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10 CFR 50.90
10 CFR 50.69

RS-20-136

October 29, 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: LaSalle County Station, Unit Nos. 1 And 2 – Response to Request for Additional Information Regarding License Amendment Requests for Amendments to Renewed Facility Operating Licenses to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," and to Adopt TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b," (EPID L-2020-LLA-0017 AND EPID-L-2020-LLA-0018)

References:

1. Letter from D. Murray (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Application to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b,'" dated January 31, 2020. (ML20035E577)

2. Letter from D. Murray (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory commission, "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,'" dated January 31, 2020. (ML20031E699)

3. Letter from B. Vaidya (Project Manager, U.S Nuclear Regulatory Commission) to B. Hanson (Exelon Generation Company, LLC), "LaSalle County Station, Unit Nos. 1 And 2 – Request for Additional Information Regarding License Amendment Requests for Amendments to Renewed Facility Operating Licenses to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors' and to Adopt TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion

Times - RITSTF Initiative 4b (EPID L-2020-LLA-0017 AND EPID-L-2020-LLA-0018),'" dated September 29, 2020 (ML20247J408)

In References 1 and 2, Exelon Generation Company, LLC (EGC) submitted two license amendment requests (LARs) to the U.S. Nuclear Regulatory Commission (NRC) for Renewed Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station, Units 1 and 2 (LSCS):

(1) Request to modify (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20031E699) the LaSalle licensing basis to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR) Section 69 (50.69), "Risk-Informed Categorization and Treatment of Structures, Systems and Components [SSCs] for Nuclear Power Reactors." The proposed changes are based on Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," dated July 2005 (ADAMS Accession No. ML052910035).

(2) Request to revise (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20035E577) the LaSalle TS requirements to permit the use of risk-informed completion times for actions to be taken when limiting conditions for operation are not met. The proposed changes are based on Technical Specifications Task Force Traveler (TSTF-505, Revision 2, "Provide Risk Informed Extended Completion Times – RITSTF Initiative 4b," dated July 2, 2018 (ADAMS Accession No. ML18269A041).

The NRC has reviewed References 1 and 2 and conducted a Regulatory Audit from June 15 through June 19, 2020. The NRC has identified areas where additional information is needed to complete its review.

On September 21, 2020, the NRC held a teleconference with EGC to clarify this request to ensure that EGC understood the NRC request.

Attachments 1 through 3 to this letter contains the NRC's request for additional information along with EGC's response.

EGC has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the NRC in References 1 and 2. The supplemental information provided in this letter does not affect the bases for concluding that the proposed license amendments do not involve a significant hazards consideration. Furthermore, the supplemental information provided in this letter does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendments.

There are no regulatory commitments contained in this letter.

Should you have any questions regarding this submittal, please contact Jason Taken at 630-806-9804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 29th day of October 2020.

Respectfully,

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

Attachment 1: Response to Request for Additional Information
Attachment 2: Revision to Attachment 5 of TSTF-505 RICT LAR for New Implementation Items Related to Resolution of Fire F&Os
Attachment 3: Revision to Section 2.3 of 10 CFR 50.69 LAR for New Implementation Items Related to Resolution of Fire F&Os

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

ATTACHMENT 1

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

Response to Request for Additional Information

DRA/APLA(C) RAI 01 – OPEN FIRE PRA FACTS AND OBSERVATIONS (F&OS)

[*Applicable for TSTF-505 and 10 CFR 50.69*]

LAR Enclosure 2, Table E2-2, presents the dispositions for three facts and observations (F&Os) that remain open after the Independent Assessment (IA) performed for closure of F&Os; two that remain open (i.e., 1-19, 4-17) and one partially resolved (i.e., 6-11). For F&Os 4-17 and 6-11, the licensee states the items will be resolved prior to TSTF-505 implementation, however, Attachment 5 of the LAR that includes a table of implementation items to be completed prior to implementation of the RICT program does not include these items. In light of these observations provide the following:

a) Regarding F&O 4-17, the LARs dispositions state that [t]his item will be resolved prior to TSTF-505 implementation [and 10 CFR 50.69 implementation]" for each respective LAR. Furthermore, the licensee states that the impact of this issue is "judged to be minimal." However, it is not clear to NRC staff the impact on the RICT calculations. Therefore, address the following:

    i. Provide justification (e.g., description and results of a sensitivity study) that any needed adjustments made to the fire PRA (FPRA) identified from review of the plant-specific data on the fire suppression and detection systems cannot impact the RICT calculations performed to support TSTF-505 or does not adversely impact the importance measures and risk metrics used to assess structures, systems, and components (SSC) categorization for 10 CFR 50.69.

    ii. Alternatively, propose a mechanism that ensures the review of plant-specific data for fire suppression and detection systems is performed and any update needed to the FPRA is completed prior to implementation of the RICT program and 10 CFR 50.69 (e.g., include as an implementation item in the respective LARs).

b) Regarding F&O 6-11, the LARs dispositions discuss that a review will be performed to verify consistency with NEI 00-01, Revision 3, prior to implementation of the RICT program [and 10 CFR 50.69 implementation] for each respective LAR. However, no commitment to complete an implementation item for this F&O is made in either LAR. Also, the NRC staff notes that in the event the review cannot verify the circuit analysis was performed in accordance with the requirements of NEI-00-01, Revision 3, then adjustments to the FPRA model may be needed.

    i. Provide sufficient justification to support the conclusion provided in Table E2-2 that any revisions to the cable selection based upon historical methods used during the time of the analysis development and more recent guidance (i.e., NEI 00-01, Revision 3) has no impact on the TSTF-505 RICT calculations performed to support TSTF-505 or does not adversely impact the importance measures and risk metrics used to assess SSC categorization for 10 CFR 50.69.

    ii. Alternatively, propose a mechanism that ensures the review of the circuit analysis to the requirements of NEI-00-01, Revision 3, is performed and any needed update to the FPRA is completed prior to implementation of the RICT program and 10 CFR 50.69 (e.g., include as an implementation item in the respective LARs).

<u>EGC RESPONSE:</u>

Parts (a)(i) & (b)(i)

Not applicable – See responses to Parts (a)(ii) and (b)(ii) for details.

Parts (a)(ii) & (b)(ii)

<u>TSTF-505 RICT LAR</u>

Attachment 5 of the TSTF-505 RICT LAR has been updated with new implementation items related to resolution of Fire F&O 4-17 and Fire F&O 6-11, as summarized in Attachment 2 of this letter. The changes to Attachment 5 of the TSTF-505 RICT LAR are identified with **bold** text.

Table APLA-01.1 summarizes the new TSTF-505 RICT implementation items related to resolution of these Fire F&Os.

TABLE APLA-01.1
NEW IMPLEMENTATION ITEMS FOR TSTF-505 RICT LAR

Source	Description	Implementation Item
Enclosure 2, Table E2-2, Fire F&O 4-17	As stated in Table E2-2 of the TSTF-505 RICT LAR for Fire F&O 4-17.	As stated in Table E2-2 of the TSTF-505 RICT LAR for Fire F&O 4-17.
Enclosure 2, Table E2-2, Fire F&O 6-11	As stated in Table E2-2 of the TSTF-505 RICT LAR for Fire F&O 6-11.	As stated in Table E2-2 of the TSTF-505 RICT LAR for Fire F&O 6-11.

<u>10 CFR 50.69 LAR</u>

Section 2.3 of the 10 CFR 50.69 LAR has been updated with new implementation items related to resolution of Fire F&O 4-17 and Fire F&O 6-11, as summarized in Attachment 3 of this letter. The changes to Section 2.3 of the 10 CFR 50.69 LAR are identified with **bold** text.

Table APLA-01.2 summarizes the new 10 CFR 50.69 implementation items related to resolution of these Fire F&Os.

TABLE APLA-01.2
NEW IMPLEMENTATION ITEMS FOR 10 CFR 50.69 LAR

Source	Description	Implementation Item
Attachment 3, Fire F&O 4-17	As stated in Attachment 3 of the 10 CFR 50.69 LAR for Fire F&O 4-17.	As stated in Attachment 3 of the 10 CFR 50.69 LAR for Fire F&O 4-17.
Attachment 3, Fire F&O 6-11	As stated in Attachment 3 of the 10 CFR 50.69 LAR for Fire F&O 6-11.	As stated in Attachment 3 of the 10 CFR 50.69 LAR for Fire F&O 6-11.

<u>DRA/APLA(B) RAI 02 – FIRE HAZARDS</u>

<u>[*Applicable for TSTF-505 and 10 CFR 50.69*]</u>

RG 1.200 states "NRC reviewers, [will] focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application." Some concerns are not always readily identified in F&Os by the peer review teams but are considered potential key assumptions by the NRC staff because using more defensible and less simplified assumptions could substantively affect the fire risk and overall risk profile of the plant. The NRC staff notes that the calculated results of the PRAs are used directly to calculate a RICT and evaluate the categorization of SSCs. Specifically, for the TSTF-505 application, the PRA results are relied upon considerably to determine how long SSCs controlled by a TS can remain inoperable; therefore, the NRC staff requests additional information for the following areas.

TSTF-505 LAR, Enclosure 9, Section 4, and Attachment 6 of the 10 CFR 50.69 LAR discusses that the LaSalle FPRA was guided using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by NRC. The licensee further states in both of the LARs that "[f]ire PRA methods were based on …, other more recent NUREGs, (e.g., NUREG-7150 […]), and published frequently asked questions (FAQs) for the Fire PRA." Furthermore for the TSTF-505 LAR, Part (e), of the proposed TS 5.5.17 ("Risk Informed Completion Time Program") states, in part, "[m]ethods 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."

The integration of NRC-accepted FPRA methods and studies described below that are relevant to these submittals could potentially impact the TSTF-505 RICT calculations, categorization of SSCs, and risk metrics for total core damage frequency (CDF) and total large early release frequency (LERF):

- NUREG-2178, Volume 1, "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE)," dated April 2016 (ADAMS Accession No. ML16110A140).
- NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE)," dated December 2016 (ADAMS Accession No. ML16343A058).

a) There have been changes to the FPRA methodology since the last full-scope peer review of the LaSalle FPRA in 2015. For each of the above NRC-accepted FPRA methods and studies, the NRC staff requests the licensee address one of the following for parts (i) through (iii) of this RAI:

    i. Discuss how the FPRA method/study had been incorporated into the LaSalle FPRA and, as applicable, summarize the changes made to the FPRA model. Indicate whether this change was PRA maintenance or a PRA upgrade as defined in ASME/ANS RA-Sa-2009, Section 1-5.4, as qualified by RG 1.200, Revision 2, along with a justification for the determination. If this change constitutes a PRA upgrade, discuss the focused-scope (or full-scope) peer review(s) that has been performed to evaluate the change, and provide any open F&Os and associated dispositions from this peer review(s) in accordance with RG 1.200, Revision 2.

    OR

ii. If the FPRA method/study has not been incorporated into the LaSalle FPRA, provide a detailed justification for why the integration of the FPRA method/study would not change the conclusions of the LARs, and subsequently not impact the TSTF-505 RICT calculations, categorization of SSCs, and risk metrics for total CDF and total LERF. As part of this justification, identify any FPRA methodologies used in the LaSalle FPRA that are no longer accepted by the NRC staff (e.g., guidance provided in FAQ 08-0046, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems," ADAMS Accession No. ML093220426, has been retired by letter dated July 1, 2016, (ADAMS Accession No. ML16167A444). Provide technical justification for its use in TSTF-505 RICT calculations and the categorization of SSCs and evaluate the significance of its use on the risk metrics provided in Enclosure 5 of the TSTF-505 LAR and Attachment 2 of the 50.69 LAR.

OR

iii. Propose a mechanism that ensures the FPRA method/study (or other NRC acceptable method) will be integrated into the LaSalle FPRA prior to implementation of the RICT program and 10 CFR 50.69. If this FPRA update is determined to be a PRA model upgrade per the ASME/ANS PRA standard, include in this mechanism a process for conducting a focused-scope peer review and ensure any findings are closed by using an approved NRC process.

b) Application of Minimum Joint Human Error Probability (HEP) Values in the Fire PRA

Section 6.2 of NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines (ADAMS Accession No. ML12216A104)" cites NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," which advises that minimum joint human error probabilities (JHEP), "not be below ~1.0E-05 since it is typically hard to defend other independent failure modes that are not usually treated…" NUREG-1921 also states in part, "while it might be reasonable to adopt some sort of limit, it needs to be done carefully, so that the results of PRAs are not distorted by arbitrary assignments of probabilities. As discussed in detail later on, any limiting values should be consistent within the context of the scenarios in which they are applied." Furthermore, Table 4-4 of Electrical Power Research Institute (EPRI) 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. For the NRC staff to assess the HRA and application of JHEPs in the FPRA model for technical acceptability consistent with RG 1.200, Revision 2, provide the following:

i. If a minimum JHEP value less than 1E-05 was used in the FPRA, then provide sufficient justification (e.g., results of a sensitivity study, basis for inapplicability of NUREG-1792, etc.) to confirm that the minimum JHEP value is reasonable and sufficient for the assigned human failure events (HFEs).

ii. Alternatively, propose a mechanism that ensures the JHEP values used in the FPRA are consistent with the guidance in NUREG-1792 for JHEP floor values prior to implementation of the risk-informed applications.

c) <u>Well-Sealed Motor Control Center (MCC) Cabinets</u>

Guidance in FAQ 08-0042 from Supplement 1 of NUREG/CR-6850 applies to electrical cabinets below 440 volts (V). With respect to Bin 15 as discussed in Chapter 6 of NUREG/CR-6850, it clarifies the meaning of "robustly or well-sealed." Thus, for cabinets of 440 V or less, fires from well-sealed cabinets do not propagate outside the cabinet. For cabinets of 440 V and higher, the original guidance in Chapter 6 remains and requires that Bin 15 panels which "house circuit voltages of 440 V or greater are counted because an arcing fault could compromise panel integrity (an arcing fault could burn through the panel sides, but this should not be confused with the high energy arcing fault type fires)." Fire PRA FAQ 14-0009, "Treatment of Well-Sealed MCC [motor control center] Electrical Panels Greater than 440V" (ADAMS Accession No. ML15119A176) provides the technique for evaluating fire damage from MCC cabinets having a voltage greater than 440 V. Therefore, propagation of fire outside the ignition source panel must be evaluated for all MCC cabinets that house circuits of 440 V or greater.

i. Describe how fire propagation outside of well-sealed MCC cabinets greater than 440 V is evaluated.

ii. If well-sealed cabinets less than 440 V are included in the Bin 15 count of ignition sources, provide justification for using this approach. Justification should be sufficient in detail to ascertain that there is no adverse impact to the FRPA model used for performing RICT calculations and the categorization of SSCs.

d) <u>PRA Treatment of Fire Dependencies Between Units 1 and 2</u>

Many plants have Units 1 and 2 adjoined and, thus, have common areas. For these plants, the risk contribution from fires originating in one unit must be addressed for impacts to the other unit given the physical proximity of the other unit and common areas. Therefore, address the following for Units 1 and 2 adjacent and common areas.

i. Discuss how the risk contribution of fires originating in one unit is addressed for the other unit given impacts due to the physical proximity of equipment and cables. Include identification of locations where a fire in one unit can affect components in the other unit and explain how the risk contributions of such scenarios are allocated for a RICT calculation, the categorization of SSCs, and RG 1.174 risk thresholds.

<u>EGC RESPONSE:</u>

a) <u>Recently Published FPRA Methods & Studies</u>

Parts (a)(i) & (a)(ii)

<u>NUREG-2178, Vol. 1 – Refining And Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE)</u>

A draft version of NUREG-2178, Vol. 1 [2-1] was incorporated into the LaSalle Fire PRA that was peer reviewed in December 2015 [2-2]. The peer reviewed Fire PRA model utilized the updated heat release rate (HRR) distributions for the electrical cabinets specified in the draft version of the report ("default" cable loading assumed for all electrical cabinets). The peer review did not identify any issues with the application of the draft report (i.e., no F&Os related to application of NUREG-2178).

Once the report was officially issued, the Fire PRA model was reviewed and updated (as necessary) to ensure that the correct final values were used in the fire modeling calculations. Since the methodology was reviewed as part of the Fire PRA Peer Review, updating the heat release rate values is considered "PRA maintenance".

<u>NUREG-2180, Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE)</u>

N/A – LaSalle does not have any incipient fire detection systems. Therefore, NUREG-2180 is not applicable.

Part (a)(iii)

Not applicable – See response to parts (a)(i) & (a)(ii) for details.

b) <u>Application of Minimum Joint Human Error Probability (HEP) Values in the Fire PRA</u>

Part (b)(i)

The HRA Dependency Analysis methodology used for the LaSalle Fire PRA has been evaluated as part of the December 2015 Peer Review [2-2] and the methodology meets Capability Category II (CC II) of the ASME/ANS PRA Standard. The Dependency Module of the HRA Calculator (HRAC) was used to develop the list of combinations and each combination was reviewed and adjusted (as necessary) to ensure accurate levels of dependency, order of actions, timing, and other criteria.

For the Fire PRA HRA Dependency Analysis, two floor values were considered for the nominally calculated joint human error probabilities (JHEPs). For JHEP values less than 1E-06, a minimum (floor) JHEP of 1E-06 was used, unless the timeframe for completing one or more actions in the combination was longer than 15 hours, for which a lower floor JHEP of 5E-07 was used.

Section 6.2 of NUREG-1921 [2-3] acknowledges that the floor value of 1E-05 stated in NUREG-1792 [2-4] is a suggestion and that use of the 1E-05 floor value can introduce skewing of risk metrics and importances as seen in the Significance Determination Process (SDP), which is a "delta" type calculation that is similar to TSTF-505 RICT. Furthermore, the JHEP lower bound application is the same as that used for the Full Power Internal Events (FPIE) PRA, which is recommended in NUREG-1921.

<u>TSTF-505 RICT Assessment</u>

A sensitivity analysis was performed to assess the potential impact on the TSTF-505 RICT calculations. The sensitivity analysis utilized the 1E-05 floor JHEP recommended in NUREG-1921 (instead of the 1E-06 / 5E-07 floor JHEP values used in the base Fire PRA model).

Table APLA-02-B.1 summarizes the results of this sensitivity on the TSTF-505 RICT calculations. The sample Technical Specification (TS) cases were selected because they represent a sample of the cases that would likely change as a result of this sensitivity analysis.

As shown in Table APLA-02-B.1, the delta Fire CDF and Fire LERF values are essentially unchanged from the base Fire PRA model (which uses the 1E-06 / 5E-07 JHEP floor values). Since the delta Fire CDF and Fire LERF values are essentially unchanged, the RICT estimates (in days) are also essentially unchanged.

Therefore, the JHEP floor values used in the Fire PRA do not impact the results of the TSTF-505 RICT calculations and use of the lower JHEP floor values (i.e., 1E-06 / 5E-07) are appropriate for implementation of the TSTF-505 RICT program.

<u>10 CFR 50.69 Assessment</u>

A sensitivity analysis was performed to assess the potential impact on the 10 CFR 50.69 SSC categorizations. The sensitivity analysis utilized the 1E-05 floor JHEP recommended in NUREG-1921 (instead of the 1E-06 / 5E-07 floor JHEP values used in the base Fire PRA model).

The importance measures from this sensitivity case were compared against the results using the base Fire PRA. The screening criteria specified in Section 5.1 of NEI 00-04 [2-5] is summarized below:

- Sum of Fussel-Vesely (F-V) for all basic events modeling the SSC of interest, including common cause > 0.005
- Maximum of component basic event Risk Achievement Worth (RAW) values > 2
- Maximum of applicable common cause basic events RAW values > 20

For the purposes of this sensitivity analysis, since there may be several independent & common cause basic events modeled for a specific SSC, the F-V criterion was lowered to 0.001 to ensure that all potentially risk-significant basic events are evaluated (i.e., lower criterion of 0.001 used to evaluate list of significant basic events, then those basic events were grouped together by SSC to determine if the SSC meets the bulleted criteria from Section 5.1 of NEI 00-004).

Based on the results of this sensitivity analysis, no additional component-related basic events met the screening criteria previously specified. Therefore, the SSC categorizations would not change from the categorizations made using the base Fire PRA model and the JHEP floor values (i.e., 1E-06 / 5E-07) used in the base Fire PRA have no impact on 10 CFR 50.69 categorization results.

Part (b)(ii)

Not applicable – See response to part (b)(i) for details.

TABLE APLA-02-B.1
JHEP FLOOR VALUE RICT CALCULATION SESNTIVITY RESULTS FOR FIRE PRA
(FPIE & SEISMIC RESULTS REMAIN UNCHANGED FOR SENSITIVITY ANALYSIS)

		BASE RESULTS (JHEP FLOOR VALUE < 1E-05)				SENSITIVITY RESULTS (JHEP FLOOR = 1E-05)	
CASE	**DESCRIPTION**	**DELTA FPIE CDF (/YR)**	**DELTA FIRE CDF (/YR)**	**DELTA SEISMIC CDF (/YR)**	**RICT ESTIMATE (DAYS)**	**DELTA FIRE CDF (/YR)**	**RICT ESTIMATE (DAYS)**
T.S. 3.3.5.1.B	Reactor Protection System (RPS) instrumentation - one or more required channels inoperable	2.00E-10	0.00E+00	1.10E-05	30.0	0.00E+00	30.0
T.S. 3.3.5.1.C	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	8.45E-06	1.18E-04	1.10E-05	26.6	1.18E-04	26.6
T.S. 3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	2.09E-06	4.05E-05	1.10E-05	30.0	4.56E-05	30.0
T.S. 3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	2.37E-06	4.05E-05	1.10E-05	30.0	4.56E-05	30.0
T.S. 3.3.5.1.F	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	2.37E-06	4.98E-05	1.10E-05	30.0	5.48E-05	30.0
T.S. 3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable	6.22E-07	3.36E-05	1.10E-05	30.0	3.34E-05	30.0
T.S. 3.5.1.B	High Pressure Core Spray (HPCS) System inoperable.	2.09E-06	4.05E-05	1.10E-05	30.0	4.56E-05	30.0
T.S. 3.5.1.C	Two low pressure ECCS injection/spray subsystems inoperable.	7.87E-06	2.92E-04	1.10E-05	11.7	2.86E-04	12.0

c) Well-Sealed Motor Control Center (MCC) Cabinets

Part (c)(i)

The LaSalle Fire PRA assumes that all MCC cabinets are not well-sealed (i.e., all fires originating from MCC cabinets are assumed to damage external targets). Given that MCC fires are not risk-significant in the Fire PRA, addition refinement using the guidance specified in FAQ 14-0009 is not necessary.

Part (c)(ii)

Not applicable – See part (c)(i) for details.

d) PRA Treatment of Fire Dependencies Between Units 1 and 2

Part (d)(i)

For each fire postulated in the LaSalle Fire PRA, a Fire Initiating Event Decision Tree (FIEDT) was used to determine the most challenging initiating event sequence based on the fire-induced failures associated with the fire scenario (e.g., loss of offsite power, loss of service water, turbine trip, etc.). At a minimum, all fires, regardless of location and target set, are assumed to result in a turbine trip in the analyzed unit (i.e., a fire in Unit 1 is assumed to result in a turbine trip in Unit 2 for the Unit 2 Fire PRA model, and vice versa). The contribution of Unit 1 fires to Unit 2 Fire CDF and Fire LERF is approximately 5% and 3%, respectively.

The fire scenario development methodology used in the Fire PRA requires that all targets within the zone of influence (ZOI), regardless of location within the respected units, be selected and analyzed for fire-induced failure. For example, for fire scenarios postulated in common areas, where targets from both units are within the ZOI, all targets are failed due to fire-induced damage and the impact on the unit's fire risk was evaluated based on the component and cable selection associated with that unit.

REFERENCES:

[2-1] U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research and Electrical Power Research Institute (EPRI), Refining And Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE) Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume, NUREG-2178, Volume 1, DRAFT, March 2015.

[2-2] LaSalle Nuclear Generating Station Fire PRA Peer Review Report Using ASME PRA Standard Requirements, February 2016.

[2-3] U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research and Electrical Power Research Institute (EPRI), Fire Human Reliability Analysis Guidelines, NUREG-1921, July 2012.

[2-4] U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, Good Practice for Implementing Human Reliability Analysis (HRA), NUREG-1792, April 2005.

[2-5] Nuclear Energy Institute, 10 CFR 50.69 SSC Categorization Guideline, NEI 00-04, Rev. 0, July 2005.

DRA/APLA RAI 03 – PRA CONFIGURATION AND CONTROL

[*Applicable for TSTF-505 and 10 CFR 50.69*]

Section 2.3.4 of NEI 06-09, Revision 0-A, specifies that "[c]riteria shall exist in PRA configuration risk management to require PRA model updates concurrent with implementation of facility changes that significantly impact RICT calculations."

TSTF-505 LAR, Enclosure 7, states that if a plant change or a discovered condition is identified and can have significant impact to the RICT program calculations, then an unscheduled update of the PRA models will be implemented. More specifically, the LAR states that if the plant changes meet specific criteria defined in the plant PRA and update procedures, including criteria associated with consideration of the cumulative risk impact, then the change will be incorporated into applicable PRA models without waiting for the next periodic PRA update.

Neither the TSTF-505 or 10 CFR 50.69 LARs explains under what conditions an unscheduled update of the PRA model will be performed or the criteria defined in the plant procedures that will be used to initiate the update. Therefore, describe the conditions under which an unscheduled PRA update (i.e., more than once every two refueling cycles) would be performed and the criteria that would be used to require a PRA update. In the response, describe what is meant by significant impact to the RICT program calculations or SSC categorization results.

EGC RESPONSE:

The EGC Risk Management FPIE & FPRA Model Update procedures require an evaluation of plant changes or discovered conditions (tracked as Updating Requirement Evaluations [UREs]) against an extensive list of criteria including change in CDF/LERF. A Risk Management Engineer will evaluate each URE to determine whether the MOR should be updated expeditiously or the update can be delayed to the next periodic update. This determination will be made based on whether the PRA model fidelity (representation of the as-built, as operated plant) without the update is adequate to support PRA applications that are currently in effect. This is determined either by qualitative screening or Working model updates for potentially significant changes.

Some of the PRA Unscheduled Update Criteria are listed below

- CDF or LERF change >25%
- ΔCDF>1E-5
- ΔLERF>1E-6
- Significant change in accident class or sequence (greater than factor of 2 increase in an accident class that contributes >5% risk)
- Configuration risk increase factors that could breach the color thresholds used in Maintenance Rule a(4).

These evaluations, particularly the check on significant sequences and configuration risk, ensure changes that could significantly impact RICT calculations initiate an unscheduled PRA model update or result in administrative limits on the RICT program per EGC procedures (for example, limiting the use of RICT to LCOs where the impact of the condition is not significant).

DRA/APLA RAI 04 – SYSTEM AND SURROGATE MODELING USED IN THE PRA MODELS

[*Applicable for TSTF-505*]

The NRC safety evaluation (SE) to NEI 06-09, Revision 0, specifies that the LAR should provide a comparison of the TS functions to the PRA modeled functions and that justification be provided to show that the scope of the PRA model is consistent with the licensing basis assumptions. Table E1-1 in Enclosure 1 to the LAR identifies each TS LCO proposed to be included in the RICT program and describes how the systems and components covered in the TS LCO are implicitly or explicitly modeled in the PRA. For some TS LCO conditions, the table explains that the associated SSCs are not modelled in the PRAs but will be conservatively represented using a surrogate event. For certain LCOs it is unclear to the NRC staff whether the modeling (i.e., depth, surrogates used) will be acceptable to address the associated RICTs calculated. Therefore, address the following:

a) For TS LCO 3.3.6.1 (Primary Containment Isolation Instrumentation), Condition A, (Primary containment instrumentation – one or more channels inoperable) in LAR Table E1-1, the logic for primary containment isolation is not modeled in detail and, therefore, a surrogate event will be used. The table states that the surrogate event will either be "failure of containment or failure of the frontline system." It is not clear to NRC staff what system is being referred to by the phrase "frontline system." Confirm/identify the frontline system(s) intended to be referred to in the table (e.g., Primary Containment Isolation valves (PCIVs)) and discuss how the failure of the function(s) compares to failure of containment.

b) For TS LCO 3.6.1.3, PCIV, Condition A (One or more penetration flow paths with one PCIV inoperable for reasons other than Condition D), in LAR Table E1-1, it states "not all PCIVs are modeled, therefore, a surrogate of a pre-existing containment failure is chosen." It is not clear to the NRC staff what pre-existing containment failure will be chosen and how LaSalle will ensure that it represents the equivalent release as an open PCIV. Therefore, discuss the pre-existing containment failure or criteria used to determine which surrogate will be identified for an inoperable PCIV and provide sufficient justification to ensure that the surrogate will reflect the function covered by the TS LCO condition. In the discussion include whether this failure assumes a large early release.

EGC RESPONSE:

a) “Frontline system” in the context of this application was meant to be interpreted as explicitly modeled containment isolation valves. TS LCO 3.3.6.1 Condition A consists of 5 main functions:

   1) Function 1 – Main Steam Line Isolation
   2) Function 2 – Primary Containment Isolation
   3) Function 3 – Reactor Core Isolation Cooling System Isolation
   4) Function 4 – Reactor Water Cleanup System Isolation
   5) Function 5 – RHR Shutdown Cooling System Isolation

   Of these, Function 1 – Main Steam Line Isolation will be mapped to the MSIVs associated with the failed instrumentation. For all other Functions the pre-existing containment failure, discussed in the response (b) below, is used as a surrogate.

b) Two methods are combined in the LaSalle fault tree analysis to quantify the Containment Isolation failure:

   Method 1) Isolation System Failures: Probability of valves being in the open position are calculated using the fault tree and are assumed to lead to a large containment failure, if the valves also fail to close, i.e., failure of containment isolation.

   Method 2) Pre-existing Isolation Failures: The quantitative estimates performed by EPRI in support of the NRC risk assessment programs lead to estimates of large and small containment isolation failures.

   For all penetrations not related to the MSIVs, an event assuming a pre-existing containment failure will be used (Method 2) as a conservative surrogate for the LCO condition. The pre-existing containment failure probability was derived with data by PNL for the NRC (see EPRI Risk Impact Assessment of Extended Integrate Leak Rate Test Intervals, TR-101824) plus the use of NUREG-1493. LaSalle ILRT results indicate that LaSalle is not an outlier and generic industry results are appropriate. The 2.3E-3 industry probability is used to define the operating experience-based failure probability of being larger than “small” failure pathway for containments.

   Isolation failures (IS) for Classes I and III are treated as any failure of containment isolation greater than 2" in diameter. The basis for excluding less than 2-inch penetration lines is NUREG / CR-3539 and 5565. The Level 2 (LERF) models containment isolation failure conservatively for all core damage events as leading directly to a release from the containment drywell. Any radionuclide release that may result is classified as an "early" release relative to core damage; however, if core damage is substantially delayed from accident initiation the release may still be considered as an intermediate or late release, e.g. delayed Station Blackout Sequences.

APLA RAI 05 – PRA MODEL UNCERTAINTY ANALYSIS

[*Applicable for TSTF-505 and 10 CFR 50.69*]

The NRC SE to NEI 06-09, Revision 0, specifies that the LAR should identify key assumptions and key sources of uncertainty and assess/disposition each as to their impact on the risk managed technical specifications (RMTS) application. LAR Enclosure 9, Tables E9-1, E9-2, and E9-3 provided the key assumptions and sources of uncertainty identified for the internal events PRA (IEPRA), transition to the Real Time Risk (RTR) model, and the FPRA and included the dispositions for each source of uncertainty for this TSTF-505 application. Furthermore, as part of its audit of the LAR, the NRC staff reviewed the PRA analyses that support the uncertainty analysis for the LaSalle IEPRA, FPRA, and risk applications cited in the LAR. Upon review of the dispositions provided in LAR Tables E9-1, E9-2 and E9-3, it was unclear to the NRC staff how the licensee concluded there was no impact on the RICT calculations. In light of these observations, address the following:

a) For the TSTF-505 and 10 CFR 50.69 LARs, the licensee identifies cable selection as a key source of FPRA modeling uncertainty because of conservatisms in the approach (i.e., lack of cable data). The LARs discuss that "an informed approach was used in developing the assumed [cable] routing" and that other modeling assumptions provide some offsetting effects; therefore, the licensee concludes that Unknown Location (UNL) modeling uncertainty has no impact on the RICT program calculations and SSC categorization. It is not clear to the NRC staff how this assumption (i.e., cable selection) was concluded to have no impact on the risk-informed applications, given a sensitivity study demonstrated that its impact on risk is "moderate." Therefore, provide a description and results of the selective sample of sensitivities performed to demonstrate that the conservatisms are bounding and do not adversely impact the results of future SSC categorization and future RICT calculations.

b) For the TSTF-505 and 10 CFR 50.69 LARs, the licensee identifies vapor suppression capability following vessel rupture failure as a key source of uncertainty. The TSTF-505 LAR dispositions this uncertainty by stating, "[a]lthough 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." The 10 CFR 50.69 LAR dispositions this uncertainty by stating, "[t]herefore, the uncertainty associated with this model uncertainty is negligible within the 50.69 application." It appears to the NRC staff that the results of the sensitivity study using upper bound values from NUREG/CR-6595 validate the concern that this source of uncertainty impacts the RICT calculations and potentially the SSC categorization results. In light of the staff concerns, address the following:

    i. Given the uncertainty indicated by "recent MAAP [modular accident analysis program]" runs cited in the LAR and the results of the sensitivity study on vapor suppression failure following vessel failure, provide sufficient justification to support the conclusion that the upper bound values used are not realistic and would result in overly-conservative RICT estimates and SSC categorization results.

c) The LaSalle FPRA uncertainty for the TSTF-505 LAR and 10 CFR 50.69 does not discuss the uncertainty associated with modeling the Hardened Containment Vent System (HCVS) in the FPRA, nor does the LAR discuss how this uncertainty will be treated in the RICT program. In light of the NRC staff concerns, address the following:

i. Discuss the uncertainty associated with modelling the HCVS in the FPRA along with justification that the modelling of the HCVS is sufficient for the TSTF-505 and 10 CFR 50.69 applications given the impact that credit for HCVS has on the fire CDF.

ii. If in the response to part (i) above, the modelling of the HCVS in the FPRA cannot be justified for this application given the significant impact that HCVS modeling credit has on the fire CDF, propose a mechanism that addresses this source of modelling uncertainty in the RICT and SSC categorization programs.

d) For the TSTF-505 LAR, Enclosure 9, Table E9-1, identifies credit for survivability of the emergency core cooling system (ECCS) after containment venting as a key source of uncertainty. It is not clear to the NRC staff how this key source of uncertainty was concluded to have negligible impact on RICT calculations and no further treatment (i.e., risk management actions, RMAs) is needed. The NRC staff acknowledges that increasing the conditional failure probability of the ECCS to 1.0 is bounding. Provide the following additional information:

i. Discuss any applicable RMAs for the TS LCOs impacted by this uncertainty (e.g., TS LCO Conditions 3.3.5.1.C, 3.3.5.1.D, 3.5.1.A, and 3.5.1.C). Include justification for why the RMAs are sufficient to address this uncertainty, specifically discuss why additional RMAs or other measures are not needed to reduce the risk of the impacted TS LCOs.

ii. If in response to part (i) above, it cannot be justified that the applicable RMAs are sufficient to address this uncertainty, then include additional RMAs or other measures that will be used to reduce the risk impact of the cited uncertainty for the applicable TS LCOs (e.g., include as an implementation item in Attachment 5 of the LAR).

EGC RESPONSE:

a) Assumed Cable Routing: Unknown Locations (UNLs)

The components assumed to be failed in select Physical Analysis Units (PAUs) analyzed in the Fire PRA due to unknown cable selection are known as Unknown Location (UNL) components. The associated basic events are dispositioned as “Y3” and components within the same unit / system / division are grouped together to create a single surrogate UNL component (e.g., Unit 2 Division 1 CRD).

A rigorous and high-confidence method was applied to identify PAUs in which a fire could impact these components (i.e., a “credit by exclusion” method is used where PAUs that don’t contain cables associated with the unit / system / division of the UNL component would exclude the UNL component from the scenario’s target set). For each UNL component, LaSalle’s cable routing database (i.e., SLICE), plant drawings, and general plant knowledge were used to develop the list of PAUs that contain cables associated with the unit / system / division of the UNL component. For the identified PAUs, all fire scenarios within the PAU with target damage beyond the ignition source would assume the UNL component is failed as part of the scenario's target set.

TSTF-505 RICT Assessment

A sensitivity analysis was performed to assess the potential impact on the TSTF-505 RICT calculations. Given the concern of conservative assumptions masking RICT impacts, the sensitivity analysis consisted of removing all UNL failures from the Fire PRA model such that the delta risk can be maximized if a UNL component is assumed to be unavailable during the RICT configuration (i.e., no fire-induced failure of the components previously using assumed routing).

Table APLA-05-A.1 summarizes the results of the "No UNL" sensitivity on the TSTF-505 RICT calculations for a sample of Technical Specification (TS). These cases were selected because they represent the most likely candidates to change as a result of the sensitivity (e.g., DG auto-start function is included as a UNL component, so T.S. LCOs related to the DGs were examined).

Based on the results presented in Table APLA-05-A.1, the RICT estimates from the base model remain unchanged for all cases analyzed as part of the "No UNL" sensitivity, except for TS 3.5.1.C, which experienced a substantial increase in the RICT estimate. Removal of the UNL components from the Fire PRA model increased the amount of credit given to support functions, which resulted in the lower delta fire risk metrics for many cases (and by extension the longer RICT estimates). However, this sensitivity analysis represents the most optimistic outlook since all UNL components are assumed to be unaffected by fire-induced impacts, which is not a realistic assumption. Therefore, the current UNL (assumed routing) methodology used in the Fire PRA is not overly conservative and the RICT calculations are not adversely impacted.

10 CFR 50.69 Assessment

A sensitivity analysis was performed to assess the potential impact on the 10 CFR 50.69 SSC categorizations. Given the concern of conservative assumptions masking SSC categorizations, the sensitivity analysis consisted of removing all UNL failures from the Fire PRA model such that the SSC importances can be maximized if a UNL component is assumed to be failed solely due to random failures (i.e., no fire-induced failure of the components previously using assumed routing).

The importances measures from this sensitivity case were compared against the results using the base Fire PRA. The screening criteria specified in Section 5.1 of NEI 00-04 [5-1] is summarized below:

- Sum of Fussel-Vesely (F-V) for all basic events modeling the SSC of interest, including common cause > 0.005
- Maximum of component basic event Risk Achievement Worth (RAW) values > 2
- Maximum of applicable common cause basic events RAW values > 20

For the purposes of this sensitivity analysis, since there may be several independent & common cause basic events modeled for a specific SSC, the F-V criterion was lowered to 0.001 to ensure that all potentially risk-significant basic events are evaluated (i.e., lower criterion of 0.001 used to evaluate list of significant basic events, then those basic events were grouped together by SSC to determine if the SSC meets the bulleted criteria from Section 5.1 of NEI 00-004).

Table APLA-05-A.2 discusses the basic events that did not meet the "F-V ≥ 0.001" criterion in the base Fire CDF results, but exceeded that criterion in the "No UNL" sensitivity case (i.e., potential candidate for "safety-significant" classification). Table APLA-05-A.3 presents similar information, but from the perspective of Fire CDF RAW values (rather than F-V values). Tables APLA-05-A.4 and APLA-05-A.5 presents the Fire LERF results from a F-V perspective and RAW perspective.

Based on the results presented in these tables, the SSC categorizations would not change from the categorizations made using the base Fire PRA model. Therefore, the conservatism associated with assumed routing approach used in the base Fire PRA has no impact on 10 CFR 50.69 categorization results.

TABLE APLA-05-A.1
NO UNL RICT CALCULATION SESNTIVITY RESULTS FOR LASALLE FIRE PRA
(FPIE & SEISMIC RESULTS REMAIN UNCHANGED FOR SENSITIVITY ANALYSIS)

		BASE RESULTS				NO UNL SENSITIVITY RESULTS	
CASE	**DESCRIPTION**	**DELTA FPIE CDF (/YR)**	**DELTA FIRE CDF (/YR)**	**DELTA SEISMIC CDF (/YR)**	**RICT ESTIMATE (DAYS)**	**DELTA FIRE CDF (/YR)**	**RICT ESTIMATE (DAYS)**
T.S. 3.3.5.1.B	Reactor Protection System (RPS) instrumentation - one or more required channels inoperable	2.00E-10	0.00E+00	1.10E-05	30.0	0.00E+00	30.0
T.S. 3.3.5.1.C	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	8.45E-06	1.18E-04	1.10E-05	26.6	3.28E-05	30.0
T.S. 3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	2.09E-06	4.05E-05	1.10E-05	30.0	3.15E-05	30.0
T.S. 3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	2.37E-06	4.05E-05	1.10E-05	30.0	3.15E-05	30.0
T.S. 3.3.5.1.F	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	2.37E-06	4.98E-05	1.10E-05	30.0	3.53E-05	30.0
T.S. 3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable	6.22E-07	3.36E-05	1.10E-05	30.0	1.43E-05	30.0
T.S. 3.5.1.B	High Pressure Core Spray (HPCS) System inoperable.	2.09E-06	4.05E-05	1.10E-05	30.0	3.15E-05	30.0
T.S. 3.5.1.C	Two low pressure ECCS injection/spray subsystems inoperable.	7.87E-06	2.92E-04	1.10E-05	11.7	8.69E-05	30.0
T.S. 3.8.1.A	One required offsite circuit inoperable	2.20E-06	8.30E-06	1.10E-05	30.0	1.01E-05	30.0

TABLE APLA-05-A.1
NO UNL RICT CALCULATION SESNTIVITY RESULTS FOR LASALLE FIRE PRA
(FPIE & SEISMIC RESULTS REMAIN UNCHANGED FOR SENSITIVITY ANALYSIS)

		BASE RESULTS				NO UNL SENSITIVITY RESULTS	
CASE	**DESCRIPTION**	**DELTA FPIE CDF (/YR)**	**DELTA FIRE CDF (/YR)**	**DELTA SEISMIC CDF (/YR)**	**RICT ESTIMATE (DAYS)**	**DELTA FIRE CDF (/YR)**	**RICT ESTIMATE (DAYS)**
T.S. 3.8.1.B	One required Division 1 or 2 DG inoperable OR required opposite unit Division 2 DG inoperable	5.08E-06	9.37E-06	1.10E-05	30.0	5.95E-06	30.0
T.S. 3.8.1.D	Two required offsite circuits inoperable	2.20E-06	8.31E-06	1.10E-05	30.0	1.06E-05	30.0
T.S. 3.8.1.E[(1)]	One required offsite circuit inoperable AND one required Division 1, 2, or 3 DG inoperable	1.37E-05[(1)]	1.69E-05[(1)]	2.20E-06[(1)]	11.1[(1)]	1.69E-05[(1)]	11.1[(1)]

<u>Note to Table APLA-05-A.1:</u>

[(1)] For T.S. 3.8.1.E, LERF is the limiting metric (rather than CDF, which is the case for all other T.S. cases listed in this table).

TABLE APLA-05-A.2
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON "NO UNL" SENSITIVITY ANALYSIS
(FIRE CDF: FUSSELL-VESELY RESULTS)

BASIC EVENT	DESC	BASE F-V	F-V SIG?	NO UNL F-V	F-V SIG?	DISCUSSION
2CNFLMLLOCA--PCC	CCF (PLUGGING) OF ECCS SUCT STRAINERS (LOCA)	6.10E-04	No	1.10E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with ECCS Suction strainers satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The "No UNL" sensitivity does not change this characterization.
2DGHB-2413---K--	4.16 kVAC CB 2AP04E-10 (2413) DG0 TO SWGR 241Y FAILS TO CLOSE	6.80E-04	No	1.06E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Bus 2AP04E satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The "No UNL" sensitivity does not change this characterization.
2FWAV2FW005--M--	FW MDRFP 2FW01PC FEED REG AOV 2FW005 MUA	5.30E-04	No	1.76E-03	Yes	**No Change in SSC Categorization** Cumulative F-V values for MDRFP AOV 2FW005 does not meet the "F-V ≥ 0.005" criterion in the base Fire PRA model or in the "No UNL" sensitivity.
2VDDMDG2V01YBD--	VD DG2B ROOM VENT BAL DAMPER 2VD01YA FAILS TO OPEN ON DEMAND	9.20E-04	No	1.62E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generator 2B Room Ventilation equipment satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The "No UNL" sensitivity does not change this characterization.

TABLE APLA-05-A.2
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON "NO UNL" SENSITIVITY ANALYSIS
(FIRE CDF: FUSSELL-VESELY RESULTS)

BASIC EVENT	DESC	BASE F-V	F-V SIG?	NO UNL F-V	F-V SIG?	DISCUSSION
2VDDMDG2V03YBD--	VD DG2B ROOM VENT BAL DAMPER 2VD03YA FAILS TO OPEN ON DEMAND	9.20E-04	No	1.62E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generator 2B Room Ventilation equipment satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The "No UNL" sensitivity does not change this characterization.
BDCBS125-ALL-FCC	CCF OF 125 VDC UNIT 2 DIV 1 & 2 & 3 AND UNIT 1 DIV 1 & 2	8.50E-04	No	1.56E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with all divisions of 125V DC Power satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The "No UNL" sensitivity does not change this characterization.
BDGDG-0-2A--FXCC	CCFTR OF DIESEL GENERATORS DG0 & DG2A - FIRE 24 HOURS	7.80E-04	No	1.28E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The "No UNL" sensitivity does not change this characterization.
BDGDG-0-2B--FXCC	CCFTR OF DIESEL GENERATORS DG0 & DG2B - FIRE 24 HOURS	9.60E-04	No	1.55E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The "No UNL" sensitivity does not change this characterization.

TABLE APLA-05-A.2
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON “NO UNL” SENSITIVITY ANALYSIS
(FIRE CDF: FUSSELL-VESELY RESULTS)

BASIC EVENT	DESC	BASE F-V	F-V SIG?	NO UNL F-V	F-V SIG?	DISCUSSION
BDGDG1A-0-2BFXCC	CCFTR OF DIESEL GENERATORS DG1A & DG0 & DG2B - FIRE 24 HOURS	7.90E-04	No	1.12E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The “No UNL” sensitivity does not change this characterization.
BDGDG1A2A2B-FXCC	CCFTR OF DIESEL GENERATORS DG1A & DG2A & DG2B - FIRE 24 HOURS	7.30E-04	No	1.01E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The “No UNL” sensitivity does not change this characterization.
BDGDG-2A-2B-FXCC	CCFTR OF DIESEL GENERATORS 2A & 2B - FIRE 24 HOURS	8.00E-04	No	1.31E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The “No UNL” sensitivity does not change this characterization.

TABLE APLA-05-A.3
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON "NO UNL" SENSITIVITY ANALYSIS (FIRE CDF: RISK ACHIEVEMENT WORTH RESULTS)

BASIC EVENT	DESC	BASE RAW	RAW SIG?	NO UNL RAW	RAW SIG?	DISCUSSION
2ACHB-2412---D--	4.16 kVAC CB 2AP04E-12 (2412) SAT 242 TO SWGR 241Y FAILS TO OPEN	1.78E+00	No	2.21E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Bus 2AP04E satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
2ACHB-2422---D--	4.16 kVAC CB 2AP06E-13 (2422) SAT 242 TO SWGR 242Y FAILS TO OPEN	1.68E+00	No	2.06E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Bus 2AP06E satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
2ACHB-2425---D--	4.16 kVAC CB 2AP06E-4 (2425) SWGR 242Y / 242X X-TIE FAILS TO OPEN	1.68E+00	No	2.06E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Bus 2AP06E satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.

TABLE APLA-05-A.3
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON "NO UNL" SENSITIVITY ANALYSIS (FIRE CDF: RISK ACHIEVEMENT WORTH RESULTS)

BASIC EVENT	DESC	BASE RAW	RAW SIG?	NO UNL RAW	RAW SIG?	DISCUSSION
2ACHB2AP04E2-D--	4.16 kVAC CB 2AP04E-2 SWGR 241Y TO 480 VAC SWGR 233 FAILS TO OPEN	1.78E+00	No	2.21E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Bus 2AP04E satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
2ACHB2AP06E3-D--	4.16 kVAC CB 2AP06E-3 SWGR 242Y TO 480 VAC SWGR 234X / 234Y FAILS TO OPEN	1.68E+00	No	2.06E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Bus 2AP06E satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
2DGDG-DG2A---M--	DG2A DIESEL GENERATOR 2DG01K MUA	1.87E+00	No	2.28E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfy the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.

TABLE APLA-05-A.3
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON “NO UNL” SENSITIVITY ANALYSIS
(FIRE CDF: RISK ACHIEVEMENT WORTH RESULTS)

BASIC EVENT	DESC	BASE RAW	RAW SIG?	NO UNL RAW	RAW SIG?	DISCUSSION
2DGHB-2413---K--	4.16 kVAC CB 2AP04E-10 (2413) DG0 TO SWGR 241Y FAILS TO CLOSE	1.91E+00	No	2.42E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Bus 2AP04E satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The increase in RAW due to the “No UNL” sensitivity does not change this characterization.
2DGHB-2423---K--	4.16 kVAC CB 2AP06E-11 (2423) DG2A TO SWGR 242Y FAILS TO CLOSE	1.79E+00	No	2.25E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Bus 2AP06E satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The increase in RAW due to the “No UNL” sensitivity does not change this characterization.
2RHXVLCIF92AXV--	RH LPCI A HEADER ISOL MANUAL VALVE 2E12-F092A SPUR CLOSES	1.66E+00	No	2.22E+00	Yes	**No Change in SSC Categorization** Although the “RAW ≥ 2” criterion is exceeded in the “No UNL” sensitivity (and not exceeded in the base Fire PRA model), this RHR Manual Valve would likely be selected as risk-significant by its association with the RHR system. Therefore, the “No UNL” sensitivity likely won’t change this characterization.

TABLE APLA-05-A.3
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON “NO UNL” SENSITIVITY ANALYSIS (FIRE CDF: RISK ACHIEVEMENT WORTH RESULTS)

BASIC EVENT	DESC	BASE RAW	RAW SIG?	NO UNL RAW	RAW SIG?	DISCUSSION
2VDDG2ATRN--M--	VD DG2A VENTILATION TRAIN MUA	1.91E+00	No	2.45E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generator 2A Room Ventilation equipment satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The increase in RAW due to the “No UNL” sensitivity does not change this characterization.
2VDFNCS2VD03CA--	VD DG2A ROOM COOLNG FAN 2VD03C FAILS TO START	1.90E+00	No	2.43E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generator 2A Room Ventilation equipment satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The increase in RAW due to the “No UNL” sensitivity does not change this characterization.
2VDFNCS2VD03CX-F	VD DG2A ROOM COOLNG FAN 2VD03C FAILS TO RUN - FIRE 24 HOURS	1.89E+00	No	2.41E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generator 2A Room Ventilation equipment satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The increase in RAW due to the “No UNL” sensitivity does not change this characterization.

TABLE APLA-05-A.3
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON "NO UNL" SENSITIVITY ANALYSIS (FIRE CDF: RISK ACHIEVEMENT WORTH RESULTS)

BASIC EVENT	DESC	BASE RAW	RAW SIG?	NO UNL RAW	RAW SIG?	DISCUSSION
BDGDG-0-2A---ACC	CCFTS OF DIESEL GENERATORS DG0 & DG2A	1.43E+01	No	2.37E+01	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
BDGDG-0-2A--FXCC	CCFTR OF DIESEL GENERATORS DG0 & DG2A - FIRE 24 HOURS	1.62E+01	No	2.62E+01	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
BDGDG-0-2B---ACC	CCFTS OF DIESEL GENERATORS DG0 & DG2B	1.77E+01	No	2.82E+01	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.

TABLE APLA-05-A.3
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON “NO UNL” SENSITIVITY ANALYSIS (FIRE CDF: RISK ACHIEVEMENT WORTH RESULTS)

BASIC EVENT	DESC	BASE RAW	RAW SIG?	NO UNL RAW	RAW SIG?	DISCUSSION
BDGDG-0-2B--FXCC	CCFTR OF DIESEL GENERATORS DG0 & DG2B - FIRE 24 HOURS	1.98E+01	No	3.10E+01	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The increase in RAW due to the “No UNL” sensitivity does not change this characterization.
BDGDG1A-0-2A-ACC	CCFTS OF DIESEL GENERATORS 1A & 0 & 2A	1.42E+01	No	2.21E+01	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The increase in RAW due to the “No UNL” sensitivity does not change this characterization.
BDGDG1A-0-2AFXCC	CCFTR OF DIESEL GENERATORS DG1A & DG0 & DG2A - FIRE 24 HOURS	1.77E+01	No	2.74E+01	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the “F-V ≥ 0.005” criterion in the base Fire PRA model. The increase in RAW due to the “No UNL” sensitivity does not change this characterization.

TABLE APLA-05-A.3
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON "NO UNL" SENSITIVITY ANALYSIS
(FIRE CDF: RISK ACHIEVEMENT WORTH RESULTS)

BASIC EVENT	DESC	BASE RAW	RAW SIG?	NO UNL RAW	RAW SIG?	DISCUSSION
BDGDG1A-0-2B-ACC	CCFTS OF DIESEL GENERATORS DG1A & DG0 & DG2B	1.81E+01	No	2.63E+01	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
BDGDG1A-2A-2BACC	CCFTS OF DIESEL GENERATORS DG1A & DG2A & DG2B	1.68E+01	No	2.37E+01	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
BDGDG-2A-2B--ACC	CCFTS OF DIESEL GENERATORS 2A & 2B	1.48E+01	No	2.42E+01	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.

TABLE APLA-05-A.3
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON "NO UNL" SENSITIVITY ANALYSIS (FIRE CDF: RISK ACHIEVEMENT WORTH RESULTS)

BASIC EVENT	DESC	BASE RAW	RAW SIG?	NO UNL RAW	RAW SIG?	DISCUSSION
BDGDG-2A-2B-FXCC	CCFTR OF DIESEL GENERATORS 2A & 2B - FIRE 24 HOURS	1.65E+01	No	2.65E+01	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generators satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
BDGFL-0-1A-2BPCC	CCF (PLUGGING) OF DG COOLING PUMP STRAINERS 0A & 1A & 2B	1.33E+01	No	2.35E+01	Yes	**No Change in SSC Categorization** Although the "No UNL" sensitivity resulted in common cause failure (CCF) of the DG Cooling Water pump strainers exceeding the "RAW ≥ 20" criterion, the required sensitivity of using the 95th percentile CCF would likely cause the RAW value to exceed the criterion. Therefore, the "No UNL" sensitivity likely won't change this characterization.
BDGFL-0A-2B--PCC	CCF (PLUGGING) OF DGCWP STRAINERS 0A & 2B	1.54E+01	No	2.35E+01	Yes	**No Change in SSC Categorization** Although the "No UNL" sensitivity resulted in common cause failure (CCF) of the DG Cooling Water pump strainers exceeding the "RAW ≥ 20" criterion, the required sensitivity of using the 95th percentile CCF would likely cause the RAW value to exceed the criterion. Therefore, the "No UNL" sensitivity likely won't change this characterization.

TABLE APLA-05-A.3
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON "NO UNL" SENSITIVITY ANALYSIS (FIRE CDF: RISK ACHIEVEMENT WORTH RESULTS)

BASIC EVENT	DESC	BASE RAW	RAW SIG?	NO UNL RAW	RAW SIG?	DISCUSSION
BVDDMDG0V01YAD--	VD DG0 ROOM VENT BAL DAMPER 0VD01YA FAILS TO OPEN	1.98E+00	No	2.20E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generator 0 Room Ventilation equipment satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
BVDDMDG0V03YAD--	VD DG0 ROOM VENT OUTLET BAL DAMPER 0VD03YA FAILS TO OPEN	1.98E+00	No	2.20E+00	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generator 0 Room Ventilation equipment satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.

TABLE APLA-05-A.4
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON "NO UNL" SENSITIVITY ANALYSIS
(FIRE LERF: FUSSELL-VESELY RESULTS)

BASIC EVENT	DESC	BASE F-V	F-V SIG?	NO UNL F-V	F-V SIG?	DISCUSSION
2VDDMDG2V01YBD--	VD DG2B ROOM VENT BAL DAMPER 2VD01YA FAILS TO OPEN ON DEMAND	9.50E-04	No	1.15E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generator 2B Room Ventilation equipment satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.
2VDDMDG2V03YBD--	VD DG2B ROOM VENT BAL DAMPER 2VD03YA FAILS TO OPEN ON DEMAND	9.50E-04	No	1.15E-03	Yes	**No Change in SSC Categorization** The cumulative impact of all basic events associated with Diesel Generator 2B Room Ventilation equipment satisfies the "F-V ≥ 0.005" criterion in the base Fire PRA model. The increase in RAW due to the "No UNL" sensitivity does not change this characterization.

TABLE APLA-05-A.5
DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON "NO UNL" SENSITIVITY ANALYSIS
(FIRE LERF: RISK ACHIEVEMENT WORTH RESULTS)

BASIC EVENT	DESC	BASE RAW	RAW SIG?	NO UNL RAW	RAW SIG?	DISCUSSION
2DCBY1EBAT2CXF--	125 VDC DIV 3 BATTERY 2DC18E FAILS	1.94E+00	No	2.15E+00	Yes	**No Change in SSC Categorization** The Division 3 Battery 2DC18E would be classified as risk-significant based on Fire CDF results. Therefore, the "No UNL" sensitivity does not change this characterization.
2DCMB-DIV3---U--	480 VAC CB 2AP79E-2B MCC 243-1 TO BATT CHRG 2DC19E SPUR OPENS	1.96E+00	No	2.17E+00	Yes	**No Change in SSC Categorization** Bus 2AP79E would be classified as risk-significant based on Fire CDF results. Therefore, the "No UNL" sensitivity does not change this characterization.
2ACBS-243C---F--	4.16 kVAC SWGR 243C (2AP07E) FAILS	1.96E+00	No	2.17E+00	Yes	**No Change in SSC Categorization** Bus 2AP07E would be classified as risk-significant based on Fire CDF results. Therefore, the "No UNL" sensitivity does not change this characterization.
2ACBS-2431---F--	480 VAC MCC 243-1 (2AP79E) FAILS	1.96E+00	No	2.17E+00	Yes	**No Change in SSC Categorization** Bus 2AP79E would be classified as risk-significant based on Fire CDF results. Therefore, the "No UNL" sensitivity does not change this characterization.

b) Vapor Suppression Capability Following Vessel Rupture

As discussed in Section 8.3 of LS-MISC-046 [5-2], the recommended upper bound values from NUREG/CR-6595 for Mark II Containments are evaluated as an alternate hypothesis (i.e., sensitivity analysis uses upper bound values of 0.2 for RPV low pressure scenarios and 0.3 for RPV high pressure scenarios).

This is a very conservative representation of the failure of the vapor suppression capability as the upper bound values in NUREG/CR-6595 [5-3] are nodal values for the simplified event tree that represents the conditional probability of early containment failure due to phenomenological issues. In the LaSalle Level 2 PRA [5-4], the “Containment Intact” or CZ node evaluates the probability of early, energetic containment failures. The CZ node is used to identify those potential containment failure modes that have the following characteristics:

- They are generally energetic in nature.
- They result in potentially large drywell failures.
- They occur at or near the time of core melt progression or RPV breach.

These containment failure modes provide fission product pathway characteristics that result in the potential for substantial radionuclide releases to the Reactor Building. The failure modes that can lead to the CZ node are generally of low conditional failure probability but are considered in the Level 2 PRA for completeness. The failure at CZ node implies an energetic containment failure of the drywell. It is modeled as a large drywell failure leading to a high radionuclide release. The following failure modes are considered in the CZ node:

In-Vessel Interactions:

- H2 Production
- Steam Explosion
- Recriticality
- Bottom Head Failure Mode

RPV Breach by Debris:

- Penetration size
- Direct Containment Heating (DCH)
- RPV blowdown
- Debris Temperature

Ex-vessel Interactions:

- Missile generation
- Ex-vessel steam explosion
- Core concrete interaction
- Hydrogen burn
- Vapor suppression failure

The nodal values used in the sensitivity analysis account for several of these failure modes, not just the vapor suppression failure. For comparison, the CZ nodal value for RPV low pressure case in the LaSalle PRA is 9.9E-03 and that value is consistent with the nodal value of 1E-02 given in NUREG/CR-6595. For the high-pressure case, the LaSalle CZ nodal value is 1.2E-2. The value used for the sensitivity for NUREG/CR-6595 was 3.0E-01, which is more than an order of magnitude higher than the LaSalle nodal probability and many orders of magnitude higher than the vapor suppression failure probability.

The LaSalle Mark II design has the following design features which affect the core melt progression:

1. The drywell floor below the RPV is sunken relative to the drywell floor outside the pedestal. This creates a cavity for collection of molten debris.
2. The drywell floor within the pedestal has floor and equipment drain plates which are susceptible to failure when exposed to molten debris.
3. Once the drywell pedestal floor drains have failed there is a suppression pool bypass mechanism because the pedestal air space in the wetwell communicates with the wetwell airspace outside the pedestal.
4. The pathway for molten debris to below the drywell pedestal floor is to the wetwell pedestal floor which is also “dry,” i.e., not connected to the suppression pool water volume.

Generally, MAAP is used in the Level 2 analysis to determine release timing and magnitude. However, for failures in the CZ node, a large early release is assumed which could be a conservative assumption.

Given LaSalle’s design features previously discussed, the molten debris attack of the drywell pedestal drain plate is estimated to take:

- 30 seconds for blowdown of sufficient material to challenge the drywell shell
- 11 seconds for transport to the shell
- 7 minutes for melting of a 0.5-inch thick plate

Total time from RPV breach until radionuclide bypass of the suppression pool is therefore estimated at 7.67 minutes. This information is used in the MAAP runs supporting this Level 2/LERF evaluation. The MAAP model of this suppression pool bypass mode due to debris attack results in significantly slower consequential containment pressurization than other postulated RPV induced failure modes which lead to early energetic containment failure. Therefore, the MAAP model does include representation of a likely suppression pool bypass after vessel failure but this does not result in an energetic containment failure for the LaSalle containment design.

In conclusion, the upper bound values used in the sensitivity calculation are judged too conservative to use in TSTF-505 RICT calculations and use of the base PRA for the following reasons:

1. The upper bound values represent other failure modes besides the failure of vapor suppression

2. The end-states for the CZ nodes are already conservative in that they assume a large early release
3. The LaSalle design results in failure of the drywell pedestal drain plate as a more likely scenario leading to pool bypass but that failure mode does not represent an immediate containment challenge.
4. Failure of vapor suppression is a phenomenologically-driven failure mode, so changes in failure probability would not change the associated RMAs. Therefore, no new RMAs would be identified.

c) Hardened Containment Vent System (HCVS)

Part (c)(i)

A sensitivity analysis associated with the Hardened Containment Vent System (HCVS) was performed for the base Fire PRA because that model change reflected a significant change in the fire risk (i.e., changes in significant accident sequences and fire initiators) that warranted further evaluation for understanding the new fire risk profile. This sensitivity analysis was not identified as part of a review of potential sources of uncertainty (i.e., no qualitative uncertainties associated with the HCVS modeling in the Fire PRA).

A focused-scope peer review (FSPR) was performed in October 2017 [5-5] for the Fire PRA model crediting the HCVS due to the significant improvement in fire risk metrics and the changes in risk-significant accident sequences in the Fire PRA model. For the FSPR, PRA Standard Technical Elements FSS-C, D, H and FQ-A, B, C, D, E, F (and associated back-referenced Supporting Requirements from the Internal Events PRA Standard) were reviewed and only one Supporting Requirement (SR) was assessed as “not met” (i.e., SR QU-B3 as referenced by FQ-B1). This SR was assessed as “not met” because the chosen truncation level for the fire risk metrics did not clearly demonstrate convergence. The Findings & Observations (F&Os) associated with this SR were subsequently resolved and assessed as adequate by an independent review team as part of the September 2019 F&O Closure [5-6].

In addition, the quantitative uncertainty associated with the HCVS was assessed as part of the parametric uncertainty analysis for the Fire PRA. See the response to APLA RAI 06 for further information regarding the potential impact of the parametric uncertainty on the risk-informed applications.

Part (c)(ii)

Not applicable – See response to part (c)(i) for details.

d) ECCS Survivability Post Containment Venting

Section 8.1 of LS-MISC-046 [5-2] summarizes the sensitivity analysis performed for the source of uncertainty associated with survivability of ECCS after containment venting. As discussed in Section 8.1 of LS-MISC-046, the alternate hypothesis of guaranteed ECCS equipment failure due to steam binding (i.e., failure probability of 1.0) shows that some of the RICT calculations are sensitive to the alternate hypothesis.

However, for each Tech Spec identified with a change in the RICT estimate, the importance measures of the base and sensitivity results were reviewed to determine if the top operator actions (and by extension the Risk Management Actions (RMAs) proposed for the various configurations) would change as a result of this bounding sensitivity analysis. For the identified configurations analyzed as part of this sensitivity, operator actions related to containment venting remain to be the top risk-significant operator actions in the Fire PRA model, which is the driving force behind the change in the RICT estimates.

Given that the top operator actions haven't changed, the proposed RMAs for this configuration would also remain unchanged. 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 ECCS survivability post containment venting failure is negligible to the RICT application.

REFERENCES:

[5-1] Nuclear Energy Institute, 10 CFR 50.69 SSC Categorization Guideline, NEI 00-04, Rev. 0, July 2005.

[5-2] LaSalle County Generating Station, Assessment of Key Assumptions and Sources of Uncertainty for Risk-Informed Applications, LS-MISC-046, Rev. 1, January 2020.

[5-3] U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events, NUREG/CR-6595, Rev. 1, October 2004.

[5-4] LaSalle County Generating Station Probabilistic Risk Assessment, LaSalle Level 2 / LERF Notebook, LS-PSA-015, Rev. 6, November 2015.

[5-5] LaSalle County Generating Station, PRA Fact and Observation Independent Assessment & Focused-Scope Peer Review, Report # 032299RPT-09, Revision 0, March 2019.

[5-6] 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.

APLA RAI 06 – EVALUATING SOKC UNCERTAINTY IMPACT ON THE RISK APPLICATIONS [*Applicable for TSTF-505 and 10 CFR 50.69*]

RG 1.174 clarifies that, because of the way the acceptance guidelines in RG 1.174 have been developed, the appropriate numerical measures to use when comparing the PRA results with the risk acceptance guidelines are mean values. The risk management threshold values for the RICT program have been developed based on RG 1.174 and, therefore, the most appropriate measures with which to make a comparison are also mean values.

Point estimates are the most commonly calculated and reported PRA results. Point estimates do not account for the state-of-knowledge correlation (SOKC) between nominally independent basic event probabilities, but they can be quickly calculated. Mean values reflect the SOKC and are always larger than point estimates but require longer and more complex calculations. NUREG-1855, Revision 1, provides guidance on evaluating how the uncertainty arising from the propagation of the uncertainty in parameter values SOKC of the PRA inputs impacts the comparison of the PRA results with the guideline values. In light of these observations, address the following:

a) Provide a summary of how the SOKC investigation was performed for the base LaSalle PRA models used to support the risk-informed applications (i.e., TSTF-505 and 10 CFR 50.69).
b) Provide a summary of how the SOKC will be addressed for the risk-informed applications (i.e., based upon the risk metrics to be considered), and explain how this process/approach is consistent with NUREG-1855, Revision 1.

EGC RESPONSE:

a) Parametric Uncertainty Results in Base PRA Models

Full Power Internal Events (FPIE)

The parametric uncertainty evaluation for the FPIE PRA model is documented in Appendix A of the Summary Notebook [6-1]. Using the UNCERT software, a Monte Carlo simulation was performed for both CDF and LERF using 50,000 samples to calculate the mean risk metrics that reflect state-of-knowledge correlation (SOKC) considerations. Table APLA-06-A.1 below summarizes the results of the parametric uncertainty evaluation performed for the base FPIE PRA model.

TABLE APLA-06-A.1
PARAMETRIC UNCERTAINTY RESULTS (FPIE PRA - UNIT 2)

METRIC	POINT-ESTIMATE (/YR)	PARAMETRIC MEAN (/YR)	DELTA (/YR)	%INCREASE
CDF	2.18E-06	2.19E-06	1.00E-08	0.46%
LERF	1.34E-07	1.35E-07	1.00E-09	0.75%

Based on the results presented in Table APLA-06-A.1, the difference between the point-estimate and parametric mean values for FPIE CDF and LERF is nearly negligible (< 1% increase). Therefore, it is concluded that the point-estimate values are good representations of the mean FPIE CDF and LERF values.

Fire PRA

The parametric uncertainty evaluation for the Fire PRA model is documented in Section 4.1 and Appendix A of the Uncertainty & Sensitivity Analysis Notebook [6-2]. The Fire PRA parametric uncertainty analysis evaluated the following fire-specific parameters (in addition to the uncertainty parameters from the FPIE analysis):

- Fire Ignition Frequencies – Uses NUREG-2169 uncertainty distributions
- Non-Suppression Probabilities – Uses NUREG/CR-1278 uncertainty distributions
- Severity Factors – Uses generic FPIE lognormal uncertainty distributions
- Spurious Probabilities – Uses NUREG/CR-7150 uncertainty distributions
- Fire Human Error Probabilities – Uses EPRI HRA Calculator uncertainty distributions

Using the UNCERT software, a Monte Carlo simulation was performed for both Fire CDF and Fire LERF using 50,000 samples to calculate the mean risk metrics that reflect state-of-knowledge correlation (SOKC) considerations. Table APLA-06-A.2 below summarizes the results of the parametric uncertainty evaluation performed for the base Fire PRA model.

TABLE APLA-06-A.2
PARAMETRIC UNCERTAINTY RESULTS (FIRE PRA - UNIT 2)

METRIC	POINT-ESTIMATE (/YR)	PARAMETRIC MEAN (/YR)	DELTA (/YR)	%INCREASE
CDF	7.77E-06	7.82E-06	5.00E-08	0.64%
LERF	3.15E-07	3.21E-07	6.00E-09	1.90%

Based on the results presented in Table APLA-06-A.2, the difference between the point-estimate and parametric mean values for Fire CDF is nearly negligible (< 1% increase) and for Fire LERF is small (< 2% increase). Therefore, it is concluded that the point-estimate values are good representations of the mean Fire CDF and Fire LERF values.

b) TSTF-505 RICT Assessment

Since the TSTF-505 RICT program is a “delta” type application, where acceptability is based on the difference between a base model and the configuration-specific model with equipment unavailable, the potential increase in CDF and LERF using the parametric mean values (which accounts for SOKC uncertainties) would be reflected in both the base PRA model results and the configuration-specific PRA model results.

As demonstrated by the results presented in Tables APLA-06-A.1 and APLA-06-A.2, the parametric means are less than 2% greater than the point-estimate values for FPIE and Fire PRA CDF and LERF. Therefore, if the parametric mean values were used, the delta risk would be expected to increase marginally, which would be essentially negligible to the RICT calculations.

A sensitivity analysis was performed for a select group of Technical Specifications (TS) to determine the potential impact of using parametric mean values (instead of point-estimate values) on the RICT calculations. For the cases specified in Table APLA-06-B.1, CDF was identified as the limiting metric for calculation of the RICT estimate (in days), so only CDF results are presented in Table APLA-06-B.1. A similar sensitivity analysis for those cases with LERF as the limiting metric would be expected to result in similar findings.

For both the FPIE PRA model and Fire PRA model, Monte Carlo simulations were performed for the various cases (using 50,000 samples) in order to calculate the parametric mean values. The parametric mean values were then used to calculate the RICT estimate (in days) for the specific configuration (note: seismic penalties of 1.1E-05/yr (CDF) and 2.2E-06/yr (LERF) remain unchanged for the RICT calculations).

Table APLA-06-B.1 summarizes the results of this sensitivity analysis. The columns to Table APLA-06-B.1 are as follows:

- Case
- Description – Brief description of the Technical Specification
- FPIE Results & Fire Results (each hazard has the following columns)
    - Point-Estimate CDF
    - Delta Point-Estimate CDF compared to the Base PRA model
    - Mean CDF (accounts for SOKC uncertainties)
    - Delta Mean CDF compared to the Base PRA model
- RICT Estimates
    - RICT Estimate using point-estimate values
    - RICT Estimate using mean values
    - %Change

As shown in Table APLA-06-B.1, of the six cases analyzed, only one Technical Specification resulted in a different RICT estimate using the mean value (i.e., the RICT estimate for T.S. T.S. 3.3.5.1.C decreased by ~5% (or ~1.25 days) using the mean value). Additional sensitivities for different Technical Specifications are expected to produce similar results as those shown in Table APLA-06-B.1.

While the point-estimate risk metric values used in the base PRA models do not account for state-of-knowledge correlation (SOKC), based on the results presented in Table APLA-

06-B.1, the SOKC uncertainties are assessed to have a minimal impact on the RICT calculations and use of the point-estimate values is adequate for implementation of the TSTF-505 RICT program.

10 CFR 50.69 Assessment

As shown in Tables APLA-06-A.1 and APLA-06-A.2, the FPIE PRA and Fire PRA parametric mean values for CDF and LERF are essentially unchanged when compared to the point-estimate values. Given that the risk metrics haven't changed significantly, the resulting importance measures would likely remain unchanged. Therefore, the SSC categorizations would likely remain unchanged if the parametric mean values were used.

TABLE APLA-06-B.1
SENSITIVITY ANALYSIS RESULTS FOR POINT-ESTIMATE VS. MEAN VALUES ON RICT CALCULATIONS

CASE INFORMATION		FPIE					FIRE					RICT ESTIMATES		
CASE	DESCRIPTION	POINT-ESTIMATE CDF (/YR)	DELTA TO BASE CDF (/YR)	MEAN CDF (/YR)	DELTA TO BASE CDF (/YR)	%INCREASE (POINT-ESTIMATE VS. MEAN)	POINT-ESTIMATE CDF (/YR)	DELTA TO BASE CDF (/YR)	MEAN CDF (/YR)	DELTA TO BASE CDF (/YR)	%INCREASE (POINT-ESTIMATE VS. MEAN)	RICT ESTIMATE (POINT-ESTIMATE) (DAYS)	RICT ESTIMATE (MEAN) (DAYS)	%CHANGE
Base	Base PRA Model	1.28E-06	-	1.29E-06	-	0.47%	5.15E-06	-	5.17E-06	-	0.35%	-	-	-
T.S. 3.3.5.1.B	Reactor Protection System (RPS) instrumentation - one or more required channels inoperable	1.28E-06	0.00E+00	1.29E-06	0.00E+00	0.47%	5.15E-06	0.00E+00	5.17E-06	4.00E-09	0.43%	30.00	30.00	0.00%
T.S. 3.3.5.1.C	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	9.71E-06	8.43E-06	9.80E-06	8.52E-06	1.02%	1.21E-04	1.16E-04	1.21E-04	1.16E-04	0.00%	27.04	27.03	-0.06%
T.S. 3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	3.37E-06	2.09E-06	3.40E-06	2.11E-06	0.77%	4.57E-05	4.05E-05	4.58E-05	4.06E-05	0.28%	30.00	30.00	0.00%
T.S. 3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	3.65E-06	2.37E-06	3.67E-06	2.38E-06	0.60%	4.57E-05	4.05E-05	4.58E-05	4.06E-05	0.35%	30.00	30.00	0.00%
T.S. 3.3.5.1.F	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	3.65E-06	2.37E-06	3.69E-06	2.40E-06	1.10%	5.48E-05	4.97E-05	5.50E-05	4.99E-05	0.38%	30.00	30.00	0.00%
T.S. 3.5.1.B	High Pressure Core Spray (HPCS) System inoperable.	3.37E-06	2.09E-06	3.39E-06	2.10E-06	0.56%	4.57E-05	4.05E-05	4.58E-05	4.07E-05	0.39%	30.00	30.00	0.00%

REFERENCES:

[6-1] LaSalle County Generating Station Probabilistic Risk Analysis, Summary Notebook, LS-PSA-013, Rev. 8, November 2015.

[6-2] LaSalle County Generating Station Fire PRA, Uncertainty and Sensitivity Notebook, LS-PSA-021.12, Rev. 3, December 2019.

<u>DRA/APLA RAI 07: Key Principle 5 – Performance Monitoring</u>
<u>[Applicable for TSTF-505 and 10 CFR 50.69]</u>

For the TSTF-505 LAR, Section 2.3 of LAR Attachment 1 states that the application of a RICT will be evaluated using the guidance provided in NEI 06-09, Revision 0-A, which was approved by the NRC on May 17, 2007 (ADAMS Accession No. ML071200238). The NRC SE for NEI 06-09, Revision 0-A, states, “[t]he impact of the proposed change should be monitored using performance measurement strategies.” Furthermore, for the adoption of 10 CFR 50.69 using NEI 00-04, the guidance discusses the use of 10 CFR 50.65, the Maintenance Rule, as a way to monitor RISC-1 and RISC-2 SSCs with the clarifications listed in Section 12 of NEI 00-04. Both NEI 00-04 and NEI 06-09 consider the use of NUMARC 93-01, Revision F (ADAMS Accession No. ML18120A069), as endorsed by RG 1.160, Revision 4 (ADAMS Accession No. ML18220B281), for the implementation of the Maintenance Rule. NUMARC 93-01, Section 9.0, contains guidance for the establishment of performance criteria.

Furthermore, Section 2.3 of the TSTF-505 LAR Attachment 1 states:

> 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.

Section 3.4 of the LAR to adopt 10 CFR 50.69 states in part, “[s]ubsequent performance monitoring and PRA updates required by the rule will continue to capture this data and provide timely insights into the need to account for any important new degradation mechanisms.”

The staff position C.3.2 provided in RG 1.177 for meeting the fifth key safety principle (specifically for TSTF-505), in addition to the endorsed guidance provided in NEI 00-04 (applicable for 10 CFR 50.69), acknowledges the use of performance criteria to assess degradation of operational safety over a period of time. It is unclear to NRC staff how the licensee’s processes for each of the risk-informed applications captures performance monitoring for the SSCs within-scope of each application. In light of these observations, address either (i) or (ii) below:

i) Confirm that the LaSalle Maintenance Rule program incorporates the use of performance criteria to evaluate SSC performance as described in the NRC-endorsed guidance in NUMARC 93-01.

OR

ii) Describe the approach/method used by LaSalle for SSC performance monitoring as described in Regulatory Position C.3.2 referenced in RG 1.177 for meeting the fifth key safety principle. In the description, include criteria (e.g., qualitative or quantitative) along with the appropriate risk metrics for each application, and explain how the approach and criteria demonstrates the intent to monitor the potential degradation of SSCs for the applicable process of the risk-informed application (i.e., for 10 CFR 50.69, paragraphs (d)(1) and (e)(2) and for TSTF-505, Section 3.4 of NEI 06-09, Revision 0).

EGC RESPONSE:

i) LaSalle does not use performance criteria as described in NUMARC 93-01.

ii) LaSalle has implemented the guidance in NEI 18-10, Rev. 0, "Monitoring the Effectiveness of Nuclear Power Plant Maintenance," as a means of meeting the requirements set forth in 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." The overall purpose of NEI 18-10 is to provide utilities with a risk-informed framework that supports the implementation and monitoring of a maintenance effectiveness program that complies with 10 CFR 50.65, effectively and efficiently leverages utility resources, and is focused on equipment performance commensurate with safety. NEI 18-10 is an alternative to the NUMARC 93-01/Regulatory Guide 1.160 guidance.

Both Regulatory Guide 1.160, Rev. 4 and NUMARC 93-01, Rev 4F (endorsed by Regulatory Guide 1.160) allow for utilities to use alternative methods or approaches to ensure the requirements of 10 CFR 50.65 are being met. Regulatory Guide 1.160 Section D Implementation, Use by Applicants and Licensees:

> Applicants and licensees may voluntarily use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this regulatory guide may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged.

NUMARC 93-01 Section 2.0 Purpose and Scope:

> This guideline describes an acceptable approach to meet the Maintenance Rule. However, utilities may elect other suitable methods or approaches for implementation. This guideline does not address the many industry programs that have been put in place to upgrade maintenance and may be used when implementing the Maintenance Rule. For example, work planning and scheduling, preventive and corrective maintenance, maintenance procedures, training, post maintenance testing, work history, cause determination methods and other maintenance related programs are not discussed.

In accordance with NEI 18-10, Rev. 0, all SSC in scope of 10 CFR 50.65(b)(1) and (b)(2) are evaluated for safety significance. Safety significance is determined by a Maintenance Rule (MR) expert panel informed by importance measures; the following would be considered as potentially high safety significant (HSS) functions.

1. FV > 0.005
2. RAW > 2.0
3. Birnbaum > 1E-05/yr CDF or > 1E-06/yr LERF

All other functions in scope of MR are considered low safety significant (LSS).

SSCs that remain capable of performing their intended functions will be retained in (a)(2) status. If an event or failure occurs and an issue report (IR) is generated in the corrective action program (CAP) associated with a scoped in SSC with HSS function(s), the IR will be reviewed for HSS maintenance rule functional failure (MRFF). Any HSS MRFF will result in an immediate (a)(1) determination (i.e. every HSS function has a reliability performance criteria = 0). All IRs that represent a plant level event (PLE) will result in an immediate (a)(1) determination. For LSS functions the performance criteria is not a set number of MRFFs, but instead is when a trend in system/function performance is observed. This is still performance criteria/monitoring and when reached/observed would drive an immediate (a)(1) determination. Trends will be identified on an ongoing/continuous basis by identification through engineer SSC performance review, through OPEX review, or during the (a)(3) assessment.

An aggregate assessment of the balance between reliability and availability with be provided by CDF trending. CDF trending looks at the risk impact associated with both planned and unplanned maintenance and considers the impact of failures, as failures that occur at power result in unplanned maintenance. CDF trending also provides an aggregate assessment of maintenance planning and execution. CDF trends will be reviewed during the (a)(3) assessment for a minimum of 1) long unavailability durations, 2) peak periods of risk increase, 3) need to update PRA, and 4) multiple occurrences of the same configuration due to ineffective maintenance. If the assessment determines that the increase in CDF average values was the result of an ineffective maintenance strategy, an immediate (a)(1) determination will be performed for the contributing SSC function(s).

Any SSC function determined to be (a)(1) will result in a CAP causal determination and (a)(1) goals will be established commensurate with the SSCs safety significance and performance and corrective actions will be planned and implemented to correct the cause of the degraded performance. Corrective actions will be tracked to completion. Goals are established to bring about the necessary improvements in performance. Monitoring consists of periodic trending and evaluating performance and/or availability of the SSC function(s) comparing the results with the established (a)(1) goals to verify that the goals are being met. Monitoring also provides a means for determining the effectiveness of the corrective actions. A goal is met and monitoring of SSC function(s) against the specific goal may be discontinued if any of the following criteria are satisfied 1) acceptable performance for three surveillance periods (when periodicity is <= 6 months), 2) acceptable performance for two surveillance periods (when periodicity is >= 6 months but less than 2 cycles), or 3) any approved and documented technical assessment that assures the cause is known and corrected thus monitoring against goals is unnecessary. If any of these conditions are met, the SSC function(s) may be returned to (a)(2) status. If none of these conditions are met then additional causal determination is necessary and new corrective actions, goal setting, and monitoring will be established to drive acceptable SSC performance.

All IRs that represent a PLE and all IRs that were determined to be HSS MRFF will result in an immediate (a)(1) determination.

SSC performance monitoring is performed on an ongoing/continuous basis and if a trend is identified, an (a)(1) determination will be performed. LSS trends and CDF trending are also reviewed during the periodic (a)(3) assessment. If the assessment determines the trends were the result of an ineffective maintenance strategy, an immediate (a)(1) determination will be performed at that time.

The (a)(1) determination will document the basis for remaining in (a)(2) status or the need for goal setting and monitoring under the requirements of (a)(1). For SSC function(s) determined to be (a)(1), goals will be established commensurate with the SSCs safety significance and performance. Monitoring will verify that goals are being met and determine the effectiveness of the corrective actions.

Every IR is reviewed for PLE or HSS MRFF. Every PLE or HSS MRFF will be evaluated to determine if the maintenance strategy is still effective. Events of lower safety significance are reviewed on an ongoing/continuous bases to determine if a trend or correlation exists between the events. If one is identified, the trend will be evaluated to determine if the maintenance strategy is still effective. CDF trending is reviewed periodically to determine the balance between reliability and availability, the effectiveness of maintenance planning and execution, and peak periods of risk increase and multiple occurrences of the same configuration. CDF trending will be used to determine if the maintenance strategy is still effective.

Anytime the maintenance strategy is determined to be ineffective, the SSC function(s) will be moved to (a)(1) status and goals will be established such that monitoring can verify performance against goals and determine the effectiveness of corrective actions.

DRA/APLA RAI 08 – CREDIT OF FLEX IN THE PRA MODELS

[*Applicable for TSTF-505 and 10 CFR 50.69*]

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06,'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC staff's position concerning incorporating mitigating strategies (FLEX) into a PRA in support of risk-informed decision making in accordance with the guidance in RG 1.200, Revision 2 (ADAMS Accession No. ML090410014).

To complete the NRC staff's review of the FLEX strategies modeled in the PRA, the NRC staff requests the following information for the IEPRA (includes internal floods) and FPRA, as appropriate.

a) Clarify whether permanent or portable FLEX equipment and associated operator actions are credited in the PRAs used to support the applications, identifying the specific PRA(s) that include such credit. If FLEX is not credited in the PRAs, then no response to parts (b) and (c) of this RAI is requested. If FLEX is credited in the PRAs and this credit is not expected to impact the PRA results used in the categorization process or RICT program (e.g., permanently installed equipment, hardened vent containment), then provide sufficient justification to confirm this conclusion, and no response to parts (b) and (c) of this question is requested.

b) If the FLEX equipment or operator actions have been credited, and their inclusion is expected to impact the PRA results used in the categorization process and RICT program, provide the following information separately for the IEPRA (includes internal floods) and FPRA, as appropriate:

i. A discussion detailing the extent of incorporation, i.e. summarize the supplemental equipment and compensatory actions that have been quantitatively credited for each of the PRA models used to support both risk-informed applications.

ii. Discuss the data and failure probabilities used to support the FLEX modeling and provide the rationale for using the chosen data. Include discussion on whether the uncertainties associated with the parameter values are in accordance with the applicable supporting requirements (SRs) in the ASME/ANS PRA Standard, as endorsed by RG 1.200, Revision 2.

iii. Discuss the methodology used to assess human error probabilities for the FLEX operator actions. The discussion should include:

1. A summary of how the impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)-(j) of SR HR-G3 of the ASME/ANS RA-Sa-2009 PRA standard was evaluated.
2. Whether maintenance and testing procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and whether the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of the ASME/ANS RA-Sa-2009 PRA standard.
3. For licensee's procedures governing the initiation or entry into mitigating strategies, identify specific areas which could be ambiguous, vague, or not explicit. Provide a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

c) If the FLEX equipment or operator actions have been credited, and their inclusion is expected to impact the PRA results used in the categorization process and RICT program, provide the following information separately for the IEPRA (includes internal floods) and FPRA, as appropriate:

i. Provide an evaluation of the model changes associated with incorporating non-safety-related SSCs that were included following the FLEX mitigation strategies (permanently installed and/or portable), which demonstrates that none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences,

OR

ii. Propose a mechanism to ensure that a focused-scope peer review is performed on the model changes associated with incorporating mitigating strategies, and associated F&Os are resolved to Capability Category II prior to implementation of the 10 CFR 50.69 categorization process and RICT program.

EGC RESPONSE:

a) Currently, the portable FLEX mitigation strategies and equipment are not credited in the Full Power Internal Events (FPIE) PRA or the Fire PRA. However, if LaSalle chooses to model FLEX in the future, the PRA model updates will be performed in accordance with the ASME/ANS PRA Standard and NEI 06-09 Revision 0-A as appropriate. The HCVS system is credited in the FPIE and Fire PRA due to it being a permanently installed system and cited in the Emergency Operating Procedures. Therefore, its modeling is consistent with the methodology and guidance used for non-FLEX components. See APLA RAI 05, part C for additional details.

b) Not applicable – See response to part (a) for details.

c) Not applicable – See response to part (a) for details.

DRA/APLA RAI 09 – IMPLEMENTATION ITEMS

[*Applicable for TSTF-505 and 10 CFR 50.69*]

RG 1.174, Revision 3, and RG 1.200, Revision 2, define the quality of the PRA in terms of its scope, level of detail, and technical adequacy. The quality must be compatible with the safety implications of the proposed change and the role the PRA plays in justifying the change. If the responses to any RAIs require any follow-up actions prior to implementation of the risk-informed applications (i.e., 10 CFR 50.69 or TSTF-505), provide a list of those actions and any PRA modeling changes including any items that will not be completed prior to issuing the amendments but must be completed prior to implementing the applications.

Propose a mechanism that ensures these activities and changes will be completed and appropriately reviewed and any issues resolved prior to implementing the applications (for example, a license condition that includes all applicable implementation items and a statement that they will be completed prior to implementation).

EGC RESPONSE:

The response to DRA/APLA(C) RAI 01 summarizes the implementation items associated with the risk-informed applications (i.e., 10 CFR 50.69 or TSTF-505 RICT) and the mechanism that ensures their completion prior to implementing the application. The information is therefore not repeated in this response.

ATTACHMENT 2

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

Revision to Attachment 5 of TSTF-505 RICT LAR For New Implementation Items Related to Resolution of Fire F&Os

ATTACHMENT 2
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

REVISION TO ATTACHMENT 5 OF TSTF-505 RICT LAR FOR NEW IMPLEMENTATION ITEMS RELATED TO RESOLUTION OF FIRE F&OS

LaSalle County Station RICT Program PRA Implementation Items

The table below identifies the items that are required to be completed prior to implementation of the Risk-Informed Completion Time (RICT) Program at LaSalle County Station. The items 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.
Enclosure 2, Table E2-2, Fire F&O 4-17	**As stated in Table E2-2 of the TSTF-505 RICT LAR for Fire F&O 4-17.**	**As stated in Table E2-2 of the TSTF-505 RICT LAR for Fire F&O 4-17.**
Enclosure 2, Table E2-2, Fire F&O 6-11	**As stated in Table E2-2 of the TSTF-505 RICT LAR for Fire F&O 6-11.**	**As stated in Table E2-2 of the TSTF-505 RICT LAR for Fire F&O 6-11.**

ATTACHMENT 3
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

Revision to Section 2.3 of 10 CFR 50.69 LAR for New Implementation Items Related to Resolution of Fire F&Os

ATTACHMENT 3
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

REVISION TO SECTION 2.3 OF 10 CFR 50.69 LAR FOR NEW IMPLEMENTATION ITEMS RELATED TO RESOLUTION OF FIRE F&OS

2.3 DESCRIPTION OF THE PROPOSED CHANGE

EGC proposes the addition of the following condition to the renewed operating license of LSCS, Units 1 and 2, to document the NRC's approval of the use 10 CFR 50.69.

> Exelon Generation Company, LLC (EGC) is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in the EGC submittal letter dated January 31, 2020, and all its subsequent associated supplements, as specified in License Amendment No. [XXX] dated [DATE].
>
> **EGC will complete the implementation items listed in Table APLA-01.2 in Attachment 1 of EGC letter to NRC dated October 29, 2020, prior to implementation of 10 CFR 50.69 program. All issues identified 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 10 CFR 50.69 categorization process.**
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).