same Function (or in some cases, within the same variable) result in redundant automatic initiation capability being lost for the feature(s). Loss of redundant automatic initiation capability for the low pressure ECCS injection feature in both divisions occurs when the initiation capability is available to less than two pumps from any single variable.

Required Action C.1 features would be those that are initiated by Functions 1.c, and 2.c (i.e., low pressure ECCS). For Functions 1.c and 2.c, redundant automatic initiation capability is lost if the Function 1.c and Function 2.c channels are inoperable. Since each inoperable channel would have Required Action C.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated Division to be declared inoperable. However, since channels in both Divisions are inoperable, and the Completion Times started concurrently for the channels in both Divisions, this results in the affected portions in both Divisions being concurrently declared inoperable. For Functions 1.c

(continued)

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function when trip capability is not maintained.

ACTIONS

C.1 and C.2 (continued)

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition G must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would either cause the initiation or would not necessarily result in a safe state for the channel in all events.

D.1, D.2, D.3, and D.4

Required Action D.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, channels within the LPCS and LPCI Pump Discharge Flow—Low (Bypass) Functions, the Injection Line Pressure—Low (Injection Permissive), and the Reactor Steam Dome Pressure—Low (Injection Permissive) Functions result in redundant automatic initiation capability being lost for the feature(s). Loss of redundant automatic initiation capability for the low pressure ECCS injection feature in both divisions occurs when the initiation capability is available to less than two pumps from any single variable. For the purposes of this Condition, the injection permissives on Reactor Steam Dome Pressure—Low and Injection Line Pressure—Low are considered the same variable. Similarly, Functions 1.e, 1.f, and 2.e are all minimum flow functions and considered the same variable.

For Required Action D.1, the features would be those that are initiated by Functions 1.d, 1.e, 1.f, 1.g, 2.d, 2.e, and 2.f (e.g., low pressure ECCS). Redundant automatic initiation capability is lost if three of the four channels associated with Functions 1.e, 1.f, and 2.e are inoperable. For Function 1.d, redundant automatic initiation capability is lost if two Function 1.d channels are inoperable concurrent with either two inoperable Function 2.d channels or one inoperable Function 2.f channel. For Function 2.d, redundant automatic initiation capability is lost if two Function 2.d channels are inoperable concurrent with two

(continued)

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function when trip capability is not maintained.

BASES

ACTIONS

D.1, D.2, D.3, and D.4 (continued)

actions are taken if multiple, inoperable channels within the Reactor Steam Dome Pressure—Low (Injection Permissive) Function result in automatic initiation capability being lost for the features in one division. For Required Action D.2, the features would be those that are initiated by Functions 1.d and 2.d (e.g., low pressure ECCS). For Functions 1.d and 2.d, automatic initiation capability is lost in one division if two Function 1.d or two Function 2.d channels are inoperable. In this situation, (loss of automatic initiation capability), the 7 day allowance of Required Action D.4 is not appropriate and the features associated with the inoperable channels must be declared inoperable within 24 hours after discovery of loss of initiation capability for features in one division. For Functions 1.g and 2.f, an allowable out of service time of 24 hours is provided by Required Action D.3.

The Completion Time is intended to allow 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." For Required Action D.1, the Completion Time only begins upon discovery that three channels of the Pump Discharge Flow—Low (Bypass) Function cannot be automatically initiated due to inoperable channels or upon discovery of a loss of redundant initiation capability for the Reactor Steam Dome Pressure—Low (Injection Permissive) and Injection Line Pressure—Low (Injection Permissive) Functions (as described above). The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels. For Required Action D.2, the Completion Time only begins upon discovery that two Function 1.d or two Function 2.d channels cannot be automatically initiated due to inoperable channels. The 24 hour Completion Time from discovery of loss of initiation capability for features in one division is acceptable because of the redundancy of the ECCS design, as shown in the reliability analysis of Reference 4.

If the instrumentation that controls the pump minimum flow valve is inoperable such that the valve will not automatically open, extended pump operation with no injection path available could lead to pump overheating and

(continued)

BASES

ACTIONS

D.1, D.2, D.3, and D.4 (continued)

failure. If there were a failure of the instrumentation such that the valve would not automatically close, a portion of the pump flow could be diverted from the reactor injection path, causing insufficient core cooling. These consequences can be averted by the operator's manual control of the valve, which would be adequate to maintain ECCS pump protection and required flow. Furthermore, other ECCS pumps would be sufficient to complete the assumed safety function if no additional single failure were to occur. If a Reactor Vessel Pressure—Low (Injection Permissive) Function channel is inoperable, another channel exists to ensure the injection valves in the ECCS division can still open. The 7 day Completion Time of Required Action D.4 to restore the inoperable channel to OPERABLE status is reasonable based on the remaining capability of the associated ECCS subsystems, the redundancy available in the ECCS design, and the low probability of a DBA occurring during the allowed out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition G must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

E.1 and E.2

Required Act
actions are
channels wit
automatic in
Automatic initiation capability is lost if either (a) one or more Function 4.a channels and one or more Function 5.a channels are inoperable and untripped, (b) one or more Function 4.b channels and one or more Function 5.b channels are inoperable and untripped, or (c) one Function 4.d channel and one Function 5.d channel are inoperable and untripped.

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action E.2 is not appropriate, and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability in both trip systems.

(continued)

BASES

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

ACTIONS

E.1 and E.2

The Completion Time is intended to allow 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." For Required Action E.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable, untripped channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status if both HPCS and RCIC are OPERABLE. If either HPCS or RCIC is inoperable, the time is shortened to 96 hours. If the status of HPCS or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCS or RCIC inoperability. However, total time for an inoperable, untripped channel cannot exceed 8 days. If the status of HPCS or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable, untripped channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action E.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition G must be entered and its Required Action taken.

or in accordance with the Risk Informed Completion Time Program

(continued)

or in accordance with the Risk Informed Completion Time Program

or in accordance with the Risk Informed Completion Time Program

or in accordance with the Risk Informed Completion Time Program

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ACTIONS

F.1 and F.2 (continued)

channel to OPERABLE status if both HPCS and RCIC are OPERABLE (Required Action F.2). If either HPCS or RCIC is inoperable, the time is reduced to 96 hours. If the status of HPCS or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCS or RCIC inoperability. However, total time for an inoperable channel cannot exceed 8 days. If the status of HPCS or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition G must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

...ociated Completion Time not ...ay be incapable of ... and the supported feature(s) associated with the inoperable untripped channels must be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each ECCS instrumentation Function are found in the SRs column of Table 3.3.5.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours as follows: (a) for Functions 3.c, 3.d, 3.e, and 3.f; and (b) for Functions other than 3.c, 3.d, 3.e, and 3.f provided the associated Function or redundant Function maintains ECCS initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic initiation capability for the RCIC System. In this case, automatic initiation capability is lost if two Function 1 parallel contacts (channels) in the same trip system are inoperable and untripped. In this situation (loss of automatic initiation capability), the 24 hour allowance of Required Action B.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour after discovery of loss of RCIC initiation capability.

The Completion Time is intended to allow 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." For Required Action B.1, the Completion Time only begins upon discovery that the RCIC System cannot be automatically initiated due to two inoperable, untripped Reactor Vessel Water Level—Low Low, Level 2 channels (parallel contacts) in the same trip system. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not credited in any accident or transient analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition E must be entered and its Required Action taken.

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

(continued)

BASES

ACTIONS

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

The Completion Time is intended to allow 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." For Required Action D.1, the Completion Time only begins upon discovery that the RCIC System cannot be automatically aligned to the suppression pool due to two inoperable, untripped channels in the same Function. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

within 24 hours

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not assumed in any accident or transient analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2.1, which performs the intended function of the channel (shifting the suction source to the suppression pool). Alternatively, Required Action D.2.2 allows the manual alignment of the RCIC suction to the suppression pool, which also performs the intended function. If Required Action D.2.1 or D.2.2 is performed, measures should be taken to ensure that the RCIC System piping remains filled with water. If it is not desired to perform Required Actions D.2.1 and D.2.2 (e.g., as in the case where shifting the suction source could drain down the RCIC suction piping), Condition E must be entered and its Required Action taken.

E.1

With any Required Action and associated Completion Time not met, the RCIC System may be incapable of performing the intended function, and the RCIC System must be declared inoperable immediately.

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

(continued)

BASES

ACTIONS (continued)

required Surveillances specified in LCO 3.6.4.1, "Secondary Containment," LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIV)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," can be performed without inducing an isolation of the MSIVs. The 4 hour and 12 hour allowances provide sufficient time to safely perform the testing. The 12 hour allowance also provides sufficient time to identify and correct minor reactor building ventilation system problems. Since the design of the Unit 1 and Unit 2 reactor buildings is such that they share a common area of the refuel floor (i.e., the reactor buildings are not separated on the refuel floor), operation of either unit's ventilation system will affect the other unit's building differential pressure. Performance of testing to verify secondary containment integrity requirements and minor correctable problems could require a dual unit outage (without the Notes).

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours, depending on the Function (12 hours for those Functions that have channel components common to RPS instrumentation and 24 hours for those Functions that do not have channel components common to RPS instrumentation), has been shown to be acceptable (Refs. 9 and 10) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Action taken.

(continued)

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

ACTIONS (continued)

A.1

With one or more channels of a Function inoperable, the Function may not be capable of performing the intended function. Therefore, only 1 hour is allowed to restore the inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition B must be entered and its Required Action taken.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

B.1

If any Required Action and associated Completion Time is not met, the associated Function may not be capable of performing the intended function. Therefore, the associated DG(s) are declared inoperable immediately. This requires entry into applicable Conditions and Required Actions of LCO 3.8.1 and LCO 3.8.2, which provide appropriate actions for the inoperable DG(s).

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each LOP Instrumentation Function are located in the SRs column of Table 3.3.8.1-1.

(continued)

BASES (continued)

APPLICABILITY

All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3 when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, the ADS function is not required when pressure is ≤ 150 psig because the low pressure ECCS subsystems (LPCS and LPCI) are capable of providing flow into the RPV below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS—Shutdown."

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable HPCS subsystem. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable HPCS subsystem and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

or in accordance with the Risk Informed Completion Time Program

A.1

If any one low pressure ECCS injection/spray subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE subsystems concurrent with a LOCA may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is based on a reliability study (Ref. 12) that evaluated the impact on ECCS availability by assuming that various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (i.e., Completion Times).

or in accordance with the Risk Informed Completion Time Program

B.1 and B.2

If the HPCS System is inoperable, and the RCIC System is immediately verified to be OPERABLE (when RCIC is required to be OPERABLE), the HPCS System must be restored to OPERABLE status within 14 days. In this Condition, adequate

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

core cooling is ensured by the OPERABILITY of the redundant and diverse low pressure ECCS injection/spray subsystems in conjunction with the ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Immediate verification of RCIC OPERABILITY is therefore required when HPCS is inoperable and RCIC is required to be OPERABLE. This may be performed by an administrative check, by examining logs or other information, to determine if RCIC is out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the RCIC System. However, if the OPERABILITY of the RCIC System cannot be immediately verified and RCIC is required to be OPERABLE, Condition D must be entered. If a single active component fails concurrent with a design basis LOCA, there is a potential, depending on the specific failure, that the minimum required ECCS equipment will not be available. A 14 day Completion Time is based on the results of a reliability study (Ref. 12) and has been found to be acceptable through operating experience.

or in accordance with the Risk Informed Completion Time Program

C.1

With two ECCS injection subsystems inoperable or one ECCS injection and the low pressure ECCS spray subsystem (LPCS) inoperable, at least one ECCS injection/spray subsystem must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced in this Condition because a single failure in one of the remaining OPERABLE subsystems concurrent with a design basis LOCA may result in the ECCS not being able to perform its intended safety function. Since the ECCS availability is reduced relative to Condition A, a more restrictive Completion Time is imposed. The 72 hour Completion Time is based on a reliability study, as provided in Reference 12.

(continued)

BASES

ACTIONS (continued)

D.1

If any Required Action and associated Completion Time of Condition A, B, or C are not met, the plant must be brought to a MODE in which the overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 15) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

The LCO requires six ADS valves to be OPERABLE to provide the ADS function. Reference 11 contains the results of an evaluation of the effect of one required ADS valve being out of service. Per this evaluation, operation of only five ADS valves will provide the required depressurization. However, overall reliability of the ADS is reduced because a single failure in the OPERABLE ADS valves could result in a reduction in depressurization capability. Therefore, operation is only allowed for a limited time. The 14 day Completion Time is based on a reliability study (Ref. 12) and has been found to be acceptable through operating experience. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

(continued)

BASES (continued)

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable RCIC system. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable RCIC system and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCOO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

or in accordance with the Risk Informed Completion Time Program

A.1 and A.2

If the RCIC System is inoperable during MODE 1, or MODES 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCS System is immediately verified to be OPERABLE, the RCIC System must be restored to OPERABLE status within 14 days. In this Condition, loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high RPV pressure since the HPCS System is the only high pressure system assumed to function during a loss of coolant accident (LOCA). OPERABILITY of the HPCS is therefore immediately verified when the RCIC System is inoperable. This may be performed as an administrative check, by examining logs or other information, to determine if the HPCS is out of service for maintenance or other reasons. Verification does not require performing the Surveillances needed to demonstrate the OPERABILITY of the HPCS System. If the OPERABILITY of the HPCS System cannot be immediately verified, however, Condition B must be entered. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCS) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of RPV water level. Therefore, a limited time is allowed to restore the inoperable RCIC to OPERABLE status.

The 14 day Completion Time is based on a reliability study (Ref. 3) that evaluated the impact on ECCS availability, assuming that various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (AOTs). Because of the similar functions of the HPCS and RCIC, the AOTs (i.e., Completion Times) determined for the HPCS are also applied to RCIC.

(continued)

BASES

ACTIONS

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

verified locked closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

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

With the air lock inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate containment overall leakage rates using current air lock leakage test results. An evaluation is acceptable since it is overly conservative to immediately declare the primary containment inoperable if both doors in the air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed) primary containment remains OPERABLE, yet only 1 hour (according to LCO 3.6.1.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the primary containment air locks must be verified closed. This Required Action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1.1, which require that primary containment be restored to OPERABLE status within 1 hour.

Additionally, the air lock must be restored to OPERABLE status within 24 hours (Required Action C.3). The 24 hour Completion Time is reasonable for restoring the inoperable air lock to OPERABLE status considering that at least one door is maintained closed in the air lock.

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

(continued)

or in accordance with the Risk Informed Completion Time Program

BASES

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

ACTIONS

A.1 and A.2 (continued)

penetration should be the closest available one to the primary containment. The Required Action must be completed within the 4 hour Completion Time (8 hours for main steam lines). The specified time period of 4 hours is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. For main steam lines, an 8 hour Completion Time is allowed. The Completion Time of 8 hours for the main steam lines allows a period of time to restore the MSIVs to OPERABLE status given the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown.

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following an accident, and no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those devices outside the primary containment and capable of being mispositioned are in the correct position. The Completion Time for this verification of "once per 31 days for isolation devices outside primary containment" is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. For devices inside the primary containment, the specified time period of "prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days," is based on engineering judgment and is considered reasonable in view of the inaccessibility of the devices and the existence of other administrative controls ensuring that device misalignment is an unlikely possibility.

following isolation

Condition A is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two or more PCIVs. For penetration flow paths with one PCIV, Condition C provides appropriate Required Actions.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

When one or more penetration flow paths with one PCIV inoperable, except for MSIV leakage rate or hydrostatically tested line leakage rate not within limit, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. The Completion Time of 4 hours for valves other than EFCVs and in penetrations with a closed system is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. The Completion Time of 72 hours for penetrations with a closed system is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. The closed system must meet the requirements of Reference 5. The Completion Time of 72 hours for EFCVs is also reasonable considering the instrument and the small pipe diameter of penetration (hence, reliability) to act as a penetration isolation boundary and the small pipe diameter of the affected penetration. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following an accident are isolated. This Required Action does not require any testing or valve manipulation. Rather, it involves verification that these devices outside containment and capable of potentially being mispositioned are in the correct position. The Completion Time of "once per 31 days" is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low.

following isolation

(continued)

BASES

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

ACTIONS

A.1 (continued)

with one of the four vacuum breakers inoperable, 72 hours is allowed to restore the inoperable vacuum breaker to OPERABLE status so that plant conditions are consistent with those assumed for the design basis analysis. The 72 hour Completion Time is considered acceptable due to the low probability of an event in which the remaining vacuum breaker capability would not be adequate.

B.1

If a required suppression chamber-to-drywell vacuum breaker is inoperable for opening and is not restored to OPERABLE status within the required Completion Time, the plant must be brought to a condition in which the overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 3) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one vacuum breaker not closed, communication between the drywell and suppression chamber airspace exists, and, as a result, there is the potential for primary containment overpressurization due to this bypass leakage if a LOCA were to occur. Therefore, both manual isolation valves in the affected vacuum breaker line must be closed. A short time is allowed to close the manual valves due to the low probability of an event that would pressurize primary containment. The required 4 hour Completion Time is considered adequate to perform this activity. With both

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

suppression pool temperature is calculated to remain below the design limit.

The RHR Suppression Pool Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, a minimum of one RHR suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below the design limits (Ref. 1). To ensure that these requirements are met, two RHR suppression pool cooling subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE, assuming the worst case single active failure. An RHR suppression pool cooling subsystem is OPERABLE when the pump, a heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE. Management of gas voids is important to RHR Suppression Pool Cooling System OPERABILITY.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause both a release of radioactive material to primary containment and a heatup and pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the RHR Suppression Pool Cooling System is not required to be OPERABLE in MODE 4 or 5.

ACTIONS

or in accordance with the Risk Informed Completion Time Program

A.1

With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time is acceptable in light of the redundant RHR suppression pool cooling capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

capacity of the RHR Suppression Pool Spray System is adequate to maintain the primary containment conditions within design limits. The time history for primary containment pressure is calculated to demonstrate that the maximum pressure remains below the design limit.

The RHR Suppression Pool Spray System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In the event of a DBA, a minimum of one RHR suppression pool spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure below the design limits (Ref. 1). To ensure that these requirements are met, two RHR suppression pool spray subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR suppression pool spray subsystem is OPERABLE when one of the pumps and associated piping, valves, instrumentation, and controls are OPERABLE. Management of gas voids is important to RHR Suppression Pool Spray System OPERABILITY.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining RHR suppression pool spray subsystems OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

or in accordance with the Risk Informed Completion Time Program

With one RHR suppression pool spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE RHR suppression pool spray subsystem is adequate to perform the primary containment bypass leakage mitigation function.

However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment bypass mitigation capability. The 7 day Completion Time was chosen in light of the redundant RHR suppression pool spray capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

(continued)

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

ACTIONS

A.1

Required Action A.1 is intended to handle the inoperability of one RHRSW subsystem. The Completion Time of 7 days is allowed to restore the RHRSW subsystem to OPERABLE status. With the unit in this condition, the remaining OPERABLE RHRSW subsystem is adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE RHRSW subsystem could result in loss of RHRSW function. The Completion Time is based on the redundant RHRSW capabilities afforded by the OPERABLE subsystem and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.9, be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

B.1

If one RHRSW subsystem is inoperable and not restored within the provided Completion Time, the plant must be brought to a condition in which the overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 6) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

ACTIONS

A.2 (continued)

supported features, or both, that are associated with the other division that has offsite power, results in starting the Completion Time for the Required Action.

Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before the unit is subjected to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection may have been lost for the required feature's function; however, function is not lost. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours.

With one required offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the plant safety systems. In this condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E distribution system.

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

(continued)

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

ACTIONS (continued)

B.3.1 and B.3.2 (continued)

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG(s) does not exist on the OPERABLE DG(s), SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DGs, the other DGs are declared inoperable upon discovery, and Condition F, G or I of LCO 3.8.1 is entered, as applicable.

Once the failure is repaired, and the common cause failure no longer exists, Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG(s), performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of those DG(s). In the event the inoperable DG(s) is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the station corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

If while a DG is inoperable, a new problem with the DG is discovered that would have prevented the DG from performing its specified safety function, a separate entry into Condition B is not required. The new DG problem should be addressed in accordance with the station corrective action program.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable time to confirm that the OPERABLE DG(s) are not affected by the same problem as the inoperable DG.

B.4

In this condition, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E distribution system. The 14 day Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

(continued)

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

ACTIONS

C.3.1 and C.3.2 (continued)

affected by the same problem as the inoperable DG.

C.4

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 72 hours. In this condition, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E distribution system. The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

D.1 and D.2

Required Action D.1 addresses actions to be taken in the event of concurrent failure of redundant required features.

(continued)

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

ACTIONS

D.1 and D.2 (continued)

to a total loss of the immediately accessible offsite power sources.

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

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

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

With two of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria. According to Regulatory Guide 1.93 (Ref. 6), with the available offsite AC sources two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

E.1 and E.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition E are modified by a Note to indicate that when Condition E is entered with no AC source to any

(continued)

ACTIONS

E.1 and E.2 (continued)

required division (i.e., the division is de-energized), Actions for LCO 3.8.7, "Distribution Systems—Operating," must be immediately entered. This allows Condition E to provide requirements for the loss of an offsite circuit and one required unit DG without regard to whether a division is de-energized. LCO 3.8.7 provides the appropriate restrictions for a de-energized division.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition E for a period that should not exceed 12 hours. In Condition E, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition D (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

Conditi

Condition F is modified by a Note indicating that this Condition is not applicable during installation of Division 2 CSCS isolation valves during a single Unit 1 Refueling Outage completed prior to July 1, 2024, and during a single Unit 2 Refueling Outage completed prior to July 1, 2023, while the outage Unit is in MODE 4, 5, or defueled. For Unit 1, the one-time use of this Note will occur during Refueling Outage 18, 19, or 20. For Unit 2, the one-time use of this Note will occur during Refueling Outage 17, 18, or 19. When the Division 2 DGs are inoperable during the CSCS isolation valve maintenance, Conditions B and G provide appropriate Required Actions.

F.1

With two required unit DGs inoperable or both required Division 2 DGs inoperable, there is no more than two

BASES

ACTIONS

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

Required Action A.2 requires that the battery float current be verified as less than or equal to 2 amps. This indicates that, if the battery has been discharged as a result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 12 hour period the battery float current is not less than or equal to 2 amps, this indicates there may be additional battery problems and the battery must be declared inoperable.

Alignment (editorial)

Required Action A.3 limits the restoration time for the inoperable battery charger to 7 days. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., balance of plant non-Class 1E battery charger). The 7 day Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

or in accordance with the Risk Informed Completion Time Program

B.1

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

If one of the Division 1 or 2 125 VDC electrical power subsystems is inoperable for reasons other than Condition A (e.g., inoperable battery), the remaining DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure could, however, result in the loss of minimum necessary DC electrical subsystems, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 7) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

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

(continued)

BASES

ACTIONS (continued)

C.1

If the Division 3 battery cannot be maintained OPERABLE, the required Division 3 battery charger cannot be restored, or the Division 3 DC electrical power subsystem is inoperable for reasons other than Condition A (e.g., inoperable battery), the HPCS System may be incapable of performing its intended function and must be immediately declared inoperable. This declaration also requires entry into applicable Conditions and Required Actions of LCO 3.5.1, "ECCS—Operating."

D.1

If the Division 1 250 VDC battery cannot be maintained OPERABLE, the required 250 VDC battery charger cannot be restored, or the Division 1 250 VDC electrical power subsystem is inoperable for reasons other than Condition A (e.g., inoperable battery), the RCIC System and the RCIC DC powered PCIVs may be incapable of performing their intended functions and must be immediately declared inoperable. This declaration also requires entry into applicable Conditions and Required Actions of LCO 3.5.3, "RCIC System," and LCO 3.6.1.3, "PCIVs."

E.1

If the opposite unit Division 2 battery cannot be maintained OPERABLE, the required opposite unit Division 2 battery charger cannot be restored, or the opposite unit Division 2 125 VDC electrical power subsystem is inoperable for reasons other than Condition A (e.g., inoperable battery), certain redundant Division 2 features (e.g., a standby gas treatment subsystem) will not function if a design basis event were to occur. Therefore, a 7 day Completion Time is provided to restore the opposite unit Division 2 125 VDC electrical power subsystem to OPERABLE status. The 7 day Completion Time takes into account the capacity and capability of the remaining DC electrical power subsystems, and is based on the shortest restoration time allowed for the systems affected by the inoperable DC electrical power subsystem in the respective system specifications.

(continued)

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

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

With one or more Division 1 and 2 required AC buses, load centers, motor control centers, or distribution panels inoperable and a loss of function has not yet occurred, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels must be restored to OPERABLE status within 8 hours.

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

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

b. The low potential for an event in conjunction with a single failure of a redundant component in the division with AC power. (The redundant component is verified OPERABLE in accordance with Specification 5.5.12, "Safety Function Determination Program (SFDP).")

(continued)

BASES

ACTIONS

B.1

With one or more Division 1 and 2 DC electrical distribution subsystems inoperable and a loss of function has not yet occurred, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required DC electrical power distribution subsystem(s) must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

(continued)

or in accordance with the Risk Informed Completion Time Program

A Note clarifies that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

BASES

ACTIONS (continued)

D.1

With one or more required opposite unit Division 2 AC or DC electrical power distribution subsystems inoperable and a loss of function has not yet occurred, certain redundant Division 2 features (e.g., a standby gas treatment subsystem) will not function if a design basis event were to occur. Therefore, a 7 day Completion Time is provided to restore the required opposite unit Division 2 AC and DC electrical power distribution subsystems to OPERABLE status. The 7 day Completion Time takes into account the capacity and capability of the remaining AC and DC electrical power distribution subsystems, and is based on the shortest restoration time allowed for the systems affected by the inoperable AC and DC electrical power distribution subsystems in the respective system specifications.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.8.1 be entered and Required Actions taken if the inoperable opposite unit AC electrical power distribution subsystem results in an inoperable required offsite circuit. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. This Completion Time is modified by a Note to clarify that the Risk Informed Completion Time Program is not applicable to a Required Action associated with a Condition that represents a loss of safety function.

E.1 and

If the
cannot
Complet
which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With the Division 3 electrical power distribution system inoperable (i.e., one or both Division 3 AC or DC electrical power distribution subsystems inoperable), the Division 3

(continued)

ATTACHMENT 4

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

Cross-Reference of TSTF-505 and LaSalle County Station Technical Specifications

ATTACHMENT 4
Cross-Reference of TSTF-505 and LaSalle County Station Technical Specifications

Tech Spec Description	TSTF-505 TS	LSCS TS	Apply RICT?	Comments
Completion Times	**1.3**	**1.3**		
Example 1.3-8	1.3-8	1.3-8		TSTF-505 changes are incorporated.
Standby Liquid Control (SLC) System	**3.1.7**	**3.1.7**		
One SLC subsystem inoperable.	3.1.7.B.1	3.1.7.A.1	Yes	TSTF-505 changes are incorporated.
Reactor Protection System (RPS) Instrumentation	**3.3.1.1**	**3.3.1.1**		
One or more required channels inoperable.	3.3.1.1.A.1 3.3.1.1.A.2	3.3.1.1.A.1 3.3.1.1.A.2	Yes Yes	TSTF-505 changes are incorporated. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. Condition C would be entered with a 1 hour Completion Time and no RICT.
One or more Functions with one or more required channels inoperable in both trip systems.	3.3.1.1.B.1 3.3.1.1.B.2	3.3.1.1.B.1 3.3.1.1.B.2	Yes Yes	TSTF-505 changes are incorporated. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. Condition C would be entered with a 1 hour Completion Time and no RICT.
Feedwater System and Main Turbine High Water Level Trip Instrumentation	**3.3.2.2**	**3.3.2.2**		
One or more feedwater system and main turbine high water level trip channels inoperable.	3.3.2.2.A.1	3.3.2.2.A.1	Yes	TSTF-505 changes are incorporated. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. Condition B would be entered with a 2 hour Completion Time and no RICT.
Feedwater system and main turbine high water level trip capability not maintained.	-----------	3.3.2.2.B.1	No	TSTF-505 changes are excluded due to loss of function.

ATTACHMENT 4
Cross-Reference of TSTF-505 and LaSalle County Station Technical Specifications

Tech Spec Description	TSTF-505 TS	LSCS TS	Apply RICT?	Comments
Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT) Instrumentation	**3.3.4.2**	**3.3.4.2**		LSCS TS align with BWR/6, NUREG-1434
One or more channels inoperable.	3.3.4.2.A.1 3.3.4.2.A.2	3.3.4.2.A.1 3.3.4.2.A.2	Yes Yes	TSTF-505 changes are incorporated. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. Condition B would be entered for loss of one function with a 72 hour Completion Time and no RICT. Condition C would be entered for a loss of both functions with a 1 hour Completion Time and no RICT.
Emergency Core Cooling System (ECCS) Instrumentation	**3.3.5.1**	**3.3.5.1**		LSCS TS align with BWR/6, NUREG-1434
As required by Required Action A.1 and referenced in Table 3.3.5.1-1. (Functions 1.a, 1.b, 2.a, and 2.b; 3.a and 3.b).	3.3.5.1.B.3	3.3.5.1.B.3	Yes	TSTF-505 changes are incorporated. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. A Note is added which prohibits applying a RICT when a loss of function occurs.
As required by Required Action A.1 and referenced in Table 3.3.5.1-1. (Functions 1.c, 1.d, 1.g, 2.c, 2.d, 2.f; 3.c, and 3.h).	3.3.5.1.C.2	3.3.5.1.C.2	Yes	TSTF-505 changes are incorporated with the exception of TSTF-505 R2 function 1.d. TSTF-505 R2 function 1.d corresponds to LSCS TS Table 3.3.5-1 function 1.d, which is directed to LSCS TS Condition D, see 3.3.5.1.D.4 below. TSTF-505 R2 functions 1.g, 2.f, and 3.h correspond to LSCS TS Table 3.3.5-1 functions 1.h, 2.g, and 3.f, respectively. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. A Note is added which prohibits applying a RICT when a loss of function occurs.

ATTACHMENT 4
Cross-Reference of TSTF-505 and LaSalle County Station Technical Specifications

Tech Spec Description	TSTF-505 TS	LSCS TS	Apply RICT?	Comments
	-----------	3.3.5.1.D.3	Yes	LSCS TS Required Action 3.3.5.1.D.3 is modified by a note limiting applicability to Functions 1.g and 2.f. Required Action 3.3.5.1.D.3 is a plant specific Required Action with a restoration action and allowed outage time of 24 hours. LSCS proposes to apply a RICT to LSCS TS 3.3.5.1, Required Action D.3. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. A Note is added which prohibits applying a RICT when a loss of function occurs.
As required by Required Action A.1 and referenced in Table 3.3.5.1-1. (Functions 3.d and 3.e).	3.3.5.1.D.2.1	-----------	No	The LSCS TS do not contain this TS. Therefore, a change is not proposed to the LSCS TS.
As required by Required Action A.1 and referenced in Table 3.3.5.1-1. (Functions 1.e, 1.f, 2.e, 3.f, and 3.g).	3.3.5.1.E.2	3.3.5.1.D.4	Yes	TSTF-505 changes are incorporated. TSTF-505 R2 functions 3.f and 3.g correspond to LSCS TS Table 3.3.5-1 functions 3.d and 3.e, respectively. TSTF-505 R2 function 1.d corresponds to LSCS TS Table 3.3.5-1 function 1.d, which is directed to LSCS TS Condition D, discussed above. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. A Note is added which prohibits applying a RICT when a loss of function occurs.
As required by Required Action A.1 and referenced in Table 3.3.5.1-1. (Functions 4.a, 4.b, 4.d, 5.a, 5.b, and 5.d).	3.3.5.1.F.2	3.3.5.1.E.2	Yes	TSTF-505 changes are incorporated. RICT insert format is modified from TSTF-505 R2 to align with LSCS TS 1.2, "Logical Connectors," direction to only use first level logic for Completion Time. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. A Note is added which prohibits applying a RICT when a loss of function occurs.

ATTACHMENT 4
Cross-Reference of TSTF-505 and LaSalle County Station Technical Specifications

Tech Spec Description	TSTF-505 TS	LSCS TS	Apply RICT?	Comments
As required by Required Action A.1 and referenced in Table 3.3.5.1-1. (Functions 4.c, 4.e, 4.f, 4.g, 4.h, 5.c, 5.e, 5.f, and 5.g).	3.3.5.1.G.2	3.3.5.1.F.2	Yes	TSTF-505 changes are incorporated. RICT insert format is modified from TSTF-505 R2 to align with LSCS TS 1.2, "Logical Connectors," direction to only use first level logic for Completion Time. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. A Note is added which prohibits applying a RICT when a loss of function occurs.
Reactor Core Isolation Cooling (RCIC) System Instrumentation	**3.3.5.2**	**3.3.5.3**		
As required by Required Action A.1 and referenced in Table 3.3.5.3-1. (Function 1).	3.3.5.2.B.2	3.3.5.3.B.2	Yes	TSTF-505 changes are incorporated. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. A Note is added which prohibits applying a RICT when a loss of function occurs.
As required by Required Action A.1 and referenced in Table 3.3.5.3-1. (Functions 3 and 4).	3.3.5.2.D.2.1	3.3.5.3.D.2.1	Yes	TSTF-505 changes are incorporated. LSCS TS Table 3.3.5.3-1 does not contain the TSTF-505 R2 Function 4. LSCS TS Table 3.3.5.3-1 Function 3, alone, references LSCS TS Condition 3.3.5.3.D. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. A Note is added which prohibits applying a RICT when a loss of function occurs.

ATTACHMENT 4
Cross-Reference of TSTF-505 and LaSalle County Station Technical Specifications

Tech Spec Description	TSTF-505 TS	LSCS TS	Apply RICT?	Comments
Primary Containment Isolation Instrumentation	3.3.6.1	3.3.6.1		
One or more channels inoperable.	3.3.6.1.A.1	3.3.6.1.A.1	Yes	TSTF-505 changes are incorporated. RICT insert format is modified from TSTF-505 R2 to align with LSCS TS 1.2, "Logical Connectors," direction to only use first level logic for Completion Time. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. Condition B would be entered with a 1 hour Completion Time and no RICT.
Loss of Power (LOP) Instrumentation	3.3.8.1	3.3.8.1		
One or more channels inoperable.	3.3.8.1.A.1	3.3.8.1.A.1	Yes	TSTF-505 changes are incorporated. Under certain circumstances, with more than one required channel inoperable, a loss of function can occur. A Note is added which prohibits applying a RICT when a loss of function occurs.
ECCS – Operating	3.5.1	3.5.1		LSCS TS align with BWR/6, NUREG-1434
One low pressure ECCS injection/spray subsystem inoperable.	3.5.1.A.1	3.5.1.A.1	Yes	TSTF-505 changes are incorporated.
High Pressure Core Spray (HPCS) System inoperable.	3.5.1.B.2	3.5.1.B.2	Yes	TSTF-505 changes are incorporated.
Two low pressure ECCS injection/spray subsystems inoperable.	3.5.1.C.1	3.5.1.C.1	Yes	TSTF-505 changes are incorporated.
One ADS valve inoperable.	3.5.1.E.1	3.5.1.E.1	Yes	TSTF-505 changes are incorporated.

ATTACHMENT 4
Cross-Reference of TSTF-505 and LaSalle County Station Technical Specifications

Tech Spec Description	TSTF-505 TS	LSCS TS	Apply RICT?	Comments
One ADS valve inoperable AND One low pressure ECCS injection/spray subsystem inoperable.	3.5.1.F.1 3.5.1.F.2	-----------	No	The LSCS TS do not currently contain this TS. Therefore, a change is not proposed to the LSCS TS.
RCIC System	**3.5.3**	**3.5.3**		LSCS TS align with BWR/6, NUREG-1434
RCIC System inoperable.	3.5.3.A.2	3.5.3.A.2	Yes	TSTF-505 changes are incorporated.
Primary Containment Air Lock	**3.6.1.2**	**3.6.1.2**		
Primary containment air lock inoperable for reasons other than Condition A or B.	3.6.1.2.C.3	3.6.1.2.C.3	Yes	TSTF-505 changes are incorporated. Compliance with the remaining portions of LCO Condition 3.6.1.2 ensure that there is a physical barrier (i.e., closed door) and an acceptable overall leakage from containment. Thus, the function is still maintained. Required Action C.1 of LCO Condition 3.6.2 requires the condition to be assessed in accordance with TS 3.6.1, "Containment Integrity" (i.e., "Initiate action to evaluate overall containment leakage rate per LCO 3.6.1" with a Completion Time of Immediately.)
Primary Containment Isolation Valves (PCIVs)	**3.6.1.3**	**3.6.1.3**		
One or more penetration flow paths with one PCIV inoperable for reasons other than Condition D.	3.6.1.3.A.1	3.6.1.3.A.1	Yes	TSTF-505 changes are incorporated. RICT insert format is modified from TSTF-505 R2 to align with LSCS TS 1.2, "Logical Connectors," direction to only use first level logic for Completion Time.
[One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.]	3.6.1.3.E.1	-----------	No	The LSCS TS do not contain this TS. Therefore, a change is not proposed to the LSCS TS.

ATTACHMENT 4
Cross-Reference of TSTF-505 and
LaSalle County Station Technical Specifications

Tech Spec Description	TSTF-505 TS	LSCS TS	Apply RICT?	Comments
Suppression Chamber-to-Drywell Vacuum Breakers	**3.6.1.8**	**3.6.1.6**		
One suppression chamber-to-drywell vacuum breaker inoperable for opening.	3.6.1.8.A	3.6.1.6.A	Yes	TSTF-505 changes are incorporated.
Residual Heat Removal (RHR) Suppression Pool Cooling	**3.6.2.3**	**3.6.2.3**		
One RHR suppression pool cooling subsystem inoperable.	3.6.2.3.A.1	3.6.2.3.A.1	Yes	TSTF-505 changes are incorporated.
Residual Heat Removal (RHR) Suppression Pool Spray	**3.6.2.4**	**3.6.2.4**		
One RHR suppression pool spray subsystem inoperable.	3.6.2.4.A.1	3.6.2.4.A.1	Yes	TSTF-505 changes are incorporated.
Residual Heat Removal Service Water (RHRSW) System	**3.7.1**	**3.7.1**		
One RHRSW pump in each subsystem inoperable.	3.7.1.B.1	-----------	No	The LSCS TS do not contain this TS. Therefore, a change is not proposed to the LSCS TS.
One RHRSW subsystem inoperable for reasons other than Condition A.	3.7.1.C.1	3.7.1.A.1	Yes	TSTF-505 changes are incorporated.
AC Sources – Operating	**3.8.1**	**3.8.1**		
One required offsite circuit inoperable.	3.8.1.A.3	3.8.1.A.3	Yes	TSTF-505 changes are incorporated.
One required Division 1, or 2 DG inoperable OR_Required opposite unit Division 2 DG inoperable	3.8.1.B.4	3.8.1.B.4	Yes	TSTF-505 changes are incorporated.

ATTACHMENT 4
Cross-Reference of TSTF-505 and LaSalle County Station Technical Specifications

Tech Spec Description	TSTF-505 TS	LSCS TS	Apply RICT?	Comments
Required Division 3 DG inoperable OR One required Division 1, 2, or 3 DG inoperable and the required opposite unit Division 2 DG inoperable.	-----------	3.8.1.C.4	Yes	LSCS TS 3.8.1.C is a plant specific condition with a restoration action and allowed outage time of 72 hours. LSCS proposes to apply a RICT to LSCS TS 3.8.1, Required Action C.4. LSCS TS Condition 3.8.1.C includes one Division 1, 2, or 3 DG inoperable and the required opposite unit Division 2 DG inoperable. As such, under certain circumstances, a loss of function can occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when a loss of function occurs.
Two required offsite circuits inoperable.	3.8.1.C.2	3.8.1.D.2	Yes	TSTF-505 changes are incorporated.
One required offsite circuit inoperable AND One required Division 1, 2, or 3 DG inoperable.	3.8.1.D.1 3.8.1.D.2	3.8.1.E.1 3.8.1.E.2	Yes Yes	TSTF-505 changes are incorporated.
One required automatic load sequencer inoperable.	3.8.1.F.1	-----------	No	The LSCS TS do not contain this TS. Therefore, a change is not proposed to the LSCS TS.
DC Sources – Operating	**3.8.4**	**3.8.4**		
One required Division 1, 2, or 3 125 VDC battery charger on one division inoperable OR One required Division 2 or opposite unit Division 2 battery charger on one division inoperable OR One required Division 1 250 VDC battery charger inoperable.	3.8.4.A.3	3.8.4.A.3	Yes	TSTF-505 changes are incorporated.
One or two batteries on one division inoperable.	3.8.4.B.1	-----------	No	The LSCS TS do not contain this TS. Therefore, a change is not proposed to the LSCS TS.
Division 1 or 2 125 VDC electrical power subsystem inoperable for reasons other than Condition A.	3.8.4.C.1	3.8.4.B.1	Yes	TSTF-505 changes are incorporated.

Tech Spec Description	TSTF-505 TS	LSCS TS	Apply RICT?	Comments
Required Action and associated completion time of Condition A not met for the opposite unit Division 2 DC electrical power subsystem OR Opposite unit Division 2 DC electrical power subsystem inoperable for reasons other than Condition A.		3.8.4.E.1	Yes	LSCS TS 3.8.4.E is a plant specific condition with a restoration action and allowed outage time of 7 days. LSCS proposes to apply a RICT to LSCS TS 3.8.4, Required Action E.1.
Distribution Systems – Operating	**3.8.9**	**3.8.7**		
One or both Division 1 and 2 AC electrical power distribution subsystems inoperable.	3.8.9.A.1	3.8.7.A.1	Yes	TSTF-505 changes are incorporated. The condition specified is one or more AC electrical power distribution subsystems inoperable. As such, under certain circumstances, a loss of function can occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when a loss of function occurs.
One or more AC vital buses inoperable.	3.8.9.B.1	-----------	No	The LSCS TS do not contain this TS. Therefore, a change is not proposed to the LSCS TS.
One or both Division 1 and 2 125 V DC electrical power distribution subsystems inoperable.	3.8.9.C.1	3.8.7.B.1	Yes	TSTF-505 changes are incorporated. The condition specified is one or more DC electrical power distribution subsystems inoperable. As such, under certain circumstances, a loss of function can occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when a loss of function occurs.
One or more required opposite unit Division 2 AC or DC electrical power distribution subsystems inoperable.	-----------	3.8.7.D.1	Yes	LSCS TS 3.8.7.D is a plant specific condition with a restoration action and allowed outage time of 7 days. LSCS proposes to apply a RICT to LSCS TS 3.8.7, Required Action D.1.

ATTACHMENT 4
Cross-Reference of TSTF-505 and
LaSalle County Station Technical Specifications

Tech Spec Description	TSTF-505 TS	LSCS TS	Apply RICT?	Comments
Programs and Manuals	**5.5**	**5.5**		
Programs and Manuals - Risk Informed Completion Time Program	[NEW TS] 5.5.15	[NEW TS] 5.5.17		The LSCS TS do not currently contain this program. The new RICT Program will be added to the LSCS TS 5.5.17 consistent with TSTF-505 R2.

ATTACHMENT 5

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

LaSalle County Station RICT Program PRA Implementation Item

ATTACHMENT 5
LaSalle County Station RICT Program PRA Implementation Item

The table below identifies the item that is required to be completed prior to implementation of the Risk-Informed Completion Time (RICT) Program at LaSalle County Station. The item identified below will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the RICT Program.

Source	Description	Implementation Item
Enclosure 1, Table E1-1, TS 3.3.2.2.A	One or more feedwater system and main turbine high water level trip channels inoperable	SSCs are not modeled. The model will be updated to include these SSCs prior to exercising the RICT program for this TS. The PRA Success Criteria will match the Design Success Criteria.

ATTACHMENT 6

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

Proposed Renewed Facility Operating License (RFOL) Changes (Mark-Ups)

LaSalle County Station Unit 1 RFOL Page 1 (no changes) and Page 9
LaSalle County Station Unit 2 RFOL Page 1 (no changes) and Page 10
INSERTS

No changes this page, provided for information

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

RENEWED FACILITY OPERATING LICENSE

Renewed License No. NPF-11

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for a renewed license filed by the applicant* complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the LaSalle County Station, Unit 1 (the facility), has been substantially completed in conformity with Construction Permit No. CPPR-99 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   E. Exelon Generation Company, LLC is technically qualified to engage in the activities authorized by this renewed operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. Exelon Generation Company, LLC has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this renewed license will not be inimical to the common defense and security or to the health and safety of the public;

*The Nuclear Regulatory Commission approved the transfer of the license from Commonwealth Edison Company to Exelon Generation Company, LLC on August 3, 2000.

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.15.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from 1998, the date of the most recent successful tracer gas test, as stated in the December 9, 2003 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.15.d, shall be within 24 months, plus 6 months allowed by SR 3.0.2, as measured from the date of the most recent successful pressure measurement test, or within 6 months if not performed previously.

(46) License Renewal License Conditions

(a) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and licensee commitments as listed in Appendix A of the "Safety Evaluation Report Related to the License Renewal of LaSalle County Station, Units 1 and 2," are collectively the "License Renewal UFSAR Supplement." This Supplement is henceforth part of the UFSAR, which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs, activities, and commitments described in this Supplement, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59, "Changes, Tests, and Experiments," and otherwise complies with the requirements in that section.

(b) The License Renewal UFSAR Supplement, as defined in license condition 46(a) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation (PEO).

1. The licensee shall implement those new programs and enhancements to existing programs no later than 6 months prior to the PEO.

2. The licensee shall complete those activities by the 6-month date prior to the PEO or to the end of the last refueling outage prior to the PEO, whichever occurs later.

INSERT 1

3. The licensee shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

No changes this page, provided for information

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE

Renewed License No. NPF-18

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for a renewed license filed by the applicant* complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the LaSalle County Station, Unit 1 (the facility), has been substantially completed in conformity with Construction Permit No. CPPR-100 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D below);

   D. There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   E. Exelon Generation Company, LLC is technically qualified to engage in the activities authorized by this renewed license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. Exelon Generation Company, LLC has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

*The Nuclear Regulatory Commission approved the transfer of the license from Commonwealth Edison Company to Exelon Generation Company, LLC on August 3, 2000.

INSERT 2

3. The licensee shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

Am. 87 03/16/95

D. The facility requires exemptions from certain requirements of 10 CFR Part 50, 10 CFR Part 70, and 10 CFR Part 73. These include:

(a) Exemptions from certain requirements of Appendices G, H and J to 10 CFR Part 50, and to 10 CFR Part 73 are described in the Safety Evaluation Report and Supplement Numbers 1, 2, 3, and 5 to the Safety Evaluation Report.

Am. 181 08/28/09

(b) DELETED

Am. 212 11/16 /17

(c) DELETED

Am. 181 08/28/09

(d) DELETED

Am. 97 04/05/96

(e) An exemption was granted to remove the Main Steam Isolation Valves (MSIVs) from the acceptance criteria for the combined local leak rate test (Type B and C), as defined in the regulations of 10 CFR Part 50, Appendix J, Option B, Paragraph III.B. Exemption (e) is described in the safety evaluation accompanying Amendment No. 97 to this License.

These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, and the rules and regulations of the Commission (except as hereinafter exempted therefrom), and the provisions of the Act.

E. Before engaging in additional construction or operational activities which may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement and its Addendum dated November 1978, and the Final Supplemental Environmental Impact Statement dated September 2016, the licensee shall provide a written notification to the Director of the Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

ATTACHMENT 6
Proposed Renewed Facility Operating License (RFOL) Changes – Mark-Ups

INSERT 1

(47) Adoption of Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times -RITSTF Initiative 4b"

Exelon is approved to implement TSTF-505, Revision 2, modifying the Technical Specification requirements related to Completion Times (CT) for Required Actions to provide the option to calculate a longer, risk-informed CT (RICT). The methodology for using the new Risk-Informed Completion Time Program is described in NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, which was approved by the NRC on May 17, 2007.

Exelon will complete the implementation item listed in Attachment 5 of Exelon letter to the NRC dated January 31, 2020, prior to implementation of the RICT Program. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the RICT Program.

INSERT 2

(36) Adoption of Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times -RITSTF Initiative 4b"

Exelon is approved to implement TSTF-505, Revision 2, modifying the Technical Specification requirements related to Completion Times (CT) for Required Actions to provide the option to calculate a longer, risk-informed CT (RICT). The methodology for using the new Risk-Informed Completion Time Program is described in NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, which was approved by the NRC on May 17, 2007.

Exelon will complete the implementation item listed in Attachment 5 of Exelon letter to the NRC dated January 31, 2020, prior to implementation of the RICT Program. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the RICT Program.

ATTACHMENT 7

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

Evaluation of Instrumentation and Control Systems

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

This Attachment provides the components available to respond to identified accident conditions. Not all components are assumed or credited in the UFSAR Chapter 15 Safety Analysis, however, the components have been confirmed to be available and useable.

The following Instrumentation Technical Specifications (TS) Sections are included in this TSTF-505 License Amendment Request (LAR) for LaSalle County Station, Units 1 and 2.

1. Reactor Protection System (RPS) Instrumentation - TS Section 3.3.1.1
2. Feedwater System and Main Turbine High Water Level Trip Instrumentation - TS Section 3.3.2.2
3. Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT) Instrumentation - TS Section 3.3.4.2
4. Reactor Core Isolation Cooling (RCIC) System Instrumentation - TS Section 3.3.5.3
5. Emergency Core Cooling System (ECCS) Instrumentation - TS Section 3.3.5.1
6. Primary Containment Isolation Instrumentation - TS Section 3.3.6.1
7. Loss-of-Power (LOP) Instrumentation - TS Section 3.3.8.1

LaSalle County Station, Units 1 and 2 TS Section 3.3 Limiting Conditions for Operation (LCOs) were developed to ensure that LaSalle County Station, Units 1 and 2 maintains necessary redundancy and diversity and complies with the "single failure" design criterion as defined in IEEE-279-1971, and the diversity requirements as defined in Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC), to Part 50 of 10 CFR, GDC-22, "Protection System Independence."

Included below is a description of the redundant and diverse means available to mitigate accidents that each identified instrumentation and control function defined in TS Section 3.3 is designed to prevent.

The following abbreviations are used within the 'Event' column of the included tables:
DBA – Design Bases Accident
IMF-AOT – Incident of Moderate Frequency Anticipated Operational Transient
II-AOT – Infrequent Incident – Anticipated Operational Transient
ATWS – Anticipated Transient Without SCRAM

1. Reactor Protection System (RPS)

Reference: TS 3.3.1.1 Reactor Protection System (RPS) Instrumentation

The RPS design creates defense-in-depth from the redundancy of the channels for each trip system. The RPS is comprised of two independent trip systems (A and B) with two logic channels in each trip system (A1 and A2, B1 and B2). The outputs of the logic channels in a trip system are combined in a one-out-of-two logic so either channel can trip the associated trip system. The tripping of both trip systems will produce a Reactor SCRAM.

Diverse inputs trip the reactor (UFSAR Table 7.2-1 and LaSalle Technical Specifications 3.3.1.1 Bases).

- Intermediate Range Monitor Neutron Flux High – 4 channels per trip system
- Intermediate Range Monitor – INOP – 4 channels per trip system
- Average Power Range Monitor Neutron Flux – High, Setdown – 3 channels per trip system

- Average Power Range Monitor Flow Biased Simulated Thermal Power – Upscale – 2 channels per trip system
- Average Power Range Monitor Fixed Neutron Flux – High – 3 channels per trip system
- Average Power Range Monitor – Inop – 3 channels per trip system
- Reactor Vessel Steam Dome Pressure – High – 2 channels per trip system
- Reactor Vessel Level – Low, Level 3 – 2 channels per trip system
- Main Steam Isolation Valve – Closure – 8 channels per trip system
- Drywell Pressure – High – 2 channels per trip system
- SCRAM Discharge Volume Water Level – High – 2 channels of each type (two float type and two level switches, meaning 4 total) per trip unit
- Turbine Stop Valve – Closure – 4 channels per trip unit
- Turbine Control Valve Fast Closure, Trip Oil Pressure – Low – 2 channels per trip unit
- Reactor Mode Switch – Shutdown Position – 2 channels per trip unit
- Manual SCRAM – 2 channels per trip unit

In addition, LaSalle County Station has redundant and diverse methods of shutting down the reactor in the unlikely event that the RPS does not SCRAM the reactor. The Alternate Rod Insertion (ARI) system provides backup capability to insert the control rods into the reactor and can be manually or automatically initiated. The Recirculation Pump Trip (RPT) breakers also trip the reactor recirculation pumps to reduce the reactor power via negative void reactivity feedback via the ATWS-RPT subsystem. LaSalle County Station also has a Standby Liquid Control System (SBLC) as an independent backup system. The system can be manually initiated via the Main Control Room keylock switches to inject boron into the Reactor Vessel and to initiate closure of the Reactor Water Clean-Up (RWCU) outboard isolation valve to prevent removal of the injected boron.

RPS Instrumentation Diversity					
TS	RPS Instrument Function	Credited Safety Analysis Event		Diverse RPS Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.1.1	1. Intermediate Range Monitors				
	a. Neutron Flux - High	15.4.1	Rod Withdrawal Error – Low Power	1) Automatic Initiation – -IRM High Neutron Flux - APRM Neutron Flux – High, Setdown 2) Manual SCRAM	II-AOT
	b. Inop	None	None	1) Manual SCRAM	-
3.3.1.1	2. Average Power Range Monitor				
	a. Neutron Flux - High, Setdown	15.4.1	Rod Withdrawal Error – Low Power	1) Automatic Initiation – -APRM Neutron Flux - High, Setdown Trip - IRM Neutron Flux - High 2) Manual SCRAM	II-AOT
	b. Flow Biased Simulated Thermal Power - Upscale	15.1.1	Loss of Feedwater Heater	1) Automatic Initiation – -APRM Flow Biased Simulated Thermal Power – Upscale - APRM Fixed Neutron Flux – High -TSV Closure 2) Manual SCRAM	IMF-AOT
		15.4.9	Control Rod Drop Accident	1) Automatic Initiation – -APRM Flow Biased Simulated Thermal Power – Upscale - APRM Fixed Neutron Flux – High - IRM Neutron Flux – High 2) Manual SCRAM	DBA
	c. Fixed Neutron Flux - High	15.1.4	Inadvertent RHR Shutdown Cooling Operation	1) Automatic Initiation - - APRM Fixed Neutron Flux -High 2) Manual SCRAM	IMF-AOT

RPS Instrumentation Diversity					
TS	RPS Instrument Function	Credited Safety Analysis Event		Diverse RPS Instrumentation	Event
		UFSAR Section	Transient / Accident		
		15.2.10	Loss of Stator Cooling	1) Automatic Initiation - - Reactor Vessel Steam Dome Pressure High Trip - APRM Fixed Neutron Flux -High 2) Manual SCRAM	IMF-AOT
		15.4.5	Recirculation Flow Control Failure with Increasing Flow	1) Automatic Initiation - - APRM Fixed Neutron Flux -Upscale 2) Manual SCRAM	IMF-AOT
	d. Inoperable	None	None	1) Manual SCRAM	-

RPS Instrumentation Diversity					
TS	**RPS Instrument Function**	**Credited Safety Analysis Event**		**Diverse RPS Instrumentation**	**Event**
		UFSAR Section	**Transient / Accident**		
3.3.1.1	3. Reactor Vessel Steam Dome Pressure - High	15.2.10	Loss of Stator Cooling	1) Automatic Initiation - -Reactor Vessel Steam Dome Pressure High -APRM Fixed Neutron Flux -High 2) Manual SCRAM	IMF-AOT
3.3.1.1	4. Reactor Vessel Level - Low (Level 3)	15.2.7	Loss of Feedwater Flow	1) Automatic Initiation - - Low Water Level (Level 3) - Main Steam Isolation Valve - Closure 2) Manual SCRAM	IMF-AOT
		15.2.8 / 15.6.6	Feedwater Line Break Outside Containment	1) Automatic Initiation - - Reactor Vessel Level – Low, Level 3 - Main Steam Isolation Valve - Closure 2) Manual SCRAM	DBA
		15.6.5	Loss of Coolant Accidents	1) Automatic Initiation - Reactor Vessel Level – Low, Level 3 - High Drywell Pressure 2) Manual SCRAM	DBA

RPS Instrumentation Diversity					
TS	RPS Instrument Function	Credited Safety Analysis Event		Diverse RPS Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.1.1	5. Main Steam Isolation Valve - Closure	15.2.4	Main Steam Isolation Valve - Closure	1) Automatic Initiation – - Main Steam Isolation Valve - Closure - Reactor Steam Dome Pressure - APRM Fixed Neutron Flux - High 2) Manual SCRAM	IMF-AOT
		15.6.4	Steam System Piping Break Outside Containment	1) Automatic Initiation – - Main Steam Isolation Valve - Closure 2) Manual SCRAM	DBA
		15.1.3	Pressure Regulator Failure	1) Automatic Initiation – - Main Steam Isolation Valve - Closure 2) Manual SCRAM	IMF-AOT
3.3.1.1	6. Drywell Pressure - High	15.6.5	Loss of Coolant Accidents	1) Automatic Initiation – - Drywell Pressure - High - Reactor Vessel Level – Low Level 3 2) Manual SCRAM	DBA

RPS Instrumentation Diversity					
TS	RPS Instrument Function	Credited Safety Analysis Event		Diverse RPS Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.1.1	7. SCRAM Discharge Volume Water Level - High				
	a. Transmitter / Trip Unit	None	None	1) Automatic Initiation – - SCRAM Discharge Volume Water Level - High 2) Manual SCRAM	-
	b. Float Switch	None	None	1) Automatic Initiation – - SCRAM Discharge Volume Water Level - High 2) Manual SCRAM	-
3.3.1.1	8. Turbine Stop Valve Closure	15.1.2A	Feedwater Controller Failure - Maximum Demand	1) Automatic Initiation – -TSV Closure Trip -Reactor Vessel Steam Dome Pressure High -APRM Fixed Neutron Flux - High 2) Manual SCRAM	IMF-AOT with bypass II-AOT without bypass
		15.1.3	Pressure Regulator Failure - Open	1) Automatic Initiation – -TSV Closure - Main Steam Isolation Valve - Closure -Reactor Vessel Steam Dome Pressure High -APRM Fixed Neutron Flux - High 2) Manual SCRAM	IMF-AOT
		15.2.3 15.2.3A	Turbine Trip	1) Automatic Initiation – -TSV Closure -Reactor Vessel Steam Dome Pressure High -APRM Fixed Neutron Flux - High 2) Manual SCRAM	IMF-AOT with bypass II-AOT without bypass

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

RPS Instrumentation Diversity					
TS	RPS Instrument Function	Credited Safety Analysis Event		Diverse RPS Instrumentation	Event
		UFSAR Section	Transient / Accident		
		15.2.5	Loss of Condenser Vacuum	1) Automatic Initiation – -TSV Closure - Main Steam Isolation Valve - Closure -Reactor Vessel Steam Dome Pressure High -APRM Fixed Neutron Flux - High 2) Manual SCRAM	IMF-AOT
		15.2.6	Loss of A-C Power	1) Automatic Initiation – - TSV Closure - Main Steam Isolation Valve - Closure - Reactor Vessel Steam Dome Pressure High 2) Manual SCRAM	II-AOT IMF-AOT
		15.3.1	Recirculation Pump Trip	1) Automatic Initiation – -TSV Closure - Main Steam Isolation Valve - Closure -Reactor Vessel Steam Dome Pressure High 2) Manual SCRAM	IMF-AOT
		15.3.2	Recirculation Flow Control Failure - Decreasing Flow	1) Automatic Initiation – -TSV Closure - Main Steam Isolation Valve - Closure -Reactor Vessel Steam Dome Pressure High 2) Manual SCRAM	IMF-AOT

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

<table>
<tr><th colspan="6">RPS Instrumentation Diversity</th></tr>
<tr><th rowspan="2">TS</th><th rowspan="2">RPS Instrument Function</th><th colspan="2">Credited Safety Analysis Event</th><th rowspan="2">Diverse RPS Instrumentation</th><th rowspan="2">Event</th></tr>
<tr><th>UFSAR Section</th><th>Transient / Accident</th></tr>
<tr><td></td><td></td><td>15.3.3</td><td>Recirculation Pump Seizure</td><td>1) Automatic Initiation –
-TSV Closure Main Steam Isolation Valve - Closure
-Reactor Vessel Steam Dome Pressure High
2) Manual SCRAM</td><td>DBA</td></tr>
<tr><td></td><td></td><td>15.2.9</td><td>Residual Heat Removal Shutdown Cooling</td><td>1) Automatic Initiation –
-TSV Closure
2) Manual SCRAM</td><td>IMF-AOT</td></tr>
<tr><td>3.3.1.1</td><td>9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low</td><td>15.2.2A</td><td>Generator Load Rejection</td><td>1) Automatic Initiation –
-Turbine Control Valve Fast Closure
-Reactor Vessel Steam Dome Pressure High
-APRM Fixed Neutron Flux - High
-Main Steam Isolation Valve – Closure
2) Manual SCRAM</td><td>IMF-AOT with bypass II-AOT without bypass</td></tr>
<tr><td></td><td></td><td>15.3.4</td><td>Recirculation Pump Shaft Break</td><td>1) Automatic Initiation –
-TCV Closure
2) Manual SCRAM</td><td>DBA</td></tr>
</table>

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Evaluation of Instrumentation and Control Systems

RPS Instrumentation Diversity					
TS	RPS Instrument Function	Credited Safety Analysis Event		Diverse RPS Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.1.1	10. Reactor Mode Switch - Shutdown Position	None	None	1) Manual SCRAM	-
3.3.1.1	11. Manual SCRAM	None	None	1) Manual SCRAM - Reactor Mode Switch to Shutdown Position	-

2. Feedwater System and Main Turbine High Water Level Trip

Reference: TS 3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation

The Feedwater System and Main Turbine High-Water Level Trip design creates defense-in-depth from the redundancy of the channels for the Trip Function.

- Trip Function has multiple channels.
- Trip Function will actuate with 2/3 tripped channels.

2/3 is defined as three channels with one trip system arranged in a two-out-of-three (energize to initiate) logic, e.g., (Channel A and B) or (Channel A and C) or (Channel B and C).

  - o Reactor Vessel Water Level (High - Level 8) - 2/3

Feedwater System and Main Turbine High Water Level Trip Instrumentation Diversity					
TS	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.2.2	Reactor Vessel Water Level - High (Level 8)	15.1.1	Loss of Feedwater Heating	1) Automatic Initiation -Reactor Vessel Water Level Turbine Trip 2) Manual Trip/Secure	IMF-AOT
		15.1.2A	Feedwater Controller Failure - Maximum Demand	1) Automatic Initiation -Reactor Vessel Water Level Turbine Trip - 2) Manual Trip/Secure	IMF-AOT with bypass II-AOT without bypass
		15.1.3	Pressure Regulator Failure - Open	1) Automatic Initiation -Reactor Vessel Water Level Turbine Trip - 2) Manual Trip/Secure	IMF-AOT
		15.2.2A	Generator Load Rejection	1) Automatic Initiation -Reactor Vessel Water Level Turbine Trip - 2) Manual Trip/Secure	IMF-AOT with bypass II-AOT without bypass
		15.2.3 15.2.3A	Turbine Trip	1) Automatic Initiation -Reactor Vessel Water Level Turbine Trip 2) Manual Trip/Secure	IMF-AOT with bypass II-AOT without bypass

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

Feedwater System and Main Turbine High Water Level Trip Instrumentation Diversity					
		Credited Safety Analysis Event			
TS	Instrument Function	UFSAR Section	Transient / Accident	Diverse Instrumentation	Event
		15.3.1	Recirculation Pump Trip	1) Automatic Initiation -Reactor Vessel Water Level Turbine Trip 2) Manual Trip/Secure	IMF-AOT
		15.3.2	Recirculation Flow Control Failure -Decreasing Flow	1) Automatic Initiation -Reactor Vessel Water Level Turbine Trip 2) Manual Trip/Secure	IMF-AOT
		15.3.3	Recirculation Pump Seizure	1) Automatic Initiation -Reactor Vessel Water Level Turbine Trip 2) Manual Trip/Secure	DBA
		15.3.4	Recirculation Pump Shaft Break	1) Automatic Initiation -Reactor Vessel Water Level Turbine Trip 2) Manual Trip/Secure	DBA

3. Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT)

Reference: TS 3.3.4.2 Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT) Instrumentation

The ATWS-RPT design creates defense-in-depth from the redundancy of the channels for the Trip Function.

- Trip Function has multiple channels.
- Trip Function will cause an Actuation with 1/2 taken twice tripped channels.
- A failed channel does cause or prevent a trip.

ATWS-RPT consists of two independent trip systems, with two channels of Reactor Steam Dome Pressure-High and two channels of Reactor Vessel Water Level – Low Low, Level 2, in each trip system. Each ATWS-RPT trip system is a one-out-of-two taken twice logic for each function. Either trip system will trip both recirculation pumps.

- o Reactor Vessel Water Level (Low Low - Level 2) – 1 of 2 channels taken twice
- o Reactor Vessel Steam Dome Pressure (High) - 1 of 2 channels taken twice

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT) Instrumentation Diversity					
TS	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.4.2	Reactor Vessel Water Level Low Low (Level 2)	15.8	Anticipated Transient Without SCRAM	1) Automatic Initiation - -ATWS-RPT 2) Manual RPT	ATWS
3.3.4.2	Reactor Vessel Steam Dome Pressure - High	15.8	Anticipated Transient Without SCRAM	1) Automatic Initiation - ATWS-RPT 2) Manual RPT	ATWS

4. Reactor Core Isolation Cooling (RCIC) System

Reference: TS 3.3.5.3 Reactor Core Isolation Cooling (RCIC) System Instrumentation

The RCIC design creates defense-in-depth because of the redundancy of the channels for the Initiation Function.

- Initiation Function has multiple channels.
- Initiation Function will cause an actuation with 1/2 and 2/4 tripped channels.
- A failed channel does not cause or prevent an initiation.

2/4 is defined as four channels, one trip system arranged in a one-out-of-two taken twice (energize to initiate) logic, e.g., (Channel A or Channel C) and (Channel E or Channel G).

2/2 is defined as two channels, one trip system arranged in a two-out-of-two taken once (energize to initiate) logic, e.g., (Channel A and Channel B).

1/2 is defined as two channels, one trip system arranged in a one-out-of-two taken once (energize to initiate) logic, e.g., (Channel A) or (Channel C).

- o Reactor Vessel Water Level (Low Low - Level 2) - 2/4
- o Reactor Vessel Water Level (High - Level 8) - 2/4
- o Condensate Storage Tank Level (Low) - 1/2
- o Manual Initiation - 1/1

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

Reactor Core Isolation Cooling (RCIC) Instrumentation Diversity					
TS	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.5.3	Reactor Vessel Water Level Low Low (Level 2)	15.1.2A	Feedwater Controller Failure - Maximum Demand	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	IMF-AOT with bypass II-AOT without bypass
		15.1.3	Pressure Regulator Failure - Open	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	IMF-AOT
		15.2.6	Loss of A-C Power	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	II-AOT IMF-AOT
		15.2.7	Loss of Feedwater Flow	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	IMF-AOT
		15.3.1	Recirculation Pump Trip	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	IMF-AOT
		15.3.2	Recirculation Flow Control Failure -Decreasing Flow	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	IMF-AOT
		15.3.3	Recirculation Pump Seizure	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	DBA
		15.2.8 / 15.6.6	Feedwater Line Break Outside Containment	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	DBA

Reactor Core Isolation Cooling (RCIC) Instrumentation Diversity					
TS	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.5.3	Reactor Vessel Water Level High (Level 8)	None	None	1) Automatic RCIC Stop - High Level (L8) 2) Manually Secure	-
3.3.5.3	Condensate Storage Tank Level - Low	None	None	1) Automatically Initiated (Swap Suction Source) 2) Manually Swap Suction Sources	-

5. Emergency Core Cooling System (ECCS)

Reference: TS 3.3.5.1 Emergency Core Cooling System Instrumentation

The ECCS design creates defense-in-depth from the redundancy of the channels for the Trip Function (ECCS System Actuation)

- Trip Function (ECCS Actuation) has multiple channels.
- Trip Function (ECCS Actuation) will cause a Trip Function with 2/2, 4/4, 2/4 or 1/2 tripped channels.
- A failed channel does cause or prevent a trip (except Manual).

2/2 is defined as two channels, one trip system arranged in a two-out-of-two taken once (energize to initiate) logic, e.g., (Channel A and Channel B).

2/4 is defined as four channels, one trip system arranged in a one-out-of-two taken twice (energize to initiate) logic, e.g., (Channel A or Channel C) and (Channel B or Channel D).

1/2 is defined as two channels, one trip system arranged in a one-out-of-two taken once (energize to initiate) logic, e.g., Channel A or Channel B).

- Low Pressure Coolant Injection (LPCI) - Loop A and Low Pressure Core Spray (LPCS) (LPCI - A and LPCS share a common logic)
  - o Reactor Vessel Water Level (Low Low Low - Level 1) - 2/4
    2 reactor vessel water level channels are combined with 2 drywell pressure channels
  - o Drywell Pressure (High) - 2/4
    2 drywell pressure channels are combined with 2 reactor vessel water level channels
  - o Manual Initiation – 1/1
- Low Pressure Coolant Injection (LPCI) - Loop B and Loop C (LPCI - B and LPCI - C share a common logic)
  - o Reactor Vessel Water Level (Low Low Low - Level 1) - 2/4
    2 reactor vessel water level channels are combined with 2 drywell pressure channels
  - o Drywell Pressure (High) - 2/4
    2 drywell pressure channels are combined with 2 reactor vessel water level channels
  - o Manual Initiation – 1/1
- High Pressure Core Spray (HPCS) System
  - o Reactor Vessel Water Level (Low Low - Level 2) - 2/4
  - o Drywell Pressure (High) - 2/4
  - o Reactor Vessel Water Level (High – Level 8) - 2/2
  - o Manual Initiation – 1/1
- Automatic Depressurization System (ADS) 2 Systems - A & B Either system A or B will cause all ADS valves to open.
  - o Reactor Vessel Water Level (Low Low Low - Level 1) - 2/2
  - o Drywell Pressure (High) – 2/4
  - o ADS Initiation Timer - 1/1
  - o Reactor Vessel Water Level (Low - Level 3 - Permissive) - 1/1
  - o LPCI-A and LPCS Pump Discharge Pressure (High) - 1/2 (System A only)
  - o LPCI-B and LPCS-C Discharge Pressure (High) - 1/2 (System B only)
  - o Manual Initiation – 2/2

  All contacts in one trip system must close to initiate ADS trip system.

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

Emergency Core Cooling System (ECCS) Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.5.1-1	1. Low Pressure Coolant Injection-A (LPCI) and Low Pressure Core Spray (LPCS) Subsystems				
	Reactor Vessel Water Level Low Low Low, Level 1	15.6.5	Loss of Coolant Accidents	1) Automatic Initiation - RX Water Vessel Low Low Low + LPCI / LPCS Time Delay Relay (Normal Or Emergency Power) + LPCI / LPCS Discharge Pressure Low 2) Manual Injection / Spray	DBA
	Drywell Pressure - High	15.6.5	Loss of Coolant Accidents	1) Automatic Initiation - Drywell Pressure High + LPCI / LPCS Time Delay Relay (Normal Or Emergency Power) + LPCI / LPCS Discharge Pressure Low 2) Manual Injection / Spray	DBA
	Manual Initiation	None	None	Manual Initiation	-
3.3.5.1-1	2. LPCI B and LPCI C Subsystems				
	Reactor Vessel Water Level Low Low Low, Level 1	15.6.5	Loss of Coolant Accidents	1) Automatic Initiation - RX Water Vessel Low Low Low + LPCI / LPCS Time Delay Relay (Normal Or Emergency Power) + LPCI / LPCS Discharge Pressure Low 2) Manual Injection / Spray	DBA
	Drywell Pressure - High	15.6.5	Loss of Coolant Accidents	1) Automatic Initiation - Drywell Pressure High + LPCI / LPCS Time Delay Relay (Normal Or Emergency Power) + LPCI / LPCS Discharge Pressure Low 2) Manual Injection / Spray	DBA
	Manual Initiation	None	None	1) Manual Initiation	-

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Evaluation of Instrumentation and Control Systems

Emergency Core Cooling System (ECCS) Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.5.1-1	3. High Pressure Core Spray (HPCS) System				
	Reactor Vessel Water Level - Low Low, Level 2	15.1.3	Pressure Regulator Failure - Open	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	IMF-AOT
		15.2.8 / 15.6.6	Feedwater Line Break	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	DBA
		15.2.7	Loss of Feedwater Flow	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	IMF-AOT
		15.3.1	Recirculation Pump Trip	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	IMF-AOT
		15.3.2	Recirculation Flow Control Failure -Decreasing Flow	1) Automatic Initiation - RX Water Level Low Low 2) Manual Initiation	IMF-AOT
		15.6.5	Loss of Coolant Accidents	1) Automatic Initiation - Drywell Pressure High 1) Manual Initiation	DBA
	Drywell Pressure - High	15.6.5	Loss of Coolant Accidents	2) Automatic Initiation - Drywell Pressure High 3) Manual Initiation	DBA

Emergency Core Cooling System (ECCS) Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event: UFSAR Section	Credited Safety Analysis Event: Transient / Accident	Diverse Instrumentation	Event
3.3.5.1-1	4. Automatic Depressurization System (ADS) Trip System A				
	Reactor Vessel Water Level - Low, Low Low, Level 1	6.3	Loss of Coolant Accidents	1) Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2) Manual Initiation	DBA
	ADS Initiation Timer	6.3	Loss of Coolant Accidents	1) Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2) Manual Initiation	DBA
	Reactor Vessel Water Level - Low, Level 3 (Permissive)	None	None	1) Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2) Manual Initiation	DBA
	LPCS Pump Discharge Pressure - High	6.3	Loss of Coolant Accidents	1) Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2) Manual Initiation	DBA
	LPCI Pump A Discharge Pressure - High	6.3	Loss of Coolant Accidents	1) Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2) Manual Initiation	DBA
	Manual Initiation	None	None	1) Manual Initiation	-

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

Emergency Core Cooling System (ECCS) Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.5.1-1	4. Automatic Depressurization System (ADS) Trip System B				
	Reactor Vessel Water Level - Low, Low Low, Level 1	6.3	Loss of Coolant Accidents	1) Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2) Manual Initiation	DBA
	ADS Initiation Timer	6.3	Loss of Coolant Accidents	1) Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2) Manual Initiation	DBA
	Reactor Vessel Water Level - Low, Level 3 (Permissive)	None	None	1) Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2) Manual Initiation	DBA
	LPCS Pump Discharge Pressure - High	6.3	Loss of Coolant Accidents	1) Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2) Manual Initiation	DBA
	LPCI Pump A Discharge Pressure - High	6.3	Loss of Coolant Accidents	1) Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2) Manual Initiation	DBA
	Manual Initiation	None	None	1) Manual Initiation	-

6. Primary Containment Isolation

Reference: TS 3.3.6.1 Primary Containment Isolation Instrumentation

The isolation actuation design creates defense-in-depth from the redundancy of the channels for each Trip Function.

- Each Trip Function has multiple channels
- Each Trip Function will cause an isolation actuation with 2/2, 2/4, or 1/1 tripped channels.
- A failed channel does not prevent a trip, but may cause a trip depending on the logic design

2/2 is defined as four channels, two trip systems, two channels per trip system arranged in a two-out-of-two taken once (de-energize to trip) logic, e.g., (Channel A and Channel B) or (Channel C and Channel D).

2/4 is defined as four channels, two trip systems, two channels per trip system arranged in a one-out-of-two taken twice (de-energize to trip) logic, e.g., (Channel A or Channel C) and (Channel C or Channel D).

1/1 is defined as two channels, two trip systems, one channel per trip system arranged in a one-out-of-one taken once (de-energize to trip) logic, e.g., (Channel A) or (Channel D).

- Main Steam Line Isolation
    - Reactor Vessel Water Level (Low Low Low - Level 1) - 2/4*
    - Main Steam Line Pressure (Low) - 2/4*
    - Main Steam Line Flow (High) - This instrumentation uses 16 flow channels, four for each steam line. One channel from each steam line inputs to one of four trip strings. Two trip strings make up each trip system, and both trip systems must trip to cause an MSL isolation. Each trip string has four inputs (one per MSL), any one of which will trip the trip string. The trip strings within a trip system are arranged in a one-out-of-two logic. Therefore, this is effectively a one-out-of-eight taken twice logic arrangement to initiate isolation of the MSIVs.**
    - Condenser Vacuum (Low) - 2/4*
    - Main Steam Line Tunnel Differential Temperature (High) - 2/4*
    - Manual Initiation - 2/4***

    * The outputs from the same channels are arranged into two 2/2 trip systems to isolate all MSL drain valves. One 2/2 trip system is associated with the inboard valve and the other 2/2 trip system is associated with the outboard valves.

    ** The outputs from the 16 flow channels are connected into two two-out-of-two trip systems (effectively two one-out-of-four taken twice logic), with one trip system isolating the inboard MSL drain valve and the other trip system isolating the outboard MSL drain valves.

    *** To close the MSL drain valves, all channels in both trip system must actuate (i.e., both channels from each of the two associated switch and push buttons are required to actuate the inboard valve trip system and both channels from each of the two associated switch and push buttons are required to actuate the outboard valve trip system).

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

- Primary Containment Isolation
  - Reactor Vessel Water Level (Low Low - Level 2) - 2/2*
  - Drywell Pressure (High) - 2/2*
  - Reactor Building Ventilation Exhaust Plenum Radiation (High) -2/2*
  - Fuel Pool Ventilation Exhaust Radiation (High) -2/2*
  - Reactor Vessel Water Level (Low Low Low, Level 1) -2/2*
  - Reactor Vessel Water Level (Low, Level 3) – 2/2*
  - Manual Initiation – 2/2*

  * The outputs from the same channels are arranged into two 2/2 trip systems to isolate all penetrations. One 2/2 trip system is associated with all automatic inboard PCIVs and the other 2/2 trip system is associated with all automatic outboard PCIVs.

- Reactor Core Isolation Cooling (RCIC) System Isolation
  - RCIC Steam Line Flow (High) - 1/1
  - RCIC Steam Line Flow (Timer) - 1/1
  - RCIC Steam Supply Pressure (Low) – 2/2*
  - RCIC Turbine Exhaust Diaphragm Pressure (High) – 2/2*
  - RCIC Equipment Room Temperature (High) –
  - RCIC Steam Line Tunnel Temperature (High) - 1/1
  - RCIC Steam Line Tunnel Differential Temperature (High) – 1/1
  - Drywell Pressure (High) – 2/4**
  - Manual initiation - 1/1 with 1 channel***

  * The functions receive input from four steam supply pressure channels and four turbine exhaust diaphragm pressure channels. The outputs from the channels are arranged into two 2/2 trip systems, each trip system isolating the inboard or outboard RCIC steam valves.

  **The Drywell Pressure function receives inputs from four drywell pressure channels. The outputs for these channels feed into two one-out-of-two trip systems with coincident RCIC Steam Supply Pressure

  ***The manual initiation contains one channel that isolates the outboard RCIC steam valve only (provided an automatic initiation signal is present).

- Reactor Water Cleanup (RWCU) System Isolation
  - Differential Flow (High) - 1/1 two channels
  - Differential Flow (Timer) – 1/1
  - RWCU Heat Exchanger Area Temperature (High) – 1/1 with 4 channels
  - RWCU Heat Exchanger Area Ventilation Differential Temperature (High) – 1/1 with 4 channels
  - RWCU Pump and Valve Area Temperature (High) – 1/1 with 6 channels
  - RWCU Pump and Valve Area Differential Temperature (High) – 1/1 with 6 channels
  - RWCU Holdup Pipe Area Temperature (High) – 1/1 with 4 channels
  - RWCU Holdup Pipe Area Ventilation Differential Temperature (High) – 1/1 with 4 channels
  - RWCU Filter / Demineralizer Valve Room Area Temperature (High) – 1/1 with 4 channels
  - RWCU Filter / Demineralizer Valve Room Area Ventilation Differential Temperature (High) – 1/1 with 4 channels

- Reactor Vessel Water Level (Low Low Level 2) – 2/4
- Standby Liquid Control System Initiation – 1/2
- Manual Initiation – 1/1

- RHR SDC System Isolation
  - Reactor Vessel Water Level (Low - Level 3) - 2/4
  - Reactor Vessel Pressure (high) – 2/2
  - Manual Initiation – 2/2

Primary Containment Isolation Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.6.1-1	Main Steam Line Isolation				
	Reactor Vessel Water Level - Low Low Low, Level 1	15.6.4	Steam System Piping Break Outside Containment	1) Automatic Initiation - RX Water Level Low Low Low 2) Manual Isolation	DBA
		15.6.5	Loss of Coolant Accidents	1) Automatic Initiation - RX Water Level Low Low Low 2) Manual Isolation	DBA
	Main Steam Line Pressure - Low	15.1.3	Pressure Regulator Failure - Open	1) Automatic Initiation - MSL Pressure Low 2) Manual Isolation	IMF-AOT
	Main Steam Flow - High	15.6.4	Steam System Pipe Break Outside Containment	1) Automatic Initiation - MSL Flow High 2) Manual Isolation	DBA
	Condenser Vacuum - Low	15.2.5	Loss of Condenser Vacuum	1) Automatic Initiation - Condenser Vacuum Low 2) Manual Isolation	IMF-AOT
	Main Steam Tunnel Differential Temperature - High	None	None	1) Automatic Initiation - MSL Differential Temp High 2) Manual Isolation	-
	Manual Initiation	None	None	1) Manual Initiation	-
3.3.6.1-1	Primary Containment Isolation				
	Reactor Vessel Water Level - Low Low (Level 2)	None	None	1) Automatic Initiation - RX Water Level Low Low 1) Manual Isolation	-

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

Primary Containment Isolation Instrumentation Diversity					
		Credited Safety Analysis Event			
TS Table	Instrument Function	UFSAR Section	Transient / Accident	Diverse Instrumentation	Event
-	Drywell Pressure - High	None	None	1) Automatic Initiation - Drywell Pressure High 2) Manual Isolation	-
	Reactor Building Ventilation Exhaust Plenum Radiation – High	None	None	1) Automatic Initiation - High Radiation 2) Manual Isolation	-
	Fuel Pool Ventilation Exhaust Radiation - High	None	None	1) Automatic Initiation - High Radiation 2) Manual Isolation	-
	Reactor Vessel Water Level – Low Low Low, Level 1	None	None	1) Automatic Initiation - RX Water Level Low Low Low 2) Manual Isolation	-
	Reactor Vessel Water Level – Low, Level 3	None	None	1) Automatic Initiation - RX Water Level Low 2) Manual Isolation	-
	Manual Initiation	None	None	1) Manual Initiation	-

Primary Containment Isolation Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.6.1-1	Reactor Core Isolation Cooling (RCIC) System Isolation				
	RCIC Steam Line Flow - High	None	None	1) Automatic Initiation - RCIC Steam Flow High + RCIC Steam Line Flow Timer 2) Manual Isolation	-
	RCIC Steam Line Flow - Timer	None	None	1) Automatic Initiation - RCIC Steam Flow High + RCIC Steam Line Flow Timer 2) Manual Isolation	-
	RCIC Steam Supply Pressure - Low	None	None	1) Automatic Initiation - RCIC Steam Supply Pressure Low 2) Manual Isolation	-
	RCIC Turbine Exhaust Diaphragm Pressure - High	None	None	1) Automatic I Initiation - RCIC Turbine Exhaust Diaphragm Pressure High 2) Manual Isolation	-
	RCIC Equipment Room Temperature - High	None	None	1) Automatic Initiation - RCIC Area Temperature High + Area Temperature Time Delay 2) Manual Isolation	-
	RCIC Equipment Room Differential Temperature - High	None	None	1) Automatic Initiation - RCIC Equipment Room Area Temperature High +Area Temperature Time Delay 2) Manual Isolation	-

Primary Containment Isolation Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
	RCIC Steam Line Tunnel Temperature - High	None	None	1) Automatic Initiation - RB Pipe Chase Area Temperature High +Area Temperature Time Delay 2) Manual Isolation	-
	RCIC Steam Line Tunnel Differential Temperature - High	None	None	1) Automatic Initiation - RB General Area Temperature High +Area Temperature Time Delay 2) Manual Isolation	-
	Drywell Pressure - High	None	None	1) Automatic Initiation - RCIC / RHR Steam Flow High 2) Manual Isolation	-
	Manual Initiation	None	None	1) Manual Initiation	-
3.3.6.1-1	Reactor Water Cleanup (RWCU) System Isolation				
	Differential Flow - High	None	None	1) Automatic Initiation - RX Water Level Low Low 2) Manual Isolation	-
	Differential Flow - Timer	None	None	1) Automatic Initiation - RWCU Differential Flow High and Differential Flow Timer 2) Manual Isolation	-
	RWCU Heat Exchanger Area Temperature - High	None	None	1) Automatic Initiation - RWCU Heat Exchanger Room Area Temperature High and Area Temperature Time Delay 2) Manual Isolation	-
	RWCU Heat Exchanger Area Ventilation Differential Temperature - High	None	None	1) Automatic Initiation - RWCU Heat Exchanger Room Area Temperature 2) Manual Isolation	-

Primary Containment Isolation Instrumentation Diversity					
		Credited Safety Analysis Event			
TS Table	Instrument Function	UFSAR Section	Transient / Accident	Diverse Instrumentation	Event
	RWCU Pump and Valve Area Temperature - High	None	None	1) Automatic Initiation - RWCU Pump and Valve Area Temperature High and Area Temperature Time Delay 2) Manual Isolation	-
	RWCU Holdup Pipe Area Ventilation Differential Temperature – High	None	None	1) Automatic Initiation - Holdup Pipe Area Temperature 2) Manual Isolation	-
	RWCU Filter / Demineralizer Valve Room Area Temperature – High	None	None	1) Automatic Initiation - Demineralizer Valve Room Area Temperature High and Area Temperature Time Delay 2) Manual Isolation	-
	RWCU Filter / Demineralizer Valve Room Area Ventilation Differential Temperature – High	None	None	1) Automatic Initiation - Demineralizer Valve Room Area Temperature 2) Manual Isolation	-
	Reactor Vessel Water Level – Low Low, Level 2	None	None	1) Automatic Initiation - RX Water Level Low 2) Manual Isolation	-
	Standby Liquid Control System Initiation	None	None	1) Manual Initiation	DBA
	Manual Initiation	None	None	1) Manual Initiation	-

Primary Containment Isolation Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		UFSAR Section	Transient / Accident		
3.3.6.1-1	RHR Shutdown Cooling System Isolation				
	Reactor Vessel Water Level - Low (Level 3)	None	None	1) Automatic Initiation - RX Water Level Low 2) Manual Isolation	-
	Reactor Vessel Pressure High	None	None	1) Automatic Initiation - RX Vessel Pressure High 2) Manual Isolation	-
	Manual Initiation	None	None	1) Manual Initiation	-

ATTACHMENT 7
Evaluation of Instrumentation and Control Systems

7. Loss-of-Power (LOP)

Reference: TS 3.3.8.1 Loss-of-Power (LOP) Instrumentation

The LOP design creates defense-in-depth from the redundancy of the channels for the Initiation Function.

- A failed channel does not cause or prevent an initiation.

Each 4.16 kV emergency bus has its own independent LOP instrumentation and associated trip logic. The voltage for Division 1, 2, and 3, 4.16 kV buses is monitored at two levels which can be considered as two different undervoltage functions: loss of voltage and degraded voltage.

For Division 1 and 2, each Loss of Voltage and Degraded Voltage function is monitored by two instruments per bus whose output trip contacts are arranged in a two-out-of-two logic configuration per bus. The Loss of Voltage signal is generated when a loss of voltage occurs for a specific time interval. Lower voltage conditions will result in decreased trip time for the inverse time undervoltage relays. The Degraded Voltage signal is generated when a degraded voltage occurs for a specific time interval; the time interval is dependent upon whether a loss of coolant accident signal is present. The relays utilized are inverse time delay voltage relays or instantaneous voltage relays with a time delay.

For Division 3, the degraded voltage function logic is the same as Divisions 1 and 2, but the Division 3 loss of voltage function logic is different, The Division 3 DG will auto-start if either one of the two bus undervoltage relays (with a time delay) actuates and the DG output breaker will automatically close with the same undervoltage permissive provided the Division 3 bus main feeder breaker is open and the DG speed and voltage permissives are met. The Division 3 bus main feed breaker trip logic included two trip systems. Each trip system consists of an undervoltage relay on the 4.16kV bus (with time delay) and an undervoltage relay on the system auxiliary transformer (SAT) side of the main feed breaker to the 4.16kV bus (with no time delay) arranged in two-out-of-two logic. The trip setting of the SAT undervoltage relay is maintained such that it trips prior to the bus undervoltage relay. Either trip system will open (trip) the main feeder to the bus.

Accident analyses credit the loading of at least two of the DGs based on the loss-of-offsite power coincident with a loss of coolant accident (LOCA).

- Two channels of each 4.16kV Emergency Bus Undervoltage (Loss of Voltage) Function per associated emergency bus are required to be operable when the associated DG is required to be operable. This ensures that no single instrument failure can preclude the DG function.
- Two channels of each 4.16kV Emergency Bus Undervoltage (Degraded Voltage) Function per associated emergency bus are required to be operable when the associated DG is required to be operable. This ensures that no single instrument failure can preclude the DG function.

Regulatory Guide 1.174, Revision 2, Section 2.1.1 – Defense-in-Depth

Defense-in-depth consists of several elements and consistency with the defense-in-depth philosophy is maintained if the following occurs:

- **A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.**
  - Current Technical Specifications (TS) reflect this balance by allowing one sensor module or channel to be placed in trip, while preserving the fundamental safety function of the applicable system. Tripping an inoperable channel does not affect the number of channels required to provide the safety function. Even in the TS condition for two channels in a function inoperable, the fundamental safety function is preserved since sufficient operable channels remain in the function.

- **Over-reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided.**
  - No programmatic activities are relied upon as compensatory measures when one or two channels of the applicable instrumentation are inoperable. The remaining operable channels for that function are fully capable of performing the safety function of the applicable system.

- **System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).**
  - System redundancy, independence and diversity remain the same as in the as-designed condition. The number of operable functions has not been decreased (diversity), the number of minimum operable channels to perform the safety function has not been decreased, and the channels remain independent as originally designed, even with one channel inoperable.

- **Defenses against potential common-cause failures are preserved, and the potential for the introduction of new common-cause failure mechanisms is assessed.**
  - This LAR does not impact the original determination of common-cause failure for the applicable instrumentation and its functions. It may allow the allowed outage time to be extended for one or two channels in a function to be inoperable prior to placing the channel in trip. Placing the channel in trip fulfils one of the two required channels in trip needed to perform the safety function.

- **Independence of barriers is not degraded.**
  - Barriers are not affected by this LAR request.

- **Defenses against human errors are preserved.**
  - In the conditions listed in the TS, a potential extension of the allowed outage time does not change any personnel actions required when the TS Action is entered. Therefore, no change to the possibility of a human error is introduced and no change to the defenses against that potential human error have been altered.

- **The intent of the plant's design criteria is maintained.**
  - The design criteria of the applicable systems are maintained as reflected in the Updated Safety Analysis Report (UFSAR). Redundancy, diversity of signal and independence of trip channel functions are maintained with the requested change. The change requested in the LAR does not physically change the applicable systems in any way. It only allows additional time, under certain low risk conditions in accordance with the Risk Informed Completion Time (RICT) Program, to perform actions that the NRC has previously determined to be acceptable.

Therefore, the defense-in-depth principals prescribed in Regulatory Guide 1.174, Revision 2, are met.

ENCLOSURE 1

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

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

ENCLOSURE 1
List of Revised Required Actions to Corresponding PRA Functions

1. Introduction

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

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

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

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

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

ENCLOSURE 1
List of Revised Required Actions to Corresponding PRA Functions

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

Examples of calculated RICT are provided in Table E1-2 for each individual condition to which the RICT applies (assuming no other SSCs modeled in the PRA are unavailable). These example calculations demonstrate the scope of the SSCs covered by TSs modeled in the PRA. Due to the close similarity between Unit 1 and Unit 2, only the Unit 2 RICT results are shown in Table E1-2. Also note that the more limiting of the CDF and LERF RICT result is shown. Following 4b implementation, the actual RICT values will be calculated on a unit-specific basis, using the actual plant configuration and the current revision of the PRA model representing the as-built, as-operated condition of the plant, as required by NEI 06-09-A, Revision 0 and the NRC safety evaluation, and may differ from the RICTs presented.

Table E1-3 lists the TSTF-505 Rev 2 Table 1 Tech Specs that require additional justification along with a description of how the additional justification is provided in the LAR.

ENCLOSURE 1
List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
LSCS Tech Spec	**Tech Spec Description**	**SSCs Covered by TS LCO Condition**	**Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.1.7.A	One SLC subsystem inoperable	SLC trains	Yes	SLC injection capability	One of two trains	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.3.1.1.A	Reactor Protection System (RPS) instrumentation - one or more required channels inoperable	Instrumentation outlined in table 3.3.1.1-1 (see Note 1)	Not explicitly	Reactor Trip Initiation	One of two channels, taken twice	Same as Design Success Criteria	Individual RPS instrumentation inputs to the RPS logic system are not modeled in the PRA. A surrogate is chosen and it represents the common cause failure of the RPS electrical system. This is conservative and represents failure of the RPS. This event covers both Condition A and Condition B of TS 3.3.1.1.
3.3.1.1.B	Reactor Protection System (RPS) instrumentation - one or more functions with one or more required channels inoperable in both trip systems	See 3.3.1.1.A					

ENCLOSURE 1
List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
LSCS Tech Spec	**Tech Spec Description**	**SSCs Covered by TS LCO Condition**	**Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.3.2.2.A	One or more feedwater system and main turbine high water level trip channels inoperable	Feedwater system and main turbine trip instrumentation	No	The trip of two feedwater pump turbines and main turbine	Two of three subsystems	N/A	Surrogate mapping was required since the function in TS 3.3.2.2.A was not modeled. The PRA model will be updated to include this function prior to exercising the RICT program for this TS.
3.3.4.2.A	Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT) Instrumentation - one or more channels inoperable	The ATWS-RPT System includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to cause initiation of a recirculation pump trip. (see Note 2)	Yes	Recirculation Pump Trip	One of two channels, taken twice	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.3.5.1.B	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1	ECCS actuation instrumentation for low pressure core spray (LPCS), low pressure coolant injection (LPCI) and high pressure core spray (HPCS) (See Notes 3, 4)	Yes	Initiate ECCS (HPCS, LPCS, and LPCI)	One of two channels, taken twice for LPCI, LPCS, and HPCS	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis

ENCLOSURE 1
List of Revised Required Actions to Corresponding PRA Functions

3.3.5.1.C	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1	ECCS actuation instrumentation for low pressure core spray (LPCS), low pressure coolant injection (LPCI) and high pressure core spray (HPCS) (See Notes 3, 4)	Not explicitly	Initiate ECCS (HPCS, LPCS, and LPCI)	One of two channels, taken twice for LPCI, LPCS, and HPCS	Same as Design Success Criteria	The purpose of this time delay is to stagger the start of the two ECCS pumps that are in each of Divisions 1 and 2, thus limiting the starting transients on the 4.16 kV emergency buses. This Function is only necessary when power is being supplied from the standby power sources (DG). The failure of a LPCI Pump Start—Time Delay Relay could result in the failure of the two low pressure ECCS pumps, powered from the emergency bus, to perform their intended function and is assumed to potentially overload the DG and thus fail the DG. Therefore, for this mapping the time delay relay is mapped to failure of the LPCS pump and LPCI A pump and the Division 1 DG. This is conservative mapping given that the time delay relay itself is not modeled in the PRA.

ENCLOSURE 1
List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
LSCS Tech Spec	**Tech Spec Description**	**SSCs Covered by TS LCO Condition**	**Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1	ECCS actuation instrumentation for low pressure core spray (LPCS), low pressure coolant injection (LPCI) and high pressure core spray (HPCS) (See Notes 3, 4)	Not explicitly	Initiate ECCS (HPCS, LPCS, and LPCI)	One of two channels, taken twice for LPCI, LPCS, and HPCS	Same as Design Success Criteria	This fail to open of the min flow valve is not modeled in the PRA nor is the automatic instrumentation associated with this design basis function. This function is assumed in design basis needed to protect the pump. Therefore, a surrogate is used and it is mapped to the frontline system (i.e., the LPCS pump fail to run).
3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1	ADS initiation logic and instrumentation (See Notes 3, 4)	Yes	Initiate ADS	One of two trains	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis

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List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
LSCS Tech Spec	**Tech Spec Description**	**SSCs Covered by TS LCO Condition**	**Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.3.5.1.F	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1	ADS initiation logic and instrumentation	Yes	Initiate ADS	One of two trains	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.3.5.3.B	Reactor Core Isolation Cooling (RCIC) System Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.3-1	Reactor Vessel Water Level-Low Low, Level 2	Yes	RCIC initiation	Two of four channels, two channels per division	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.3.5.3.D	Reactor Core Isolation Cooling (RCIC) System Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.3-1	CY Condensate Storage Tank Level Sensors	Yes	Initiate swap suction source from the CST to the Suppression Pool	One of two channels	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis

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List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
LSCS Tech Spec	**Tech Spec Description**	**SSCs Covered by TS LCO Condition**	**Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.3.6.1.A	Primary containment instrumentation - one or more channels inoperable	Sensors, relays and switches that are necessary to cause initiation	Not explicitly	Automatic isolation of Primary Containment Isolation valves	Logic strings depend based on subsystem logic structure	Same as Design Success Criteria	The logic for primary containment isolation is not modeled in detail. Therefore, a surrogate is chosen that represents either a failure of containment or failure of the frontline system.
3.3.8.1.A	Loss of Power (LOP) instrumentation - one or more channels inoperable	The LOP System includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to trip offsite power circuits and start the emergency diesel generators. (See Note 5)	Not explicitly	Undervoltage sensing capability	Two of two logic per division	Same as Design Success Criteria	Individual instrument channels for loss of power instrumentation is not modeled. Therefore, a surrogate relay is chosen that fails the DG start mode or undervoltage relay.
3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable	Three LPCI trains and one LPCS train	Yes	Low pressure injection into the RPV	Two of four subsystems	One of four subsystems (See Note 8)	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

ENCLOSURE 1
List of Revised Required Actions to Corresponding PRA Functions

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

LSCS Tech Spec	Tech Spec Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.5.1.B	High Pressure Core Spray (HPCS) System inoperable	HPCS components	Yes	High pressure injection into the RPV	One of one train	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.5.1.C	Two low pressure ECCS injection/spray subsystems inoperable	Three LPCI trains and one LPCS train	Yes	Low pressure injection into the RPV	Two of four subsystems	One of four subsystems (See Note 8)	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.5.1.E	One required ADS valve inoperable	ADS valves and supporting components	Yes	Vessel depressurization	Five of seven ADS valves	Two of seven ADS valves (Reference 4)	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis

ENCLOSURE 1
List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
LSCS Tech Spec	**Tech Spec Description**	**SSCs Covered by TS LCO Condition**	**Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.5.3.A	RCIC System inoperable	RCIC components	Yes	Supply high pressure makeup water to the RPV.	One of one train	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.6.1.2.C	Primary containment air lock	Primary containment air lock equipment	No	Primary Containment boundary maintained	One of two doors maintain boundary	N/A	See Note 7. The airlocks are not modeled so a large pre-existing leak failure will be used as a conservative surrogate for the RICT calculation.
3.6.1.3.A	One or more penetration flow paths with one PCIV inoperable for reasons other than Condition D	Primary Containment Isolation Valves	Not explicitly	To limit fission product release during and following postulated Design Basis Accident (DBAs) to within limits	One of two isolation valves per penetration	Same as Design Success Criteria	Not all primary containment isolation valves are modeled. Therefore, a surrogate of a pre-existing containment failure is chosen.

ENCLOSURE 1
List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
LSCS Tech Spec	**Tech Spec Description**	**SSCs Covered by TS LCO Condition**	**Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.6.1.6.A	One suppression chamber-to-drywell vacuum breaker inoperable for opening	Four lines with two vacuum breakers in series per line	Not explicitly	Relieve vacuum in the drywell	Three of four required	Same as Design Success Criteria	The opening function of the suppression chamber to drywell vacuum breakers is not modeled in the PRA. The vapor suppression function is modeled and is used as a surrogate here.
3.6.2.3.A	One RHR suppression pool cooling subsystem inoperable	RHR pumps, valves and heat exchangers (Seen Note 6)	Yes	Removal of heat from the Suppression Pool	One of two trains	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.6.2.4.A	One RHR suppression pool spray subsystem inoperable	RHR pumps, valves and heat exchangers	Yes	Removal of heat from the Suppression Pool and suppression pool airspace	One of two trains	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis

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List of Revised Required Actions to Corresponding PRA Functions

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

LSCS Tech Spec	Tech Spec Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.7.1.A	One RHRSW subsystem inoperable	RHR pumps, valves and heat exchangers	Yes	To provide cooling water for the Residual Heat Removal (RHR) System heat exchangers	One of two trains	One of two trains	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.8.1.A	One required offsite circuit inoperable	The Unit Station Auxiliary Transformer (SAT) and the opposite unit SAT and associated breakers and offsite power supplies	Yes	Supply AC loads during normal operation	One offsite source	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.8.1.B	One required Division 1 or 2 DG inoperable OR required opposite unit Division 2 DG inoperable	EDGs and their support systems	Yes	Supply AC loads during abnormal operation	One diesel per division	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
LSCS Tech Spec	**Tech Spec Description**	**SSCs Covered by TS LCO Condition**	**Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.8.1.C	Required Division 3 DG inoperable OR One required Division 1, 2, or 3 DG inoperable and the required opposite unit Division 2 DG inoperable	See 3.8.1.A and 3.8.1.B					
3.8.1.D	Two required offsite circuits inoperable						
3.8.1.E	One required offsite circuit inoperable AND one required Division 1, 2 or 3 DG inoperable						
3.8.4.A	One required Division 1, 2, or 3 125 VDC battery charger on one division inoperable OR One required Division 2 or opposite unit Division 2 battery charger on one division inoperable OR One required Division 1 250 VDC battery charger inoperable	Battery chargers	Yes	To provide DC loads during normal operation	One per required division	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis

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List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
LSCS Tech Spec	**Tech Spec Description**	**SSCs Covered by TS LCO Condition**	**Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.8.4.B	Division 1 or 2 125 VDC electrical power subsystem inoperable for reasons other than Condition A	The Division 1 and 2 125 VDC batteries and interconnecting cabling	Yes	To provide DC loads during normal operation	One of two subsystems	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.8.4.E	Required Action and associated Completion Time of Condition A not met for the opposite unit Division 2 DC electrical power subsystem OR Opposite unit Division 2 DC electrical power subsystem inoperable for reasons other than Condition A	The Division 2 125 VDC batteries and interconnecting cabling	Yes	To provide DC loads during normal operation	One subsystem	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.8.7.A	One or both Division 1 and 2 AC electrical power distribution subsystem inoperable	4.16kV buses, 600V load centers and distribution panels, and 120V panels	Yes	AC power distribution to the required Divisional Loads	One subsystem	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis

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List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
LSCS Tech Spec	**Tech Spec Description**	**SSCs Covered by TS LCO Condition**	**Modeled in PRA**	**Function Covered by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Comments**
3.8.7.B	One or both Division 1 and 2 125 VDC electrical power distribution subsystems are inoperable	Two divisions of DC distribution	Yes	DC power distribution to the required Divisional Loads	One subsystem	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis
3.8.7.D	One or more required opposite unit Division 2 AC or DC electrical power distribution subsystems inoperable	The opposite unit Division 2 125 VDC batteries and interconnecting cabling and the opposite unit 4.16kV buses, 480V load centers, distribution panels, and 120V panels	Yes	AC and DC power distribution to the required Divisional Loads	One subsystem	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis

Notes:

1. The reactor protection system is made up of two independent trip systems (A and B). Each trip system contains 2 logic channels (A1, A2 and B1, B2). The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor SCRAM. Each channel contains the various functional inputs to RPS such Reactor level, MSIV closure, etc. Loss of any functional input does not prevent the channel from responding to other inputs. Use of an electrical SCRAM failure as a surrogate for a non-modeled functional input is conservative as it encompasses loss of all the inputs to all channels rather than any single input to a channel A.

2. ATWS-RPT system instrumentation is part of the redundant reactivity control system and has 2 independent trip systems each composed of two channels of each functional input. Each trip system uses a 2-out-of-2 logic for each function. Thus, either two Reactor Water Level – Low Low, Level 2 or two Reactor Vessel Steam Dome Pressure – High signals are needed to trip a trip system. Either trip system will trip both recirculation pump fast speed breakers.

3. The control logic for LPCS automatic initiation occurs for low reactor water level or high drywell pressure. Reactor Vessel water level – Low Low Low (Level 1) is sensed by two trip units in the Reactor Instrumentation System. Drywell pressure – High signals are sent from the Reactor Instrumentation System (RIS) to two high drywell pressure relay contacts. The trip unit outputs are in a two-out-of-two logic. The outputs of the four trip units (two trip units from each of the two variables) are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for automatic initiation.
   Automatic initiation of LPCI occurs for conditions of Reactor Vessel Water Level – Low Low Low, Level 1 or Drywell Pressure – High. Reactor vessel water level is monitored by two redundant differential pressure transmitters per division and drywell pressure is monitored by two redundant pressure transmitters per division, each providing input to a trip unit. The outputs of the four Division 2 LPCI (loops B and C) trip units (two trip units from each of the two variables) are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic. The Division 1 LPCI (loop A) receives its initiation signal from the LPCS logic, which uses a similar one-out-of-two taken twice logic.
   Automatic initiation of HPCS occurs for conditions of Reactor Vessel Water Level – Low Low, Level 2 or Drywell Pressure – High. Reactor vessel water level is monitored by four redundant differential pressure transmitters and drywell pressure is monitored by four redundant pressure transmitters, each providing input to a trip unit. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each variable.

4. Individual pieces of instrumentation such as a pressure transmitter may be shared by multiple design basis functions.

5. Each 4.16 kV emergency bus has its own independent LOP instrumentation and associated trip logic. The voltage for the Division 1, 2, and 3 buses is monitored at two levels, which can be considered as two different undervoltage functions: loss of voltage and degraded voltage. For Division 1 and 2, each loss of voltage and degraded voltage function is monitored by two instruments per bus whose output trip contacts are arranged in a two-out-of-two logic configuration per bus. The loss of voltage signal is generated when a loss of voltage occurs for a specific time interval. Lower voltage conditions will result in decreased trip times for the inverse time undervoltage relays. The degraded voltage signal is generated when a degraded voltage occurs for a specified time interval; the time interval is dependent upon whether a loss of coolant accident signal is present. The relays utilized are inverse time delay voltage relays or instantaneous voltage relays with a time delay.

6. The RHR system contains three separate pump trains, two of which contain heat exchangers for heat removal. The third pump train is for LPCI functions only.

7. This condition will be modeled as early containment bypass as a conservative surrogate in the PRA. Compliance with the remaining portions of LCO Condition 3.6.1.2 ensure that there is a physical barrier (i.e., closed door) and an acceptable overall leakage from containment. Thus, the function is still maintained. Required Action C.1 of LCO Condition 3.6.1.2 requires the condition to be assessed in accordance with TS 3.6.1.1, "Primary Containment" (i.e., "Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.1" with a Completion Time of Immediately.)

8. The success criteria for the ability of the Low Pressure ECCS systems to provide inventory makeup are based on NEDO-24708A (Figures 3.5.2.1-5.1 to 5.8) and confirmed with additional MAAP 4.0.5 calculations (LS14017 and LS14017A).

Table E1-2: Example RICT Calculations

Tech Spec	LCO Condition	RICT Estimate[1] (Days)
3.1.7.A	One SLC subsystem inoperable	30.0
3.3.1.1.A	Reactor Protection System (RPS) instrumentation - one or more required channels inoperable	0.4 [2]
3.3.1.1.B	Reactor Protection System (RPS) instrumentation - one or more functions with one or more required channels inoperable in both trip systems	0.4 [2]
3.3.2.2.A	One or more feedwater system and main turbine high water level trip channels inoperable	30.0 [3]
3.3.4.2.A	Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT) Instrumentation - one or more channels inoperable	30.0
3.3.5.1.B	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	30.0
3.3.5.1.C	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	26.6
3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	30.0
3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	30.0
3.3.5.1.F	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	30.0
3.3.5.3.B	Reactor Core Isolation Cooling (RCIC) System Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	30.0
3.3.5.3.D	Reactor Core Isolation Cooling (RCIC) System Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	30.0
3.3.6.1.A	Primary containment instrumentation - one or more channels inoperable	30.0
3.3.8.1.A	Loss of Power (LOP) instrumentation - one or more channels inoperable	30.0
3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable	30.0
3.5.1.B	High Pressure Core Spray (HPCS) System inoperable.	30.0
3.5.1.C	Two low pressure ECCS injection/spray subsystems inoperable.	11.7
3.5.1.E	One required ADS valve inoperable	30.0
3.5.3.A	RCIC System inoperable	30.0
3.6.1.2.C	Primary containment air lock	30.0
3.6.1.3.A	One or more penetration flow paths with one PCIV inoperable for reasons other than Condition D	30.0
3.6.1.6.A	One suppression chamber-to-drywell vacuum breaker inoperable for opening	3.3[2]
3.6.2.3.A	One RHR suppression pool cooling subsystem inoperable	30.0
3.6.2.4.A	One RHR suppression pool spray subsystem inoperable	30.0
3.7.1.A	One RHRSW subsystem inoperable	30.0
3.8.1.A	One required offsite circuit inoperable	30.0
3.8.1.B	One required Division 1 or 2 DG inoperable OR required opposite unit Division 2 DG inoperable.	30.0
3.8.1.C	Required Division 3 DG inoperable OR One required Division 1, 2, or 3 DG inoperable and the required opposite unit Division 2 DG inoperable	30.0
3.8.1.D	Two required offsite circuits inoperable	30.0
3.8.1.E	One required offsite circuit inoperable AND one required Division 1, 2 or 3 DG inoperable.	11.1

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Table E1-2: Example RICT Calculations		
Tech Spec	**LCO Condition**	**RICT Estimate[1] (Days)**
3.8.4.A	One required Division 1, 2, or 3 125 VDC battery charger on one division inoperable OR One required Division 2 or opposite unit Division 2 battery charger on one division inoperable OR One required Division 1 250 VDC battery charger inoperable	3.0[2]
3.8.4.B	Division 1 or 2 125 VDC electrical power subsystem inoperable for reasons other than Condition A	3.1[2]
3.8.4.E	Required Action and associated Completion Time of Condition A not met for the opposite unit Division 2 DC electrical power subsystem OR Opposite unit Division 2 DC electrical power subsystem inoperable for reasons other than Condition A	30.0
3.8.7.A	One or both Division 1 and 2 AC electrical power distribution subsystem inoperable.	0.0[2]
3.8.7.B	One or both Division 1 and 2 125 VDC electrical power distribution subsystems are inoperable.	0.1[2]
3.8.7.D	One or more required opposite unit Division 2 AC or DC electrical power distribution subsystems inoperable	30.0

Table E1-2 Notes:

1. RICTs are based on the internal events, internal flood, and internal fire PRA model calculations with seismic penalties. RICTs calculated to be greater than 30 days are capped at 30 days based on NEI 06-09-A. RICTs are rounded to nearest tenth of a day.
2. Per NEI 06-09-A, for cases where the total CDF or LERF is greater than 1E-03/yr or 1E-04/yr, respectively, the RICT Program will not be entered.
3. Not explicitly modeled, surrogate modeling was used to represent the TS function.

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List of Revised Required Actions to Corresponding PRA Functions

Table E1-3 lists the TSTF-505 Rev 2 Table 1 Tech Specs that require additional justification along with a description of how the additional justification is provided in the LAR.

Table E1-3: TSTF-505 Rev 2 Table 1 Technical Specifications (TS) that Require Additional Justification			
TS Description	**TSTF-505 TS**	**LSCS TS**	**Additional Justification**
Source Range Monitor Instrumentation - One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below.	3.3.1.2.A	3.3.1.2.A	N/A – TSTF-505 changes are excluded.
Feedwater and Main Turbine High Water Level Trip Instrumentation - Two or more feedwater and main turbine high water level trip channels inoperable.	3.3.2.B	----	N/A – TSTF-505 changes are excluded.
End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation - One or more required channels inoperable.	3.3.4.1.A.1 3.3.4.1.A.2	3.3.4.1.A	N/A – TSTF-505 changes are excluded.
Low-Low-Set (LLS) Instrumentation	3.3.6.3	----	N/A – TSTF-505 changes are excluded.
Loss of Power (LOP) Instrumentation - One or more channels inoperable.	3.3.8.1.A	3.3.8.1.A	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when a loss of function occurs.

Table E1-3: TSTF-505 Rev 2 Table 1 Technical Specifications (TS) that Require Additional Justification			
TS Description	**TSTF-505 TS**	**LSCS TS**	**Additional Justification**
Primary Containment Air Lock - Primary containment air lock inoperable for reasons other than Condition A or B.	3.6.1.2.C.3	3.6.1.2.C	TSTF-505 changes are incorporated. Compliance with the remaining portions of LCO Condition 3.6.1.2 ensure that there is a physical barrier (i.e., closed door) and an acceptable overall leakage from containment. Thus, the function is still maintained. Required Action C.1 of LCO Condition 3.6.2 requires the condition to be assessed in accordance with TS 3.6.1, "Containment Integrity" (i.e., "Initiate action to evaluate overall containment leakage rate per LCO 3.6.1" with a Completion Time of Immediately.)
Primary Containment Isolation Valves (PCIVs) - One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.	3.6.1.3.E.1	3.6.1.3.E	N/A – TSTF-505 changes are excluded.
Reactor Building-to-Suppression Chamber Vacuum Breakers	3.6.1.7	----	N/A – TSTF-505 changes are excluded.
Main Turbine Bypass System - Requirements of the LCO not met or Main Turbine Bypass System inoperable.	3.7.7.A	3.7.7.A	N/A – TSTF-505 changes are excluded.

2. References

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

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

3. LS-PSA-003, Rev 6, "LaSalle Success Criteria Notebook", dated December 11, 2017.

4. LS-PSA-005.09, "Automatic Depressurization System (ADS) Systems Notebook," November 2015.

ENCLOSURE 2

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

ENCLOSURE 2
Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

1. Introduction

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

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

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

Section 2 of this enclosure describes requirements related to the scope of the LSCS PRA models. Section 3 addresses the technical adequacy of the internal events PRA for this application. Section 4 similarly addresses the technical adequacy of the Fire PRA for this application. Section 5 lists references used in the development of this enclosure.

All of the PRA models described below have been peer reviewed, and the review and closure of finding-level F&Os from the peer review have been independently evaluated to confirm that the associated model changes did not constitute a model upgrade. Sections 3 and 4 provide the disposition of all open peer review F&O findings and F&O suggestions that were associated with Supporting Requirements (SRs) assessed as "Not Met" or Capability Category (CC) I following the closure review, including the disposition of the open F&O relative to this application. Note that all open F&Os that represent a gap to meeting CC II, regardless of whether it is categorized as a finding or suggestion, are dispositioned in this application. The resolved findings and the basis for resolution are documented in the LSCS PRA documentation and the F&O Closure Review reports (References 9, 10 and 12).

2. Requirements Related to Scope of LAS PRA Models

The PRA models discussed in this enclosure have been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference 4) consistent with NRC RIS 2007-06 (Reference 14).

Both the LSCS internal events PRA model and the LSCS Fire PRA model are at-power models. The models are capable of quantifying Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). Internal flooding is included in both the CDF and LERF internal events PRA models.

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

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

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

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

The LAS PRA model for internal events received a formal industry peer review in April 2008 (Reference 6). The LSCS full power internal events (FPIE) (including internal flooding) Peer Review was performed using the NEI 05-04 process (Reference 7), the ASME PRA Standard ASME RA-Sc-2007 (Reference 15) and Regulatory Guide 1.200, Rev. 1 (Reference 8). The Peer Review found that 91% of the SRs evaluated met Capability Category II or better. There were twenty (20) SRs that were assessed as "Not Met" and seven (7) SRs that were assessed as meeting only Capability Category I. Of the 27 SRs which were assessed as not meeting Capability Category II or better, two (2) were related to Internal Flooding SRs. Many of these findings, leading to the open SRs, were related to documentation issues.

The 2008 FPIE Peer Review findings were addressed through several periodic PRA updates and the resolutions to the findings were reviewed by an independent review team (Reference 9). The independent review team concluded that for the FPIE PRA, one finding was dispositioned as "partially resolved" and one finding was still "open". All other findings were dispositioned as "resolved". The "partially resolved" finding was identified by the independent review team as requiring enhanced documentation.

Eleven (11) F&Os associated with SRs assessed as less than Capability Category II (i.e., SRs assessed as "Not Met" or Capability Category I) were categorized as suggestions rather than findings. The resolution of these suggestion level F&Os was not previously independently reviewed, so a supplemental FPIE F&O Closure Review was performed in conjunction with the Fire PRA closure (Reference 10). The supplemental review concluded that 8 suggestions assessed as less than Capability Category II were dispositioned as "resolved".

Also, a gap analysis to Regulatory Guide 1.200 Rev.2 (Reference 4) and the current ASME/ANS PRA Standard RA-Sa-2009 (Reference 5) was performed as part of the 2014 PRA update. The gap analysis is documented in LS-PSA-016, Rev. 3 (Reference 16) and the identified gaps were mostly related to unresolved F&Os at the time of the 2014 PRA update. A separate line item in Table E2-1 is documented for those gaps identified as part of the self-assessment.

Table E2-1 provides a listing of all findings and suggestions associated with SRs assessed as less than CC II that were identified during the PRA Peer Review that remain "open" (including those that may be only "partially resolved") at the time of this submittal. The F&Os discussed in Table E2-1 represent the gaps to meeting Capability Category II for the FPIE PRA model.

As documented in Table E2-1, only two FPIE findings and three suggestions remain open and an assessment with respect to the impact on this application is also provided.

Given the resolution of the remaining "partially resolved" finding and SR assessed as less than Capability Category II, it is concluded that the LSCS internal events PRA (including internal flooding) will be of adequate technical capability to support the TSTF-505 program.

Table E2-1

LASALLE FPIE / INTERNAL FLOODING PRA PEER REVIEW - OPEN FACTS AND OBSERVATIONS - FINDINGS AND SUGGESTIONS LINKED TO SUPPORTING REQUIREMENTS ASSESSED AS LESS THAN CAPABILITY CATEGORY II (POST F&O CLOSURE)

F&O ID	Originating SR(s)	Finding Details	Basis for Significance	Possible Resolution	Status	Disposition from F&O Closure Review	Maintenance vs. Upgrade	Impact to TSTF-505 Implementation
IE-D3-01 (Finding)	IE-D3 AS-C3 SC-C3 CY-C3 HR-I3 DA-D3 IF-F3 QU-E2 LE-G4	The Summary Notebook includes information that attempts to identify the key sources of uncertainty in the initiating event analysis. However, with the changes to eliminate "key" from the SR definition, this SR cannot be considered met.	Section 4 of the LS-PSA-013 notebook discusses the industry "key sources of uncertainty" per EPRI guidance. However, the current analysis does not fully meet the requirements of RG 1.200, which requires a discussion of sources of model uncertainty and related assumptions. Also, there may be some plant-specific assumptions made that may not be fully captured by the generic list of potential sources of uncertainty.	Expand the existing treatment of sources of uncertainty to consider sources of model uncertainty and related assumptions. Consideration should also be given to potential plant-specific assumptions that should also be noted as sources of uncertainty. NUREG-1855 and an upcoming EPRI Treatment of Uncertainty report should provide more guidance on how to meet this SR in the future.	Partially Resolved	<u>LaSalle Assessment:</u> The uncertainty analysis was updated as part of the 2011A PRA update. The uncertainty analysis follows the current industry guidance as documented in NUREG-1855 and associated EPRI reports to identify both generic and plant specific modeling uncertainties. The uncertainty analysis is documented in LS-PSA-013, LaSalle PRA Summary Notebook. Appendix B of the LaSalle PRA Summary Notebook provides postulated modeling uncertainties identified through a systemic structured process using a methodology developed by EPRI. <u>Independent Team Assessment:</u> The team reviewed the uncertainty discussions in the LaSalle Summary Notebook LS-PSA-013. The team concluded that all originating SRs are met except LE-G4. The team agrees that this information in the summary notebook fulfills the requirements of SR IE-D3. For the other SRs, the requirements were to document the key assumptions and key sources of uncertainty within the subject analyses. Tables in Appendix B of LS-PSA-013 and the discussion in the individual notebooks meet these requirements. For QU-E2, the requirement is to identify key assumptions made in the development of the PRA model. This is considered met by the tables in Appendix B emphasizing the key findings. For LE-G4, the requirement is to document key assumptions and key sources of uncertainty associated with the LERF analysis, including results and important insights from sensitivity studies. Note the increased requirement associated with this SR. The team could not find a systematic treatment in the notebook (LS-PSA-015 or Appendix B of LS-PSA-013 that includes the results and insights. The one guidance driven sensitivity study did not address plant specific results and insights. Note the current ASME/ANS PRA standard has removed the word "key".	Maintenance – improved documentation and sensitivity analysis. <u>Independent Team Assessment:</u> Agree with PRA maintenance	Additional documentation of LERF key sources of uncertainty including results and important insights are needed to fully close out this F&O. However, this issue does not impact RICT calculations or this license amendment request (LAR). The model sources of uncertainty, both generic and plant-specific, as they impact this risk-informed application are specifically addressed in Enclosure 9 of this LAR.

Table E2-1

LASALLE FPIE / INTERNAL FLOODING PRA PEER REVIEW - OPEN FACTS AND OBSERVATIONS - FINDINGS AND SUGGESTIONS LINKED TO SUPPORTING REQUIREMENTS ASSESSED AS LESS THAN CAPABILITY CATEGORY II (POST F&O CLOSURE)

F&O ID	Originating SR(s)	Finding Details	Basis for Significance	Possible Resolution	Status	Disposition from F&O Closure Review	Maintenance vs. Upgrade	Impact to TSTF-505 Implementation
SY-A4-01 (Suggestion)	SY-A4	Perform plant walkdowns with system engineers AND plant operators. Better document the walkdowns performed in support of the PRA and refence those walkdowns in each system notebook to achieve Capability Category II.	Enhance PRA technical capability.	Perform plant walkdowns with system engineers AND plant operators following next update.	Open	LaSalle Assessment: The current LaSalle PRA is based on the PRA model developed by Sandia National Laboratory and documented in NUREG/CR-4832, Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP). The RMIEP PRA model for LaSalle was a very detailed PRA with respect to system modeling and as stated in Volume 1 of NUREG/CR-4832, when developing system fault trees, "exacts locations were obtained for all components represented in the fault trees." Since the development of the RMIEP PRA model for LaSalle, the model has been updated and modification to systems have been included. These updates included system manager interviews to confirm plant design and operations. Detailed plant walkdowns have been performed as part of the Internal Flooding analysis and are documented in LS-PSA-012, LaSalle PRA, Internal Flood Analysis, Volume 2, Flood Walkdown Notebook. In addition, numerous plant walkdowns have been performed in support of the LaSalle Fire PRA and are documented in the Fire PRA Notebooks. Additionally, there is a Site Risk Management Engineer (SRME) at LaSalle who directly interfaces with Operations and System Engineers to ensure the LaSalle PRA model continues to represent the as-built, as-operated plant. Therefore, given the methodical and detailed approach to the LaSalle RMIEP PRA model development, the continued efforts to update the LaSalle PRA to ensure consistency with plant design and the daily interactions of the SRME with plant operations and engineer staff, it is judged that these efforts are equivalent to the intent of the Supporting Requirement (SR) and no gap exists in system modeling as a result of not having detailed plant walkdowns documented for each system. Independent Team Assessment: Open, Not Reviewed	Maintenance – improvement in PRA documentation and confirmation of results Independent Team Assessment: Open, Not Reviewed	While it is judged that this Finding has no impact on the PRA results and therefore, no impact on TSTF-505 implementation, this F&O will be resolved with a PRA update and system walkdowns will be conducted and documented with System Engineers and Plant Operators prior to implementation of TSTF-505.

Table E2-1

LASALLE FPIE / INTERNAL FLOODING PRA PEER REVIEW - OPEN FACTS AND OBSERVATIONS - FINDINGS AND SUGGESTIONS LINKED TO SUPPORTING REQUIREMENTS ASSESSED AS LESS THAN CAPABILITY CATEGORY II (POST F&O CLOSURE)

F&O ID	Originating SR(s)	Finding Details	Basis for Significance	Possible Resolution	Status	Disposition from F&O Closure Review	Maintenance vs. Upgrade	Impact to TSTF-505 Implementation
DA-C8-01 (Finding)	DA-C8	Basic events used to model the standby status of various plant systems use a mix of plant-specific operational data and engineering judgment. For the plant service water system and several other systems, standby estimates have been determined from procedures and operating data. For other components, assumptions are used (e.g., 50% probability of either of two pumps in a system is in standby). So, overall the LaSalle PRA has some Capability Category (CC) II attributes and some CC I attributes.	Current approach of assuming standby time does not meet the requirements of the Supporting Requirement. The use of actual plant data could result in small changes in PRA results.	Collect plant specific data for all of the basic events that reflect standby status to meet CC II requirements.	Open	LaSalle Assessment: A complete review of the LaSalle PRA model was conducted to identify systems that use assumptions for standby estimates of components. The only instance identified was related to the Turbine Building Closed Cooling Water (TBCCW) system. In this case, it was assumed that there was a 50% probability that either the A or B train TBCCW pump would be in standby. Further discussion with plant personnel regarding standby times for the TBCCW pumps revealed that these pumps are swapped every six months, therefore the standby time of 50% is a reasonable assumption. The PRA system notebooks will be updated during a PRA update to document the approach used to determine standby times for components. Additionally, as required by procedures, actual plant operating experience will be used to update the standby basic event probabilities. Independent Team Assessment: Open, Not Reviewed	Maintenance – updated basic event probabilities to reflect current plant operating experience. Independent Team Assessment: Open, Not Reviewed	This issue has minimal impact on the TSTF-505 application as the plant-specific configuration is specifically accounted for in the RICT calculations. Further, review of this issue has determined that assumptions used in the PRA model are consistent with the plant operating practices.

Table E2-1

LASALLE FPIE / INTERNAL FLOODING PRA PEER REVIEW - OPEN FACTS AND OBSERVATIONS - FINDINGS AND SUGGESTIONS LINKED TO SUPPORTING REQUIREMENTS ASSESSED AS LESS THAN CAPABILITY CATEGORY II (POST F&O CLOSURE)

F&O ID	Originating SR(s)	Finding Details	Basis for Significance	Possible Resolution	Status	Disposition from F&O Closure Review	Maintenance vs. Upgrade	Impact to TSTF-505 Implementation
DA-C6-01 (Suggestion)	DA-C7 DA-C9 DA-C10	LS-PSA-010, Component Data Notebook, Appendix C, states, "No actual data or estimates for these parameters are provided by system managers. Data from the MSPI basis document, scoping and performance criteria document, and 2003 data notebook is used." As the was obtained from Maintenance Rule and MSPI sources, the techniques used to obtain this data are probably consistent with the guidance in this supporting requirement, but this cannot be positively determined. Similarly, for SR DA-C7, it is unable to be determined if surveillance tests, planned and unplanned maintenance activities were based on actual plant experience. For SR DA-C9, the reviewers were unable to conclude whether plant specific operational records were used to determine standby time. Similarly, for DA-C10, it is not clear how surveillance tests were used.	This appears to be primarily a documentation issue, as it is expected that the assumptions used to collect data for Maintenance Rule and MSPI are similar to those required by the ASME standard. However, it is possible that some differences in methodology could exist between these programs and the PRA.	Enhance the documentation in LS-PSA-010 to specifically discuss how the data provided by the Maintenance Rule and MSPI programs are consistent with the requirements of SRs DA-C6, DA-C7 and DA-C9. If differences do exist, then adjust the collected raw data to meet the requirements of these SRs.	Open	LaSalle Assessment: During the 2006 LaSalle PRA update, a complete set of plant specific system operating data was not collected from the System Managers. Instead the update relied on MSPI and Maintenance Rule data. In 2011 update, plant specific operating data was obtained from the System Managers and collected in accordance with SR DA-C6, DA-C7, DA-C9 and DA-C10. With regards to potential component failures, the PRA Analyst reviews failure data as documented in Issue Reports and can further discuss component failures with the System Managers to ensure proper categorization in accordance with the PRA Standard. The documentation in LS-PSA-10, LaSalle PRA Component Data Notebook, will be updated during a PRA update to better discuss data collection and how it meets the SRs noted in this suggestion. Note that during a PRA update, current plant specific operating data will again be obtained and incorporated into the PRA model. Independent Team Assessment: Open, Not Reviewed	Maintenance – updated basic event probabilities to reflect current plant operating experience. Independent Team Assessment: Open, Not Reviewed	This issue has minimal impact on the TSTF-505 application as the plant-specific data was updated during the 2011 and 2014 PRA updates. Further, LaSalle will be updating the plant-specific data during a PRA update before implementation of TSTF-505.

Table E2-1

LASALLE FPIE / INTERNAL FLOODING PRA PEER REVIEW - OPEN FACTS AND OBSERVATIONS - FINDINGS AND SUGGESTIONS LINKED TO SUPPORTING REQUIREMENTS ASSESSED AS LESS THAN CAPABILITY CATEGORY II (POST F&O CLOSURE)

F&O ID	Originating SR(s)	Finding Details	Basis for Significance	Possible Resolution	Status	Disposition from F&O Closure Review	Maintenance vs. Upgrade	Impact to TSTF-505 Implementation
IF-C3b-01 (Suggestion)	IF-C3b	Address potential unavailability of barriers that affect the propagation of water in order to meet the CC II requirements of the ASME Standard.	This is a suggestion since it is considered a documentation issue. The flood scenarios analyzed in detail are so large (i.e., typically involving draining the lake into the Turbine building until it fills) that structural analysis of non-flood doors and any difference in flood propagation will have no significant impact.	If reasonable, discuss in the analysis the insignificance of barrier unavailability in light of the currently modeled scenarios.	Open	LaSalle Assessment: As noted by the Peer Review Team, this is considered a documentation issue. The LaSalle internal flood analysis revealed that the CDF due to flooding is dominated by very large floods involving draining the large flood into the Turbine Building. Therefore, refinement of barriers (e.g. non-flood doors) would have a very small impact on the overall flood analysis results. This suggestion will be resolved during a update and a full discussion of flood barrier and propagation will be provided in the LaSalle PRA Internal Flood Notebook. Independent Team Assessment: Open, Not Reviewed	Maintenance - Update of documentation Independent Team Assessment: Open, Not Reviewed	This open issue has no impact on the TSTF-505 implementation as it is a documentation issue. However, this suggestion will be resolved during a PRA update.

Table E2-1

LASALLE FPIE / INTERNAL FLOODING PRA PEER REVIEW - OPEN FACTS AND OBSERVATIONS - FINDINGS AND SUGGESTIONS LINKED TO SUPPORTING REQUIREMENTS ASSESSED AS LESS THAN CAPABILITY CATEGORY II (POST F&O CLOSURE)

F&O ID	Originating SR(s)	Finding Details	Basis for Significance	Possible Resolution	Status	Disposition from F&O Closure Review	Maintenance vs. Upgrade	Impact to TSTF-505 Implementation
URE LS2020-0001 (LS Update Requiring Evaluation Tracking ID) Gaps from Self-Assessment [16]	IFSO-A3 IFSN-A7 IFQU-A3	As part of the Self-Assessment performed during the 2014 FPIE PRA update, the following gaps to RG 1.200 (Rev. 2) and the ASME/ANS PRA Standard were identified: 1. IFSO-A3 Further documentation clarification is required for those flood locations that are screened out based on the quantitative screening criteria described in the PRA Standard. 2. IFSN-A7 Further documentation clarification is required for justification of crediting EQ limits for ensuring operability of instrumentation given spray-induced impacts. 3. IFQU-A3 Further documentation clarification is required for those flood locations that are screened out based on the quantitative screening criteria described in the PRA Standard.	N/A	N/A	Open	LaSalle Assessment These identified gaps are primarily documentation issues. These gaps will be resolved during a update and additional justifications will be provided in the LaSalle PRA Internal Flood Notebook.	Maintenance - Update of documentation	These open issues have no impact on the TSTF-505 implementation as they are primarily documentation issues. However, these gaps will be resolved during a PRA update.

4. Scope and Technical Adequacy of LSCS Fire PRA Model

The LSCS Fire PRA (FPRA) Peer Review (Reference 13) was performed in December 2015 using the NEI 07-12 Fire PRA peer review process (Reference 11), the ASME/ANS PRA Standard, ASME/ANS RA-Sa-2009 (Reference 5) and Regulatory Guide 1.200, Rev. 2 (Reference 4). The purpose of this review was to establish the technical adequacy of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used. The 2015 LSCS FPRA Peer Review was a full-scope review of the LAS at-power FPRA against all technical elements in Part 4 of the ASME/ANS PRA Standard (Reference 5), including the referenced internal events supporting requirements (SRs). The findings were addressed in subsequent FPRA updates and a F&O Closure Review was performed by an independent review team in October 2017 (Reference 10). Following that F&O Closure Review, sixteen (16) of the findings were dispositioned as "partially resolved" or "open".

In addition, during the F&O Closure Review, a Focused Scope Peer Review was conducted against the technical element Fire Risk Quantification (FQ) due to the large change in CDF and LERF as a result of the resolution of many technical F&Os. The Focused Scope Peer Review resulted in all SRs, except for one, being met at Capability Category II. Three finding level F&Os were identified including one related to the SR that was not met. The Focused Scope Peer Review is documented in the independent review team report (Reference 10).

In September 2019, another F&O Closure Review (Reference 12) was conducted to independently review the changes made to the LSCS FPRA. This most recent LSCS FPRA update addressed 17 F&O findings, including those findings identified during the focused-scope peer review in October 2017, as well as one suggestion level F&O. As a result of this review, all 18 F&Os reviewed were dispositioned as resolved by the Independent Review Team.

Table E2-2 provides a listing of all findings that were identified during the Fire PRA Peer Review and still remain as "open" (including those that may be only "partially resolved") and one suggestion related to a SR assessed as "Not Met". The F&Os discussed in Table E2-2 represent the gaps to meeting Capability Category II for the Fire PRA model.

As documented in Table E2-2, an assessment with respect to the impact on this application is also provided.

Given the resolution of the remaining "partially resolved" or "open" findings that may impact RICT calculations, the LSCS Fire PRA will be of adequate technical capability to support the TSTF-505 program.

Table E2-2

LASALLE FIRE PRA PEER REVIEW - OPEN FACTS AND OBSERVATIONS - FINDINGS AND SUGGESTIONS LINKED TO SUPPORTING REQUIREMENTS ASSESSED AS LESS THAN CAPABILITY CATEGORY II (POST F&O CLOSURE)

F&O ID	Originating SR(s)	Finding Details	Basis for Significance	Possible Resolution	Status	Disposition from F&O Closure Review	Maintenance vs. Upgrade	Impact to TSTF-505 Implementation
1-19 (Suggestion)	CS-A1	The peer review examined the cable selection package for offsite power loss switchyard breaker (OCB 4-6). The circuit evaluation package includes two pages of notes regarding interlock evaluations and the notes and assumptions associated with the interlocks. For example, a note is made that "the interlock associated with trip and lockout of SAT 242. Cables that can cause relay to actuate are to be included with SAT 242". The FPRA development team indicated that this impact for SAT 242 is addressed by the FPRA, but that no systematic review of the circuit evaluation package notes was performed.	A review of circuit evaluation notes and assumptions is important to ensure that FPRA plant response model identifies cables whose fire-induced failure could adversely affect selected equipment and/or credited functions in the Fire PRA plant response model.	Perform a systematic review of the circuit evaluation package notes and assumptions to ensure that FPRA plant response model identifies cables whose fire-induced failure could adversely affect selected equipment and/or credited functions in the Fire PRA plant response model.	Open, Not Reviewed	LaSalle Assessment: As suggested by the Peer Review Team, a systematic review of the circuit evaluation package notes and assumptions was performed to ensure that the Fire PRA plant response model identifies cables whose fire-induced failure could adversely affect selected equipment and/or credited functions. The systematic review is documented in Appendix A of the Detailed Circuit Analysis Notebook (LS-PSA-021.03.01) and the model was updated appropriately. Independent Team Assessment: Open, Not Reviewed	Maintenance – update of documentation Independent Team Assessment: Open, Not Reviewed	This item has no impact on the TSTF-505 as it has been resolved, just not reviewed and closed by the Independent Assessment Team.

Table E2-2

LASALLE FIRE PRA PEER REVIEW - OPEN FACTS AND OBSERVATIONS - FINDINGS AND SUGGESTIONS LINKED TO SUPPORTING REQUIREMENTS ASSESSED AS LESS THAN CAPABILITY CATEGORY II (POST F&O CLOSURE)

F&O ID	Originating SR(s)	Finding Details	Basis for Significance	Possible Resolution	Status	Disposition from F&O Closure Review	Maintenance vs. Upgrade	Impact to TSTF-505 Implementation
4-17 (Finding)	FSS-D7	There is no generic estimate or plant-specific value assigned to the non-suppression probability. (This F&O originated from SR FSS-D7)	The non-suppression values are only based on the NUREG/CR-6850 generic values for unreliability with no account for unavailability.	Assign a plant-specific unavailability value for the credited suppression and detection systems to be included to the non-suppression probabilities. Alternatively, assign a generic estimate for unavailability and perform a review of the suppression and detection systems for outlier behavior relative to system unavailability.	Open, Not Reviewed	LaSalle Assessment: Currently, the FPRA model uses the generic unreliability estimates from NUREG/CR-6850 for automatic detection and suppression systems and this is also assumed to encompass unavailability based on interviews with the site. To meet Capability Category II, a review of plant-specific data is required in order to determine if automatic detection and suppression systems exhibit outlier behavior. As part of the 2015 Fire PRA development, the LaSalle fire protection engineer reviewed the NUREG/CR-6850 unreliability data and estimated that the values also were adequate to reflect both unavailability and unreliability. Additional reviews of plant data will be conducted during a Fire PRA update. It is expected that the data used from NUREG/CR-6850 is conservative; however, additional data reviews will be conducted prior to implementation of the RICT license amendment to document the validity of this assumption. Independent Team Assessment: Open, Not Reviewed	LaSalle Assessment: Maintenance - Similar to PRA Standard Example 3 (updating data) and Example 9 (correcting an omission). Independent Team Assessment: Open, Not Reviewed	The impact on the RICT LAR is judged to be minimal. However, plant-specific data will be reviewed and refined data for automatic detection and suppression systems will be incorporated into the FPRA model during a Fire PRA update if necessary. This item will be resolved prior to TSTF-505 implementation.

Table E2-2

LASALLE FIRE PRA PEER REVIEW - OPEN FACTS AND OBSERVATIONS - FINDINGS AND SUGGESTIONS LINKED TO SUPPORTING REQUIREMENTS ASSESSED AS LESS THAN CAPABILITY CATEGORY II (POST F&O CLOSURE)

F&O ID	Originating SR(s)	Finding Details	Basis for Significance	Possible Resolution	Status	Disposition from F&O Closure Review	Maintenance vs. Upgrade	Impact to TSTF-505 Implementation
6-11 (Finding)	CS-A1 CS-A2 CS-A3	The cable selection work performed related to the cable data in the Fire Safe Shutdown report pre-dates NEI-00-01 guidance and was done to the standards at that time. No other information is currently available regarding the circuit analysis techniques used for the Fire Safe Shutdown Report. In general, the MSO circuit analysis work was performed using NEI-00-01, Revision 2 or Revision 3 (depending upon the particular package).	Potentially Risk Significant	Revises cable selection to be consistent with NEI-00-01 Rev. 3	Partially Resolved	LaSalle Assessment: Although the circuit analysis techniques originally used for the Fire Safe Shutdown Analysis (SSA) predate NEI 00-01, the approaches used were acceptable for the plant licensing basis and were representative of those that became the basis for NEI 00-01 Rev. 0. Through the years the circuit analysis and cable selection that supports the SSA has received numerous NRC inspections as well as internal reviews. These inspections and reviews substantiate the technical integrity of the SSA cable selection thereby supporting its use for applicable equipment modeled in the FPRA. With regards to NEI 00-01 Rev. 2 vs. Rev. 3, LaSalle performed a gap assessment to examine the impact of changes associated with Rev. 3 (tracked via IR 01277652). Action items identified included review of the new MSO scenarios for applicability to LaSalle and performance of corrective actions as needed, and to determine if any revisions to NEI 00-01 circuit failure criteria or MSO list guidance affected any MSO evaluations and to resolve any such impacts. As documented in Calculation L-003779 Revision 0 (November 2012), the plant completed their MSO scenario analysis, providing MSO resolutions in accordance with SECY-08-0093, RG 1.189 Revision 2, and NEI 00-01 Rev. 3. These actions were completed prior to the 2015 FPRA update. Any supplementary circuit analysis performed strictly to support the FPRA was performed consistent with NEI 00-01 Rev. 3 and circuit analysis performed subsequent to the Peer Review has been performed consistent with NEI 00-01 Rev. 3. Based on the above, the cables selected and incorporated into the Fire PRA from the SSA and the MSO evaluations present a technically appropriate basis for cable selection for the Fire PRA. To address this technical aspect, a new item has been added to the list of assumptions and uncertainties as documented in Section 2.2 of the Cable Selection Notebook. Also, additional justification for the present state of the FPRA has been added to Section 3.1 of the Cable Selection Notebook. Finally, prior to implementation of the RICT license assessment, the recommendation of the Independent Review Team will be performed as noted below to confirm the technical adequacy of cable selection and circuit analysis to support the Fire PRA technical adequacy.	LaSalle Assessment: Maintenance – Any revision to cable selection based on historical methods used during the time of the development of the SSA and the more recent guidance of NEI 00-01 Rev. 3 is anticipated to have negligible impacts on the risk significant accident or accident progression sequences. This is similar to PRA Standard Example 9 for correcting an omission or reflecting new knowledge. Independent Team Assessment: Agree with PRA Maintenance.	There is no impact on TSTF-505 implementation as this issue will be resolved prior to implementation.

Table E2-2

LASALLE FIRE PRA PEER REVIEW - OPEN FACTS AND OBSERVATIONS - FINDINGS AND SUGGESTIONS LINKED TO SUPPORTING REQUIREMENTS ASSESSED AS LESS THAN CAPABILITY CATEGORY II (POST F&O CLOSURE)

F&O ID	Originating SR(s)	Finding Details	Basis for Significance	Possible Resolution	Status	Disposition from F&O Closure Review	Maintenance vs. Upgrade	Impact to TSTF-505 Implementation
						Independent Team Assessment: The Independent Review Team identified that although updates have been made based upon more recent guidance (e.g. ground equivalent hot shorts, additional MSOs), the base analysis did predate revision 0 of NEI 00-01. It is reasonable to expect that the methodology for the SSA was consistent with the state of the art practices at the time that ultimately became NEI 00-01 R0. However, a review must be performed to confirm that the LaSalle SSA was performed to this expected pedigree. In order to confirm that the circuit analysis performed to support the LaSalle SSA is not an outlier, the Independent Review Team recommended that a review should be conducted to verify that the analysis was performed in accordance with the requirements of NEI-00-01. A review of a representative sample will provide sufficient assurance. A sample should include each of the typical component types (e.g. pump, MOV, AOV, logic circuit) and should include both AC and DC circuit examples.		

5. References

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

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

3. Regulatory Guide (RG) 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018.

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

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6. LaSalle Unit 2 Nuclear Plant PRA Peer Review Report Using ASME PRA Standard Requirements, July 2008.

7. NEI 05-04, "Process for Performing PRA Peer Reviews Using the ASME PRA Standard," Revision 1, Draft G, November 2007.

8. Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, January 2007.

9. LaSalle County Generation Station Unit 2, PRA Facts and Observations Independent Assessment Report Using NEI 05-04/07-12/12-06 Appendix X, June 2017.

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

12. LaSalle Units 1 & 2, Fire PRA Finding & Suggestion Level Fact and Observation Closure by Independent Assessment, Report Number # 032362-RPT-01, Revision 0, November 2019.

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14. NRC Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," March 22, 2007.

15. ASME RA-Sc-2007, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, August 2007.

16. LaSalle County Generating Station Probabilistic Risk Assessment Self-Assessment of the LaSalle PRA Against the Combined ASME/ANS PRA Standard Requirements, LS-PSA-016, Rev. 3, November 2015.

ENCLOSURE 3

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

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

This enclosure is not applicable to the LaSalle County Station submittal.

EGC is not proposing to use any PRA models in the LaSalle County Station Risk-Informed Completion Time Program for which a PRA standard, endorsed by the NRC in RG 1.200, Revision 2 does not exist.

ENCLOSURE 4

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

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

ENCLOSURE 4
Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models

1 Introduction and Scope

Topical Report NEI 06-09-A, Revision 0 (Reference [1]), as clarified by the Nuclear Regulatory Commission (NRC) final safety evaluation (Reference [2]), requires that the License Amendment Request (LAR) provide a justification for exclusion of risk sources from the Probabilistic Risk Assessment (PRA) model based on their insignificance to the calculation of configuration risk as well as discuss conservative or bounding analyses applied to the configuration risk calculation. This enclosure addresses this requirement by discussing the overall generic methodology to identify and disposition such risk sources. This enclosure also provides the LaSalle County Station (LSCS) specific results of the application of the generic methodology and the disposition of impacts on the LSCS Risk Informed Completion Time (RICT) Programs. Section 3 of this enclosure presents the plant-specific bounding analysis of seismic risk to LSCS. Section 4 of this enclosure presents the justification for excluding analysis of high wind risk to LSCS. Section 5 presents the justification for excluding External Flooding for LSCS. Section 6 of this enclosure presents the justification for excluding analyses of other external hazards from the LSCS PRA.

Topical Report NEI 06-09-A does not provide a specific list of hazards to be considered in a RICT Program. However, non-mandatory Appendix 6-A in the ASME/ANS PRA Standard (Reference [3]) provides a guide for identification of most of the possible external events for a plant site. Additionally, NUREG-1855 [4] provides a discussion of hazards that should be evaluated to assess uncertainties in plant PRAs and support the risk-informed decision-making process. This information was reviewed for the LSCS site and augmented with a review of information on the site region and plant design to identify the set of external events to be considered. The information in the UFSAR regarding the geologic, seismologic, hydrologic, and meteorological characteristics of the site region as well as present and projected industrial activities in the vicinity of the plant were also reviewed for this purpose. No new site-specific and plant-unique external hazards were identified through this review. The list of hazards in Appendix 6-A of the PRA Standard were considered for LSCS as summarized in Table E4-11.

The scope of this enclosure is consideration of the hazards in Table E4-11 for LSCS. As explained in subsequent sections of this enclosure, risk contribution from seismic events is evaluated quantitatively, and the other listed external hazards are evaluated and screened as having low risk.

2 Technical Approach

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

Consistent with NUREG-1855 (Reference [4]), external hazards may be addressed by:

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

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

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

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

Hazard Screening

The first step in the evaluation of an external hazard is screening based on an estimation of a bounding core damage frequency (CDF) for beyond design basis hazard conditions. An example of this type of screening is reliance on the NRC's 1975 Standard Review Plan (SRP) (Reference [5]), which is acknowledged in the NRC's Individual Plant Examination of External Events (IPEEE) procedural guidance (Reference [6]) as assuring a bounding CDF of less than 1E-6/yr for each hazard. The bounding CDF estimate is often characterized by the likelihood of the site being exposed to conditions that are beyond the design basis limits and an estimate of the bounding conditional core damage probability (CCDP) for those conditions. If the bounding CDF for the hazard can be shown to be less than 1E-6/yr, then beyond design basis challenges from that hazard can be screened out and do not need to be addressed quantitatively in the RICT Program.

The basis for this is as follows:

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

The LSCS IPEEE hazard screening analysis (Reference [7]) has been updated to reflect current LSCS site conditions. The results are discussed in Section 6 of this Enclosure and show that all the events listed in Table E4-11 can be screened except seismic events for LSCS.

Hazard Analysis - CDF

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

Evaluate Bounding LERF Contribution

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

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

The approaches used for seismic LERF is described in Section 3.

Risks from Hazard Challenges

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

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

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

Section 3 of this enclosure provides the analysis for the LSCS sites with respect to the beyond design basis seismic hazard, and Section 4 provides an analysis for the extreme winds hazard. Section 5 address the analysis of External Flooding for LSCS. Section 6 of this enclosure provides an analysis of the representative external hazards for the LSCS sites.

3 Seismic Risk Contribution Analysis

LaSalle Seismic Assessment

The TSTF-505 program requires accounting for seismic risk contribution in calculating extended risk informed technical specification (TS) completion times (CT, also referred to as Allowed Outage Time, AOT).

A seismic PRA (SPRA) was developed for LSCS in 1993 (Reference [8]), cited in the LSCS IPEEE (Individual Plant Examination for External Events), (Reference [7]) and reviewed by the NRC (Reference [9]). However, since the LSCS SPRA was not maintained, an alternative approach is taken to provide an estimate of SCDF based on the current LSCS seismic hazard curve and assuming the seismic capacity of a component whose seismic failure would lead directly to core damage.

The approach taken for the estimate of SLERF is performed by using an estimated average Seismic Conditional Large Early Release Probability (SCLERP), based on the spectrum of SCDF accident sequence types, and multiplying the estimated SCDF by the average SCLERP estimate.

Input and Assumptions

Hazard Curve:

The LSCS seismic hazard is defined by the seismic hazard curve provided to the NRC in Reference [10] using the seismic hazard curve per Reference [11]

PGA Metric:
The ground motion metric used to define the seismic hazard in this analysis is peak ground acceleration (PGA).

High Confidence of a Low Probability of Failure (HCLPF):

The assumed limiting plant seismic capacity has a high confidence of low probability of failure (HCLPF) value of 0.30g PGA as cited in GI-199, Table B.2, Plant Level Fragility Data (Reference [12]). This value is consistent with the LSCS IPEEE review level earthquake (RLE) of 0.30g PGA as specified Reference [6] [NUREG-1407].

The uncertainty parameter for seismic capacity is represented by a composite beta factor (βc) of 0.4. This is a commonly-accepted approximation and is consistent with the value used in GI-199, Table C.1, Bases for Establishing Plant-Level Fragility Curves Parameters from IPEEE Information (Reference [12]).

Convolution to Determine SCDF:

The estimation of SCDF in this calculation is performed by a mathematical convolution of the PGA-based seismic hazard curve and the LSCS PGA-based plant HCLPF from Reference [12]. This convolution estimation approach is a common analysis in approximating an SCDF for use in risk-informed decision making (e.g., it is commonly used in RICT seismic penalty calculations; the NRC used this approach in the GI-199 risk assessment in absence of a current full-scope SPRA.

SLERF:

The LSCS SLERF for this risk evaluation is obtained by multiplying the calculated SCDF by an average seismic conditional large early release probability (SCLERP). The average SCLERP is estimated using information from both the LSCS IPEEE SPRA (References [8] and [9]) and results from quantification of the LSCS FPIE PRA model (Reference [13]).

Consideration of S-LOOP:

The analysis also assesses the incremental risk associated with seismic-induced LOOP that may occur from seismic events below the LSCS seismic design basis. The analysis compares a convolution estimation of seismic-induced LOOP frequency with the random LOOP frequencies from the LSCS FPIE PRA. This analysis aspect and approach has been used in past RICT seismic penalty calculations.

Calculations

The general approach to estimation of the SCDF is to use the plant level HCLPF and convolve the corresponding failure probabilities as a function of seismic hazard level with the seismic hazard curve frequencies of occurrence. This is a commonly used approach to estimate SCDF when a seismic PRA is not available and is also the approach that was used in the Vogtle pilot TSTF-505 license amendment request submittal (Reference [14]) and a previous Exelon TSTF-505 submittal for Calvert Cliffs (Reference [15]). The detailed calculations for SCDF and SLERF are documented in Reference [16]. The key elements of that SCDF calculation are discussed below.

Seismic Hazard and Intervals

The seismic hazard curve in units of g (PGA) from Reference [10] is shown in Table E4- 1. Several points have been interpolated to provide values at convenient seismic hazard points. Linear interpolation is used to calculate the exceedance frequencies for these additional magnitude points. The mean fractile occurrence frequencies of Table E4- 1 are used in the calculations here; use of mean values is a typical and expected PRA practice. Table E4- 2 shows the seismic interval magnitude range, representative g-level, and annual initiator frequency for each of the 8 seismic hazard intervals. Eight seismic hazard intervals have been used to be consistent with the Exelon fleet Phase 1 SPRA models (Reference [17] for LSCS). This is an acceptable and common number of seismic hazard intervals based on industry guidance provided in the EPRI SPRA Implementation Guide used in industry SPRAs (Reference [18]).

The representative g-level for seismic hazard intervals %G1 through %G7 is calculated using a geometric mean approach, i.e., the square root of the product of the g-level values at the beginning and end of the interval. For seismic hazard interval %G8, the representative g-level is estimated as 1.1 times the g-level at the beginning of the interval since the interval has no upper limit. This is common practice in industry SPRAs Reference [19].

The seismic hazard interval annual initiating event frequency is calculated (except for the final interval) by subtracting the mean exceedance frequency associated with the g-interval (high) end point from the mean exceedance frequency associated with the g-interval beginning point. The frequency of the last seismic hazard interval is the exceedance frequency at the beginning point of that interval. This is common practice in industry SPRAs (Reference [18]).

The portion of the LSCS seismic hazard frequency below the 0.1g point for the start of the first interval, which is consistent with the LSCS operating basis earthquake (OBE), is a non-significant contribution to the calculated SCDF and SLERF. The limiting HCLPF value for LSCS (i.e., HCLPF=0.3g, PGA) is sufficiently higher than the OBE and because the HCLPF represents a 1% failure probability (on the Mean hazard curve fractile) it is determined that there is no significant risk associated with the portion of the hazard curve below the OBE. In fact, the plant can reasonably be expected to remain online for seismic events below the OBE. Not explicitly including seismic events below the OBE is common practice in industry SPRAs (References [18]).

TABLE E4-1
LASALLE SEISMIC HAZARD DATA[1]

(from Reference [11] Table A-1a, Mean and Fractile Seismic Hazard Curves for PGA at LaSalle)

PGA (g)	Mean	0.05	0.16	0.50	0.84	0.95
0.0005	8.53E-02	5.20E-02	6.54E-02	8.60E-02	9.93E-02	9.93E-02
0.001	7.24E-02	3.63E-02	5.27E-02	7.34E-02	9.24E-02	9.93E-02
0.005	2.80E-02	1.08E-02	1.72E-02	2.64E-02	3.84E-02	5.12E-02
0.01	1.52E-02	5.75E-03	8.85E-03	1.38E-02	2.07E-02	3.09E-02
0.015	1.03E-02	3.63E-03	5.66E-03	9.24E-03	1.38E-02	2.22E-02
0.03	4.52E-03	1.34E-03	2.01E-03	3.73E-03	6.45E-03	1.13E-02
0.05	1.98E-03	4.56E-04	7.13E-04	1.44E-03	2.96E-03	5.75E-03
0.075	9.50E-04	1.60E-04	2.80E-04	6.26E-04	1.44E-03	2.88E-03
0.1	5.57E-04	7.55E-05	1.42E-04	3.52E-04	8.60E-04	1.67E-03
0.15	2.61E-04	2.60E-05	5.42E-05	1.62E-04	4.19E-04	8.23E-04
0.3	6.66E-05	4.31E-06	1.08E-05	3.79E-05	1.13E-04	2.25E-04
0.5	2.01E-05	1.11E-06	3.01E-06	1.02E-05	3.47E-05	6.93E-05
0.75	6.42E-06	2.88E-07	8.35E-07	3.09E-06	1.10E-05	2.29E-05
1	2.59E-06	8.60E-08	2.84E-07	1.18E-06	4.50E-06	9.37E-06
1.5	6.60E-07	1.02E-08	4.90E-08	2.76E-07	1.13E-06	2.53E-06
3	6.03E-08	1.82E-10	1.32E-09	1.55E-08	9.24E-08	2.68E-07
5	9.37E-09	9.11E-11	1.29E-10	1.40E-09	1.31E-08	4.56E-08
7.5	1.86E-09	9.11E-11	1.11E-10	2.35E-10	2.32E-09	9.79E-09
10	5.42E-10	8.12E-11	9.11E-11	1.20E-10	6.93E-10	2.96E-09

[1] Interpolated exceedance frequency values (using linear straight line interpolation) were calculated in this analysis for use in calculation of the hazard interval frequencies for 0.7g, 0.9g, 1.1g and 1.3g. These specific PGA points are not listed in Reference [11]. Interpolations performed only for PGA mean values for this evaluation.

Table E4- 2
Seismic Hazard Interval Frequencies

Hazard Interval ID	Seismic IE Interval Range (g, PGA)	Seismic IE Interval Representative Magnitude (g, PGA)	Interval Frequency (1/yr)
%G1	0.1 - 0.3	0.17	4.90E-04
%G2	0.3 - 0.5	0.39	4.65E-05
%G3	0.5 - 0.7	0.59	1.09E-05
%G4	0.7 - 0.9	0.79	5.03E-06
%G5	0.9 - 1.1	0.99	1.92E-06
%G6	1.1 - 1.3	1.20	7.72E-07
%G7	1.3 - 1.5	1.40	7.72E-07
%G8	>1.5	1.65	6.60E-07

Seismic Failure Probabilities

The seismic failure probability of the LaSalle limiting plant HCLPF for each hazard interval is calculated using the following equations:

Fragility (i.e., failure probability) = Φ [ln(A/Am)/ßc], where

Φ is the standard lognormal distribution function

A is the g level in question

Am is the median seismic capacity

The uncertainty parameters (betas) are related as follows:

ßc = (ßu^2 + ßr^2)^0.5.

Additionally, HCLPF and Am are related as follows:

Am = HCLPF / (exp (-1.65(ßr + ßu)))

The above fragility relationships are used in calculation Reference [16] to determine the plant level seismic-induced failure probability as a function of seismic hazard interval. The following Table shows the LSCS limiting plant HCLPF information.

TABLE E4- 3
LASALLE LIMITING PLANT HCLPF INFORMATION

Source	HCLPF[2]	Am	ßr	ßu	ßc
NRC GI-199[3]	0.3g PGA	0.763	0.283	0.283	0.40

Whether the ßr and ßu values are equal or different (e.g., (0.32 + 0.24)^0.5 = 0.4) does not change the calculated mean fragility probabilities as long as they produce the same ßc value. With all parameters specified, the interval-specific failure probabilities are obtained as defined above. The plant level HCLPF fragility mean failure probabilities as a function of hazard interval can be seen in Table E4- 4.

Seismic Core Damage Frequency

The SCDF for each hazard interval is computed as the product of the hazard interval initiating event frequency (/yr) and the plant level HCLPF fragility failure probability for that same hazard interval. The results per hazard interval are then straight summed to produce the overall total SCDF across the hazard curve. The SCDF convolution calculation is summarized in Table E4- 4.

As shown in Table E4- 4, the total estimated SCDF is 1.1E-5/yr.

[2] HCLPF and Am are related as follows: HCLPF = Am (exp -1.65(ßr + ßu))

[3] LSCS limiting plant seismic HCLPF of 0.3 g (PGA) obtained from NRC GI-199 risk assessment, Table B.2, Summary of Site Types, Evaluation Methods, HCLPF, and SSE Values. (Reference [12])

Table E4- 4
Convolution Summary of LaSalle Seismic CDF

Hazard Interval ID							
%G1	%G2	%G3	%G4	%G5	%G6	%G7	%G8
Representative Magnitude (g, PGA)							
0.17	0.39	0.59	0.79	0.99	1.20	1.40	1.65
Interval Initiating Frequency (/yr)							
4.9E-4	4.7E-5	1.1E-5	5.0E-6	1.9E-6	7.7E-7	7.7E-7	6.6E-7
Mean Fragility Probability[4]							
1.1E-4	4.5E-2	2.6E-1	5.4E-1	7.5E-1	8.7E-1	9.3E-1	9.7E-1
Convolved SCDF per Interval (/yr)[5]							
5.2E-8	2.1E-6	2.9E-6	2.7E-6	1.4E-6	6.7E-7	7.2E-7	6.4E-7
% SCDF Contribution per interval							
0.5%	18.7%	25.6%	24.2%	12.8%	6.0%	6.4%	5.7%
TOTAL CONVOLVED SCDF ACROSS HAZARD CURVE (/YR)							**1.1E-5**

As can be seen from Table E4- 4, the conditional core damage probability (i.e., "Mean Fragility Probability" row) of the final hazard interval is not precisely 1.0 but rather slightly below at 0.97. This is due to usage of the seismic hazard intervals consistent with the LSCS Phase 1 SPRA as opposed to performing the convolution out to many g's. This is a negligible impact on the calculated SCDF of 1.1E-5/yr. If the representative magnitude of the final hazard interval were increased to a higher value or the convolution performed all the way out to 10g's (using all the data points of Table E4- 1) the calculated SCDF would change in the third decimal place.

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[4] Fragility (i.e., failure probability) = Φ[ln(A/Am)/ßc]. Where: A is the g level in question; Am is the median seismic capacity in units of g; and the composite beta ßc = (ßu^2 + ßr^2)^0.5 (Reference [18])

[5] The convolution is the seismic initiating event annual frequency of the hazard interval multiplied by the plant level HCLPF fragility failure probability for that same hazard interval. The results per hazard interval are then straight summed to produce the overall total SCDF across the hazard curve.

Seismic Large Early Release Frequency

The LSCS SLERF for this risk evaluation is obtained by multiplying the estimated SCDF shown in Table E4- 4 (1.1E-5/yr) by the average seismic conditional large early release probability (SCLERP). An estimate of the average SCLERP is calculated using

- Seismic insights from the LSCS RMIEP SPRA (Reference [8]); and
- Level 2 PRA accident sequence progression information from the quantification results of the current LSCS FPIE PRA (Reference [13]) adjusted to reflect the influence of seismic-induced failures.

This SLERF methodology is discussed below according to the following topics:

- Overview of LSCS 1993 RMIEP seismic PRA
- Spectrum of seismic-induced core damage accident sequence types
- CLERP as a function of seismic core damage accident sequence type
- Application of SLERF in RICT Calculations

<u>Overview of LSCS 1993 RMIEP Seismic PRA</u>

The 1993 LSCS SPRA was performed for the USNRC by Lawrence Livermore National Laboratory (LLNL) staff in support of NRC NUREG/CR-4832 / UCID-21245, Analysis of LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP). Volume 8 of the NUREG/CR-4832 contained the LSCS seismic analysis, i.e. level 1 SPRA. This document will be hereafter referred to as the LSCS RMIEP SPRA. A Level 2 SPRA was not included in the scope. As previously noted the LSCS RMIEP SPRA was cited in the LSCS IPEEE submittal to the USNRC (Reference [7]).

The LSCS RMIEP SPRA includes the following three fundamental technical areas:

- Seismic hazard analysis
- Response and fragility analysis
- SPRA systems and accident sequence analysis

The seismic hazard analysis was performed using methodologies contained in NUREG/CR-5250 (1989) with adjustments made based on site-specific data available in the LSCS UFSAR. This analysis covered the typical aspects of seismic sources, attenuation and site amplification. Hazard exceedance curves and ground motions were provided. The SPRA analysis was performed for the g-PGA (peak ground acceleration) motion metric.

The fragility analysis was primarily plant-specific. Structural fragility analyses were performed based on the Seismic Safety Margins Research Program (SSMRP) developed by LLNL in NUREG/CR-

2015 for the NRC. Structural fragilities were performed for various structures including the reactor building, diesel generator building, containment and its internal structures (e.g. reactor supports), and the CST. Equipment fragility calculations for select equipment were developed by extrapolating design information and covered hundreds of equipment items and the typical spectrum of equipment types (e.g., pumps, heat exchangers, tanks and accumulators, buses and transformers, circuit breakers, reactor internals, NSSS piping, non-NSSS piping, valves, cable trays, strainers, etc.). Some generic equipment fragility values were also utilized.

The PGA seismic hazard curve and fragility information was integrated into the LSCS RMIEP internal events at-power PRA event trees for the quantification of the SPRA. All systems dependent upon offsite power were removed from the event trees in the SPRA. The LSCS RMIEP SPRA explicitly quantified the following types event trees and associated accident sequences:

- Large LOCA
- Medium LOCA
- Small LOCA
- Loss of Offsite Power

Spectrum of Seismic-Induced Core Damage Accident Sequence Types

The estimation of an average SCLERP requires as an input the assessment of the contribution of different accident sequence types to seismic core damage frequency (SCDF). The contribution of various accident sequence types (or accident classes) to core damage frequency at a given plant is not necessarily the same between FPIE PRA and other hazard (e.g., seismic) PRAs. Although the LSCS RMIEP SPRA was performed prior to the development of current SPRA methods and standards, that study does provide useful insights into seismic accident sequences. Therefore, the results from the LSCS RMIEP SPRA are used here to define the spectrum of seismic-induced accident sequences types to SCDF.

The categories of SCDF sequence types considered are as follows:

- Seismic-LOOP with early loss of injection: These are seismic-induced loss of offsite power scenarios with RPV coolant injection failure at t=0.

- Seismic-LOOP with loss of containment cooling: These are seismic-induced loss of offsite power scenarios with RPV coolant makeup initially successful but containment cooling (e.g., RHR) is not successful. Adequate core cooling is subsequently failed (e.g., harsh environment in reactor building) due to primary containment overpressurization and failure.

- Seismic-LOOP with Seismic-LOCA and early loss of injection: These are scenarios with a seismic-induced LOOP and seismic-induced LOCA (small, medium or large) and RPV coolant injection failure at t=0. [Note: Given the very high capacity of NSSS piping in comparison to the very low capacity of offsite power, the contribution of Seismic-LOCA with offsite AC available to seismic risk is negligible and is not listed here as a separate category.]

- Seismic-LOOP with Seismic-LOCA and loss of containment cooling: These are scenarios with a seismic-induced LOOP and seismic-induced LOCA (small, medium or large) with RPV coolant makeup initially successful but containment cooling (e.g., RHR) is not successful. Adequate core cooling is subsequently failed (e.g., harsh environment in reactor building) due to primary containment overpressurization and failure. [Note: Given the very high capacity of NSSS piping in comparison to the very low capacity of offsite power, the contribution of Seismic-LOCA with offsite AC available to seismic risk is negligible and is not listed here as a separate category.]

- Seismic-ATWS Unmitigated: These are seismic-induced failure to SCRAM scenarios with failure of reactivity control (e.g., failure of standby liquid control). These accidents proceed with high reactor power discharge into containment resulting in dynamic loading and failure of the primary containment. Adequate core cooling is failed (e.g., harsh environment in reactor building) upon primary containment failure. [Note: Given the very low probability of random failure to SCRAM, seismic-induced events with random failure to SCRAM are encompassed by this accident class category.]

- Direct to Core Damage: These are scenarios with significant seismic-induced failures that are modeled directly as core damage. Such scenarios include key structural failures (e.g., RPV support failure, reactor building and control building structural failure) and ISLOCA scenarios.

- Seismic-Transients with early loss of injection: These sequences have offsite power available (and thus potential use of balance of plant (BOP) equipment, e.g., Feedwater), and RPV coolant injection failure at t=0. [Note: The LSCS RMIEP SPRA specifically presumes a LOOP occurs and thus precludes use of BOP systems; the risk contribution to SCDF from seismic "transients" is typically very small.]

- Seismic-Transients with loss of containment cooling: These sequences have offsite power available (and thus potential use of BOP equipment, e.g., Feedwater), RPV coolant makeup is initially successful but containment cooling (e.g., RHR) is not successful. Adequate core cooling is subsequently failed (e.g., harsh environment in reactor building) due to primary containment overpressurization and failure. [Note: The LSCS RMIEP SPRA specifically presumes a LOOP occurs and thus precludes use of BOP systems; the risk contribution to SCDF from seismic "transients" is typically very small.]

The above accident sequence categories cover the key critical safety functions (reactivity control, core cooling, RPV and primary containment integrity) and are sufficient to describe the spectrum of SCDF accident sequences.

Based on the seismic accident sequence results in the LSCS RMIEP seismic analysis, the spectrum of LSCS SCDF accident sequence types is summarized in Table E4- 5, along with their percentage contributions to the LSCS RMIEP seismic CDF results. The largest contribution (77%) to SCDF was from seismic-induced LOOP scenarios with early loss of coolant injection (e.g., seismic-induced loss of 125V DC, seismic-induced loss of AC buses); this result is typical of BWR SPRAs. The next two most dominant accident sequence types were S-LOOP with long term loss of containment cooling (11%) and S-LOOP with S-LOCA with early loss of injection (10%). The high contribution from these second and third sequence types are due to four comparatively low fragilities and conservative modeling of their effects, i.e., Am=1.0g for RHR, LPCS, HPCS and RCIC pump suction strainers and thus all core cooling (injection) is modeled as failed and since the RHR pump suction strainers have the same low fragilities, loss of containment cooling occurs (primary containment venting is assumed failed).

CLERP as a Function of SCDF Accident Sequence Type

The next step in the estimation of an average seismic CLERP is to estimate the CLERP for each SCDF accident sequence type. A given accident sequence type may not result in a core damage event until well after the PRA "Early" release time frame (defined in the LSCS FPIE PRA as ≤ 5 hours from the time of the cue for a General Emergency declaration; per Section 5.4.1 of Reference [20]. Conversely, some accident sequence types would, by PRA convention, be modeled directly as a LERF, such as a station blackout scenario with failure to manually isolate containment isolation valves that are initially open and do not automatically isolate as designed. Seismic CLERP as a function of SCDF accident sequence type is summarized in Table E4- 5.

Table E4- 5:

Spectrum of SCDF Accident Sequences and Associated SCLERP

LSCS RMIEP L1 SPRA Accident Sequence Type	%SCDF[6]	SCLERP[7]	Comment
S-LOOP with early loss of injection	77%	2.0E-02	Based on CLERP results for LOOP with no injection at t=0 accidents with no AC recovery (i.e., Class IBE) and no coolant injection recovery in LaSalle FPIE Level 2 PRA. LaSalle Mark II primary containment does not have steel shell liner with air gap (such as in Mark I containments) and thus likelihood of a "High" magnitude release for an unmitigated core damage accident is lower in comparison to a Mark I containment design.
S-LOOP with loss of containment cooling	11%	5.0E-02	Declaration of a general emergency would be in accordance with LaSalle Emergency Action Levels. However, the LaSalle PRA includes a 5% probability (basic event ID "GEN-EMERG") that the General Emergency declaration is delayed and thus can result in an "Early" release for these sequences (refer to Appendix G of the LaSalle Level 2 PRA notebook, Reference [20]. Using the 5E-02 SCLERP value is conservative because it does not account for the primary containment failure location in reducing release magnitude (i.e., if failure occurs in the wetwell airspace the release would be scrubbed and not a "High" magnitude release).
S-LOOP with S-LOCA with early loss of injection	10%	1.9E-02	SCLERP would be similar to LaSalle FPIE LOOP early loss of injection case above except the probabilities of containment failure due to certain energetic phenomena (e.g., direct containment heating; high pressure blowdown overwhelming vapor suppression) are much lower likelihood (or even precluded) given the LOCA condition.
S-LOOP with S-LOCA and loss of containment cooling	2%	5.0E-02	Same basis discussed above for S-LOOP with loss of containment cooling.

6 Results from the LSCS RMIEP SPRA are used here to estimate the percentage contributions of the spectrum of seismic-induced accident sequences types to total SCDF. (Reference [8]).

7 These are FPIE CLERP estimates using information from the LSCS FPIE PRA that are adjusted to reflect seismic considerations (e.g., no credit for recovery of offsite power or injection), yielding the "SCLERP" label.

Table E4- 5:

Spectrum of SCDF Accident Sequences and Associated SCLERP

LSCS RMIEP L1 SPRA Accident Sequence Type	%SCDF[6]	SCLERP[7]	Comment
S-ATWS unmitigated	< 1%	5.0E-02	Based on review of the LaSalle Level 2 PRA, the majority of unmitigated ATWS scenarios would be either Moderate/Early or Low/Early given the likely primary containment failure location is in the wetwell airspace and would be scrubbed. The 0.05 SCLERP is a nominal value of those AWTS sequences that would result in H/E without credit for injection systems (refer to Table 6.6-6 of the LaSalle Level 2 PRA Notebook, Reference [20].
Scenarios direct to core damage (e.g., RPV support failure, RB and CB structural failures, ISLOCA)	< 1%	1	SCLERP slightly conservative (e.g., some such contributors would not necessarily be directly High Magnitude/Early release, e.g., CB failure alone).
S-Transients (no LOOP) with early loss of injection	<1%	2.0E-02	Based on LSCS FPIE PRA, LOOP with no injection at t=0 accidents with no AC recovery (i.e., Class IBE) and no coolant injection recovery in Level 2 FPIE PRA.
S-Transients (no LOOP) with loss of containment cooling	<1%	5.0E-02	Same basis discussed above for S-LOOP with loss of containment cooling.
Sequence-Weighted Average SCLERP	---	0.02	Sum of (%SCDF x SCLERP) over all sequence types

The sequence-weighted average SCLERP over the SCDF accident sequence contributions and assigned SCLERPs is estimated as 0.02. Refer to later discussion regarding conservative increase in this value for use in RICT calculations. In addition, the following discussions regarding seismic-induced structural failure of primary containment and primary containment isolation failure (both random and seismic-induced) are provided to support the reasonableness of the average SCLERP estimation (e.g., there are no normally-open AC-powered MOV PCIVs):

- Primary Containment Structural Fragility: The primary containment structural fragility information in the RMIEP study does not report the HCLPF and Am values in g, PGA but rather in structural strength (e.g., ksi). The primary containment, as suspected, is sufficiently strong that seismic-induced failure of the primary containment does not even appear in the RMIEP SPRA quantified results. Similarly, the RPV pedestal fragility is sufficiently high in seismic capacity (Am >4g, PGA based on Reference [8] Table 8.2 and using SA/PGA information from Figures 5.9 and 5.10 from Reference [8]) that this structural fragility also does not appear in the RMIEP SPRA quantified results. As such, the seismic capacity of the primary containment and the RPV pedestal are sufficiently high that they are non-significant contributors to SLERF and do not change the average SCLERP calculation previously discussed.

- Primary Containment Isolation Random Failure: Random failure of primary containment isolation is already included in the average SCLERP estimation discussed previously. A significant fraction (approximately 25%) of the FPIE PRA LERF cutset results (biased to reflect seismic consideration, as discussed previously) used in the SCLERP calculation for early loss of injection scenarios involve non-seismic failures of primary containment isolation (e.g., pre-existing unisolated containment, basic event 2CNHU-PREINIT).

- Primary Containment Isolation Fragility: Seismic-induced failure of primary containment isolation is very low likelihood and encompassed by the SCLERPs used in Table E4- 7. The primary containment isolation valves (PCIVs) modeled in the LSCS L2 PRA containment isolation fault tree are summarized in Table E4- 6. Note that the containment isolation fault tree also includes contributions for pre-existing containment leakage and various containment hatches not properly closed (the probabilities of these potential pre-existing contributors are not influenced by the seismic event). As can be seen from Table E4- 6, the containment isolation valves of interest to the LERF risk metric are air-operated valves (AOVs), most normally-closed at-power, that fail-safe closed on loss of pneumatic or electric power (e.g., seismic-induced LOOP). Successful primary containment isolation for seismic-induced accidents is not dependent upon pneumatic supply, electric power, or containment isolation signals (i.e., ~99% of SCDF involves seismic-induced LOOP and the PCIVs fail-safe closed under such conditions). The pre-existing primary containment leakage probability (that is conservatively modeled as a LERF release) is higher in probability.

  The PCIVs have very high seismic capacities such that seismic loading will have a negligible likelihood of failing the PCIVs in the open position. These PCIVs are AOVs that fail-safe closed via internal spring force inside the AOV operator. Once closed, these valves do not need to open again during or after the seismic event. Therefore, they do not meet the definition of an "active" valve per the air operated valve equipment class (per the EPRI SQUG Generic Implementation Procedure, GIP, and EPRI NP-7149 Seismic Adequacy of Equipment Classes). The spring will successfully cause the PCIVs to shut at accelerations much greater than those associated with the functional failure capacity used to determine the fragility of active valves. As such, these PCIVs are essentially inactive valves, which are inherently rugged as there is not a credible seismic failure mechanism that would prevent the valves from failing shut as desired. In addition, both in-series AOV PCIVs in a penetration line would have to seismically fail to fail-safe closed to result in an open release pathway.

Table E4- 6

Summary of LaSalle Level 2 PRA Primary Containment Isolation

Penetration[8],[9]	Penetration Pathway Normal Status	Valve ID	Valve Type	SPRA Comment
Main Steam Line A (M-1)	Normally open at-power	1(2)B21-F022A 1(2)B2-1F028A	AOV AOV	Loss of pneumatic force to AOVs upon seismic-LOOP (PC isolation signal not necessary). AOVs fail-safe closed and not required to re-open. As such, fragility would be assessed as "rugged".
Main Steam Line B (M-2)	Normally open at-power	1(2)B21-F022B 1(2)B2-1F028B	AOV AOV	Loss of pneumatic force to AOVs upon seismic-LOOP (PCIV signal not necessary). AOVs fail-safe closed and not required to re-open. As such, fragility would be assessed as "rugged".
Main Steam Line C (M-3)	Normally open at-power	1(2)B21-F022C 1(2)B2-1F028C	AOV AOV	Loss of pneumatic force to AOVs upon seismic-LOOP (PCIV signal not necessary). AOVs fail-safe closed and not required to re-open. As such, fragility would be assessed as "rugged".
Main Steam Line D (M-4)	Normally open at-power	1(2)B21-F022D 1(2)B21-F028D	AOV AOV	Loss of pneumatic force to AOVs upon seismic-LOOP (PCIV signal not necessary). AOVs fail-safe closed and not required to re-open. As such, fragility would be assessed as "rugged".

[8] This Table lists the primary containment pathways modeled in the containment isolation fault tree of the LSCS Level 2 PRA. (Reference [20])

[9] The LSCS primary containment isolation fault tree logic also includes reactor well drain (penetration M-65) but this pathway has two in-series manual valves that are normally locked-closed during power operation (per LSCS procedure LOP-LV-01(2)M) and they would not be opened during power operation.

Table E4- 6

Summary of LaSalle Level 2 PRA Primary Containment Isolation

Penetration[8],[9]	Penetration Pathway Normal Status	Valve ID	Valve Type	SPRA Comment
Main Steam Drain Line (M-22)	• Normally open at power • At-power PRA uses 1E-2 probability that in drain mode at time of event.	1(2)B21-F016 1(2)B21-F019	MOV MOV	Loss of motive (electric) power results in MOVs fail as-is. These MOVs are 3" Gate valves. Shutdown and post-accident position is closed. The associated SCLERP for this line could be assumed equal to the PRA-modelled 1E-2 probability that the lines is open at the time of the event (given that loss of AC power can reasonable be assumed for much of the seismic hazard curve). This low value of 1E-02 does not affect the results of this assessment of primary containment isolation which is already significantly below the average SCLERP used in the RICT calculations.
Drywell Purge / Inerting Supply (M-20)	• Normally closed at-power • At-power PRA uses 1E-2 probability that operator has DW purge open at time of event	1(2)VQ029 1(2)VQ030 1(2)VQ042	AOV AOV AOV	Loss of pneumatic force to AOVs upon seismic-LOOP (PCIV signal not necessary). AOVs fail-safe closed and not required to re-open. As such, fragility would be assessed as "rugged".

Table E4- 6

Summary of LaSalle Level 2 PRA Primary Containment Isolation

Penetration[8],[9]	Penetration Pathway Normal Status	Valve ID	Valve Type	SPRA Comment
Drywell Purge Exhaust (Vent) to SGTS (M-21)	• Normally closed at-power • At-power PRA uses 1E-2 probability that operator has DW purge open at time of event	1(2)VQ034 1(2)VQ036	AOV AOV	Loss of pneumatic force to AOVs upon seismic-LOOP (PCIV signal not necessary). AOVs fail-safe closed and not required to re-open. As such, fragility would be assessed as "rugged".
Wetwell Purge / Inerting Supply (M-66)	• Normally closed at-power • At-power PRA uses 1E-2 probability that operator has WW purge open at time of event	1(2)VQ026 1(2)VQ027 1(2)VQ043	AOV AOV AOV	Loss of pneumatic force to AOVs upon seismic-LOOP (PCIV signal not necessary). AOVs fail-safe closed and not required to re-open. As such, fragility would be assessed as "rugged". 1(2)VQ026 and 027 are on 26" Purge Supply. 1(2)VQ043 is 8" N2 Inerting Supply.
Wetwell Purge Exhaust (Vent) to SGTS (M-67)	• Normally closed at-power • At-power PRA uses 1E-2 probability that operator has WW purge open at time of event	1(2)VQ031 1(2)VQ040	AOV AOV	Loss of pneumatic force to AOVs upon seismic-LOOP (PCIV signal not necessary). AOVs fail-safe closed and not required to re-open. As such, fragility would be assessed as "rugged".

Table E4- 6

Summary of LaSalle Level 2 PRA Primary Containment Isolation

Penetration[8],[9]	Penetration Pathway Normal Status	Valve ID	Valve Type	SPRA Comment
Drywell Floor Drain (M-98)	• Normally closed at power but valves auto-open on level to permit drainage. • At-power PRA uses 1E-2 probability that in drain mode at time of event.	1(2)RF012 1(2)RF013	AOV AOV	Loss of pneumatic force to AOVs upon seismic-LOOP (PCIV signal not necessary). AOVs fail-safe closed and not required to re-open. As such, fragility would be assessed as "rugged".
Drywell Equip. Drain (M-96)	• Normally closed at power but valves auto-open on level to permit drainage. • At-power PRA uses 1E-2 probability that in drain mode at time of event.	1(2)RE024 1(2)RE025	AOV AOV	Loss of pneumatic force to AOVs upon seismic-LOOP (PCIV signal not necessary). AOVs fail-safe closed and not required to re-open. As such, fragility would be assessed as "rugged".

Table E4- 6

Summary of LaSalle Level 2 PRA Primary Containment Isolation

Penetration[8],[9]	Penetration Pathway Normal Status	Valve ID	Valve Type	SPRA Comment
Pre-Existing Containment Failure[10]	• This is a PRA-modelled pre-initiator for incorrect valve alignment and is modelled with a 2.3E-3 probability.	Various valves	Various valves	Incorrect line-up of PCIVs results in a containment failure. The associated SCLERP is equal to the PRA-modelled failure probability. This negligible value of 2.3E-03 does not affect the results of this assessment of primary containment isolation.
Hatches or Doors are not Properly Sealed[10]	• This is a PRA-modelled pre-initiator for incorrect valve alignment and is modelled with a 3.0E-6 probability.	Various Hatches & Doors	Various Hatches & Doors	Incorrect sealing (or closure) of containment hatches or doors results in a containment failure. The associated SCLERP is equal to the PRA-modelled failure probability. This negligible value of 3.0E-06 does not affect the results of this assessment of primary containment isolation.

Some primary containment penetrations use motor operated valves (MOV) for containment isolation which would require electric power for closure and for an isolation signal. However, such PCIV MOVs are not significant to LERF for one or more of the following reasons:

- MOV in closed position during at-power operation and at the time of the seismic event (e.g., main steam line drains)
- Very small line (e.g., 1" diameter instrument gas line)
- AOV or check valve PCIV in-series with the MOV
- Penetration is a closed-loop system that would not represent a LERF (i.e., High magnitude release) pathway (e.g., RWCU)

Based on the information in Table E4- 5, an estimate of CLERP based on the LSCS RMIEP SPRA results is 0.02, i.e., seismic LERF is equal to 2% of the seismic CDF estimate).

[10] The primary containment isolation fault tree also models contributions of pre-existing primary containment leakage and various containment hatches not properly sealed.

As discussed previously, the calculated average SCLERP estimate contains conservatisms but to address the potential uncertainty of newer ground motions and structural fragility calculations if a more recent SPRA were performed (and thus potential increase in the risk contribution of direct to core damage accident sequence types) the average SCLERP for use in RICT calculations is increased to 0.2. This conservative increase by a factor of 10x is not based on specific knowledge or expectations that such direct to core damage scenarios would increase (or that any increase would be significant) in risk contribution if a new SPRA were performed but is made merely as a nominal conservative factor increase. The conservative increase to a 0.2 average SLCERP is not expected to over-penalize RICT calculations.

Therefore, a conservative estimate of SLERF is:

SLERF = 1.1E-5/yr (SCDF from Table E4- 4) **x 0.2** (CLERP) **= 2.2E-6/yr.**

The above estimated SLERF will be used for the base case SLERF value for RICT calculations that apply when the primary containment is inerted. If a RICT is being entered during a period when the primary containment is de-inerted, a different SLERF penalty of 1.1E-05/yr (SCDF = SLERF, CLERP = 1.0) will be applied to address the increased potential for hydrogen deflagration events in the primary containment. This is deemed conservative since the LSCS Level 2 FPIE PRA (Reference [20]) credits steam inerting in the primary containment with an estimated 0.5 probability that the steam inerting would fail to mitigate the hydrogen deflagration event. Given the uncertainty in the steam inerting value of 0.5 and the small time frame for potential de-inerted conditions, a conservative assumption for CLERP of 1.0 will apply when the primary containment is de-inerted.

To summarize application of primary containment SCLERP values:

- **SCLERP of 0.2 will apply when the primary containment is inerted.**
- **SCLERP of 1.0 will apply when the primary containment is de-inerted.**

Application of SLERF in RICT Calculations

The SLERF estimate documented above is conservatively used in the RICT process. Conservatism in the RICT process derives from the proposed approach to apply the total estimated annual seismic LERF as a delta SLERF in each RICT calculation, regardless of the duration of the completion time. The total estimated annual seismic CDF and LERF will be applied starting at time zero for each RICT calculation.

Seismic Summary

Estimates of SCDF and SLERF have been derived for use in the LSCS TSTF-505 program. Since the estimates are intended to be treated as conservative values in the RICT calculations for that program, the results for the case of plant level HCLPF = 0.30g PGA with ßc = 0.4 will be used, i.e., SCDF = 1.1E-5/yr and SLERF = 2.2E-6/yr.

- When containment is inerted then a penalty of SCDF = 1.1E-5/yr and SLERF = 2.2E-6/yr is used.
- When containment is de-inerted then the penalty of SCDF = 1.1E-5/yr and SLERF of 1.1E-5/yr is used to address the increased potential for hydrogen deflagration events in the primary containment.

  Note; RICT calculations use the formulaic construct of: ΔCDF x Time in Configuration (same formula for ΔLERF metric). In the case of the seismic risk contribution to the RICT calculations, the total SCDF and total SLERF seismic penalties are treated as ΔSCDF and ΔSLERF. In effect this approach is assuming the base seismic risk is negligible, which has the effect of producing conservative ΔSCDF and ΔSLERF values.

4 Extreme Winds Analysis

This Section provides an analysis of the High Winds / Tornados risk impact for LSCS. As described in Section 7 of the LaSalle IPEEE submittal (Reference [7]), LaSalle Station performed a bounding analysis for high winds and tornado. Extreme winds were removed from further consideration after the plant structures were evaluated. The median frequency of CDF due to tornadoes was calculated to be 3E-7/year. Due to the conservatism introduced in the bounding analyses by neglecting the plant system failures and consequence analysis and due to the low CDFs resulting from the bounding analyses, the IPEEE study concluded that high winds and tornadoes do not present a significant contributor to plant risk. Various later high wind studies have been performed for LaSalle and have likewise concluded low risk from this hazard.

Wind Pressure

As discussed in Section 3.3.2.1 of the LaSalle UFSAR (Reference [21]), the wind loading design parameters of the LaSalle design-basis tornado include:

- maximum rotational velocity of 300 mph
- translational velocity of 60 mph
- external pressure drop of 3 psi at the vortex within a 3 second interval, and
- a radius of maximum wind speed of 227 feet

Tornado wind speed hazard curve information for LaSalle (as well as other U.S. nuclear plants) are provided in Table 6-1 of NUREG/CR-4461, Rev. 2 (Reference [22]). The NUREG/CR-4461 tornado hazard estimation methodologies include accepted practices and consider uncertainties. The Enhanced Fujita Scale based tornado hazard curve for the LaSalle plant shows that the annual frequency of occurrence of the design-basis tornado on the LaSalle site is <1E-7/yr. Table 6-1 of NUREG/CR-4461 provides a 1E-05, 1E-06 and 1E-07 annual exceedance frequency data point for each of the U.S. nuclear plant sites. The 1E-07 exceedance frequency of the Enhanced Fujita Scale hazard curve for the LaSalle plant is 220 mph. Thus, even when not including the translational velocity, the LaSalle design-basis tornado wind speed is below 1E-07/yr, using the Enhanced Fujita Scale.

Tropical storms (i.e., hurricanes) are not a concern at LaSalle due its location (i.e., >600 miles inland from the coast). Straight winds (e.g., due to thunderstorms) are typically in the 50 – 70 mph range, although in rare cases may be over 100 mph. However, the hazard curve for straight winds tails off very quickly, such that below approximately 1E-3/yr, straight winds do not affect the overall wind hazard for areas with hurricane and/or tornado hazards (Reference [23]).

High Wind Missiles

LaSalle's design missile spectra and characteristics are provided in Section 3.5.1.4 of the UFSAR [21] and summarized in Table E4- 7.

Table E4- 7
Design Basis Tornado Missiles

Missile	Physical Properties	Impact Velocity (mph)
Wood Plank	4" x 12" x 12'	225
Automobile Wt. 4000 lbs	20 ft^2 front area	50

The maximum height reached by the automobile is 25 feet above grade (Reference [21]).

Extreme winds were screened in the LaSalle IPEEE; the median frequency of CDF was calculated to be 3E-7/yr [7]. Subsequent to the IPEEE, LaSalle performed tornado missile protection (TMP) evaluations in response to the NRC issued Regulatory Issue Summary (RIS) 2015-06 (Reference [24]). As a result of the LaSalle TMP project, two systems were determined to be vulnerable to tornado missiles and not in conformance with the LaSalle design and license bases. Specifically, the Main Control Room (MCR) HVAC (System VC) and Auxiliary Electrical Equipment Room (AEER) HVAC (System VE) systems are located on the 796' and 802' levels in the Auxiliary Building (AB) (References [25], [26]). Note that 480V MCC 236X-1 was initially determined to be non-conforming but was subsequently determined to be protected from tornado missiles (Reference [27]). The floor above this level (815') is only 6" thick reinforced concrete (Reference [28]), which is not evaluated to protect against design basis vertical missiles.

However, only a few missiles types are capable of perforating or failing a 6" concrete floor. Analyses were performed in support of the NEI 17-02 methodology (Reference [29]) to determine the missile types that could perforate or fail 4" and 8" reinforced concrete roofs. Missile speeds of 153 mph (2/3 x 230 mph)[11] were analyzed. Based on the analysis results, the following missiles are able to perforate or fail a 6" concrete slab (Reference [29]):

- Rebar
- Utility Pole
- 6" Pipe
- Large Container
- Large Equipment
- Pallet Rack
- Vehicle

With the exception of the rebar, all of the above missile types are heavy, with the lightest one being 284 lbs (6" pipe); the rest are greater than 675 lbs. Per Section 3.5.1.4 of the SRP (Spectrum II) [30], the Utility Pole (Missile D – 1500 lb) and Automobile (Missile F – 4000 lb) do not need to be considered as credible missiles greater than 30 feet above grade. The 815' level of the barrier is greater than 100' above grade; therefore, it is unlikely that any of the other heavy missiles could achieve that height in a tornado. Although a limited number of these missile types may exist in elevated structures that are not protected against tornados (the Turbine Building, the refuel floor of the Reactor Buildings, and upper levels of the Aux Building), they would still need to be lifted well above the 815' level to achieve the speeds necessary to damage or perforate the 6" floor.

In order for any of the above missile types to achieve a horizontal velocity of 230 mph (with the equivalent vertical velocity of 153 mph), the wind speed would need to be greater than 360 mph. This is based on the missile velocity fractions (of tornado wind speed) provided in the SRP (Missile Spectrum A) (Reference [30]). The frequency of 360 mph tornado wind speeds at LaSalle is significantly less than 1E-6/yr (Reference [22]).

Other missiles capable of causing damage to the VC/VE systems, that are not in the SRP (e.g., large equipment, pallet rack), would be expected to have similar fractions as the SRP missiles (i.e., 0.2 – 0.6). Even if a horizontal missile speed of 153 mph were considered (i.e., such that vertical and horizontal missiles speeds were equal), only the steel rod would require wind speeds less than 360 mph; the steel rod would need a wind speed of 255 mph, which has an annual exceedance frequency at the LaSalle site less than 1E-6/yr (Reference [22]).

Therefore, the frequency of tornado wind speeds needed to develop missiles (design basis and beyond) capable of penetrating the 6" 815' level floor above the vulnerable SSCs, is much less than 1E-6/yr.

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[11] RG 1.76 (Reference [41]) considers that missiles have vertical velocities equal to 67% of horizontal velocities.

In the unlikely event that a tornado missile penetrated the 815' floor, the VC and VE systems are each spatially separated by train, with at least 20' between components from different trains, and more than 40' for many (Reference [31]). Therefore, a single missile causing the failure of both trains is not likely. However, the VC and VE systems are co-located, such that a missile impacting one train of either VC or VE could conceivably result in failure of the same train of the other system (e.g., if a missile strikes a VC Train 'A' SSC, VE Train 'A' may also be impacted). Additionally, there is one section of common return ducting for each system that could be struck by a single missile, potentially resulting in the failure of both trains of one system.

Loss of ventilation to the control room and/or the AEER would not result in immediate failure to any safety related or risk significant components. Several hours would elapse before the affected rooms would reach temperatures that could potentially result in higher failure rates for components. Procedures are available to mitigate failures:

- If a single train of either VC and/or VE were failed due to a tornado missile strike, operations would start the standby train(s). Normally in a dual unit LOOP (DLOOP), the operating VC and VE trains will stop and various dampers will isolate. Operations is directed by procedures to start a train of VC and VE following a DLOOP, regardless of whether there has been tornado missile damage. Numerous alarms are available to provide cues to the operators, as well as procedural guidance, including abnormal operation procedure LOA-TORN-001 (Reference [32]).
- If either the VC or VE systems are completely failed, LOA-FSG-005 (Reference [33]) provides direction to open doors and align temporary cooling for both the AEER and the MCR. Equipment (e.g., door stops, portable fans) is staged in boxes within the plant.

As discussed above, failure of ventilation will result in room heatup, but not imminent equipment failure. In fact, even after equipment design temperatures are reached, failure is not guaranteed, but degradation occurs and failure is more likely. Based on room heat-up calculations and the LaSalle UFSAR Fire Protection Report, it is conservatively assumed that at least 2 hours are available to perform the above actions before design temperatures in the MCR or AEER are reached. Estimates from these calculations indicate that several additional hours are available.

Therefore, tornado missiles are screened from consideration at LaSalle. The plant is designed to protect against tornado missiles. Although vulnerabilities were found that do not conform to the design basis (References [25], [26]), the as-built and operated plant affords significant protection of these vulnerabilities from tornado missiles. Specifically,

- There is protection to the vulnerable SSCs by 6" reinforced concrete floors above the level where these components are located.
- The height missiles must achieve (~100' above grade) in order to potentially penetrate the 6" floor severely limits the number and type of missiles that would pose a threat. It is extremely unlikely that any missiles capable of penetration the 6" floor would achieve the elevation necessary, even in very intense tornadoes.
- The tornado wind speeds necessary to generate missile velocities capable of penetrating the 6" barrier have a frequency much less than 1E-6/yr.

- Failure of the vulnerable equipment does not result in the direct failure of SSCs needed to mitigate core damage following a tornado event.
    - Substantial time is available before the affected room(s) heat up to potentially damaging temperatures; even then the equipment would not fail immediately.
    - Procedures are available to establish alternate cooling for the affected rooms.
- There are no vulnerabilities to tornado missiles at LaSalle that would specifically affect containment integrity and large early release probability.

Configuration Specific Considerations

An assessment of the high wind and tornado screening was performed considering SSCs out of service for maintenance. Based on the considerable missile protection and the limited vulnerabilities at LaSalle, high wind and tornado risk would not be significantly affected by allowed maintenance configurations.

5 External Flooding Assessment

This Section provides an analysis of the external flooding risk impact for LaSalle.

Current Risk Basis

The external flooding hazard at the site was recently updated as a result of the post-Fukushima 50.54(f) Request for Information. A flood hazard reevaluation report (FHRR) was submitted to NRC for review on March 12, 2014 (Reference [34]). The results indicate that flooding from all mechanisms except local intense precipitation (LIP) and probable maximum storm surge (PMSS) were bounded by the current licensing basis (CLB). Only LIP and PMSS require evaluation in a Focused Evaluation (FE) to determine if the plant's current design basis bounds the reevaluated flood parameters.

The LIP maximum still water elevation levels (SWELs) exceed the CLB at LaSalle and cause ponding to occur at the plant's exterior doors. An evaluation was performed in the plant's FE, submitted to NRC on March 8, 2017 (Reference [35]), to determine the amount of water capable of entering the plant between the exterior doors and their thresholds. It was determined that no significant water accumulation is estimated in Unit 2 in rooms or buildings housing SSCs. This is due to the configuration of the plant where water will be intercepted by stairwells or floor drains. In Unit 1, water will accumulate in the Low-Pressure Core Spray (LPCS)/Reactor Core Isolation Cooling (RCIC) Pump Cubicle and the RHR A Cubicle which are the only areas of the plant with safety-related SSCs impacted by a LIP event. The SSCs in these areas are at least 18 inches off the floor and would require approximately 28,613 gallons of water to accumulate to this height. Given the flooding depths and duration of the flood, it was calculated that the ingress would be limited to 7,477 gallons during a LIP event. Therefore, the available physical margin (APM) is adequate to ensure there would be no loss of safety related SSCs given a LIP event (Reference [35]).

The PMSS SWELs estimated in the FHRR were not bounded by the CLB, however, the reevaluated levels were approximately 9 feet below site grade. The wind-generated wave run-up at the lake screen house and the core standby cooling system (CSCS) inlet structure is 710.6 feet and 712 feet, respectively. The ground surface elevation around these two structures is approximately 713.8 feet and therefore, water will be contained in the intake flume. The APM of 3.2 feet and 1.8 feet are adequate to conclude there will be no impacts to SR SSCs due to a PMSS event (Reference [35]).

The results of the FE were submitted to NRC for review and a staff assessment was issued on August 23, 2017 (Reference [36]). The NRC acknowledged the results presented in the FE concluding that there were no impacts to SR SSCs from the LIP and PMSS events and the design basis of the plant is adequate to mitigate the effects from external flood causing mechanisms with sufficient margin. The SE stated the post-Fukushima response requirements for external flooding were met and the 10 CFR50.54(f) request for information is closed.

Challenges Posed

Weather induced Loss of Offsite Power (LOSP) is a potential challenge.

Disposition for the RICT Program

As described above, flooding from a LIP or PMSS event are bounded by the plant's current design and there are no postulated impacts to SR SSCs from either mechanism. Therefore, the risk from external flooding is considered negligible and can be screened from inclusion in the RICT program. The flood hazards were reevaluated using modern day guidance for developing a plant's design basis and the design of the plant was determined to be able to mitigate the effects of the flood causing mechanisms with permanently installed passive flood protection (e.g. exterior doors and plant grade) that require no manual action for success. For the RICT program, there are no configuration specific considerations related to the screening assessment.

6 Evaluation of External Event Challenges and IPEEE Update Results

This Section provides an evaluation of other external hazards. The results of the assessment of these hazards is provided in Table E4- 11. Table E4- 12 provides the summary criteria for screening of the hazards listed in Table E4- 11.

Hazard Screening

The IPEEE for LSCS Units 1 and 2 provides an assessment of the risk to LSCS associated with other external hazards. Additional analyses have been performed since the IPEEE to provide updated risk assessments of various hazards, such as aircraft impacts, industrial facilities and pipelines, and external flooding. These analyses are documented in the UFSAR.

Table E4- 11 reviews and provides the bases for the screening of external hazards, identifies any challenges posed, and identifies any additional treatment of these challenges, if required. The conclusions of the assessment, as documented in Table E4- 11, assure that the hazard either does not present a design-basis challenge to LSCS or is adequately addressed in the PRA.

Impacts to RICT

In the application of Risk-Informed Completion Times, a significant consideration in the screening of external hazards is whether particular plant configurations could impact the decision on whether a particular hazard that screens under the normal plant configuration and the base risk profile would still screen given the particular configuration. The external hazards screening evaluation for LSCS has been performed accounting for such configuration-specific impacts. This evaluation involves several steps.

As a first step in this screening process, hazards that screen for one or more of the following criteria (as defined in Table E4- 12) still screen regardless of the configuration, as these criteria are not dependent on the plant configuration.

- The occurrence of the event is of sufficiently low frequency that its impact on plant risk does not appreciably impact CDF or LERF. (Criterion C2)
- The event cannot occur close enough to the plant to affect it. (Criterion C3)
- The event which subsumes the external hazard is still applicable and bounds the hazard for other configurations (Criterion C4)
- The event develops slowly, allowing adequate time to eliminate or mitigate the hazard or its impact on the plant. (Criterion C5)

The next step in the screening process is to consider the remaining hazards (i.e., those not screened per the above criteria) to consider the impact of the hazard on the plant given particular configurations for which a RICT is allowed. For hazards for which the ability to achieve safe shutdown may be impacted by one or more such plant configurations, the impact of the hazard to particular SSCs is assessed and a basis for the screening decision applicable to configurations impacting those SSCs is provided.

As noted above, the configurations to be evaluated are those involving unavailable SSCs whose LCOs are included in the RICT program.

Seismic-Induced Loss of Offsite Power Challenges

Past TSTF-505 applications have also included evaluation of any incremental risk associated with challenges to the facility that do not exceed the design capacity and the past submittals have focused on the challenge of seismically-induced LOOP. The methodology for computing the seismically-induced LOOP frequency is to convolve the LSCS mean seismic hazard curve with the offsite power fragility curve. Past TSTF-505 applications have approached this conservatively by performing the convolution over the entire hazard curve (not just below the design basis). That same approach is used here.

The offsite power failure probabilities for a given seismic interval are represented by failure of ceramic insulators in the power distribution system. The fragility data from Table A-0-4 of the RASP Handbook, Volume 2 (Reference [37]) is used to compute the probabilities of failure of offsite power. The data is shown below in Table E4- 8.

Table E4- 8

LaSalle Offsite Power Fragility Data

Source	HCLPF	Am	ßr	ßu
NRC RASP Handbook[12]	0.1g PGA	0.3g	0.3	0.45

Table E4- 9 provides the LOOP seismic-induced failure probability for each seismic interval based on the fragility of offsite power from Table E4- 8 and the seismic-induced LOOP frequencies for each seismic interval using the seismic hazard data from Table E4- 1.

[12] Reference [37]

Given the mean frequency and failure probability for each seismic interval, it is straightforward to compute the estimated frequency of seismically induced loss of offsite power for the LSCS site by taking the product of the interval frequency and the offsite power failure probability.

Table E4- 9

LaSalle Seismic-Induced LOOP Frequency Estimate

Seismic Interval (g, PGA)	Interval Representative Magnitude (g, PGA)	Interval Frequency (/yr)	Offsite Power Failure Probability	Seismic Interval LOOP Frequency (/yr)
%G1[13]: 0.1 - 0.3	0.17	4.90E-04	1.55E-01	7.60E-05
%G2: 0.3 - 0.5	0.39	4.65E-05	6.82E-01	3.17E-05
%G3: 0.5 - 0.7	0.59	1.09E-05	8.95E-01	9.80E-06
%G4: 0.7 - 0.9	0.79	5.03E-06	9.64E-01	4.85E-06
%G5: 0.9 - 1.1	0.99	1.92E-06	9.87E-01	1.89E-06
%G6: 1.1 - 1.3	1.20	7.72E-07	9.95E-01	7.68E-07
%G7: 1.3 - 1.5	1.40	7.72E-07	9.98E-01	7.70E-07
%G8: >1.5	1.65	6.60E-07	9.99E-01	6.59E-07
Total Seismic LOOP Frequency =				**1.26E-04**

As shown in Table E4- 9, the total seismic LOOP frequency is the sum of interval frequencies, or approximately 1.3E-4/yr. Note that this overstates the below-design challenge rate. In Table E4- 9, the %G1 seismic interval from 0.1g to 0.3g represents approximately 60% of the total seismic-induced LOOP frequency. The %G1 interval seismic-induced LOOP probability is conservative and the calculated contribution in this interval would be reduced if 2 or more intervals were used in the calculation. However, the splitting of the %G1 interval is not necessary to show this below-design seismic-induced challenge is not significant in comparison to the random LOOP frequency. In addition, if this interval was bounded by the SSE of 0.2g (as opposed to extending up to 0.3g PGA) the below-design seismic-induced LOOP frequency would be shown to be even lower.

[13] The %G1 interval encompasses the "below design basis" portion of the seismic hazard.

The internal events PRA models LOOP from plant-centered, switchyard-centered, grid-related, and weather-related events. Based on the LSCS internal events PRA, the total frequency of unrecovered loss of offsite power (i.e., the sum of the frequency times the non-recovery probability at 24 hours over these LOOP events), is 1.2E-3/yr, as shown in Table E4- 10.

Table E4- 10

Loss of Offsite Power (LOOP) Non-Recovery Frequency

LOOP Contributor	LOOP Contributor Frequency [14]	Probability of Non-Recovery by 24 hours[15]	24-hr Non-Recovered LOOP Frequency
Plant-Centered	1.62E-03	4.85E-03	7.9E-06
Switchyard-Centered	1.26E-02	1.86E-02	2.3E-04
Grid-Related	2.30E-03	1.94E-02	4.5E-05
Weather-Related	5.08E-03	1.82E-01	9.3E-04
Total			**1.2E-03**

The total (i.e., across the entire hazard curve) seismically-induced (unrecoverable) LOOP frequency is approximately 10% of the total unrecovered LOOP frequency already addressed in the PRA. The below-design seismic-induced LOOP frequency is approximately 5% of the total unrecovered LOOP frequency already addressed in the PRA; this frequency is judged to be a reasonably small fraction that it will not significantly impact the RICT Program calculations and it can be omitted.

[14] IEFs Values are the "Combined" LOOP IEF values from LS-PSA-001, Rev 7, LaSalle County Generating Station PRA IE Notebook, 2014 Update, Table 3.4-5, Posterior (per calendar year). (Reference [42])

[15] (Reference [42]) did not include these values at a duration of 24 hours. Values used here were developed by LaSalle FPIE PRA Model Owner using the same parameters in (Reference [42]) Appendix E.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
Aircraft impacts	A direct or indirect (i.e. skidding impact) collision of a portion of or an entire aircraft with one or more structures at or in the area surrounding the plant site.	Y	PS2 PS4	In the NRC Staff Evaluation of the IPEEE (Reference [9]), a probabilistic bounding analysis was performed for aircraft impact. The median frequency of CDF was calculated as 5E-7/year **(PS4)**. From Section 3.5.1.6, Aircraft Hazards of the LaSalle County Station (LSCS) UFSAR, the airports and airways in the vicinity of the site are described in Subsection 2.2.2.5 of the UFSAR (Reference [21]) **(PS2).** a. There are no federal airways or airport approaches passing within 2 miles of the station. The closest airway corridor is 3 miles away from the station. b. There are no commercial airports existing within 10 miles of the site and there is only one private airstrip within 5 miles. c. The projected landing and take-off operations out of those airports located within 10 miles of the site are far less than 500 d^2 per year, where d is the distance in miles. The projected operations per year for airports located outside of 10 miles is less than 1000 d^2 per year.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
				d. The only military facility within 10 miles of the site is the Illinois Army Reserve National Guard Training Facility. It is located approximately 1 mile northwest of LSCS cooling lake. There are no airstrips at the Training Facility. Based on this review, the Aircraft Impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Avalanche	A rapid flow of a large mass of accumulated frozen precipitation down a sloped surface.	Y	C3	The mid-western location of LaSalle station precludes the possibility of an avalanche. Based on this review, the Avalanche impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Biological events	The accumulation or deposition of vegetation or organisms (e.g., zebra mussels, clams, fish) on an	Y	C5	Hazard is slow to develop and can be identified via monitoring and managed via standard maintenance process. Actions committed to and completed by LaSalle station in response to Generic Letter 89-13

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
	intake structure or internal to a system that uses an intake structure.			provide on-going control of biological hazards. These controls are described in Exelon procedure ER-AA-340, "GL 89-13 Program Implementing Procedure" (Reference [38]). Based on this review, the Biological Event impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Coastal erosion	The wearing away of a shoreline due to wave action, tidal currents, wave currents, drainage, or winds.	Y	C3	The mid-western location of LaSalle station precludes the possibility of coastal erosion. Based on this review, the Coastal Erosion impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Drought	An extended period of months or years when a region experiences a	Y	C5	Drought is a slowly developing hazard allowing time for orderly plant reductions, including shutdowns. Based on this review, the Drought impact hazard can be considered to be negligible.

Table E4- 11 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**Lasalle Response**
	deficiency in its surface or underground water supply			There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
External Flooding	Accumulation of excessive water on the station grounds from various sources including Local Intense Precipitation and Snow Accumulation	Y	C1	See information in Section 5 of this Enclosure.
Extreme Wind or Tornado	Excessive winds, straight-line or tornadic	Y	C1	See information in Section 4 of this Enclosure.
Fog	Water droplets suspended in the atmosphere at or near the Earth's surface that limit visibility.	Y	C4	The principal effects of such events (such as freezing fog) would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for LaSalle. Based on this review, the Fog impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
Forest or Range Fire	Fires originating from outside the plant site boundary that are caused by the uncontrolled combustion of vegetation (e.g., trees, grasses, brush, etc.)	Y	C3	Forest fires were screened in the IPEEE (Reference [9]). The site landscaping and lack of forestation prevent such fires from posing a threat to LaSalle station. Based on this review, the Forest or Range Fire impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Frost	A thin layer of ice crystals that form on the ground or the surface of an earthbound object when the temperature of the ground or surface of the object falls below freezing.	Y	C4	The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for LaSalle. Based on this review, the Frost impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
Hail	Showery precipitation in the form of irregular pellets or balls of ice.	Y	C4	The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for LaSalle. Based on this review, the Hail impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
High summer temperature	High abnormal ambient temperatures.	Y	C1 C5	The plant is designed for this hazard **(C1)**. The principal effects of such events would result in elevated lake temperatures which are monitored by station personnel. Should the ultimate heat sink temperature exceed the prescribed temperature limit, an orderly shutdown would be initiated. In addition, plant trips due to this hazard are covered in the definition of another event in the PRA model (e.g., transients, loss of condenser) **(C4)**. Based on this review, the High Summer Temperature impact hazard can be considered to be negligible.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
				There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
High tide, Lake Level, or River Stage	The periodic maximum rise of sea level resulting from the combined effects of the tidal gravitational forces exerted by the Moon and Sun and the rotation of the Earth.	Y	C3 C5	The mid-western location of LaSalle station precludes the possibility of a high tide condition **(C3)**. High lake effects would take place slowly allowing time for orderly plant reductions including shutdowns **(C5)**. Based on this review, the High Tide, Lake Level, or River Stage impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Hurricane	An extremely large, powerful, and destructive storm resulting in strong winds, excessive rainfall, high waves, storm surge, and tornados.	Y	C3	The mid-western location of LaSalle station precludes the possibility of a hurricane. Based on this review, the Hurricane impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
Ice cover	The accumulation of frozen water on bodies of water (e.g., lakes, rivers, etc.) or on structures, systems, and components.	Y	C1 C4	Per UFSAR 2.4.7 (Reference [21]), essential for ice jam formation is a constriction to passage of flowing ice. Such a constriction does not exist in the Illinois River near the site, since the river is approximately 800 feet wide and is kept navigable by dredging when required. The lake screen house is protected against icing in the lake by provision of warming lines near the screen house **(C1)**. The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating events in the internal events PRA model for LaSalle **(C4)**. Based on this review, the Ice Cover impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Industrial or military facility accident	An accident at an offsite industrial or military facility such as a release of toxic gases, a release of	Y	C3 C1	The only military facility within 10 miles is the Illinois Army Reserve National Guard (ILARNG) Training Facility within 1 mile northwest of LaSalle Station and encompassing approximately 2560 acres. There are no missile sites, bombing ranges or runways at the facility,

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
	combustion products, a release of radioactivity, an explosion, or the generation of missiles.			but there are 5 firing ranges in the direction of north to northwest (**C3**). Hazardous chemicals used and/or stored by manufacturers within five miles of the plant were evaluated and determined to either screen from further evaluation or were determined to meet the acceptance criteria associated with Control Room operator protection as discussed in LaSalle UFSAR, Section 2.2.3 (Reference [21]). **(C1)** See also Transportation Accident Based on this review, the Industrial or Military Facility Accident impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Internal Flooding	Excessive water accumulation internal to the station buildings	N/A	N/A	The LaSalle Internal Events PRA includes evaluation of risk from internal flooding events.
Internal Fire	Fire events that are internal to the station buildings	N/A	N/A	The LaSalle Internal Fire PRA includes evaluation of risk from internal fire events.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
Landslide	A rapid flow of a large mass of earth, rock, or material other than accumulated frozen precipitation down a sloped surface.	Y	C3	The mid-western location of LaSalle station precludes the possibility of a landslide. Based on this review, the Landslide impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Lightning	An electrical discharge from a cloud to the ground or Earth-bound object.	Y	C4	Lightning strikes are not uncommon in nuclear plant experience. They can result in losses of off-site power or surges in instrumentation output if grounding is not fully effective. The latter events often lead to reactor trips. Both events are incorporated into the LaSalle internal events model through the incorporation of generic and plant specific data. Based on this review, the Lightning impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

Table E4- 11 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**Lasalle Response**
Low Lake Level or River Stage	A decrease in the water level of the lake or river used for power generation.	Y	C5	These effects would take place slowly allowing time for orderly plant reductions, including shutdowns. Based on this review, the Low Lake Level or River Stage impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Low winter temperature	Low abnormal ambient temperatures.	Y	C5 C4	The principal effects of such events would be to cause a loss of off-site power. These effects would take place slowly allowing time for orderly plant reductions, including shutdowns (**C5**). At worst, the loss of off-site power events would be subsumed into the base PRA model results (**C4**). Based on this review, the Low Winter Temperature impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Meteorite or Satellite Impact	A meteoroid or artificial satellite that releases	Y	PS4	The frequency of a meteor or satellite strike is judged to be so low as make the risk impact from such events

Table E4- 11 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**Lasalle Response**
	energy due to its disintegration in the atmosphere above the Earth's surface, direct impact with the Earth's surface, or a combination of these effects.			insignificant. This hazard also was reviewed as part of the IPEEE submittal [9] and screened based on low frequency of occurrence. Based on this review, the Meteorite or Satellite Impact impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Pipeline accident	An accident involving the rupture of a pipeline carrying hazardous materials or toxic gases.	Y	C1	Per UFSAR Section 2.2.2.3 (Reference [21]), there are no tank farms or gas pipelines within 5 miles of the site. However, there are two natural gas pipelines between 5 and 7 miles from the site and two crude oil pipelines approximately 3 miles west of the plant. There is no significant hazard from toxic releases or explosions involving these pipelines that could interact with the plant. Based on this review, the Pipeline Accident impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
Release of Chemicals in Onsite Storage	An onsite accident involving the storage or handling of hazardous materials such as a release of toxic gases, a release of combustion products, a release of radioactivity, an explosion, or the generation of missiles. In this context, an onsite release of radioactivity is assumed to be associated with low-level radioactive waste.	Y	C1	The impact of releases of hazardous materials stored on-site was evaluated in the IPEEE submittal and updated in LaSalle station's UFSAR. UFSAR Section 2.2.3 (Reference [21]) discusses toxic gas. There is no onsite storage of chlorine; sodium hypochlorite/sodium bromide biocide system is used, thus eliminating an onsite chlorine hazard. Every 3 years a survey will be conducted to re-evaluate the use of chlorine, within 5 miles of the control room, to ensure that a chlorine hazard does not exist. Every 6 years a survey will be conducted to re-evaluate the use of toxic chemicals, within 5 miles of the control room, to ensure that a toxic chemical hazard does not exist. See also Transportation Accidents. Based on this review, the Release of Chemicals in Onsite Storage impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
River diversion	The redirection of all or a portion of river flow by natural causes (e.g. a riverine embankment landslide) or intentionally (e.g. power production, irrigation, etc.).	Y	C1	Per UFSAR Section 2.4.9 (Reference [21]), the Illinois River flows in the same general location as its predecessor of nearly a million years ago. Presence of navigation locks and dams over the entire length of the river has further stabilized the river course. Based on the available evidence, no change in the regime of the river is expected. Based on this review, the River Diversion impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Sand or Dust Storm	A strong wind storm with airborne particles of sand and dust.	Y	C1	The mid-western location of LaSalle station precludes the possibility of a sandstorm. More common wind-borne dirt can occur but poses no significant risk to LaSalle station given the robust structures and protective features of the plant. Based on this review, the Sand or Dust Storm impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
Seiche	An oscillation of the surface of a landlocked body of water, such as a lake, that can vary in period from minutes to several hours.	Y	C3	Flooding due to seiches is not relevant for LaSalle station per Section 2.4.5 of the UFSAR (Reference [21]). Based on this review, the Seiche impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Seismic activity	A sudden release of energy from the Earth's crust resulting in strong ground motion.	N/A	N/A	See information in Section 3 of this Enclosure.
Snow	The accumulation of snow on structures, systems, and components	Y	C5	This hazard is slow to develop and can be identified via monitoring and managed via normal plant processes **(C5)**. See also External Flooding Based on this review, the Snow impact hazard can be considered to be negligible.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
				There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Soil shrink-swell	The relative change in volume of the soil as a result of the type of soil and the amount of moisture.	Y	C1	The potential for this hazard is low at the site, the plant design considers this hazard and the hazard is slow to develop and can be mitigated. Based on this review, the Soil Shrink-Swell Consolidation impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Storm surge	An abnormal rise in sea level accompanying a hurricane or other intense storm, whose height is the difference between the observed level of the sea surface and the level that would have occurred in the absence of the intense storm.	Y	C3	The mid-western location of LaSalle station precludes the possibility of a sea level driven storm surge. Based on this review, the Storm Surge impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

Table E4- 11 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**Lasalle Response**
Toxic Gas	An onsite accident involving the storage or handling of hazardous materials such as a release of toxic gases, a release of combustion products, a release of radioactivity, an explosion, or the generation of missiles. In this context, an onsite release of radioactivity is assumed to be associated with low-level radioactive waste.	Y	C3	UFSAR Section 2.2.3 (Reference [21]) discusses toxic gas. There is no onsite storage of chlorine; sodium hypochlorite/sodium bromide biocide system is used, thus eliminating an onsite chlorine hazard. In addition, there is no possibility of an accident that could lead to the formation of flammable clouds in the vicinity of LaSalle because (1) there is no chemical plant in the vicinity; (2) no gas pipeline passes the station; and (3) no liquefied gases are transported in the vicinity. Per the IPEEE, the bounding analysis showed that these accidents do not significantly contribute to the plant risk. See also Transportation Accidents Based on this review, the Toxic Gas impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Transportation accidents	An accident involving damage to a land-based or marine vehicle transporting hazardous materials that	Y	C1 C3	The impact of transportation accidents was evaluated in the IPEEE (Reference [7]) and in UFSAR Section 2.2.3 (Reference [21]). In the IPEEE, an evaluation was conducted to demonstrate that the probability of a

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
	may result in a release of toxic gases, a release of combustion products, or an explosion.		PS4	rail, land or waterway accident that resulted in release of toxic materials that could affect the site was less than 1E-6 /yr **(PS4)**. Per the UFSAR: <u>Flammable Vapor Clouds (delayed ignition):</u> There is no possibility of an accident that could lead to the formation of flammable clouds in the vicinity of LSCS because (1) there is no chemical plant in the vicinity; (2) no gas pipeline passes the station; and (3) no liquefied gases are transported in the vicinity **(C3)**. <u>Transportation of Toxic Chemicals:</u> The only transportation route carrying toxic chemicals which is within 5 miles of the station is the Illinois River. The toxic chemicals transported are chlorine and anhydrous ammonia. A toxic chemical analysis was performed (Reference [39]) which concluded that chlorine was an insignificant hazard to the station. For anhydrous ammonia, redundant detectors have been added on each outside air intake of the control room system. These detectors will sense ammonia concentrations at the outside air intakes from near zero ppm and higher. On detection of ammonia in the outside air, a control room annunciator alarms. Within

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
				2 minutes of detection of high ammonia concentration in the air intake, the Operator will align the control room envelope HVAC systems in recirculation mode and will don a self-contained breathing apparatus **(C1)** Explosions on the Highway: For explosions on the highway, the worst event would be an explosion from a truck carrying 43,000 pounds of TNT on County Highway 6 at the nearest location to the plant (2000 feet away). If a 43,000-pound charge of TNT explodes at this distance, the structure will receive a peak reflected pressure of 1.5 psi. This magnitude is less than the tornado design pressure **(C1)**. Explosions on the Waterway: For explosions on the waterway, the volume of a maximum tank barge is about $1.8x10^5$ ft^3. Assuming the air mix ratio is adequate for an empty gasoline barge and a detonation takes place, the energy released will be on the order of 107 kcal (Reference [40]), which is equivalent to an explosion of 10 tons of TNT. Since the Seismic Category I structures are located 4 miles away from the river, the peak reflected pressure on the structure will be less than 1 psi in case there is a detonation. Since the Seismic Category I structures have been designed for higher tornado wind pressures, the plant can withstand such a postulated explosion **(C1)**.

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
				Based on this review, the Transportation Accident impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Tsunami	A sea wave of local or distant origin that results from large-scale seafloor displacements associated with large earthquakes or major submarine slides or landslides.	Y	C3	The mid-western location of LaSalle station precludes the possibility of a tsunami. Based on this review, the Tsunami impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Turbine-generated missiles	The generation of a high-energy missile that is ejected from the turbine casing resulting from failure of a steam turbine. The turbine-generated missile may be ejected either upward (i.e., high-trajectory	Y	C1	Per the IPEEE [7], the mean CDF for turbine-generated missiles was 1E-7/yr. Turbine generated missiles are discussed in UFSAR Section 3.5.1.3 (Reference [21]). With the replacement of the Low Pressure (LP) rotors, all the turbine rotors are of the monoblock design. The monoblock rotors have very low stress level. Missile generation due to turbine failure is generally postulated to be caused by turbine overspeed. General Electric has established

Table E4- 11 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**Lasalle Response**
	missile) which may result in damage to safety-related structures, systems, and components (SSCs) from the falling missile or it may be ejected directly toward safety-related SSCs (i.e., low-trajectory missiles).			that the speed capability of these rotors is considerably higher than the maximum attainable speed of these turbine generator units. Consequently, the probability of missiles being generated is statistically insignificant. Based on this review, the Turbine-Generated Missiles impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Volcanic activity	The extrusion of magma from beneath the earth's crust that may be accompanied by the flow of lava and explosion of fragmented material (pulverized pieces of rock, bits of chilled magma), and releases of volcanic ash and dust as well as gases and steam.	Y	C3	Not applicable to the site because of location (no active or dormant volcanoes located near plant site). Based on this review, the Volcanic Activity impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Waves	An area of moving water that is raised above the main surface of an ocean, a	Y	C3 C4	Waves associated with adjacent large bodies of water are not applicable to the site **(C3)**. Waves associated

Table E4- 11
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	Lasalle Response
	lake, etc. as a result of the wind blowing over an area of fluid surface.			with external flooding are covered under that hazard **(C4)**. Based on this review, the Waves impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

Table E4- 12: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is < events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Table E4- 12: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
	PS3. Design basis event mean frequency is < 1E-5/y and the mean conditional core damage probability is < 0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009	
	PS4. Bounding mean CDF is < 1E-6/y.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

7 Conclusions

Based on this analysis of external hazards for LSCS Units 1 and 2, no additional external hazards other than seismic events need to be added to the existing PRA model. The evaluation concluded that the hazards either do not present a design-basis challenge to LSCS, the challenge is adequately addressed in the PRA, or the hazard has a negligible impact on the calculated RICT and can be excluded.

The ICDP/ILERP acceptance criteria of 1E-5/1E-6 will be used within the PARAGON framework to calculate the resulting RICT and RMAT based on the total configuration-specific delta CDF/LERF attributed to internal events and internal fire, plus the seismic delta CDF/LERF values.

8 References

[1] Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, October 12, 2012 (ADAMS Accession No. ML 12286A322).

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

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

[4] NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1, March 2017.

[5] NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," 1975.

[6] NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

[7] Commonwealth Edison, Individual Plant Examination and Individual Plant Examination (External Events) Submittal, LaSalle County Nuclear Power Station, April 28, 1994.

[8] NUREG/CR-4832, "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP), Volume 8, Seismic Analysis," November 1993.

[9] LaSalle County Station, Units 1 and 2, "NRC Staff Evaluation of the Individual Plant Examination of External Events (IPEEE) Submittal," (TAC NOS. M83634 and M83635), December 8, 2000.

[10] Exelon Generation Company, LLC, RS-14-071, "Seismic Hazard and Screening Report (Central and Eastern United States (CEUS) Sites)," Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, March 31, 2014.

[11] Lettis Consultants International (LCI), Inc., Project No. 1041, "LaSalle Seismic Hazard and Screening Report, Rev 1," February 27, 2014 [also cited herein as EPRI 2013 Revision 1].

[12] USNRC Information Notice 2010-18, "Safety/Risk Assessment Results for Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,'" in *Tables B.2, C.1, and C.2*, September 2, 2010 (ADAMS ML100270582).

[13] LSCS 2014 Full Power Internal Events (FPIE) PRA update, Unit 2, LS214A, November 2015.

[14] Vogtle Electric Generating Plant – Units 1 and 2 License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Enclosure E3), September 13, 2012, NRC ADAMS Accession # ML12258A055.

[15] Calvert Cliffs Nuclear Power Plant, Units 1 and 2 - License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b," February 25, 2016, NRC ADAMS Accession # ML16060A223.

[16] LS-MISC-027, "LaSalle External Hazards Notebook," Revision 0, November 2019.

[17] LS-PSA-20.001, Revision 0, LaSalle Seismic Probabilistic Risk Assessment, Seismic Methods and Results Summary [Notebook], January 25, 2013 [Phase 1 SPRA].

[18] Electric Power Research Institute (EPRI) TR 3002000709, "Final Report, Seismic Probabilistic Risk Assessment Implementation Guide," December 2013.

[19] FRANX: Software Manual Version 4.4, EPRI, Palo Alto: 2017.

[20] LS-PSA-015, Revision 6, LaSalle County Generating Station, Probabilistic Risk Analysis, Level 2/LERF Notebook, 2014 PRA Update, November 5, 2015.

[21] LaSalle County Station (LSCS) Updated Final Safety Analysis Report (UFSAR), Revision 23, April 2018.

[22] NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," Revision 2, February 2007.

[23] EPRI 3002003107, "High Wind Risk Assessment Guidelines," 2015.

[24] NRC Regulatory Issue Summary (RIS) 2015-06, "Tornado Missile Protection," June 10, 2015.

[25] EC Evaluation 620178, "LAS Tornado Missile Protection Evaluation and Non-Conformances," Rev. 0, February 2018.

[26] LaSalle ARs 04104391, 04104393, 04104393, 04104396, 04104397, and 04104401.

[27] LS-MISC-044, "LaSalle TMRE Vulnerable SSC Walkdown Report," Revision 0.

[28] LaSalle Station Drawing S-1079, "Auxiliary Building Floor Framing Elevation 815'-0" South Area," Revision AK.

[29] NEI 17-02, "Tornado Missile Risk Evaluator (TMRE) Industry Guidance Document," Rev. 1A, July 2018.

[30] NUREG-0800, "Standard Review Plan (SRP)," Rev. 2, July 1981.

[31] LaSalle Station Drawing M-6, "General Arrangement Reactor Building Floor Plans," Revision Y.

[32] LOA-TORN-001, "High Winds / Tornado," Revision 25, February 2019.

[33] LOA-FSG-005, "Area Ventilation," Revision 3, February 2017.

[34] LaSalle Flood Hazard Reevaluation Report (FHRR), NRC ADAMS Accession No. ML14079A425, March 12, 2014.

[35] LaSalle County Station, Units 1 and 2, Flooding Focused Evaluation Summary Submittal, NRC ADAMS Accession No. ML17067A402, March 8, 2017.

[36] NRC Letter, LaSalle County Station, Units 1 and 2, Staff Assessment of Flooding Focused Evaluation, NRC ADAMS Accession No. ML17191A323, August 23, 2017.

[37] Risk Assessment of Operational Events, Volume 2 – External Events – Internal Fires – Internal Flooding – Seismic – Other External Events – Frequencies of Seismically-Induced LOOP Events (RASP Handbook), Revision 1.02, US Nuclear Regulatory Commission, November 2017 ML17349A301.

[38] ER-AA-340, "GL 89-13 Program Implementing Procedure," Revision 9.

[39] LaSalle Design Analysis L-003414, Toxic Chemical Analysis of 2008 Offsite Chemical Survey Results.

[40] CRC, "Handbook of Chemistry and Physics," 49th Edition, 1969.

[41] Regulatory Guide 1.76, "Design-basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, March 2007.

[42] LS-PSA-001, Revision 7, LaSalle County Generating Station, Probabilistic Risk Analysis, Initiating Events Notebook, 2014 PRA Update, November 5, 2015.

ENCLOSURE 5

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

ENCLOSURE 5
Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

1. Introduction

Section 4.0, Item 6 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09-A, 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.)

The purpose of this enclosure is to demonstrate that the LaSalle County Station (LSCS) total Core Damage Frequency (CDF) and total Large Early Release Frequency (LERF) are below the guidelines established in RG 1.174. RG 1.174 does not establish firm limits for total CDF and LERF, but it recommends that risk-informed applications be implemented only when the total plant risk is no more than about 1E-4/year for CDF and 1E-5/year for LERF. Demonstrating that these limits are met confirms that the risk metrics of NEI 06-09-A can be applied to the LSCS Risk-Informed Completion Time (RICT) Program.

2. Technical Approach

Table E5-1 lists the LSCS CDF and LERF point estimate values that resulted from a quantification of the baseline internal events (including internal flooding) and fire Probabilistic Risk Assessment (PRA) models (References 5 and 6, respectively). This table also includes an estimate of the seismic contribution to CDF and LERF based on the methodology detailed in Enclosure 4, Section 3. Other external hazards are below accepted screening criteria and therefore do not contribute significantly to the totals.

Table E5-1
Total Baseline CDF/LERF

LaSalle Unit 1 Baseline CDF			**LaSalle Unit 1 Baseline LERF**	
Source	**Contribution**		**Source**	**Contribution**
Internal Events PRA	1.3E-06		Internal Events PRA	1.3E-07
Fire PRA	1.0E-05		Fire PRA	9.8E-07
Seismic	1.1E-05		Seismic	2.2E-06
Other External Events	No significant contribution		Other External Events	No significant contribution
Total Unit 1 CDF	2.2E-05		**Total Unit 1 LERF**	3.3E-06
LaSalle Unit 2 Baseline CDF			**LaSalle Unit 2 Baseline LERF**	
Source	**Contribution**		**Source**	**Contribution**
Internal Events PRA	1.3E-06		Internal Events PRA	1.3E-07
Fire PRA	7.8E-06		Fire PRA	3.2E-07
Seismic	1.1E-05		Seismic	2.2E-06
Other External Events	No significant contribution		Other External Events	No significant contribution
Total Unit 2 CDF	2.0E-05		**Total Unit 2 LERF**	2.7E-06

ENCLOSURE 5
Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

As demonstrated in Tables E5-1, the total CDF and total LERF are within the guidelines set forth in RG 1.174 and support small changes in risk that may occur during RICT entries following TSTF-505 implementation. Therefore, LSCS TSTF-505 implementation is consistent with NEI 06-09-A guidance. There will be a proceduralized check of the overall PRA results against the Reg Guide 1.174 thresholds in the PRA model update procedures.

The base internal events model for LSCS is based on Unit 2, (Reference 5). Due to design symmetry and consistent operational practices between the units, it can be considered representative of Unit 1 as well. Therefore, there is no Unit 1 specific internal events model and the Unit 1 baseline CDF and LERF are taken from the Unit 2 specific model, (Reference 5). The Seismic CDF and LERF listed in Table E5-1 are listed for the containment inerted state. For further explanation on Seismic penalties in inerted vs de-inerted states consult Enclosure 4.

3. References

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

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

3. Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.

4. Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011 (Accession No. ML10091006).

5. LS-ASM-03, Revision 0, MSPI Update Application-Specific Model (ASM) Notebook, July 2017.

6. LS-PSA-021.11, Revision 3, LaSalle County Generating Station Fire PRA Summary & Quantification Notebook, December 2019.

ENCLOSURE 6

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

<u>Justification of Application of At-Power PRA Models to Shutdown Modes</u>

This enclosure is not applicable to the LaSalle County Station submittal.

EGC is proposing to apply the Risk-Informed Completion Time Program only in Modes 1 and 2 and not in the shutdown Modes.

ENCLOSURE 7

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

PRA Model Update Process

ENCLOSURE 7
PRA Model Update Process

1. Introduction

Section 4.0, Item 8 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09-A, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide a discussion of the licensee's programs and procedures which assure the PRA models which support the RMTS are maintained consistent with the as-built/as-operated plant.

This enclosure describes the administrative controls and procedural processes applicable to the configuration control of PRA models used to support the Risk-Informed Completion Time (RICT) Program, which will be in place to ensure that these models reflect the as-built/as-operated plant. Plant changes, including physical modifications and procedure revisions, will be identified and reviewed prior to implementation to determine if they could impact the PRA models per ER- AA-600-1015, FPIE [Full Power Internal Events] PRA Model Update (Reference 3), and ER-AA-600-1061, Fire PRA Model Update and Control (Reference 4). The configuration control program will ensure these plant changes are incorporated into the PRA models as appropriate. The process will include discovered conditions associated with the PRA models, which will be addressed by the applicable site Corrective Action Program.

Should a plant change or a discovered condition be identified that has a significant impact to the RICT Program calculations as defined by the above procedures, an unscheduled update of the PRA model will be implemented. Otherwise, the PRA model change is incorporated into a subsequent periodic model update. Such pending changes are considered when evaluating other changes until they are fully implemented into the PRA models. Periodic updates are typically performed every two refueling cycles.

2. PRA Model Update Process

Internal Event, Internal Flood, and Fire PRA Model Maintenance and Update

The Fleet risk management process ensures that the applicable PRA models used for the RICT Program reflect the as-built/as-operated plant for LaSalle Units 1 and 2. The PRA configuration control process delineates the responsibilities and guidelines for updating the full power internal events, internal flood, and fire PRA models, and includes both periodic and unscheduled PRA model updates.

The process includes provisions for monitoring potential impact areas affecting the technical elements of the PRA models (e.g., due to plant changes, plant/industry operational experience, or errors or limitations identified in the model), assessing the individual and cumulative risk impact of unincorporated changes, and controlling the model and necessary computer files, including those associated with the real time risk model.

Changes that are considered an upgrade per the ASME/ANS PRA standard receive a peer review focused on those aspects of the PRA model that represent the upgrade.

ENCLOSURE 7
PRA Model Update Process

Review of Plant Changes for Incorporation into the PRA Model

1. Plant changes or discovered conditions are reviewed for potential impact to the PRA models, including the real time risk model and the subsequent risk calculations which support the RICT Program (NEI 06-09-A, Section 2.3.4, Items 7.2 and 7.3, and 2.3.5, Items 9.2 and 9.3).

2. Plant changes that meet the criteria defined in References 3 and 4 (including consideration of the cumulative impact of other pending changes) will be incorporated in the applicable PRA model(s), consistent with the NEI 06-09-A guidance. Otherwise, the change is assigned a priority and is incorporated at a subsequent periodic update consistent with procedural requirements. (NEI 06-09-A, 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-A (NEI 06-09-A, Section 2.3.4, Item 7.1, and 2.3.5, Item 9.1).

4. If a PRA model change is required for the real time risk model, but cannot be immediately implemented for a significant plant change or discovered condition, either:

   a. Interim analyses to address the expected risk impact of the change will be performed. In such a case, these interim analyses become part of the RICT Program calculation process until the plant changes are incorporated into the PRA model during the next update. The use of such bounding analyses is consistent with the guidance of NEI 06-09-A.

   OR

   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-A.

   These actions satisfy NEI 06-09-A, Section 2.3.5, Item 9.3.

3. References

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

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

3. ER-AA-600-1015, "FPIE PRA Model Update."

4. ER-AA-600-1061, "Fire PRA Model Update and Control."

ENCLOSURE 8

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

<u>Attributes of the Real Time Risk Model</u>

ENCLOSURE 8
Attributes of the Real Time Risk Model

1. Introduction

Section 4.0, Item 9 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09-A, 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 used to support the RMTS. This includes identification of how the baseline probabilistic risk assessment (PRA) model is modified for use in the configuration risk management program (CRMP) tools, quality requirements applied to the PRA models and CRMP tools, consistency of calculated results from the PRA model and the CRMP tools, and training and qualification programs applicable to personnel responsible for development and use of the CRMP tools. NEI 06-09-A, Revision 0, uses the term CRMP for the program controlling the use of RMTS. This term is also used to designate the program implementing 10 CFR 50.65(a)(4) and the monitoring program for other risk informed LARs. To avoid confusion the term RICT program is used to indicate the program required by NEI 06-09-A, Revision 0, in lieu of the term CRMP. This item should also confirm that the RICT program tools can be readily applied for each Technical Specification (TS) limiting condition for operation (LCO) within the scope of the plant-specific submittal.

This enclosure describes the necessary changes to the peer-reviewed baseline PRA models for use in the real time risk (RTR) tool to support the Risk-Informed Completion Time (RICT) Program. The process employed to adapt the baseline models 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 RICT Program are also discussed in this enclosure.

2. Translation of Baseline PRA Model for Use in Configuration Risk

The baseline PRA models for internal events, including internal flood and internal fire, are the peer-reviewed models. These models are updated when necessary to incorporate plant changes to reflect the as-built/as-operated plant. The internal flood model is integrated into the internal events model. These models will be used in the RICT Program. The models may be optimized for quantification speed but are verified to provide the same result as the baseline models in accordance with approved procedures.

The RTR tool will be used to facilitate all configuration-specific risk calculations and support the RICT Program implementation. The PRA Models utilize system initiator event fault trees so equipment unavailabilities are captured explicitly in these system initiator fault trees. Therefore, no adjustment to initiating event frequencies in required within the RTR tool.

ENCLOSURE 8
Attributes of the Real Time Risk Model

The baseline PRA models are modified as follows for use in configuration risk calculations:

- The unit availability factor is set to 1.0 (unit available).
- Maintenance unavailability is set to zero/false unless unavailable due to the configuration.
- Mutually exclusive combinations, including normally disallowed maintenance combinations, are adjusted to allow accurate analysis of the configuration.
- For systems where some trains or components are in service and some in standby or there are seasonal dependencies, the RTR tool addresses the actual configuration of the plant including as needed.
- RHR recovery terms set to one/true to remove credit.
- There are no changes in success criteria based on the time in the core operating cycle.

The configuration risk software is designed to quantify the unit-specific configuration for both internal events, including internal flooding and fire, and includes the seismic risk contribution when calculating the RMAT and RICT. Full quantifications will be used for each configuration. Pre-solved cutsets will be limited to results for specific configurations. For configurations without pre-solved cutsets the model will be quantified to produce cutsets for the previously unanalyzed configuration. If there are any changes in the underlying PRA, the PRA Results database in PARAGON will be updated in accordance with the RTR Update Procedure. The unique aspect of the configuration risk software for the RICT program is the quantification of fire risk and the inclusion of the seismic risk contribution. The other adjustments above are those used for the evaluation of risk under the 10CFR 50.65(a)(4) program.

The LaSalle County Station (LSCS) Units 1 and 2 PRA calculates Common Cause Basic Event (CCBE) probabilities from alpha factors and places the basic events under appropriate gates in the fault tree.

Adjustments to the CCF grouping or CCF probabilities are not necessary when a component is taken out-of-service for preventative maintenance:

- The component is not out-of-service for reasons subject to a potential common cause failure, and so the in-service components are not subject to increases in common cause probabilities.
- CCF relationships are retained for the remaining in-service components.
- The net failure probability for the in-service components includes the CCF contribution of the out-of-service component.

As described in Reg Guide 1.177 (Reference 6), Section A-1.3.2.2, the CCF term should be treated differently when a component is taken down for preventive maintenance (PM) than as described for failure of a component. For PMs, the common cause factor is changed so that the model represents the unavailability of the remaining component. In the example provided in Reg Guide 1.177 for a 2-train system, the CCF event can be set to zero for PMs. This is done so that the model represents the unavailability of the remaining component, and not the common cause multiplier. The LSCS approach is conservative in that for a 2-train system, the CCF event is retained for the component removed from service. Likewise, for systems with three or more trains, the CCF events that are related to the out-of-service component are retained.

The Vogtle RICT Safety Evaluation (Reference 5) describes the Vogtle approach for modeling common cause events with planned inoperability: "For planned inoperability, the licensee sets the appropriate independent failure to 'true' and makes no other changes while calculating a RICT." The LSCS approach is the same as this Vogtle approach.

It is recognized that other modifications could be made to CCF factors for planned maintenance, particularly for common cause groups of three or more components. For example, in the Safety Evaluation (SE) in the Vogtle RICT Amendment (Reference 5), the NRC identifies a possible planned maintenance CCF modification to "modify all the remaining basic event probabilities to reflect the reduced number of redundant components."

Like Vogtle, the LSCS CCF approach is a straightforward simplification that has inherent uncertainties. In the context of modifying CCF basic events for PMs, the Vogtle SE states the following:

> "The NRC staff also notes that common cause failure probability estimates are very uncertain and retaining precision in calculations using these probabilities will not necessarily improve the accuracy of the results. Therefore, the NRC staff concludes that the licensee's method is acceptable because it does not systematically and purposefully produce non-conservative results and because the calculations reasonably include common cause failures consistent with the accuracy of the estimates." (Reference 5)

The LSCS approach for CCF during PMs is the same as the Vogtle approach; therefore, the LSCS CCF approach is acceptable for RICT calculations and adjusting the common cause grouping is not necessary for PMs. However, if a numeric adjustment is performed, the RICT calculation shall be adjusted to numerically account for the increased possibility of CCF in accordance with RG 1.177, as specified in Section A-1.3.2.1 of Appendix A of the RG.

For emergent conditions where the extent of condition is not completed prior to entering into the Risk Management Action Times or the extent of condition cannot rule out the potential for common cause failure, common cause RMAs are expected to be implemented to mitigate common cause failure potential and impact, in accordance with Exelon procedures. This is in line with the guidance of NEI 06-09-A and precludes the need to adjust CCF probabilities. However, if a numeric adjustment is performed, the RICT calculation shall be adjusted to numerically account for the increased possibility of CCF in accordance with RG 1.177, as specified in Section A-1.3.2.1 of Appendix A of the RG.

3. Quality Requirements and Consistency of PRA Model and Configuration Risk Tools

The approach for establishing and maintaining the quality of the PRA models, including the configuration risk model, includes both a PRA maintenance and update process (described in Enclosure 7), and the use of self-assessments and independent peer reviews (described in Enclosure 2).

The information provided in Enclosure 2 demonstrates that the site's internal event, internal flood, and internal fire PRA models reasonably conform to the associated industry standards endorsed by Regulatory Guide 1.200 (Reference 3). This information provides a robust basis for concluding that the PRA models are of sufficient quality for use in risk-informed licensing actions.

For maintenance of an existing configuration risk model, changes made to the baseline PRA model in translation to the configuration risk model will be controlled and documented. Every PRA MOR Update results in an update to the RTR model in accordance with the FPIE and Fire PRA Update procedures. An acceptance test is performed after every configuration risk model update. This testing also verifies correct mapping of plant components to the basic events in the configuration risk model. The RTR model documentation includes changes made to the MOR model files to work with the RTR model software (e.g., quantification settings) along with verification that results are consistent between the RTR and PRA zero maintenance results. In addition, the RTR update for the MOR includes quantifying the RTR model for representative maintenance configurations and examining the results for appropriateness. These actions are procedurally controlled.

4. Training and Qualification

The PRA staff is responsible for development and maintenance of the configuration risk model. Operations and Work Control staff will use the configuration risk tool under the RICT Program. PRA Staff and Operations are trained in accordance with a program using National Academy for Nuclear Training (ACAD) documents, which is also accredited by INPO.

5. Application of the Configuration Risk Tool to the RICT Program Scope

The PARAGON software will be used to facilitate all configuration-specific risk calculations and support the RICT Program implementation. LSCS Units 1 and 2 each have their own PARAGON model. This program is specifically designed to support implementation of RMTS. PARAGON will permit the user to evaluate all plant configurations using appropriate mapping of equipment to PRA basic events. The equipment in the scope of the RICT program will be able to be evaluated in the appropriate PRA models. The RICT program will meet RG 1.174 (Reference 4) and Exelon software quality assurance requirements.

6. References

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

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

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

4. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011.

5. Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Implementation of Topical Report Nuclear Energy Institute NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specification (RMTS) Guidelines," Revision 0-A (CAC NOS. ME9555 and ME9556), ML15127a669.

6. Nuclear Regulatory Commission, Regulatory Guide 1.177, May 2011, Revision 1.

ENCLOSURE 9

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

<u>Key Assumptions and Sources of Uncertainty</u>

ENCLOSURE 9
Key Assumptions and Sources of Uncertainty

1. Introduction

The purpose of this enclosure is to disposition the impact of Probabilistic Risk Assessment (PRA) modeling epistemic uncertainty for the Risk Informed Completion Time (RICT) Program. Topical Report NEI 06-09-A (Reference 1), Section 2.3.4, item 10 requires an evaluation to determine insights that will be used to develop risk management actions (RMAs) to address these uncertainties. The baseline Internal Events PRA and Fire PRA (FPRA) models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. The approach taken is, therefore, to review these documents to identify the items which may be directly relevant to the RICT Program calculations, to perform sensitivity analyses where appropriate, to discuss the results and to provide dispositions for the RICT Program.

The epistemic uncertainty analysis approach described below applies to the internal events PRA and any epistemic uncertainty impacts that are unique to FPRA are also addressed. In addition, Topical Report NEI 06-09-A requires that the uncertainty be addressed in RICT Program Configuration Risk Management Program (CRMP), otherwise referred to as the Real-Time Risk (RTR), tools by consideration of the translation from the PRA model to the RTR model. The RTR model, also referred to as the PARAGON model, discussed in Enclosure 8 includes internal events, flooding events and fire events. The model translation uncertainties evaluation and impact assessment are limited to new uncertainties that could be introduced by application of the RTR tool during RICT Program calculations.

2. Assessment of Internal Events PRA Epistemic Uncertainty Impacts

In order to identify key sources of uncertainty for the RICT Program application, an evaluation of Internal Events baseline PRA model uncertainty was performed, based on the guidance in NUREG-1855 (Reference 2) and Electric Power Research Institute (EPRI) report 1016737 (Reference 3). As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

Parametric uncertainty was addressed as part of the LaSalle County Station (LSCS) baseline PRA model quantification (Reference 4) and the Fire PRA uncertainty evaluation (Reference 6).

Modeling uncertainties are considered in both the base PRA and in specific risk-informed applications. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the LSCS Internal Events PRA technical elements are noted in the individual notebooks. The Internal Events PRA model uncertainties evaluation is documented in Reference 4 and considers the modeling uncertainties for the base PRA by identifying assumptions, determining if those assumptions are related to a source of modeling uncertainty and characterizing that uncertainty, as necessary. EPRI compiled a listing of generic sources of modeling uncertainty to be considered for each PRA technical element (Reference 2), and the evaluation performed for LSCS (Reference 4) considered each of the generic sources of modeling uncertainty as well as the plant-specific sources.

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application (Reference 4). No specific issues of PRA completeness have been identified relative to the TSTF-505 application, based on the results of the Internal Events PRA and Fire PRA peer reviews.

Additionally, an evaluation of Level 2 internal events PRA model uncertainty was performed, based on the guidance in NUREG-1855 (Reference 2) and Electric Power Research Institute (EPRI) report 1026511 (Reference 5). The potential sources of model uncertainty in the LSCS PRA model were evaluated for the 32 Level 2 PRA topics outlined in EPRI 1026511.

A detailed review of the generic and plant-specific sources of internal events model uncertainties are discussed in LS-MISC-046 (Reference 7) and are therefore not repeated in this enclosure. The purpose of this enclosure is to summarize the key sources of uncertainty that could potentially impact the RICT calculations.

Based on following the methodology in EPRI 1016737, as supplemented by EPRI 1026511, the impact of key sources of uncertainty in the internal events PRA model on the RICT application is summarized in Table E9-1. The key sources of uncertainty identified in Table E9-1 do not present a significant impact on the LSCS RICT calculations and therefore, the internal events PRA model is capable of producing accurate RICT calculations. Note that RMAs will be developed when appropriate using insights from the PRA model results specific to the configuration.

Table E9-1

ASSESSMENT OF INTERNAL EVENTS PRA EPISTEMIC UNCERTAINTY

Source of Uncertainty and Assumptions	RICT Program Impact	Model Sensitivity and Disposition
ECCS Survivability Post Containment Venting Failure		
ECCS equipment survivability post-containment venting is treated probabilistically. Containment venting does not fail all equipment in the Reactor Building or Turbine Building.	Although the treatment is realistic, there is the potential for a non-conservative bias given the unknown phenomenological events that could be associated with containment venting (e.g., hydrogen buildup in the Reactor Buildings, harsh events due to steam release, and other unknown consequences). Therefore, the assumption of ECCS survivability post containment venting is identified as a candidate source of model uncertainty and was further evaluated with various sensitivity analyses.	For this source of model uncertainty, sensitivity analyses were performed for a select group of technical specifications within scope of the RICT application. The sensitivity analyses consisted of assuming that ECCS equipment would not survive post containment venting failure (i.e., failure probability of 1.0). Although the RICT estimates change as a result of this sensitivity, the bounding sensitivity analysis assumes that ECCS equipment would be guaranteed to fail due to steam binding, which is not a realistic assumption and use of this bounding assumption would result in overly-conservative RICT estimates. Therefore, the uncertainty associated with this model uncertainty is negligible within the RICT application.

Table E9-1

ASSESSMENT OF INTERNAL EVENTS PRA EPISTEMIC UNCERTAINTY

Source of Uncertainty and Assumptions	RICT Program Impact	Model Sensitivity and Disposition
Core Melt Arrest Prior to Vessel Failure		
Injection from these high capacity low pressure systems will preclude vessel failure if they are available following RPV depressurization given core damage occurs at high RPV pressure.	Core melt arrest prior to vessel failure may not be guaranteed with LP injection recovered after core damage, but prior to vessel failure. Therefore, the assumption of LP ECCS restoration assuring that vessel failure is avoided is identified as a candidate source of model uncertainty and was further evaluated with various sensitivity analyses.	For this source of model uncertainty, sensitivity analyses were performed for a select group of technical specifications within scope of the RICT application. The sensitivity analyses consisted of assuming that LP ECCS is inadequate and vessel failure is assumed to occur (i.e., failure probability of 1.0). Due to the small impact demonstrated by the sensitivity cases, the uncertainty associated with this model uncertainty is negligible within the RICT application.

Table E9-1

ASSESSMENT OF INTERNAL EVENTS PRA EPISTEMIC UNCERTAINTY

Source of Uncertainty and Assumptions	RICT Program Impact	Model Sensitivity and Disposition
Vapor Suppression Capabilities at Vessel Failure		
Ex-vessel core melt progression overwhelms vapor suppression noted as extremely unlikely for low pressure RPV failures modes and very unlikely for high pressure failure modes based on reference to generic studies and identification of plant-specific features. However, more recent MAAP results indicate that containment pressurization following vessel failure for wet containment conditions might be higher than what had previously been calculated or what was originally considered.	Ex-vessel core melt progression overwhelms vapor suppression capabilities (in the context of containment pressurization for wet drywell conditions) is identified as a potential candidate source of model uncertainty. It is noted that NUREG/CR-6595 (Reference 9) would indicate that upper bound values of 0.2 for low pressure scenarios and 0.3 for high pressure scenarios (for Mark II Containments) may need to be explored as an alternate hypothesis. Therefore, this item has been identified as a candidate source of model uncertainty and was further evaluated with various sensitivity analyses.	For this source of model uncertainty, sensitivity analyses were performed for a select group of technical specifications within scope of the RICT application. The sensitivity analyses consisted of using the recommended upper bound values from NUREG/CR-6595 (Reference 9) for Mark II Containments as an alternate hypothesis (i.e., sensitivity analysis uses upper bound values of 0.2 for low pressure scenarios and 0.3 for high pressure scenarios). Although the RICT estimates change as a result of this sensitivity, the bounding sensitivity analysis utilizes the upper bound values, which is not a realistic assumption and use of this bounding assumption would result in overly-conservative RICT estimates. Therefore, the uncertainty associated with this model uncertainty is negligible within the RICT application.

Table E9-1

ASSESSMENT OF INTERNAL EVENTS PRA EPISTEMIC UNCERTAINTY

Source of Uncertainty and Assumptions	RICT Program Impact	Model Sensitivity and Disposition
RHR Recovery		
Credit for repair / recovery of RHR for loss of decay heat removal sequences is credited in the base PRA models with adequate justification provided.	For the RICT calculations, repair / recovery of RHR for loss of decay heat removal sequences is conservatively not credited. Therefore, this item has been identified as a candidate source of model uncertainty and was further evaluated with various sensitivity analyses.	For this source of model uncertainty, sensitivity analyses were performed for a select group of technical specifications within scope of the RICT application. The sensitivity analyses consisted of retaining the RHR recovery failure probability used in the average maintenance model (rather than setting it to 1.0). Due to the small impact demonstrated by the sensitivity cases, the uncertainty associated with this model uncertainty is negligible within the RICT application.
Digital Feedwater Controls		
There are model uncertainties associated with modeling digital systems, such as those related to determining the failure modes of these systems and components. The reliability values from the similar vendor study demonstrating that the system performance would result in less than 0.1 transients per year are used for the key components of the system. The reliability analysis for causing plant trips performed by similar FW vendor studies is assumed to be equally applicable to the reliability of the system post plant trips that are caused by other means that do not directly affect the feedwater availability.	Digital feedwater control failure probabilities are treated probabilistically. Therefore, this item has been identified as a candidate source of model uncertainty and was further evaluated with various sensitivity analyses.	For this source of model uncertainty, sensitivity analyses were performed for a select group of technical specifications within scope of the RICT application. The sensitivity analyses consisted of increasing the failure probability associated with digital feedwater controls by a factor of 50 (i.e., from 0.01 to 0.5). Due to the small impact demonstrated by the sensitivity cases, the uncertainty associated with this model uncertainty is negligible within the RICT application.

Table E9-1

ASSESSMENT OF INTERNAL EVENTS PRA EPISTEMIC UNCERTAINTY

Source of Uncertainty and Assumptions	RICT Program Impact	Model Sensitivity and Disposition
Water Hammer Pipe Rupture		
Water hammer is a potential failure mode of important systems and can also cause a flood related event. ECCS system draindown scenarios are included in the LSCS PRA model. Subsequent starting or restarting of these systems causes a water hammer and system leak or rupture.	The water hammer evaluation represents a realistic evaluation using plant-specific water hammer evaluations performed for LSCS. However, this item has been identified as a candidate source of model uncertainty and was further evaluated with various sensitivity analyses	For this source of model uncertainty, sensitivity analyses were performed for a select group of technical specifications within scope of the RICT application. The sensitivity analyses consisted of increasing the ECCS pipe rupture failure probabilities due to water hammers by a factor of 100 (i.e., from 1E-3 to 1E-1). Due to the small impact demonstrated by the sensitivity cases, the uncertainty associated with this model uncertainty is negligible within the RICT application.
FLEX Equipment Reliability		
There are no industry-approved data sources for FLEX equipment reliability. Currently, FLEX is not credited in the base FPIE and Fire PRA models, but is included in the fault trees for sensitivity purposes. The equipment failure rate data from equivalent non-FLEX systems is used as a surrogate for the FLEX equipment modeled in the PRA (until industry approved FLEX data is developed).	Since FLEX is not currently credited in the PRA, there is no impact, but future revisions to the PRA models will credit FLEX, so CDF and LERF may be sensitive to the FLEX equipment failure probabilities. Therefore, this item has been identified as a candidate source of model uncertainty and was further evaluated with various sensitivity analyses.	For this source of model uncertainty, sensitivity analyses were performed for a select group of technical specifications within scope of the RICT application. The sensitivity analyses consisted of increasing the FLEX equipment failure probabilities by a factor of 10. Due to the small impact demonstrated by the sensitivity cases, the uncertainty associated with this model uncertainty is negligible within the RICT application.

Table E9-1

ASSESSMENT OF INTERNAL EVENTS PRA EPISTEMIC UNCERTAINTY

Source of Uncertainty and Assumptions	RICT Program Impact	Model Sensitivity and Disposition
Containment Integrity Following Vessel Failure		
There is model uncertainty regarding the subsequent treatment that increases the likelihood of LERF for this extremely rare event. A portion of the vessel rupture sequences are assumed to result in concurrent containment failure coincident with the vessel rupture.	Containment integrity following vessel rupture is treated probabilistically. Therefore, this item has been identified as a candidate source of model uncertainty and was further evaluated with various sensitivity analyses.	For this source of model uncertainty, sensitivity analyses were performed for a select group of technical specifications within scope of the RICT application. The sensitivity analyses consisted of assuming that containment failure occurs coincidently with vessel rupture. Due to the small impact demonstrated by the sensitivity cases, the uncertainty associated with this model uncertainty is negligible within the RICT application.

3. Assessment of Translation (RTR Model) Uncertainty Impacts

Incorporation of the baseline PRA models into the RTR model used for RICT Program calculations may introduce new sources of model uncertainty. Table E9-2 provides a description of the relevant model changes and dispositions of whether any of the changes made represent possible new sources of model uncertainty that must be addressed. Refer to Enclosure 8 for additional discussion on the RTR model.

Table E9-2

ASSESSMENT OF TRANSLATION UNCERTAINTY IMPACTS

RTR Model Change and Assumptions	Part of Model Affected	Impact on Model	Disposition
PRA model logic structure may be optimized to increase solution speed.	Fault tree logic model structure, affecting both internal and Fire PRAs.	The model, if restructured, will be logically equivalent and produce results comparable to the baseline PRA logic model.	Since the restructured model will produce comparable numerical results, this is not a source of uncertainty for the RICT program.
Incorporation of seismic risk bias to support RICT Program risk calculations. A conservative value for the seismic delta CDF is applicable.	Calculation of RICT and RMAT within RTR.	The addition of bounding impacts for seismic events has no impact on baseline PRA or RTR model. Impact is reflected in calculation of all RICTs and RMATs.	Since this is a bounding approach for addressing seismic risk in the RICT Program it is not a source of translation uncertainty and RICT Program calculations are not impacted, so no mandatory RMAs are required.
Set plant availability (Reactor Critical Years Factor) basic event to 1.0.	Type Code: @CRIT-FACTOR	Since the RTR model evaluates specific configurations during at-power conditions, the use of a plant availability factor less than 1.0 is not appropriate. This change allows the RTR model to produce appropriate results for specific at-power configurations.	This change is consistent with RTR tool practice; therefore, this change does not represent a source of uncertainty, and RICT program calculations are not impacted, so no mandatory RMAs are required.
RHR Recovery Terms	Basic Events: 2RHRXDHRRECLTH-- 2RHRXDHRRECLTH-F 2HRRXRHRNOTRCH-- 2RHRX-REC-AT-F--	Setting these terms to 1.0 prevents any credit being taken for RHR recovery.	Not taking credit for RHR recovery removes a potential source of uncertainty.

4. Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

The purpose of the following discussion is to address the epistemic uncertainty in the LSCS FPRA. The LSCS FPRA model includes various sources of uncertainty that exist because there is both inherent randomness in elements that comprise the FPRA and because the state of knowledge in these elements continues to evolve. The development of the LSCS FPRA was guided by NUREG/CR-6850 (Reference 10). The LSCS FPRA model used consensus models described in NUREG/CR-6850.

LSCS used guidance provided in NUREG/CR-6850 and NUREG-1855 (Reference 2) to address uncertainties associated with FPRA for the RICT Program application. As stated in Section 1.3 of NUREG-1855:

> *"Although the guidance in this report does not currently address all sources of uncertainty, the guidance provided on the uncertainty identification and characterization process and on the process of factoring the results into the decision making is generic and independent of the specific source of uncertainty. Consequently, the guidance is applicable for sources of uncertainty in PRAs that address at-power and low power and shutdown operating conditions, and both internal and external hazards."*

NUREG-1855 also describes an approach for addressing sources of model uncertainty and related assumptions. It defines:

> *"A source of model uncertainty exists when (1) a credible assumption (decision or judgment) is made regarding the choice of the data, approach, or model used to address an issue because there is no consensus and (2) the choice of alternative data, approaches or models is known to have an impact on the PRA model and results. An impact on the PRA model could include the introduction of a new basic event, changes to basic event probabilities, change in success criteria, or introduction of a new initiating event. A credible assumption is one submitted by relevant experts and which has a sound technical basis. Relevant experts include those individuals with explicit knowledge and experience for the given issue. An example of an assumption related to a source of model uncertainty is battery depletion time. In calculating the depletion time, the analyst may not have any data on the time required to shed loads and thus may assume (based on analyses) that the operator is able to shed certain electrical loads in a specified time."*

NUREG-1855 defines consensus model as:

> *"A model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that NRC has utilized or accepted for the specific risk-informed application for which it is proposed."*

The plant-specific assumptions in the LSCS FPRA (Reference 6) and the 71 generic sources of uncertainty identified in EPRI 1026511 (Reference 5) were evaluated for their potential impact on the RICT application. This guideline organizes the uncertainties in Topic Areas similar to those outlined in NUREG/CR-6850 and was used to evaluate the baseline FPRA epistemic uncertainty and evaluate the impact of this uncertainty on RICT Program calculations.

A detailed review of the generic and plant-specific sources of internal fire model uncertainties are discussed in LS-MISC-046 (Reference 7) and are therefore not repeated in this enclosure. The purpose of this enclosure is to summarize the key sources of uncertainty that could potentially impact the RICT calculations.

Table E9-3 summarizes the review for key sources of uncertainty in the internal fire PRA model for the RICT application (organized by NUREG/CR-6850 tasks).

As noted above, the LSCS FPRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by NRC. Fire PRA methods were based on NUREG/CR-6850, other more recent NUREGs, (e.g., NUREG-7150 (Reference 11)), and published "frequently asked questions" (FAQs) for the Fire PRA.

The key sources of uncertainty identified in Table E9-3 do not present a significant impact on the LSCS RICT calculations and therefore, the internal events PRA model is capable of producing accurate RICT calculations. Note that RMAs will be developed when appropriate using insights from the PRA model results specific to the configuration.

Table E9-3

ASSESSMENT OF FIRE PRA EPISTEMIC UNCERTAINTY

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
1	Analysis boundary and partitioning	This task establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	Based on a review of the assumptions and potential sources of sources of uncertainly associated with this element, it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
2	Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the Fire PRA.	The uncertainty associated with this task is related to the identification of all components that should be credited/linked in the FPRA. This source of uncertainty is reduced as a result of multiple overlapping tasks including the MSO expert panel, reviews of FPIE screened initiating events, screened containment penetrations, and screened ISLOCA scenarios. Additional internal reviews of analysis results further reduce the uncertainty associated with this task. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Table E9-3

ASSESSMENT OF FIRE PRA EPISTEMIC UNCERTAINTY

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
3	Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	Additionally, as part of the Fire PRA, some components were conservatively assumed to be failed based on lack of cable data. Components in this category are referred to as Unknown Location (UNL) components because specific cables were not identified for the components. Based on recent Fire PRA updates, the UNL components are mostly limited to Balance of Plant (BOP) systems. A sensitivity analysis was performed to measure the risk associated with the assumption that these components fail in select fire scenarios. The sensitivity removed all UNL components from every fire scenario, as described in the Uncertainty & Sensitivity Analysis Notebook. Based on the results, the inclusion of the UNL components introduces moderate risk to both Fire CDF and LERF. Although the sensitivity shows a moderate impact on Fire CDF and Fire LERF, complete removal of UNLs would not be considered realistic since those cables could be identified with detailed circuit analysis and those failures would exist in specific areas of the plant. Also, the dominant fire scenarios are undeveloped full room burnouts that when refined with detailed fire modeling and fire scenario development would reduce the overall impact of the bounding sensitivity. Given that an informed approach was used to developing the assumed routing, the methodology employed by the Fire PRA is appropriate. Based on a review of the assumptions and potential sources of uncertainty related to this element it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Table E9-3

ASSESSMENT OF FIRE PRA EPISTEMIC UNCERTAINTY

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
4	Qualitative Screening	Qualitative screening was performed; however, some structures (locations) were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the FPRA (based on industry guidance and criteria) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables identified in the prior two tasks) and consequently are expected to have a low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
5	Fire-Induced Risk Model	The internal events PRA model was updated to add fire specific initiating event structure as well as additional system logic. The methodology used is consistent with that used for the internal events PRA model development as was subjected to industry Peer Review. The developed model is applied in such a fashion that all postulated fires are assumed to generate a plant trip. This represents a source of uncertainty, as it is not necessarily clear that fires would result in a trip. In the event the fire results in damage to cables and/or equipment identified in Task 2, the PRA model includes structure to translate them into the appropriate induced initiator.	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the Fire PRA development process would have reviewed significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Table E9-3

ASSESSMENT OF FIRE PRA EPISTEMIC UNCERTAINTY

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
6	Fire Ignition Frequency	Fire ignition frequency is an area with inherent uncertainty. Part of this uncertainty arises due to the counting and related partitioning methodology. However, the resulting frequency is not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the industry generic frequency values used for the Fire PRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates.	The LaSalle Fire PRA utilized the bin frequencies from NUREG/CR-2169 (Reference 12), which represents the most current approved source for bin frequencies. As such, some of the inherent conservatism associated with bin frequencies from NUREG/CR-6850 was removed. A parametric uncertainty analysis using the Monte Carlo method is provided in the FPRA documentation. Consensus approaches are employed in the model. Based on a review of the assumptions and potential sources of uncertainty related to this element it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
7	Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	Quantitative screening criteria was defined for the LaSalle Fire PRA as the CDF / LERF contribution of zero, such that all quantified fire scenarios are retained. All of the results were retained in the cumulative CDF / LERF, therefore, no uncertainty was introduced as a result of this task. Based on the discussion above, it is concluded that the methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
8	Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development (Tasks 8 and 11). The discussion of uncertainty for both tasks is provided in the discussion for Task 11.	See Task 11 discussion.

Table E9-3

ASSESSMENT OF FIRE PRA EPISTEMIC UNCERTAINTY

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
9	Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the Fire PRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. Hot short probabilities and hot short duration probabilities as defined in NUREG-7150, Volume 2, based on actual fire test data, were used in the LSCS Fire PRA. The uncertainty (conservatism) which may remain in the Fire PRA is associated with scenarios that do not contribute significantly to the overall fire risk. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
10	Circuit Failure Mode Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability and a hot short duration probability are assigned using industry guidance published in NUREG 7150, Volume 2. The uncertainty values specified in NUREG-7150, Volume 2 are based on fire test data.	The use of hot short failure probability and duration probability is based on fire test data and associated consensus methodology published in NUREG-7150, Volume 2. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Table E9-3

ASSESSMENT OF FIRE PRA EPISTEMIC UNCERTAINTY

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
11	Detailed Fire Modeling	The application of fire modeling technology is used in the Fire PRA to translate a fire initiating event into a set of consequences (fire-induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and response of plant staff (detection, fire control, fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution (aleatory uncertainty). The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	Consensus modeling approach is used for Detailed Fire Modeling and it is concluded that the methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no additional RMAs are required to address this item.

Table E9-3

ASSESSMENT OF FIRE PRA EPISTEMIC UNCERTAINTY

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
12	Post-Fire Human Reliability Analysis	The human error probabilities (HEPs) used in the Fire PRA were adjusted to consider the additional challenges that may be present given a fire. The HEPs were obtained using the EPRI HRA Calculator (HRAC) and included the consideration of degradation or loss of necessary cues due to fire. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The HEPs include the consideration of degradation or loss of necessary cues due to fire. The fire risk importance measures indicate that the results are somewhat sensitive to HRA model and parameter values. The LSCS Fire PRA model HRA is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Assuming no credit for operator response is not realistic. However, the TSTF-505 procedure will require appropriate Risk Management Action (RMA) focus on human performance for RICT entry (e.g., including an operator briefing on the significant human actions in the PRA that are pertinent to the configuration). It is concluded that the methodology for the Post-Fire Human Reliability Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, RICT Program calculations are not impacted, and no additional RMAs are required to address this item.
13	Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic-induced fires should not be a source of model uncertainty as it is not expected to provide changes to the quantified Fire PRA model. A conservative seismic hazard penalty is already applied to all RICT calculations to account for seismic risk impact. Based on the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
14	Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit. However, the selected truncation was confirmed to be consistent with the requirements of the PRA Standard (Reference 8).	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard (Reference 8). Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Table E9-3

ASSESSMENT OF FIRE PRA EPISTEMIC UNCERTAINTY

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
15	Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty.	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty. Based on the discussion above, it is concluded that the methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
16	FPRA Documentation	This task does not introduce any new uncertainties to the fire risk.	This task does not introduce any new uncertainties to the fire risk as it outlines documentation requirements. Based on the discussion above, it is concluded that the methodology for the Fire PRA documentation task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

5. References

1. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, October 2012 (ADAMS Accession No. ML12286A322).
2. NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, Revision 1, March 2017.
3. EPRI 1016737, Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, December 2008.
4. LS-PSA-013, LaSalle County Generating Station Probabilistic Risk Analysis Summary Notebook, Revision 8, November 2015.
5. EPRI 1026511, Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty, December 2012.
6. LS-PSA-021.12, LaSalle Fire PRA Uncertainty and Sensitivity Analysis Notebook, Rev. 3, 2019.
7. LS-MISC-046, Assessment of Key Assumptions and Sources of Uncertainty for Risk-Informed Applications, Revision 0, January 2020.
8. ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RAS-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.
9. NUREG/CR-6595, An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events, Revision 1, October 2004.
10. NUREG/CR-6850 (also EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," September 2005, with Supplement 1 (EPRI 1019259), September 2010.
11. NUREG/CR-7150, Joint Asessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), October 2012.
12. NUREG-2169 / EPRI 300200936, Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, Revision 0, January 2015.

ENCLOSURE 10

License Amendment Request

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

Program Implementation

ENCLOSURE 10
Program Implementation

1. Introduction

Section 4.0, Item 11 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A (Reference 2) requires that the license amendment request (LAR) provide a description of the implementing programs and procedures regarding the plant staff responsibilities for the Risk Managed Technical Specifications (RMTS) implementation, and specifically discuss the decision process for risk management action (RMA) implementation during a Risk-Informed Completion Time (RICT).

This enclosure provides a description of the implementing programs and procedures regarding the plant staff responsibilities for the RICT Program, including training of plant personnel, and specifically discusses the decision process for RMA implementation during extended Completion Times (CT).

2. RICT Program and Procedures

Exelon will develop a program description and implementing procedures for the RICT Program. The program description will establish the management responsibilities and general requirements for risk management, training, implementation, and monitoring of the RICT program. More detailed procedures will provide specific responsibilities, limitations, and instructions for implementing the RICT program. The program description and implementing procedures will incorporate the programmatic requirements for RMTS included in NEI 06-09-A. The program will be integrated with the online work control process. The work control process currently identifies the need to enter an LCO Action statement as part of the planning process and will additionally identify whether the provisions of the RICT program are required for the planned work. The risk thresholds associated with 10CFR50.65(a)(4) will be coordinated with the RICT limits. The Maintenance Rule performance monitoring provisions and Mitigating System Performance Index (MSPI) thresholds will assist in controlling the amount of risk expended in use of the RICT program.

The Operations Department (licensed operators) is responsible for compliance with the TS and will be responsible for implementation of RICTs and RMAs. Entry into the RICT program will require management approval prior to pre-planned activities and as soon as practicable following emergent conditions.

The procedures for the RICT program will address the following attributes consistent with NEI 06-09-A:

- 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 Real-Time Risk tool.
- Guidance on recalculating RICT and risk management action time (RMAT) within 12 hours or within the most limiting front-stop CT after a plant configuration change.

- Requirements to identify and implement RMAs when the RMAT is exceeded or is anticipated to be exceeded, and to consider common cause failure potential in emergent RICTs.
- Guidance on the use of RMAs including the conditions under which they may be credited in RICT calculations.
- 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.

3. RICT Program Training

The scope of training for the RICT Program will include rules for the new TS program, Real-Time Risk tool software, TS Actions included in the program, and procedures. This training will be conducted for the following Exelon 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
- Regulatory Assurance Personnel
- Selected Maintenance Personnel
- Engineering
- Risk Management
- Other Selected Management

Corporate Personnel
- Operations Corporate Functional Area Manager
- Fleet Outages Corporate Functional Area Manager
- Licensing Management and Personnel
- Risk Management Personnel and Managers
- Training Management and Personnel
- Other Selected Management

Training will be carried out in accordance with Exelon training procedures and processes. These procedures were written based on the Institute of Nuclear Power Operations (INPO) Accreditation (ACAD) requirements, as developed and maintained by the National Academy for Nuclear Training. Exelon has planned three levels of training for implementation of the RICT Program. They are described below:

Level 1 Training

This is the most detailed training. It is intended for the individuals who will be directly involved in the implementation of the RICT Program. This level of training includes the following attributes:

- Specific training on the revised TS
- Record keeping requirements
- Case studies
- Hands-on experience with the Real-Time Risk tool for calculating RMAT and RICT
- Identifying appropriate RMAs
- Common cause failure RMA considerations in emergent RICTs
- Other detailed aspects of the RICT Program

Level 2 Training

This training is applicable to plant management positions with authority to approve entry into the RICT Program, as well as supervisors, managers, and other personnel who will closely support RICT implementation. These individuals need a broad understanding of the purpose, concepts, and limitations of the RICT Program. Level 2 training is significantly more detailed than Level 3 training (described below), but it is different from Level 1 training in that hands-on time with the Real-Time Risk tool, case studies, and other specifics are not required.

Level 3 Training

This training is intended for the remaining personnel who require an awareness of the RICT Program. These employees need basic knowledge of the RICT Program requirements and procedures. This training will cover the RICT Program concepts that are important to disseminate throughout the organization.

4. References

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

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

ENCLOSURE 11

License Amendment Request

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Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

Monitoring Program

ENCLOSURE 11
Monitoring Program

1. Introduction

Section 4.0, Item 12 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A (Reference 2) requires that the license amendment request (LAR) provide a description of the implementation and monitoring program as described in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1 (Reference 3), and NEI 06-09-A (Reference 2). (Note that RG 1.174, Revision 2 [Reference 4], issued by the NRC in May 2011, made editorial changes to the applicable section referenced in the NRC safety evaluation for Section 4.0, Item 12.)

This enclosure provides a description of the process applied to monitor the cumulative risk impact of implementation of the Risk-Informed Completion Time (RICT) Program, specifically the calculation of cumulative risk of extended Completion Times (CTs). Calculation of the cumulative risk for the RICT Program is discussed in Step 14 of Section 2.3.1 and Step 7.1 of Section 2.3.2 of NEI 06-09-A, Risk Informed Technical Specifications Initiative 4b (Reference 2). General requirements for a Performance Monitoring Program for risk-informed applications are discussed in Element 3 of Regulatory Guide 1.174 (Reference 3).

2. Description of Monitoring Program

The RICT Program will require calculation of cumulative risk impact at least every refueling cycle, not to exceed 24 months, consistent with the guidance in NEI 06-09-A (Reference 2). 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 (Reference 4), 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 (RMAs) 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.

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 (Reference 3) and NEI 06-09-A (Reference 2).

3. References

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

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

3. Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.

4. Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011.

ENCLOSURE 12

License Amendment Request

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Risk Management Action Examples

ENCLOSURE 12
Risk Management Action Examples

1. Introduction

This enclosure describes the process for identification and implementation of Risk Management Actions (RMA) applicable during extended Completion Times (CT) and provides examples of RMAs. RMAs will be governed by plant procedures for planning and scheduling maintenance activities. The procedures will provide guidance for the determination and implementation of RMAs when entering the Risk-Informed Completion Time (RICT) Program consistent with the guidance provided in NEI 06-09-A, Revision 0 (Reference 1).

2. Responsibilities

For planned entries into the RICT Program, Work Management is responsible for developing the RMAs with assistance from Operations and Risk Management. Operations is responsible for approval and implementation of RMAs. For emergent entry into extended CTs, Operations is also responsible for developing the RMAs.

3. Procedural Guidance

For planned maintenance activities, implementation of RMAs will be required if it is anticipated that the risk management action time (RMAT) will be exceeded. For emergent activities, RMAs must be implemented if the RMAT is reached. Also, if an emergent event occurs requiring recalculation of a RMAT already in place, the procedure will require a reevaluation of the existing RMAs for the new plant configuration to determine if new RMAs are appropriate. These requirements of the RICT Program are consistent with the guidance of NEI 06-09-A.

For emergent entry into a RICT, if the extent of condition is not known, RMAs related to the success of redundant and diverse SSCs and reducing the likelihood of initiating events relying on the affected function will be developed to address the increased likelihood of a common cause event.

RMAs will be implemented in accordance with current procedures (e.g., References 2, 3, 4,5) no later than the time at which an incremental core damage probability (ICDP) of 1E-6 is reached, or no later than the time when an incremental large early release probability (ILERP) of 1E-7 is reached. If, as the result of an emergent condition, the instantaneous core damage frequency (ICDF) or the instantaneous large early release frequency (ILERF) exceeds 1E-3 per year or 1E-4 per year, respectively, RMAs are also required to be implemented. These requirements are consistent with the guidelines of NEI 06-09-A.

By determining which structures, systems, or components (SSCs) are most important from a CDF or LERF perspective for a specific plant configuration, RMAs may be created to protect these SSCs. Similarly, knowledge of the initiating event or sequence contribution to the configuration-specific CDF or LERF allows development of RMAs that enhance the capability to mitigate such events. The guidance in NUREG-1855 (Reference 6) and EPRI TR-1026511 (Reference 7) will be used in examining PRA results for significant contributors for the configuration, to aid in identifying appropriate compensatory measures (e.g., related to risk-significant systems that may provide diverse protection, or important support systems or human actions). Enclosure 9 identifies several areas of uncertainty in the internal events and fire PRAs that will be considered in defining configuration-specific RMAs when entering a RICT.

ENCLOSURE 12
Risk Management Action Examples

If the planned activity or emergent condition includes an SSC that is identified to impact Fire PRA, as identified in the current Real Time Risk Program, Fire PRA specific RMAs associated with that SSC will be implemented per the current plant procedure.

It is possible to credit RMAs in RICT calculations, to the extent the associated plant equipment and operator actions are modeled in the PRA; however, such quantification of RMAs is neither required nor expected by NEI 06-09-A. Nonetheless, if RMAs will be credited to determine RICTs, the procedure instructions will be consistent with the guidance in NEI 06-09-A.

NEI 06-09-A classifies RMAs into the three categories described below:

1) Actions to increase risk awareness and control.

   - Shift brief
   - Pre-job brief
   - Training
   - Presence of system engineer or other expertise related to the activity
   - Special purpose procedure to identify risk sources and contingency plans

2) Actions to reduce the duration of maintenance activities.

   - Pre-staging materials
   - Conducting training on mock-ups
   - Performing the activity around the clock
   - Performing walk-downs on the actual system(s) to be worked on prior to beginning work

3) Actions to minimize the magnitude of the risk increase.

   - Suspend or minimize activities on redundant systems
   - Suspend or minimize activities on other systems that adversely affect the CDF or LERF
   - Suspend or minimize activities on systems that may cause a trip or transient to minimize the likelihood of an initiating event that the out-of-service component is meant to mitigate
   - Use temporary equipment to provide backup power, ventilation, etc.
   - Reschedule other risk-significant activities

Determination of RMAs involves the use of both qualitative and quantitative considerations for the specific plant configuration and the practical means available to manage risk. The scope and number of RMAs developed and implemented are reached in a graded manner.

Procedural guidance for development of RMAs in support of the RICT program builds off the RMAs developed for other processes, such as the RMAs developed under the 10CFR 50.65(a)(4) program and the protected equipment program. Additionally, Common Cause RMAs are developed to address the potential impact of common cause failures.

General RMAs are developed for input into the RICT system guidelines. These guidelines are listed in site-specific T&RMs and are developed using a graded approach. Consideration is given for system functionality and includes consideration for common cause impacts within the system. These RMAs include:

- Consideration of rescheduling maintenance to reduce risk
- Discussion of RICT in pre-job briefs
- Consideration of proactive return-to-service of other equipment
- Efficient execution of maintenance

In addition to the RMAs developed qualitatively for the system guidelines, RMAs are developed based on the Real-Time Risk tool to identify configuration-specific RMA candidates to manage the risk associated with internal events, internal flooding, and fire events. These actions include:

- Identification of important equipment or trains for protection
- Identification of important Operator Actions for briefings
- Identification of key fire initiators and fire zones for RMAs in accordance with the site Fire RMA process
- Identification of dominant initiating events and actions to minimize potential for initiators
- Consideration of insights from PRA model cutsets, through comparison of importances

Common cause RMAs are also developed to ensure availability of redundant SSCs, to ensure availability of diverse or alternate systems, to reduce the likelihood of initiating events that require operation of the out-of-service components, and to prepare plant personnel to respond to additional failures. Common cause RMAs are developed by considering the impact of loss of function for the affected SSCs.

Examples of common cause RMAs include:

- Performance of non-intrusive inspections on alternate trains
- Confidence runs performed for standby SSCs
- Increased monitoring for running components
- Expansion of monitoring for running components
- Deferring maintenance and testing activities that could generate an initiating event which would require operation of potentially affected SSCs
- Readiness of operators and maintenance to respond to additional failures
- Shift briefs or standing orders which focus on initiating event response or loss of potentially affected SSCs

Per Exelon procedure, for emergent conditions where the extent of condition is not performed prior to entering into the Risk Management Action Times or the extent of condition cannot rule out the potential for common cause failure, common cause RMAs are expected to be implemented to mitigate common cause failure potential and impact. These can include the pre-identified RMAs included in the system guidelines as discussed above, as well as alternative common cause RMAs for the specific configuration. Alternate RMAs, including both regular and common cause considerations, are developed for the specific configuration following the steps outlined above.

4. Examples

Multiple example RMAs that may be considered during a RICT Program entry to reduce the risk impact and ensure adequate defense-in-depth are provided below. Specific examples are given for unavailability of one Diesel Generator (DG), one Offsite Source, one Battery Charger, or one Residual Heat Removal (RHR) pump.

4.1. Electrical Action Statements

4.1.1 For TS action 3.8.1.B, One required Division 1, or 2 DG inoperable OR Required opposite unit Division 2 DG inoperable. Additional RMAs would include:

1. Actions to increase risk awareness and control.

    - Briefing of the on-shift Operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a Loss of Offsite Power and station blackout including bus crossties.

    - Notification of the TSO of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.

    - Proactive implementation of RMAs during times of high grid stress conditions, such as during high demand conditions.

2. Actions to reduce the duration of maintenance activities.

    - For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability.

    - Confirmation of parts availability prior to entry into a preplanned RICT.

    - Walkdown of work prior to execution.

3. Actions to minimize the magnitude of the risk increase.

    - Evaluation of weather conditions for threats to the reliability of offsite power supplies.

    - Deferral of elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers associated with the unit.

    - Deferral of planned maintenance or testing that affects the reliability of operable DGs and their associated support equipment. Treat the remaining operable DGs as protected equipment.

    - Deferral of planned maintenance or testing on redundant train safety systems. If testing or maintenance activities must be performed, a review of the potential risk impact will be performed.

    - Implementation of 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected DGs.

4.1.2 TS action 3.8.1.A, one offsite source inoperable, additional RMAs would include:

1. Actions to increase risk awareness and control.

   - Briefing of the on-shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a Loss of Offsite Power and station blackout including bus crossties.

   - Notification of the TSO of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.

   - Proactive implementation of RMAs during times of high grid stress conditions prior to reaching the RMAT, such as during high demand conditions.

2. Actions to reduce the duration of maintenance activities.

   - For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability

   - Confirmation of parts availability prior to entry into a preplanned RICT.

   - Walkdown of work prior to execution.

3. Actions to minimize the magnitude of the risk increase.

   - Evaluation of weather conditions for threats to the reliability of remaining offsite power supplies.

   - Deferral of elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers associated with the unit.

   - Protection of the remaining offsite source, including switchyard and transformer.

   - Deferral of planned maintenance or testing that affects the reliability of DGs and their associated support equipment. Treat these as protected equipment.

   - Implementation of 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected offsite source.

4.1.3 TS action 3.8.1.E, one required offsite circuit inoperable AND One required Division 1, 2, or 3 DG inoperable, additional RMAs would include:

1. Actions to increase risk awareness and control.

- Briefing of the on-shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a Loss of Offsite Power and station blackout.

- Notification of the TSO of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.

- Proactive implementation of RMAs during times of high grid stress conditions prior to reaching the RMAT, such as during high demand conditions.

- For a planned RICT, prior to removal from service the actions in the associated loss of bus procedure would be reviewed and implemented.

2. Actions to reduce the duration of maintenance activities.

   - For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability.

   - Confirmation of parts availability prior to entry into a preplanned RICT.

   - Walkdown of work prior to execution.

3. Actions to minimize the magnitude of the risk increase.

   - Deferral of elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers associated with the unit.

   - Deferral of planned maintenance or testing that affects the reliability of DGs and their associated support equipment for the remaining buses.

   - Implementation of 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected bus.

   - Place unaffected trains of systems into service. For example, if one of two instrument nitrogen compressors is powered by the affected bus, the other unaffected compressor would be placed in service to support containment atmosphere control. This would be done prior to entry into a planned RICT.

4.1.4 TS action 3.8.4.A, One required Division 1, 2, or 3 125 VDC battery charger on one division inoperable. OR One required Division 2 or opposite unit Division 2 battery chargers on one division inoperable. OR One required Division 1 250 VDC battery charger inoperable. Additional RMAs would include:

1. Actions to increase risk awareness and control.

   - Briefing of the on-shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a Loss of DC division and station blackout.

- Briefing of the on-shift operations crew concerning the impact the DC division has on the potential response to plant events such as reduced control systems.
- Prior to removal from service. If a Planned RICT, the actions in the associated loss of bus procedure would be reviewed and implemented.
- Minimize activities that could trip the unit.

2. Actions to reduce the duration of maintenance activities.
   - For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability.
   - Confirmation of parts availability prior to entry into a preplanned RICT.
   - Walkdown of work prior to execution.

3. Actions to minimize the magnitude of the risk increase.
   - Deferral of elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers associated with the unit.
   - Protection of the remaining DC electrical buses in that unit.
   - Remove nonessential loads from battery to extend time voltage will remain above minimum required level.
   - Implementation of 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected bus.

4.2. ECCS Action Statements

4.2.1 TS Action 3.5.1.A, one low pressure ECCS injection/spray subsystem inoperable, the RMAs would include the following.

1. Defer planned maintenance or testing activities on the redundant SPC loop and its associated support equipment. Treat those systems as protected equipment.
2. Defer planned maintenance or testing that affects the reliability of those safety systems that provide a defense-in-depth. If testing or maintenance activities must be performed, a review of the potential risk impact will be performed.
3. Minimize activities that could trip the unit.
4. Verify system alignment of low pressure ECCS.

5. Implement 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected ECCS loop.

5. References

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

2. Exelon Procedure OP-AA-201-012-1001, "Operations On-Line Fire Risk Management"

3. LS-CRM-08, "Development of Risk Management Actions for the Inclusion of Fire Insights into LaSalle Configuration Risk Management Program"

4. Exelon Procedure WC-AA-101-1006, "On-Line Risk Management and Assessment"

5. Exelon Procedure OP-AA-108-117, "Protected Equipment Program"

6. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," U.S. Nuclear Regulatory Commission, March 2009.

7. EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk- Informed Applications with a Focus on the Treatment of Uncertainty," Technical Update, Electric Power Research Institute, December 2012