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RS-20-139

10 CFR 50.90
10 CFR 50.69

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Response to Request for Additional Information Regarding License Amendment Requests to Adopt TSTF-505, Revision 2, and 10 CFR 50.69

References:

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. NRC, "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 April 30, 2020
2. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. NRC, "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,'" dated April 30, 2020
3. Email from J. Wiebe (U.S. NRC) to K. M. Nicely (Exelon Generation Company, LLC), "Request for Additional Information Regarding TSTF-505 and 50.69 Applications," dated October 27, 2020

In References 1 and 2, Exelon Generation Company, LLC (EGC) requested two amendments to Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. The amendment proposed in Reference 1 would modify Technical Specifications requirements to permit the use of Risk Informed Completion Times in accordance with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," (ADAMS Accession No. ML18183A493). The amendment proposed in Reference 2 would modify the CPS licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of 10 CFR 50.69, "Risk-Informed categorization and treatment of structures, systems and components for nuclear power reactors."

The NRC requested additional information that is needed to complete review of these two license amendment requests in Reference 3. In response to this request, EGC is providing the attached information.

EGC has reviewed the information supporting findings of no significant hazards consideration, and the environmental considerations, that were previously provided to the NRC in References 1 and 2. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendments do not involve a significant hazards consideration. In addition, the additional information provided in this submittal 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 concerning this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 24th day of November 2020.

Respectfully,

Patrick R. Simpson
Sr. Manager Licensing

Attachment: Response to Request for Additional Information

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Clinton Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

<u>DRA/APLA RAI 01 – Open Internal Events PRA Facts and Observations (F&O)</u>

[Applicable for TSTF-505 and 10 CFR 50.69]

a. LAR Enclosure 2, Table E2-1 presents the dispositions for two F&Os (i.e., F&Os 1-32 and 1-34) that remain open after the internal events F&O closure review performed in November 2019. These F&Os both address the same concern, stating in F&O 1-32 that "potentially risk significant combinations of HFEs [Human Error Events] are not captured through the current approach, due to the chosen truncation level for the dependency identification (5E-9 / 5E-10 for CDF/LERF) in conjunction with the elevated HEP level chosen (0.1)." The LAR states that the existing cutsets were reviewed and only a "few" dependent Human Error Probability (HEP) combinations with some level of dependency were identified. The LAR concluded that the overall risk results were not substantially impacted by the current treatment.

   Lowering the truncation level is likely to reveal further HEP combinations that require dependency analysis and possible adjustment of the joint probability for the combination. Therefore, it is not clear to NRC staff that the current treatment of HEP dependency has no impact on the RICT calculation for plant configurations and SSC categorization.

   In light of the observations above, address the following:

   i. Provide justification that the additional HEP combinations using a lower quantification level will not adversely impact the RICT calculations or SSC categorizations for each of the risk-informed applications.

   ii. Alternatively, propose a mechanism that ensures the results of the dependency analysis associated with F&O 1-32 and 1-34 are resolved prior to implementation of the RICT and SSC categorization programs.

b. The disposition for F&O 1-32 further states that "a floor value of 1E-06 or 5E-07 may be imposed on the dependent joint HEP" depending on the timing of operator actions." Section 4.4.3.2 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA) states in part, "the PRA analyst can define the significant contributors by using typical PRA criteria […], such as importance measure thresholds as well as other qualitative and quantitative considerations (ADAMS Accession No. ML051160213)." It is not clear to the NRC staff what assumptions (i.e., qualitative or quantitative considerations) were made in the internal events PRA for the application of the minimum joint HEP values and how they were determined to be sufficient. In light of the observations above, address the following:

   Confirm what minimum joint HEP values were used in the internal events PRA. If a minimum joint HEP value less than 1E-06 was used in the internal events PRA, then provide sufficient justification that demonstrates that for each joint HEP value used below 1E-6 issue-relevant human actions have been appropriately addressed. The justification should support that the internal events PRA minimum values used have no adverse impact on the RICT and SSC risk-informed applications.

Response

a) HRA Dependent Group Identification Methodology

The HRA Dependency Analysis methodology used for the Clinton FPIE PRA has been evaluated as part of the October 2009 Peer Review [1] 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.

Although additional combinations would be discovered using lower quantification truncation levels, the overall quantification time would increase substantially without introducing significant changes to the overall risk profile (i.e., risk-significant accident sequences and initiators would likely remain the same). The HRA Dependency Analysis methodology involves artificially increasing the human error probability (HEP) to a minimum value of 0.1 to ensure that dependent combinations comprised of human failure events (HFEs) with low independent HEPs are captured during the combination identification process. Reducing the truncation level used for the HRA Dependency Analysis would likely identify new combinations with several independent HFEs included in the combination (e.g., combinations of 7 or more HFEs), and the combinations would be forced to use the joint HEP (JHEP) floor value, which can artificially skew risk results.

TSTF-505 RICT Assessment

A sensitivity analysis (Sensitivity #1) was performed to evaluate the potential impact of unadjusted dependent combinations on the TSTF-505 RICT calculations. The sensitivity analysis consisted of analyzing the base FPIE cutsets with the HRA Calculator Dependency Module in order to identify new combinations that were previously unadjusted (i.e., implicitly assumed "zero dependence"). The default order and dependency level were retained for the new combinations, which is conservative since the HRA Calculator Dependency Module assumes "High" or "Complete" dependence for several of the new combinations based on similar delay times (i.e., this reflects a source of conservatism for the sensitivity analysis).

Table APLA-01-A.1 summarizes the results of this sensitivity (Sensitivity #1) on a sample of TSTF-505 RICT calculations. The sample Technical Specification (TS) cases were selected because they represent the cases where the RICT calculations produce less than 30 days.

As shown in Table APLA-01-A.1, the RICT estimates decreased by a small amount for all sample TS (i.e., largest reduction shown is ~5% reduction which equates to ~0.5-day reduction). Given the inherent conservatisms of the new dependent combinations (i.e., several of the new groups assume "Complete Dependence" by default, which could be reduced with a detailed review of the combination), the largest potential reduction in the RICT estimate is ~0.5 days, which would only be reduced with additional refinements to the new combinations.

Therefore, inclusion of new dependent combinations would not substantially impact the TSTF-505 RICT calculations and the current FPIE HRA Dependency Analysis is appropriate for implementation of the TSTF-505 RICT program.

<u>10 CFR 50.69 Assessment</u>

A sensitivity analysis was performed to evaluate the potential impact of unadjusted dependent combination on the 10 CFR 50.69 SSC categorizations. The sensitivity analysis consisted of analyzing the base FPIE cutsets with the HRA Calculator Dependency Module in order to identify new combinations that were previously unadjusted (i.e., implicitly assumed "zero dependence"). The default order and dependency level were retained for the new combinations, which is conservative since the HRA Calculator Dependency Module assumes "High" or "Complete" dependence for several of the new combinations based on similar delay times (i.e., this reflects a source of conservatism for the sensitivity analysis).

The importance measures from the base FPIE PRA model were compared against the sensitivity case, where additional dependent groups were identified and adjusted (Sensitivity #1). The screening criteria specified in Section 5.1 of NEI 00-04 is as follows:

- Sum of Fussell-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) > 2
- Maximum of applicable common cause basic events RAW > 20

For the purposes of this sensitivity analysis, since there may be several independent and 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 above).

Table APLA-01-A.2 discusses the basic events that could potentially change SSC categorizations based on meeting the criteria above in Sensitivity #1, but not in the base model (i.e., basic event is risk-significant in the sensitivity case, but not in the base model).

Based on the results, the SSC categorizations would not change from the categorizations made using the base FPIE PRA model. Therefore, the current HRA Dependency Analysis used in the base FPIE PRA model has no impact on 10 CFR 50.69 categorization results.

TABLE APLA-01-A.1

SENSITIVITY #1: ADDITIONAL DEPENDENT COMBINATIONS RICT CALCULATION SENSITIVITY RESULTS FOR FPIE PRA

(FPRA & SEISMIC RESULTS REMAIN UNCHANGED FOR SENSITIVITY ANALYSIS)

			BASE RESULTS (CL-LAR-010, Rev. 0)				APLA-01 SENSITIVITY #1 RESULTS (ADDITIONAL DEPENDENT COMBINATIONS)		
CASE	**DESCRIPTION**	**CDF vs. LERF**	**DELTA FPIE CDF**	**DELTA FIRE CDF**	**DELTA SEISMIC CDF**	**RICT ESTIMATE (DAYS) (30 DAY MAXIMUM)**	**DELTA FPIE CDF**	**RICT ESTIMATE (DAYS) (30 DAY MAXIMUM)**	**RICT % CHANGE**
3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	CDF	1.35E-05	1.12E-04	6.40E-06	27.7	1.58E-05	27.2	-1.72%
3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	CDF	2.01E-05	2.59E-04	6.40E-06	12.8	2.07E-05	12.8	-0.23%
3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable.	CDF	5.86E-05	5.00E-04	6.40E-06	6.5	7.58E-05	6.3	-2.95%
3.5.1.B	High Pressure Core Spray (HPCS) System inoperable.	CDF	1.35E-05	1.12E-04	6.40E-06	27.7	1.58E-05	27.2	-1.72%
3.5.1.F	One required ADS valve inoperable AND One low pressure ECCS injection/spray subsystem inoperable.	CDF	5.86E-05	5.03E-04	6.40E-06	6.4	7.58E-05	6.2	-2.94%
3.7.1.B	Division 1 or 2 SX subsystem inoperable.	CDF	1.68E-05	3.03E-04	6.40E-06	11.2	2.88E-05	10.8	-3.55%
3.8.1.B	One required DG inoperable.	CDF	1.14E-05	2.68E-04	6.40E-06	12.8	1.79E-05	12.5	-2.21%
3.8.1.D	One offsite circuit inoperable AND One required DG inoperable.	CDF	2.99E-05	3.33E-04	6.40E-06	9.9	5.10E-05	9.3	-5.39%
3.8.4.A	One battery charger on Division 1 or 2 inoperable.	CDF	1.80E-05	3.68E-04	6.40E-06	9.3	3.04E-05	9.0	-3.05%
3.8.9.B	One or more Division 1 or 2 uninterruptible AC bus distribution subsystems inoperable.	CDF	1.84E-04	7.26E-04	6.40E-06	4.0	1.86E-04	4.0	-0.25%

TABLE APLA-01-A.2

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS

SENSITIVITY #1 (ADDITIONAL COMBINATIONS; BASE JHEP FLOOR VALUE)

(FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
FPIE PRA						
1FXDGPRIFLEX-X--	PRIMARY FLEX DIESEL GENERATOR FAILS TO RUN	3.50E-04	1.08E-03	1.01	1.02	**No Change in SSC Categorization** Due to the lower F-V threshold, this basic event was identified as risk-significant. The FLEX Diesel Generators are risk-significant based on the Level 1 (CDF) results (this record is based on Level 2 (LERF) results). This sensitivity does not change the SSC categorization.
1RHFN-1VY02C-A--	RHR A PUMP RM COOLER FAN FAILS TO START	9.70E-04	1.11E-03	1.91	2.04	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHFN-1VY03C-A--	RHR A HX ROOM COOLER FAN FAILS TO START	9.30E-04	1.08E-03	1.88	2.01	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHMVF048AB--KCC	CC FAILURE HX BYPASS VLVS TO CLOSE	8.00E-05	1.30E-04	14.9	23.9	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

TABLE APLA-01-A.2

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS

SENSITIVITY #1 (ADDITIONAL COMBINATIONS; BASE JHEP FLOOR VALUE)

(FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
1RHMV-F064ABCDCC	CC RHR 64A, B, & C MIN FLOW MOV FAIL TO OPEN	4.00E-05	6.00E-05	14.5	23.6	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHMV-F064AB-DCC	CC RHR 64A & B MIN FLOW MOV FAIL TO OPEN	3.00E-05	5.00E-05	14.3	23.3	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHPM--2ABC--ACC	CC RHR A B&C PUMPS FAIL TO START	6.00E-05	1.00E-04	15.3	24.4	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHPMABC-LPCSACC	CC RHR A B&C/LPCS PUMPS FAIL TO START	1.10E-04	1.80E-04	16.6	25.5	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

TABLE APLA-01-A.2

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS

SENSITIVITY #1 (ADDITIONAL COMBINATIONS; BASE JHEP FLOOR VALUE)

(FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
1RHPM-ABLPCS-ACC	CC RHR A&B/LPCS PUMPS FAIL TO START	6.00E-05	1.00E-04	15.3	24.4	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHPM-C002AB-ACC	CC RHR A&B PUMPS FAIL TO START	6.50E-04	9.90E-04	19.6	29.3	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHPM-C002AB-XCC	CC RHR A&B PUMPS FAIL TO RUN	2.00E-05	3.00E-05	12.9	22.3	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1SXAVVYCLROUTDCC	CC FAILURE FOR VY COOLER DISCHARGE VALVE	3.00E-05	2.30E-04	14.8	111	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

TABLE APLA-01-A.2

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS

SENSITIVITY #1 (ADDITIONAL COMBINATIONS; BASE JHEP FLOOR VALUE)

(FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
1SXMV14AB68ABDCC	CC FAILURE FOR RH HX VALVES	1.90E-04	3.10E-04	16.4	25.8	**No Change in SSC Categorization** Given that the functions associated with the SX system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1SXPM-SX01PA-X--	PUMP 1SX01PA FAILS TO RUN	9.00E-04	1.17E-03	4.40	5.39	**No Change in SSC Categorization** Given that the functions associated with the SX system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1VDDM-VD01YA-D--	FAILURE OF DAMPER VD01YA TO OPEN	8.30E-04	1.02E-03	3.67	4.28	**No Change in SSC Categorization** Given that the functions associated with the DG system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1VDDM-VD01YB-D--	FAILURE OF DAMPER VD01YB TO OPEN	9.00E-04	1.06E-03	3.88	4.42	**No Change in SSC Categorization** Given that the functions associated with the DG system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

TABLE APLA-01-A.2

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS

SENSITIVITY #1 (ADDITIONAL COMBINATIONS; BASE JHEP FLOOR VALUE)

(FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
1VYFN-DIV12HXACC	CC DIV 1&2 HX ROOM COOLER FANS FAIL TO START	3.40E-04	5.40E-04	17.4	27.0	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1VYFN-DIV12HXXCC	CC DIV 1&2 HX ROOM COOLER FANS FAIL TO RUN	2.76E-03	4.22E-03	19.4	29.2	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1VYFN-DIV12PRACC	CC DIV 1&2 PUMP ROOM COOLER FANS FAIL TO START	3.80E-04	5.80E-04	19.0	28.5	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

b) HRA Dependency Analysis Floor Value Methodology

For the FPIE 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] acknowledges that the floor value of 1E-05 stated in NUREG-1792 [3] 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.

The FPIE PRA HRA Dependency Analysis is documented in Section 5.3 of CL-PRA-004, HRA Notebook.

TSTF-505 RICT Assessment

Two sensitivity analyses were performed to evaluate the potential impact of using a higher floor value (i.e., 1E-05) on the TSTF-505 RICT calculations:

1. Sensitivity #2: The 1E-05 floor value is applied to the base FPIE model.

2. Sensitivity #3: Sensitivity #1 from part (a) but with a floor value of 1E-05 applied to those new combinations with a JHEP below 1E-05.

Table APLA-01-B.1 summarizes the results of this sensitivity on a sample of TSTF-505 RICT calculations. The sample TS cases were selected because they represent the cases where the RICT calculations produce less than 30 days.

For Sensitivity #2 (base FPIE model using 1E-05 floor value), the delta FPIE CDF values are essentially unchanged for several of the sample TS and only two cases showed a small decrease in the RICT estimate (i.e., ~5% reduction which equates to ~0.5-day reduction). Since the delta values are essentially unchanged, the RICT estimates (in days) are also essentially unchanged.

For Sensitivity #3 (Sensitivity #1 model using 1E-05 floor value), all sample TS experienced small reductions (i.e., ~1% - 6% reductions) due to the inherent conservatisms of the default dependency levels assumed for the sensitivity analysis. Like Sensitivity #1, refinement of these default dependency levels (which typically assume "High" or "Complete" dependence) would reduce the overall impact on the RICT calculations.

Therefore, the JHEP floor values used in the FPIE PRA do not substantially 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>

Two sensitivity analyses were performed to evaluate the potential impact of using a higher floor value (i.e., 1E-05) on the 10 CFR 50.69 SSC categorizations:

1. Sensitivity #2: The 1E-05 floor value is applied to the base FPIE model.

2. Sensitivity #3: Sensitivity #1 from part (a) but with a floor value of 1E-05 applied to those new combinations with a JHEP below 1E-05.

The importance measures from the base FPIE PRA model were compared against the sensitivity cases (similar to Sensitivity #1 from part (a)). The screening criteria specified in Section 5.1 of NEI 00-04 is as follows:

- Sum of Fussell-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) > 2

- Maximum of applicable common cause basic events RAW > 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 above).

Table APLA-01-B.2 discusses the basic events that could potentially change SSC categorizations based on meeting the criteria above in Sensitivity #2, but not in the base model (i.e., basic event is risk-significant in the sensitivity case, but not in the base model).

Table APLA-01-B.3 discusses the basic events that could potentially change SSC categorizations based on meeting the criteria above in Sensitivity #3, but not in the base model (i.e., basic event is risk-significant in the sensitivity case, but not in the base model).

Based on the results presented in these two tables, the SSC categorizations would not change from the categorizations made using the base FPIE PRA model.

Therefore, the JHEP values used in the base FPIE PRA model are appropriate for implementation of the 10 CFR 50.69 program.

TABLE APLA-01-B.1

JHEP FLOOR VALUE RICT CALCULATION SENSITIVITY RESULTS FOR FPIE PRA

(FPRA & SEISMIC RESULTS REMAIN UNCHANGED FOR SENSITIVITY ANALYSIS)

			BASE RESULTS (CL-LAR-010, Rev. 0)				APLA-01 SENSITIVITY #2 RESULTS (MIN. JHEP = 1E-05 USING BASE FPIE PRA)			APLA-01 SENSITIVITY #3 RESULTS (MIN. JHEP = 1E-05 USING SENSITIVITY #1 MODEL)		
CASE	DESCRIPTION	CDF vs. LERF	DELTA FPIE CDF	DELTA FIRE CDF	DELTA SEISMIC CDF	RICT ESTIMATE (DAYS) (30 DAY MAXIMUM)	DELTA FPIE CDF	RICT ESTIMATE (DAYS) (30 DAY MAXIMUM)	RICT % CHANGE	DELTA FPIE CDF	RICT ESTIMATE (DAYS) (30 DAY MAXIMUM)	RICT % CHANGE
3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	CDF	1.35E-05	1.12E-04	6.40E-06	27.7	2.10E-05	26.2	-5.35%	2.32E-05	25.8	-6.81%
3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	CDF	2.01E-05	2.59E-04	6.40E-06	12.8	2.06E-05	12.8	-0.19%	2.11E-05	12.7	-0.38%
3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable.	CDF	5.86E-05	5.00E-04	6.40E-06	6.5	5.95E-05	6.4	-0.16%	7.65E-05	6.3	-3.07%
3.5.1.B	High Pressure Core Spray (HPCS) System inoperable.	CDF	1.35E-05	1.12E-04	6.40E-06	27.7	2.10E-05	26.2	-5.35%	2.32E-05	25.8	-6.81%
3.5.1.F	One required ADS valve inoperable AND One low pressure ECCS injection/spray subsystem inoperable.	CDF	5.86E-05	5.03E-04	6.40E-06	6.4	5.95E-05	6.4	-0.16%	7.65E-05	6.2	-3.06%
3.7.1.B	Division 1 or 2 SX subsystem inoperable.	CDF	1.68E-05	3.03E-04	6.40E-06	11.2	1.71E-05	11.2	-0.09%	2.88E-05	10.8	-3.57%
3.8.1.B	One required DG inoperable.	CDF	1.14E-05	2.68E-04	6.40E-06	12.8	1.16E-05	12.8	-0.07%	1.80E-05	12.5	-2.23%
3.8.1.D	One offsite circuit inoperable AND One required DG inoperable.	CDF	2.99E-05	3.33E-04	6.40E-06	9.9	3.04E-05	9.9	-0.13%	5.09E-05	9.3	-5.36%
3.8.4.A	One battery charger on Division 1 or 2 inoperable.	CDF	1.80E-05	3.68E-04	6.40E-06	9.3	1.84E-05	9.3	-0.09%	3.04E-05	9.0	-3.06%
3.8.9.B	One or more Division 1 or 2 uninterruptible AC bus distribution subsystems inoperable.	CDF	1.84E-04	7.26E-04	6.40E-06	4.0	1.84E-04	4.0	-0.07%	1.88E-04	4.0	-0.49%

TABLE APLA-01-B.2

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS SENSITIVITY #2 (BASE COMBINATIONS; 1E-05 JHEP FLOOR VALUE) (FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
FPIE PRA						
1APBS-S004---F--	BUS FAILURE 1C1 (1E22-S004)	0.00E+00	2.00E-05	1.26	2.19	**No Change in SSC Categorization** The cumulative impact of all basic events associated with this bus satisfies the "F-V > 0.005" criterion in the base FPRA model. This sensitivity does not change the SSC categorization.
1HPTP--N051--F--	PRESS XMTR 1E22NO51 FAILS TO PROVIDE LOGIC FOR OPENING MIN FLOW VALVE	1.00E-05	3.00E-05	1.45	2.41	**No Change in SSC Categorization** Given that the functions associated with the HPCS system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1LPPM-WTR-LG-XCC	CCF LPCS / RHR A & RHR B / C WATER LEG PUMPS FTR	9.00E-05	1.18E-03	1.03	1.42	**No Change in SSC Categorization** Given that the functions associated with the RHR & LPCS systems are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

TABLE APLA-01-B.3

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS

SENSITIVITY #3 (ADDITIONAL COMBINATIONS; 1E-05 JHEP FLOOR VALUE)

(FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
FPIE PRA						
1APBS-S004---F--	BUS FAILURE 1C1 (1E22-S004)	0.00E+00	1.00E-05	1.26	2.03	**No Change in SSC Categorization** The cumulative impact of all basic events associated with this bus satisfies the "F-V > 0.005" criterion in the base FPRA model. This sensitivity does not change the SSC categorization.
1HPTP--N051--F--	PRESS XMTR 1E22NO51 FAILS TO PROVIDE LOGIC FOR OPENING MIN FLOW VALVE	1.00E-05	3.00E-05	1.45	2.29	**No Change in SSC Categorization** Given that the functions associated with the HPCS system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHMVF048AB--KCC	CC FAILURE HX BYPASS VLVS TO CLOSE	8.00E-05	1.20E-04	14.9	21.5	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHMV-F064ABCDCC	CC RHR 64A, B, & C MIN FLOW MOV FAIL TO OPEN	4.00E-05	6.00E-05	14.5	21.3	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

TABLE APLA-01-B.3

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS

SENSITIVITY #3 (ADDITIONAL COMBINATIONS; 1E-05 JHEP FLOOR VALUE)

(FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
1RHMV-F064AB-DCC	CC RHR 64A & B MIN FLOW MOV FAIL TO OPEN	3.00E-05	5.00E-05	14.3	21.0	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHPM--2ABC--ACC	CC RHR A B&C PUMPS FAIL TO START	6.00E-05	9.00E-05	15.3	22.0	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHPMABC-LPCSACC	CC RHR A B&C/LPCS PUMPS FAIL TO START	1.10E-04	1.60E-04	16.6	23.0	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

TABLE APLA-01-B.3

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS

SENSITIVITY #3 (ADDITIONAL COMBINATIONS; 1E-05 JHEP FLOOR VALUE)

(FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
1RHPM-ABLPCS-ACC	CC RHR A&B/LPCS PUMPS FAIL TO START	6.00E-05	9.00E-05	15.3	22.0	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHPM-C002AB-ACC	CC RHR A&B PUMPS FAIL TO START	6.50E-04	8.90E-04	19.6	26.5	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1RHPM-C002AB-XCC	CC RHR A&B PUMPS FAIL TO RUN	2.00E-05	3.00E-05	12.9	20.1	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

TABLE APLA-01-B.3

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS

SENSITIVITY #3 (ADDITIONAL COMBINATIONS; 1E-05 JHEP FLOOR VALUE)

(FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
1SXAVVYCLROUTDCC	CC FAILURE FOR VY COOLER DISCHARGE VALVE	3.00E-05	2.20E-04	14.8	103	**No Change in SSC Categorization** Given that the functions associated with the SX system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1SXMV14AB68ABDCC	CC FAILURE FOR RH HX VALVES	1.90E-04	2.80E-04	16.4	23.3	**No Change in SSC Categorization** Given that the functions associated with the SX system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1VYFN-DIV12HXACC	CC DIV 1&2 HX ROOM COOLER FANS FAIL TO START	3.40E-04	4.90E-04	17.4	24.4	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

TABLE APLA-01-B.3

DISCUSSION OF BASIC EVENTS THAT MAY INFLUENCE SSC CATEGORIZATION BASED ON HRA DEPENDENCY ANALYSIS

SENSITIVITY #3 (ADDITIONAL COMBINATIONS; 1E-05 JHEP FLOOR VALUE)

(FPIE RESULTS)

BASIC EVENT ID	DESCRIPTION	BASE F-V	SENSITIVITY F-V	BASE RAW	SENSITIVITY RAW	DISCUSSION
1VYFN-DIV12HXXCC	CC DIV 1&2 HX ROOM COOLER FANS FAIL TO RUN	2.76E-03	3.80E-03	19.4	26.4	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.
1VYFN-DIV12PRACC	CC DIV 1&2 PUMP ROOM COOLER FANS FAIL TO START	3.80E-04	5.20E-04	19.0	25.8	**No Change in SSC Categorization** Given that the functions associated with the RHR system are risk-significant in the base model supporting the LAR application, all components necessary for performing the function would be classified as risk-significant. This sensitivity does not change the SSC categorization.

DRA/APLA RAI 02 – Peer Review History for the Internal Events, including Internal Flooding, PRA

[Applicable for TSTF-505 and 10 CFR 50.69]

The ASME/ANS RA-Sa-2009 PRA standard defines PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of ASME/ANS RA-Sa-2009 PRA Standard states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this Standard.

LAR Enclosure 2 states that the last full scope peer review for the internal events PRA was conducted in October 2009 and that F&O closure reviews to close out F&Os from the 2009 review were conducted in December 2018 and November 2019. The LAR does not discuss the internal events and internal flood PRA model changes made between October 2009 and when the F&O closure reviews were performed in to improve the model or to incorporate changes to reflect the as-built, as-operated plant. Given the significant length of time between the last full scope peer review and the F&O closure reviews, address the following:

Summarize the significant model changes performed for the internal events (including internal flood) PRA since October 2009 and for each change justify why or why not the change meets the definition of a PRA upgrade as defined in the ASME/ANS RA-Sa-2009 PRA Standard (e.g., Changing to different PRA software or a different HRA methods are examples of possible PRA upgrades).

Response

There have been four FPIE model updates since the October 2009 peer review of the 2006 PRA model (i.e., CL06C). A summary of model changes made to the FPIE models are documented in the introductory pages of the Summary Notebook (CL-PRA-013). Additionally, changes are tracked in the Clinton Updating Requirements Evaluation (URE) database which tracks PRA observations and open items identified in between PRA model updates.

Table APLA-02-1 summarizes the important model changes made to the internal events model (including internal flooding) since the October 2009 peer review. For each model change, a discussion is provided to identify whether the change is classified as "Maintenance" vs. "Upgrade", as defined in Appendix 1-A of the ASME/ANS PRA Standard. In addition, each model change references an Appendix 1-A "Example" that best relates to the Clinton-specific model change.

As summarized in Table APLA-02-1, all FPIE PRA model changes since the October 2009 peer review are classified as "Maintenance". None of the model changes meet the definition and criteria of "Upgrade" as defined in the ASME/ANS PRA Standard.

TABLE APLA-02-1

SUMMARY OF INTERNAL EVENTS PRA MODEL CHANGES SINCE OCTOBER 2009

#	DESCRIPTION OF MODEL CHANGE	MAINTENANCE VS. UPGRADE
2011 PRA Update		
1	Initiating event frequencies were updated using Bayesian techniques and the most recent Clinton operation experience and the most current generic data.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #2 & #3 (new generic and/or plant-specific data).
2	Revised component failure data including extensive use of plant-specific component failure data gathered from the site.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #2 & #3 (new generic and/or plant-specific data).
3	Individual component random failure probabilities were updated using Bayesian techniques, using the provided plant-specific data (where applicable) and the most current generic data sources.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #2 & #3 (new generic and/or plant-specific data).
4	Removal of Division 3 diesel cross-tie human failure event (HFE)	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #6 & #7 (logic model refinement; no new methodologies employed).
5	Updated HRA calculations (independent & dependent HFEs) using EPRI HRA Calculator software (no change in HRA methodology)	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #11 (new software; no new methodologies employed).
6	Expanded use of Cause-Based Decision Tree Method (CBDTM) for assessing cognitive errors.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #21 (enhancing completeness & expanding use of a previously used methodology).
7	Addressed Internal Flooding Analysis F&Os and refined flood scenarios based on confirmatory walkdowns of significant flood scenarios.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #6 & #7 (logic model refinement; no new methodologies employed).
8	Incorporation of RAT plant modification (3 RATs from 1 RAT).	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #6, #7, & #9 (logic model refinement – change reflects new knowledge; no new methodologies employed).
2014 PRA Update (Data Update Only)		
9	Removed the ADS inhibit step from the EOPs for non-ATWS accident scenarios.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #22 (change reflects procedure change – no new methods).
10	Updated Division 3 diesel cross-tie human failure probability (HEP) based on revision to the procedure which allows use of the HPCS diesel generator to supply Division 1 or Division 2 electrical divisions.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #22 (change reflects procedure change – no new methods).
11	Changes to Class IIID to enhance modeling of vapor suppression failures for Medium and Large LOCAs.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #9 (change reflects new knowledge).
12	Use of lower truncation limits to ensure adequate convergence.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #28 (change in truncation limit – no new method).

TABLE APLA-02-1

SUMMARY OF INTERNAL EVENTS PRA MODEL CHANGES SINCE OCTOBER 2009

#	DESCRIPTION OF MODEL CHANGE	MAINTENANCE VS. UPGRADE
2017 PRA Update (Revision A)		
13	Initiating event frequencies were updated using Bayesian techniques and the most recent Clinton operating experience and the most current generic data.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #2 & #3 (new generic and/or plant-specific data).
14	Revised component failure data including extensive use of plant-specific component failure data gathered from the site.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #2 & #3 (new generic and/or plant-specific data).
15	Individual component random failure probabilities were updated using Bayesian techniques, using the provided plant-specific data (where applicable) and the most current generic data sources.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #2 & #3 (new generic and/or plant-specific data).
16	FLEX Credited in the base model with detailed HEPs.	**Maintenance** No major event tree or fault tree changes required. The FLEX-related HEP calculations utilize the same HRA methodology as all other HEPs developed for the PRA. Incorporation of FLEX is consistent ASME/ANS PRA Standard Appendix 1-A, Example #8 (completeness; similar fault tree logic as other equivalent systems modeled in the PRA).
17	Updated LOOP & ERAT event trees to include RCIC short term operation.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #8 (completeness; similar fault tree logic as other equivalent systems modeled in the PRA).
18	Updated independent & dependent HEPs.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #2 & #3 (new generic and/or plant-specific data).

TABLE APLA-02-1

SUMMARY OF INTERNAL EVENTS PRA MODEL CHANGES SINCE OCTOBER 2009

#	DESCRIPTION OF MODEL CHANGE	MAINTENANCE VS. UPGRADE
2017 PRA Update (Revision B)		
19	Updated RHR A & B room cooling logic to include common cause failures precluding credit for operator action for opening doors for room cooling (consistent with independent failures).	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #25 (new CCF using existing methodology).
20	Reduced times credited in the offsite power non-recovery probabilities to account for the time required to realign buses to offsite power sources given successful recovery of offsite power.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #2 & #3 (new generic and/or plant-specific data).
21	Inclusion of water hammer sequences due to loss of offsite power (LOOP) conditions of failure of the water leg pumps.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #6 (correction of model logic errors).
22	Correction of the RAT-LOOP logic so that random failures of the RAT and ERAT are properly propagated through the LOOP event tree.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #6 (correction of model logic errors).
23	Inclusion of updated HRA dependency analyses for CDF and LERF.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Examples #2 & #3 (new generic and/or plant-specific data).
24	Use of lower truncation limit for LERF to meet convergence criteria.	**Maintenance** Consistent with ASME/ANS PRA Standard Appendix 1-A, Example #28 (change in truncation limit – no new method).

<u>DRA/APLA RAI 03 - System and Surrogate Modeling Used in the PRA Models</u>

[Applicable for TSTF-505]

The NRC SE for NEI 06-09 specifies that the LAR should provide a comparison of the Technical Specifications (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 of the LAR identifies each TS Limited conditions of operation (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 certain TS LCO Conditions, the table explains that the associated SSCs are not modeled in the PRAs but will be conservatively represented using a surrogate event. For some LCOs, the LAR did not provide enough description of the surrogate PRA modeling that will be used in the RICT calculations for NRC staff to understand whether the modeling will be acceptable. Therefore, address the following:

> LAR Table E1-1 states for TS LCO 3.3.6.5 (Relief and Low-Low Set instrumentation) Condition A (One trip system inoperable) that the "[r]elief function" is not modeled and that the "Low-Low Set Point values are used as surrogate for the trip system." It is not clear to the NRC staff how Clinton identified that the Low-Low Set Point values proposed to be used as a surrogate reflect the failure of the function covered by the applicable TS LCO Condition. Therefore:
>
> Confirm which SSC (i.e., Low-Low Set instrumentation or over pressurization relief function) is not explicitly included in the PRA model. For the confirmed SSC that is not explicitly modeled, identify the surrogate proposed to be used in the PRA model and provide justification that ensures the surrogate appropriately represents the failure of the design function associated with TS LCO Condition 3.3.6.4.A [sic].

<u>Response</u>

The relief or overpressure protection function of the SRVs is modeled in the PRA. Failure of sufficient SRVs to open on high reactor pressure is considered to cause a large LOCA in the PRA model. The pressure instrumentation providing the signal to the SRVs to open on high reactor pressure is not modeled, therefore the SRVs associated with the LLS instrumentation are used as a conservative surrogate for any LLS instrumentation failure.

The only SRVs included in the system model for depressurization are those SRVs that can be supplied compressed air from the backup air bottles. SRVs which are supplied by the air bottles are the seven (7) ADS SRVs and the two (2) LLS SRVs. Five of the SRVs are equipped with a Low-Low Set (LLS) feature incorporated into the Relief function. This causes two LLS valves (one on each division) to be opened at a lower pressure than the relief mode pressure setpoints and results in all of the LLS valves to stay open longer, such that reopening of more than one SRV is prevented on subsequent actuation. These Div 1 and Div 2 LLS valves are used as a surrogate for the relief and low-low set instrumentation.

<u>DRA/APLA RAI 04 – Total Risk Consideration of State-of-Knowledge Correlation and Modeling Updates</u>

[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.

LAR Enclosure 5, Section 2 states that the total CDF and LERF values presented in Enclosure 5 for Clinton Power Station (CPS) are "point estimate values." NRC staff notes that for CPS, the total CDF of 8.8E05 per year and begins to approach the RG 1.174, Revision 3 guidelines for total CDF without considering the risk increase due to SOKC. The NRC staff notes based on RG 1.174 and Section 6.4 of NUREG-1855, Revision 1, for a Capability Category II risk evaluation, the mean values of the risk metrics (total and incremental values) need to be compared against the risk acceptance guidelines. Additionally, NRC staff notes that the current PRA models might potentially be updated in response to information requests (e.g., response to DRA/APLA RAI 01 regarding unresolved F&Os, DRA/APLA RAI 05 regarding FLEX modeling, etc.). Accordingly, an increase in CDF due to any updates in combination with an increase resulting from SOKC could potentially impact the conclusions of these risk-informed applications.

In light of the observations above, address the following:

a) Provide a summary of how the SOKC investigation was performed for the base Clinton 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 (i.e., this should include increases due to potential updates to PRA models that may be identified in response to NRC staff RAIs affecting the total risk for Unit 1 for conformance with the RG 1.174 risk acceptance criteria for CDF and LERF. Also, include identification of the fire PRA parameters for which SOKC was applied in the parametric uncertainty analysis of fire events.

Response

a) The Parametric Uncertainty evaluations for the FPIE & FPRA models are discussed below.

Full Power Internal Events (FPIE)

The parametric uncertainty evaluation for the FPIE PRA model is documented in Appendix G of the Quantification Notebook [4]. 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 SOKC considerations.

As summarized in the Quantification Notebook, the point-estimate and the correlated mean values were substantially different for the base uncertainty evaluation due to the calculation process sampling a significant number of basic events with mean probabilities greater than 1.0. This overestimation was due to conservative specified uncertainty distributions (e.g., lognormal error factor of 10). Therefore, additional refinements were performed to reduce the overestimation of the base parametric uncertainty evaluation (e.g., changing the uncertainty distributions to ensure that sampled mean probabilities are less than 1.0).

Table APLA-04-A.1 below summarizes the results of the updated parametric uncertainty evaluation performed for the base FPIE PRA model.

TABLE APLA-04-A.1

PARAMETRIC UNCERTAINTY RESULTS FOR BASE FPIE PRA MODEL

(UPDATED TO REDUCE OVERESTIMATION FROM ORIGINAL EVALUATION)

METRIC	POINT-ESTIMATE (/YR)	PARAMETRIC MEAN (/YR)	DELTA (/YR)	%INCREASE
CDF	3.33E-06	3.38E-06	5.00E-08	1.5%
LERF	1.65E-07	1.69E-07	4.00E-09	2.4%

Based on the results presented in Table APLA-04-A.1, the difference between the point-estimate and parametric mean values for FPIE CDF & LERF is small (< 3% increase). Therefore, it is concluded that the point-estimate values are good representations of the mean FPIE CDF and FPIE LERF risk metrics.

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 [5]. 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 SOKC considerations. Table APLA-04-A.2 below summarizes the results of the parametric uncertainty evaluation performed for the base FPRA model.

TABLE APLA-04-A.2

PARAMETRIC UNCERTAINTY RESULTS FOR BASE FPRA MODEL

METRIC	POINT-ESTIMATE (/YR)	PARAMETRIC MEAN (/YR)	DELTA (/YR)	%INCREASE
CDF	7.75E-05	7.88E-05	1.26E-06	1.6%
LERF	5.30E-06	5.34E-06	3.60E-08	0.7%

Based on the results presented in Table APLA-04-A.2, the difference between the point-estimate and parametric mean values for Fire CDF is small (< 2% increase) and for Fire LERF is nearly negligible (< 1% increase). Therefore, it is concluded that the point-estimate values are good representations of the mean Fire CDF and Fire LERF risk metrics.

Total (Including Seismic Penalties)

Table APLA-04-A.3 summarizes the total CDF and LERF for all hazards using the point-estimate and parametric mean values. As shown in Table APLA-04-A.3, total CDF and LERF conform with the RG 1.174 risk acceptance guidance (i.e., CDF < 1E-04 and LERF < 1E-05 per year) using both the point-estimate values and the parametric mean values.

Based on the results presented in Table APLA-04-A.3, the difference between the point-estimate and parametric mean values for total CDF & total LERF is small (< 2% increase). Therefore, it is concluded that the point-estimate values are good representations of the mean CDF and LERF risk metrics.

TABLE APLA-04-A.3

COMPARISON OF POINT-ESTIMATE & PARAMETRIC MEAN VALUES

METRIC	POINT-ESTIMATE (/YR)	PARAMETRIC MEAN (/YR)	DELTA (/YR)	%INCREASE
CDF				
FPIE	3.33E-06	3.38E-06	5.00E-08	1.5%
FPRA	7.75E-05	7.88E-05	1.26E-06	1.6%
Seismic	6.40E-06	6.40E-06	-	-
Total	**8.72E-05**	**8.86E-05**	**1.31E-06**	**1.5%**
LERF				
FPIE	1.65E-07	1.69E-07	4.00E-09	2.4%
FPRA	5.30E-06	5.34E-06	3.60E-08	0.7%
Seismic	1.60E-06	1.60E-06	-	-
Total	**7.07E-06**	**7.11E-06**	**4.00E-08**	**0.6%**

b) Potential Impact on Risk-Informed Applications

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 Table APLA-04-A.3, the combined parametric means are less than 3% greater than the point-estimate values for combined 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 TS to determine the potential impact of using parametric mean values (instead of point-estimate values) on the RICT calculations. The sample TS cases were selected because they represent a subset of the cases where the RICT calculations produce less than 30 days.

For the cases specified in Table APLA-04-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-04-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 6.4E-06/yr (CDF) and 1.6E-06/yr (LERF) remain unchanged for the RICT calculations).

For the sample RICT calculations summarized in CL-LAR-010 [6], the unfactored Minimum Cutset Upper Bound (MCUB) results were used, which presents a slight conservatism for the Fire PRA results given that the unfactored results tend to be larger than the factored results. For this sensitivity analysis, the factored MCUB results for the FPIE & Fire PRA were used as a better representation of the point-estimate results such that any comparisons to the parametric mean results are more accurate. Therefore, the RICT estimates in Table APLA-04-B.1 are slightly larger than the estimates documented in CL-LAR-010.

Table APLA-04-B.1 summarizes the results of this sensitivity analysis. The columns to Table APLA-04-B.1 are as follows:

- Case
- Description – Brief description of the TS
- Limiting Metric – CDF or LERF
- FPIE Results & Fire Results (each hazard has the following columns)
    - Point-Estimate (using the factored MCUB results)
    - Delta to Base Point-Estimate
    - Parametric Mean (accounts for SOKC uncertainties)
    - Delta to Parametric Mean
    - %Increase (Point-Estimate vs. Parametric Mean)
- RICT Estimates
    - RICT Estimate using point-estimate values
    - RICT Estimate using parametric mean values
    - %Change

As shown in Table APLA-04-B.1, nearly all cases showed negligible change in the RICT estimates (i.e., < 0.5 day reduction). The case with the largest reduction in the RICT estimate was TS 3.8.4.A, which had a total reduction of 0.3 days (~3%). Additional sensitivities for different TS are expected to produce similar results as those shown in Table APLA-04-B.1.

While the point-estimate risk metric values used in the base PRA models do not account for SOKC, based on the results presented in Table APLA-04-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.

<u>10 CFR 50.69 Assessment</u>

As shown in Tables APLA-04-A.1 and APLA-04-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 have not 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-04-B.1

SENSITIVITY ANALYSIS RESULTS FOR POINT-ESTIMATE VS. MEAN VALUES ON RICT CALCULATIONS

CASE INFORMATION			FPIE					FIRE					TOTAL (INCLUDING SEISMIC)		
CASE	DESCRIPTION	LIMITING METRIC	POINT-ESTIMATE (/YR)(1)	DELTA TO BASE CASE (/YR)	MEAN (/YR)	DELTA TO BASE CASE (/YR)	%INCREASE (POINT-ESTIMATE VS. MEAN)	POINT-ESTIMATE (/YR) (1)	DELTA TO BASE CASE (/YR)	MEAN (/YR)	DELTA TO BASE CASE (/YR)	%INCREASE (POINT-ESTIMATE VS. MEAN)	RICT ESTIMATE (POINT-ESTIMATE)	RICT ESTIMATE (PARAMETRIC MEAN)	%CHANGE
Base	Base PRA Model (Zero Maintenance)	CDF	3.32E-06	-	3.37E-06	-	1.35%	6.97E-05	-	7.06E-05	-	1.41%	-	-	
3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable.	CDF	6.19E-05	5.86E-05	6.21E-05	5.87E-05	0.27%	5.01E-04	4.32E-04	5.01E-04	4.30E-04	-0.14%	7.35	7.37	0.31%
3.5.1.F	One required ADS valve inoperable AND One low pressure ECCS injection/spray subsystem inoperable.	CDF	6.20E-05	5.87E-05	6.21E-05	5.87E-05	0.13%	5.04E-04	4.34E-04	5.07E-04	4.36E-04	0.60%	7.31	7.28	-0.41%
3.7.1.B	Division 1 or 2 SX subsystem inoperable.	CDF	2.01E-05	1.68E-05	2.04E-05	1.70E-05	1.44%	3.37E-04	2.67E-04	3.43E-04	2.72E-04	1.78%	12.56	12.34	-1.78%
3.8.1.B	One required DG inoperable.	CDF	1.48E-05	1.15E-05	1.50E-05	1.16E-05	1.28%	3.11E-04	2.41E-04	3.11E-04	2.41E-04	0.23%	14.11	14.11	0.05%
3.8.1.D	One offsite circuit inoperable AND One required DG inoperable.	CDF	3.33E-05	3.00E-05	3.38E-05	3.05E-05	1.59%	3.69E-04	2.99E-04	3.76E-04	3.05E-04	1.87%	10.88	10.68	-1.87%
3.8.4.A	One battery charger on Division 1 or 2 inoperable.	CDF	2.14E-05	1.81E-05	2.17E-05	1.83E-05	1.17%	3.88E-04	3.18E-04	3.99E-04	3.28E-04	2.89%	10.66	10.35	-2.96%

<u>Note to Table ALPA-04-B.1</u>

(1) For the sample RICT calculations summarized in CL-LAR-010 [6], the unfactored MCUB results were used, which presents a slight conservatism for the Fire PRA results given that the unfactored results tend to be larger than the factored results. For this sensitivity analysis, the factored MCUB results for the FPIE & Fire PRA were used as a better representation of the point-estimate results such that any comparisons to the parametric mean results are more accurate.

DRA/APLA RAI 05 – Credit for FLEX Equipment and Actions

[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 were 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) The ASME/ANS RA-Sa-2009 PRA standard defines PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of ASME/ANS RA-Sa-2009 states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this Standard.

   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.

<u>Response</u>

a) FLEX is credited in the Clinton FPIE PRA (including Internal Flooding) & FPRA models.

b) <u>FLEX PRA Modeling</u>

   i. <u>FLEX Mitigation Strategies and Compensatory Actions</u>

   The following FLEX mitigating strategies are credited in the PRA models:

   1. <u>Electrical Strategy:</u> Deployment and alignment of portable FLEX 480V diesel generators.

- FLEX Diesel Generators:

  Two (2) 480 VAC diesel-powered generators are included as part of the FLEX strategies equipment. The primary FLEX generator and distribution panel is permanently housed in the Unit 2 side of the Control and Diesel Generator Buildings; therefore, deployment will not be impeded by a beyond design basis external event (BDBEE). The connecting cabling is pre-routed from the vicinity of the primary FLEX switchgear to the vicinity of the Division 1 480 VAC unit substations which will have connection points for an external source of power. The alternate FLEX Diesel Generator is in a robust enclosure next to the Screen House.

2. Water Strategy: Deployment and alignment of portable diesel-driven FLEX pumps to support Suppression Pool Cooling (SPC), Spent Fuel Pool makeup, Suppression Pool makeup, and RPV makeup.

   - FLEX Diesel-Driven Pumps

     The FLEX system provides alternate injection options via two (2) diesel-driven low pressure high capacity self-prime pumps. These pumps are designed for emergency cooling in case of a catastrophic event. The primary and alternate pumps can provide approximately 3000 gpm with approximately 2000 gpm provided for RHR Heat Exchanger cooling and 1000 gpm to the RPV (if needed). Current pump specifications are 3150 gpm at 172 psig. Along with these pumps are two hose trailers which contain hoses, fittings, and tools necessary for the strategies associated with the portable FLEX pumps. These are stored in a robust FLEX storage building next to the Screen House.

The following FLEX mitigating strategies / compensatory actions (outlined in the 4306.01 procedures) are discussed below:

- RCIC Operation (RPV Injection) – Credited in the PRA

  RCIC suction from the RCIC tank is assumed to be unavailable (i.e., not credited). RCIC can operate with suction from the Suppression Pool for up to 8 hours without Suppression Pool Cooling. Given success of Suppression Pool Cooling, RCIC will operate beyond its 24-hour mission time. Also, the FLEX diesel generators provide AC charging support for extended DC power support for RCIC.

- Suppression Pool Cooling – Credited in the PRA

  The use of an RHR Heat Exchanger to provide Suppression Pool Cooling requires 2000 gpm for the tube side (FLEX diesel-driven pumps with suction from UHS) and 1500 gpm for the shell side (Suppression Pool

Clean-up and Transfer (SF) pump with suction from the Suppression Pool).

- Low Pressure Injection using FLEX Pumps – Not credited in the PRA

  When RCIC is no longer available, the low pressure RPV makeup can be provided using SRVs to control RPV pressure. The preferred method for low pressure RPV makeup is available once 480VAC power is available to the SF pumps. The SF pump can supply enough suppression pool water to the RPV via LPCI or the SDC return line to make-up for boil-off and system leakage. Alternate low pressure RPV injection strategies exist (e.g., FLEX pump connected to UHS and LPCS and/or RHR C injection lines).

- Containment Venting – Not Required

  Based on MAAP case CL06017A, which modeled loss of decay heat removal, containment pressure reaches the containment vent pressure of 45 psig approximately 23 hours after event initiation. If left unmitigated, containment pressure reaches the ultimate primary containment pressure of 76 psig at approximately 34 hours after event initiation. Therefore, a FLEX-specific containment venting strategy is not required over the 24 hour PRA mission time.

ii. FLEX Equipment Failure Rates

The portable FLEX diesel generators and pumps are not like other installed plant equipment, so the generic failure rates in NUREG/CR-6928 are not directly applicable. However, since industry data for FLEX equipment is currently being collected and is anticipated to be available prior to implementation of the TSTF-505 RICT program at Clinton, an assumption was made in the short-term to estimate the FLEX equipment unreliability failure probabilities.

A factor of two is applied to the Clinton-specific unreliability failure probabilities for similar equipment in order to estimate the FLEX equipment unreliability failure probabilities. This escalation is a reasonable approximation of the unreliability of FLEX until industry data is published. Table APLA-05-1 below summarizes the unreliability failure probabilities that are used for FLEX equipment.

The uncertainties associated with the data values are based upon the uncertainty parameters from the generic data and are in accordance with the ASME/ANS PRA Standard. Use of these values should provide a reasonable approximation of the reliability of the FLEX equipment until industry-approved data becomes available for FLEX equipment.

In addition to the FLEX unreliability failure probabilities, the PRAs include a bounding "FLEX fails after alignment" failure probability of 0.1, which fails all

FLEX strategies. This bounding failure accounts for the potential uncertainty with the FLEX strategies.

As documented in Section 8.7 of CL-MISC-033, a sensitivity analysis was performed to assess the potential impact on the TSTF-505 RICT calculations. The FLEX equipment failure probabilities were escalated by a factor of five (which equates to a factor of 10 increase when compared to the base non-FLEX equipment failure rates used for non-FLEX equipment). For the FLEX diesel generators, the FTS failure rate increased to 6.15E-02 per demand and the FTR failure rate increased to 3.08E-01 per 24 hours. For the FLEX diesel-driven pumps, the FTS failure rate increased to 3.15E-02 per demand and the FTR failure rate increased to 5.25E-02 per 24 hours. Given the magnitude of these failure rates, a factor of 10 on the non-FLEX equipment failure rates is assessed to be suitable for assessing potential impacts on TSTF-505 RICT calculations.

However, an additional sensitivity analysis was performed for which the FLEX equipment failure rates used in the CL-MISC-033 sensitivity analysis (i.e., the values discussed in the previous paragraph) were increased by a factor of two (2). These elevated FLEX equipment failure rates reflect a factor of 20 applied to the non-FLEX equipment failure rates used as a starting point for developing the FLEX equipment failure rates. With a factor of 20 applied, the values used in this sensitivity analysis should represent the upper percentile range of possible values given the lack of industry-approved failure rates for comparison.

Table APLA-05-2 summarizes the results of the "factor of 20" sensitivity analysis results for a sample of TSTF-505 RICT calculations. The sample TS cases were selected because they represent the cases that would likely be impacted the most by the alternate hypothesis (i.e., the base RICT estimate was either right at the maximum 30 day cutoff, or significantly less than the maximum 30 day cutoff).

Although the "factor of 20" sensitivity shows a decrease in RICT estimates in two cases, the sensitivity failure rates are excessively conservative and this sensitivity distorts the overall risk profile (e.g., for this sensitivity, the FLEX DG FTS failure probability is 1.23E-01 per demand and the FTR failure probability is 6.15E-01 per 24 hours).

Therefore, the results of the first sensitivity analysis (factor of 10 increase over non-FLEX equipment failure rates) are more appropriate for comparison to the base model RICT calculations, as discussed in CL-MISC-033.

In addition, once industry-approved FLEX failure data becomes available, this topic will no longer be a source of uncertainty that requires special consideration (i.e., uncertainty will be addressed by the parametric uncertainty evaluations performed on the base PRA models). In terms of identification of RMAs, see response to part (d) for details.

TABLE APLA-05-1

FLEX UNRELIABILITY FAILURE PROBABILITIES

COMPONENT	FAILURE MODE	NUREG/CR-6928 CASE	BAYESIAN UPDATED PROBABILITY	FACTOR	MISSION TIME	FLEX PROBABILITY
FLEX Diesel Generators	FTS	Emergency Diesel Generator (EDG) Fail to Start & Fails to Run < 1 Hour	6.15E-03 (per demand)	2	1 (demand)	1.23E-02
	FTR	Emergency Diesel Generator (EDG) Fail to Run > 1 Hour	1.28E-03 (per hour)	2	24 (hours)	6.15E-02
					Total	**7.38E-02**
FLEX Diesel-Driven Pumps	FTS	Engine Driven Pump Fails to Start, Normally Standby & Engine Driven Pump Fails to Run, Early Term, Normally Standby	3.15E-03	2	1 (demand)	6.30E-03
	FTR	Engine Driven Pump Fails to Run, Late Term	2.18E-04	2	24 (hours)	1.05E-02
					Total	**1.68E-02**

TABLE APLA-05-2

FLEX EQUIPMENT FAILURE RATE RICT CALCULATION SENSITIVITIES

TECH SPEC	TS/LCO CONDITION	BASE MODEL RICT ESTIMATE (DAYS)	SENSITIVITY #1 (FACTOR OF 10) RICT ESTIMATE (DAYS)	SENSITIVITY #2 (FACTOR OF 20) RICT ESTIMATE (DAYS)
3.3.6.5.A	Relief and Low-Low Set Instrumentation - One trip system inoperable.	30.00	30.00	30.00
3.6.2.3.A	One RHR suppression pool cooling subsystem inoperable.	30.00	30.00	30.00
3.8.1.A	One offsite circuit inoperable.	30.00	30.00	30.00
3.8.1.D	One offsite circuit inoperable AND One required DG inoperable.	9.88	9.12	7.47
3.8.4.A	One battery charger on Division 1 or 2 inoperable.	9.30	8.63	7.16
3.8.7.A	Division 1 or 2 inverter inoperable.	30.00	30.00	30.00
3.8.9.B	One or more Division 1 or 2 uninterruptible AC bus distribution subsystems inoperable.	3.98	3.98	3.98

iii. FLEX Human Reliability Analysis

1. FLEX HRA Methodology

   All human error probabilities (HEPs) associated with FLEX human failure events (HFEs) are evaluated using the same methodology used for non-FLEX HFEs in the Clinton PRAs, as documented in the Clinton HRA Notebooks (FPIE: CL-PRA-004; FPRA: CL-PRA-021.09). The EPRI HRA Calculator (HRAC) was used to calculate and document the impact of plant-specific and scenario-specific performance shaping factors specified in the ASME/ANS PRA Standard Supporting Requirement (SR) HR-G3(a)-(j).

   For the actions discussed in Sections 7.5.4 and 7.5.5 of NEI 16-06, engineering judgement and conservative assumptions were used to assess the cognitive and execution portions of the FLEX HEPs (e.g., in the execution assessment, error of commission (EOC) overrides were used when a standard option did not apply). The Clinton HRA Notebooks contain the details of the HEP calculations.

2. Pre-Initiator Identification

   The maintenance and testing procedures have not been reviewed for potential pre-initiator human failures. However, as previously discussed in part (b)(ii) of this response, the PRAs include a bounding "FLEX fails after alignment" failure (probability of 0.1), which fails all FLEX strategies. This bounding failure accounts for the potential uncertainty with the FLEX strategies (including the potential for pre-initiators that are not currently modeled).

3. Ambiguity of Entry Conditions

   The FLEX procedures (4306 series) provide a clear direction as to when the FLEX mitigation strategies would be pursued and implemented. However, as previously discussed in part (b)(ii) of this response, the PRAs include a bounding "FLEX fails after alignment" failure (probability of 0.1), which fails all FLEX strategies. This bounding failure accounts for the potential uncertainty with initiation of the FLEX strategies.

Additionally, a sensitivity analysis was performed to evaluate the potential impact of FLEX-related HEPs on the TSTF-505 RICT calculations. For this sensitivity analysis, the 95th percentile HEP values for the independent & dependent HEPs associated with FLEX HFEs were selected as the conservative and bounding case. The 95th percentile independent HEPs range from ~1E-02 to 2.5E-01 and the 95th percentile dependent HEPs range from ~2E-06 to 3E-02.

Table APLA-05-3 summarizes the results of the 95th percentile sensitivity analysis results for a sample of TSTF-505 RICT calculations. The sample TS cases were selected because they represent the cases that would likely be impacted the most by the alternate hypothesis (i.e., the cases where FLEX could provide the maximum benefit to the RICT calculations).

Although the sensitivity analysis using the 95th percentile HEPs shows a decrease in RICT estimates for two of the seven cases, using the 95th percentile HEPs for the FLEX HFEs in the base RICT calculations would be overly conservative and could mask key risk insights for specific configurations. Therefore, the FLEX-related HEPs used in the base PRAs are appropriate for use in the TSTF-505 RICT program.

TABLE APLA-05-3

FLEX 95TH% HEP RICT CALCULATION SENSITIVITY RESULTS

TECH SPEC	TS/LCO CONDITION	BASE MODEL RICT ESTIMATE (DAYS)	SENSITIVITY RICT ESTIMATE (DAYS)
3.3.6.5.A	Relief and Low-Low Set Instrumentation - One trip system inoperable.	30.00	30.00
3.6.2.3.A	One RHR suppression pool cooling subsystem inoperable.	30.00	30.00
3.8.1.A	One offsite circuit inoperable.	30.00	30.00
3.8.1.D	One offsite circuit inoperable AND One required DG inoperable.	9.88	7.44
3.8.4.A	One battery charger on Division 1 or 2 inoperable.	9.30	7.17
3.8.7.A	Division 1 or 2 inverter inoperable.	30.00	30.00
3.8.9.B	One or more Division 1 or 2 uninterruptible AC bus distribution subsystems inoperable.	3.98	3.98

c) <u>PRA Maintenance vs. Upgrade</u>

Incorporation of FLEX into the Clinton PRA models reflects plant modifications and procedure changes in order to maintain the models as representative of the as-built, as-operated plant. The ASME/ANS PRA Standard defines "PRA Upgrade" as a change that meets one of the following criteria:

- New methodology
- Change in scope that impacts the significant accident sequences or the significant accident progression sequences
- Change in capability that impacts the significant accident sequences or the significant accident progression sequences

The Clinton FLEX model logic changes (as they relate to the "PRA Upgrade" criteria) are discussed below.

New Methodology

Consistent with Table A-1 of RG 1.200, Rev. 2, the term "new method" refers to an analysis method (i.e., not documentation method) that is new to the subject PRA even if the method itself is not new and has been applied in other PRAs. This term also encompasses newly developed methods in the industry that have been implemented in the base PRA in question.

No new methods were used for the development of the FLEX fault tree logic, component data, or human reliability analysis. The FLEX fault tree was created in a consistent manner as existing fault trees. Failure rates for the FLEX equipment are discussed in part (b)(ii) of this response. The human reliability analysis for FLEX mitigating strategies is discussed in part (b)(iii) of this response and the methodology employed is identical to the methodology used for developing non-FLEX HEPs.

Change in Scope

The Scope attribute is defined consistent with Section C of RG 1.2001:

> *"The scope of the PRA ...is defined in terms of (1) the metrics used to characterize risk, (2) the plant operating states for which the risk is to be evaluated, and (3) the causes of initiating events (hazard groups) that can potentially challenge and disrupt the normal operation of the plant and, if not prevented or mitigated, would eventually result in core damage and/or a large release."*

The scope of the model remains unchanged with the incorporation of FLEX.

Change in Capability

Consistent with concepts in RG 1.200, Rev. 2 as well as the basis for Capability Category distinctions in the PRA Standard, this term is defined in terms of degree of analysis detail and plant-specific realism. Implementation of this criterion in the context of determining whether a specific PRA change represents an upgrade is whether the change would increase the Capability Category (from Not Met or CC-I to CC-II) for one or more SRs.

There were no changes in the capability assessment for any of the PRA Standard Supporting Requirements as a result of the inclusion of FLEX into the PRA models.

DRA/APLA RAI 06 - Performance Monitoring

[Applicable for TSTF-505 and 10 CFR 50.69]

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 RG 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 Clinton 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 Clinton 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).

<u>Response</u>

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

ii) CPS has implemented the guidance in NEI 18-10, Revision 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/RG 1.160 guidance.

Both RG 1.160, Revision 4, and NUMARC 93-01, Revision 4F (endorsed by RG 1.160), allow for utilities to use alternative methods or approaches to ensure the requirements of 10 CFR 50.65 are being met. RG 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, Revision 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 an equivalent of a reliability performance criterion = 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 will 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 basis 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-07 - PRA MODEL UNCERTAINTY ANALYSIS RESULTS

[Applicable for TSTF-505 and 10 CFR 50.69]

The NRC staff safety evaluation to NEI 06-09, Revision 0, specifies that the LAR should identify key assumptions and sources of uncertainty and to assess/disposition each as to their impact on the RMTS application. LAR Enclosure 9, Tables E9-1, E9-2, and E9-3 identify the key assumptions and sources of uncertainty for the internal events PRA, transition to the RTR model, and the fire PRA and provides dispositions for each source of uncertainty for this TSTF-505 application. NRC staff reviewed the dispositions provided in LAR Tables E9-1, E9-2 and E9-3 to the key assumptions and sources of modeling uncertainty and noted that not all uncertainties that appeared to have the potential to impact the RICT calculations seemed fully resolved. NRC staff notes that LAR Tables E9-1, E9-2, and E9-3 do not address certain sources of uncertainty identified in those reports that appear to NRC staff to have the potential to impact the RICT application. Therefore, address the following:

a) LAR Enclosure 9, Table E9-3 identifies cable selection as a source of fire PRA modeling uncertainty because of conservatisms in the approach. The LAR states that some components were conservatively assumed to be failed based on lack of cable data in certain locations. Such components were referred to as Unknown Location (UNL) components which were stated to be "mostly limited" to Balance of Plant systems. The LAR states that two sensitivity analysis were performed to determine the impact of the treatment of UNL components. Based on the sensitivity studies, the LAR states "it is concluded that the methodology for the Cable selection task does not introduce any epistemic uncertainties." The LAR does not provide the results of the uncertainty analyses. NRC staff notes that conservatism in PRA modeling could have a nonconservative impact on the RICT calculations while having only a small or modest impact on total CDF or LERF. It is not clear to NRC staff that the modeling assumption applied to untraced cables has no impact on the RICT program calculations. If an SSC is part of a system not credited in the fire PRA or it is supported by a system that is assumed to always fail, then the risk increase due to taking that SSC out of service is masked. Therefore, address the following:

i. Identify the systems or components that are assumed to be always failed in the PRA or not included in the PRA (due to lack of cable tracing or other reasons). Justify that this assumption has an inconsequential impact on the RICT calculations.

ii. If in part (i) above, it is be determined that the cited assumption has a consequential impact on the estimated RICTs, then identify what programmatic changes will be considered to compensate for this uncertainty and the basis for their consideration.

b) LAR Enclosure 9, Table E9-3 identifies post-fire HRA as a source of fire PRA modeling uncertainty because fire HEPs must be adjusted to consider the additional challenges present given a fire. The LAR states that industry consensus modeling approaches are used and concludes that this source of uncertainty "does not introduce any epistemic uncertainties that would require sensitivity treatment" and that additional RMAs were, therefore, not required. Accordingly, it Is not clear why the need for additional RMAs to mitigate the impact of this uncertainty was not considered. Therefore, address the following:

i. Justify that the uncertainty associated with post-fire HRA modeling does not have a consequential impact on calculated RICTs for components supporting TS LCO Conditions in the RICT program.

ii. If in response to part (i) above: it is determined that the cited source of uncertainty has a consequential impact on the estimated RICTs, then identify what programmatic changes will be considered to compensate for this uncertainty.

<u>Response</u>

a) <u>UNL Methodology</u>

The components assumed to be failed in select Physical Analysis Units (PAUs) analyzed in the FPRA 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, Clinton'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.

UNL Sensitivity Analyses (Base Model)

The sensitivity analyses performed on the base average-maintenance model are summarized in Section 4.2.1 of CL-PRA-021.12, Uncertainty & Sensitivity Analysis Notebook [5].

The first sensitivity analysis consisted of removing all UNL failures from the FPRA model, which estimates the potential conservatisms of the UNL treatment (i.e., UNL components are never failed by fire-induced impacts). The results of this sensitivity analysis showed a maximum decrease of ~5-6% in Fire CDF and Fire LERF, indicating that removing the UNLs would not have a substantial impact on the FPRA results.

The second sensitivity analysis consisted of expanding the UNL failures to additional PAUs using more conservative assumptions, which estimates the potential non-conservatisms of the UNL treatment (i.e., UNL components are failed in more fire scenarios). The results of this sensitivity analysis showed that a maximum increase of ~12-15% in Fire CDF and Fire LERF, indicating that expanding the UNL failures could have a significant impact on the FPRA results. However, this sensitivity analysis assumed that a majority of the UNL components are failed in every PAU, which would be an unrealistic assumption. With further refinements to the expanded UNL treatment, the change in fire risk would likely decrease to similar levels shown in the first sensitivity analysis.

Therefore, the UNL treatment used in the FPRA is assessed to be appropriate to avoid overly conservative results that mask key risk insights.

TSTF-505 RICT Assessment

Table APLA-07-A.1 summarizes the results of the assumed routing task for the Clinton FPRA. Most of the UNL components are associated with systems that are not within the scope of the TSTF-505 RICT program and do not serve support functions for the TSTF-505 RICT LCOs. Failure of these out-of-scope UNL components would increase the importance of redundant support functions, thus ensuring that any associated RICT calculation would result in a larger delta risk and therefore a smaller RICT estimate.

The columns to Table APLA-07-A.1 are as follows:

- UNL Component ID – Surrogate component ID to represent the unit / system / division of all components included in this surrogate
- Description – Brief description of the UNL component
- PAUs – PAUs where cables related to the UNL are assumed to be present (i.e., UNLs failed for all scenarios with damage beyond the ignition source within the listed PAU)
- Routing Comment – Brief comment describing any changes made to the UNL assumed routing assessment for the UNL component

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 FPRA 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-07-A.2 summarizes the results of the "No UNL" sensitivity on the TSTF-505 RICT calculations for a sample of cases. The sample TS cases were selected because they represent the cases where the RICT calculations produce less than 30 days.

Based on the results presented in Table APLA-07-A.2, the RICT calculations produced a larger RICT estimate for nearly all TS LCOs analyzed as part of the "No UNL" sensitivity (TS 3.8.1.D produced a slightly lower RICT estimate due to rounding errors). Removal of the UNL components from the FRA model increased the amount of credit given to support functions, which resulted in the lower delta fire risk metrics (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 FPRA is not overly conservative and the RICT calculations are not adversely impacted.

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TABLE APLA-07-A.1

UNL COMPONENTS IN THE CLINTON FPRA

UNL COMPONENT ID	DESCRIPTION	PAUS	COMMENT
AC_N1E_DIV1_UNL	AC Power - Non-Safety (Non-Class 1E) (Division 1)	A-1a, A-1b, A-2k, A-2m, A-2n, A-3d, A-3f, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1g, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, F-1a, F-1m, F-1p, M-2c, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1p, R-1q, R-1r, R-1s, R-1t, SY-1b, T-1a, T-1f, T-1h, TY-1a, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
AC_N1E_DIV2_UNL	AC Power - Non-Safety (Non-Class 1E) (Division 2)	A-1b, A-2k, A-3d, A-3e, A-3f, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1g, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, F-1a, F-1m, F-1p, M-2c, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1p, R-1q, R-1r, R-1s, R-1t, SY-1b, T-1a, T-1f, T-1h, T-1k, TY-1a, TY-1b, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
CA_DIV1_UNL	Condenser Vacuum (Division 1)	A-2k, CB-1f, CB-2, CB-3a, CB-4, CB-7, R-1c, R-1i, R-1p, R-1q, R-1t, T-1a, T-1f, T-1g, T-1h, T-1j, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
CA_DIV2_UNL	Condenser Vacuum (Division 2)	A-1a, A-1b, A-2k, A-3d, A-3f, CB-1f, CB-2, CB-3a, CB-4, R-1i, R-1p, R-1q, R-1t, T-1a, T-1f, T-1g, T-1h, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
CB_DIV1_UNL	Condensate Booster (Division 1)	A-2k, CB-1f, CB-2, CB-3a, CB-4, R-1c, R-1i, R-1p, R-1t, T-1a, T-1b, T-1e, T-1f, T-1g, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
CB_DIV2_UNL	Condensate Booster (Division 2)	A-1a, A-1b, A-3d, A-3f, CB-2, CB-3a, CB-4, R-1c, R-1i, R-1p, R-1t, T-1a, T-1b, T-1e, T-1f, T-1g, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
CD_DIV1_UNL	Condensate (Division 1)	A-1a, A-1b, A-2k, CB-1f, CB-2, CB-3a, CB-4, R-1c, R-1i, R-1p, R-1t, T-1a, T-1c, T-1e, T-1f, T-1h, T-1j, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
CD_DIV2_UNL	Condensate (Division 2)	A-1a, A-1b, A-3d, A-3f, CB-2, CB-3a, CB-4, R-1c, R-1i, R-1p, R-1t, T-1a, T-1c, T-1f, T-1h, T-1j, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
CD_UNL	Condensate (All Divisions)	A-1a, A-1b, A-2k, A-3d, A-3f, CB-1f, CB-2, CB-3a, CB-4, R-1c, R-1i, R-1p, R-1t, T-1a, T-1c, T-1e, T-1f, T-1h, T-1j, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
CI_IB_UNL	Containment Isolation - Inboard Isolation Signals	A-3f, C-1, C-2, CB-2, CB-3e	Assumed to be routed in the same PAUs as N081B/C Level Transmitters.
CI_OB_UNL	Containment Isolation - Outboard Isolation Signals	A-2k, A-2n, C-1, C-2, CB-1f, CB-1g, CB-3f, CB-4, CB-6b	Assumed to be routed in the same PAUs as N081A/D Level Transmitters.
CI_RE_DIV1_UNL	Containment Isolation - Equipment Drains (Division 1)	A-1b, A-2d, A-2k, A-2m, A-2n, A-3d, A-3f, C-1, C-2, CB-1f, CB-2, CB-3a, CB-4, F-1a, F-1m, F-1p, R-1i, R-1m, R-1o, R-1p, R-1t, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
CI_RE_DIV2_UNL	Containment Isolation - Equipment Drains (Division 2)	A-2k, A-3d, A-3e, A-3f, C-1, C-2, CB-1f, CB-2, CB-3a, CB-4, F-1a, F-1m, F-1p, R-1i, R-1m, R-1o, R-1p, R-1t, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.

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TABLE APLA-07-A.1

UNL COMPONENTS IN THE CLINTON FPRA

UNL COMPONENT ID	DESCRIPTION	PAUS	COMMENT
CI_RF_DIV1_UNL	Containment Isolation - Floor Drains (Division 1)	A-1a, A-1b, A-2b, A-2k, A-2m, A-2n, A-3d, A-3e, A-3f, C-1, C-2, CB-1e, CB-1f, CB-2, CB-3a, CB-4, F-1a, F-1m, F-1p, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1p, R-1t, T-1a, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
CI_RF_DIV2_UNL	Containment Isolation - Floor Drains (Division 2)	A-1a, A-1b, A-3d, A-3f, C-1, C-2, CB-2, CB-3a, CB-4, F-1a, F-1m, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1t, T-1a, T-1f, T-1h, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
CW_DIV1_UNL	Circulating Water (Division 1)	A-1a, A-1b, A-2k, A-2m, A-3d, A-3f, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, F-1m, F-1p, M-2c, R-1c, R-1i, R-1m, R-1o, R-1p, R-1t, T-1a, T-1e, T-1f, T-1h, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
CW_DIV2_UNL	Circulating Water (Division 2)	A-2k, A-3d, A-3e, A-3f, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, F-1a, F-1m, F-1p, M-2c, R-1c, R-1i, R-1m, R-1o, R-1p, R-1t, T-1a, T-1d, T-1e, T-1f, T-1h, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
CY_DIV1_UNL	Cycled Condensate / Condensate Makeup (Division 1)	A-1b, A-2b, A-2e, A-2k, A-2m, A-2n, C-2, CB-1e, CB-1f, CB-2, CB-3a, CB-4, F-1m, F-1p, R-1c, R-1i, R-1p, R-1q, R-1t, T-1a, T-1f, T-1h, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
DC_N1E_DIV1_UNL	DC Power - Non-Safety (Division 1)	A-2k, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1g, CB-2, CB-3a, CB-4, CB-7, R-1a, R-1C, R-1h, R-1i, R-1m, R-1p, R-1q, R-1r, R-1t, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
DC_N1E_DIV2_UNL	DC Power - Non-Safety (Division 2)	A-1b, A-3d, A-3f, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1g, CB-2, CB-3a, CB-4, R-1i, R-1p, R-1t, T-1a, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
ERAT_SVC_UNL	ERAT Static VAR Compensator (SVC)	TY-1b	Assumed to be limited to ERAT PAU.
FC_DIV1_UNL	Fuel Pool Cooling (Division 1)	A-1b, A-2d, A-2k, A-2m, A-2n, C-1, C-2, CB-1e, CB-1f, CB-2, CB-3a, CB-4, CB-7, D-4a, F-1a, F-1i, F-1m, F-1p, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1p, R-1t	PAUs based on SLICE data query using System Code & Segregation Codes.
FC_DIV2_UNL	Fuel Pool Cooling (Division 2)	A-2k, A-3d, A-3e, A-3f, A-3g, C-2, CB-1f, CB-2, CB-3a, CB-4, F-1a, F-1i, F-1m, F-1p, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o	PAUs based on SLICE data query using System Code & Segregation Codes.
FP_DIV1_UNL	Fire Protection (Division 1)	A-1a, A-1b, A-2b, A-2d, A-2e, A-2k, A-2m, A-2n, A-3d, A-3e, A-3f, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1g, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, CB-7, F-1a, F-1m, F-1p, M-2a, M-2c, M-4, R-1c, R-1i, R-1m, R-1o, R-1p, R-1q, R-1r, R-1s, R-1t, SB-1, SY-1b, T-1a, T-1e, T-1f, T-1g, T-1h, T-1k, T-1m, TY-1a, TY-1b, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.

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Response to Request for Additional Information

TABLE APLA-07-A.1

UNL COMPONENTS IN THE CLINTON FPRA

UNL COMPONENT ID	DESCRIPTION	PAUS	COMMENT
FP_DIV2_UNL	Fire Protection (Division 2)	A-1a, A-2k, A-3d, A-3e, A-3f, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, D-10, F-1a, F-1m, F-1p, M-2c, R-1c, R-1i, R-1p, R-1t, T-1a, T-1f, TY-1b, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
FW_DIV1_UNL	Feedwater (Division 1)	A-1b, A-2b, A-2f, A-2k, A-2n, C-2, CB-1e, CB-1f, CB-2, CB-3a, CB-4, CB-7, R-1c, R-1i, R-1p, R-1t, T-1a, T-1f, T-1h, T-1m	PAUs based on SLICE data query using System Code & Segregation Codes.
FW_DIV2_UNL	Feedwater (Division 2)	A-1a, A-1b, A-2f, A-2k, A-3a, A-3d, A-3f, C-2, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, CB-7, R-1i, R-1p, R-1t, T-1a, T-1e, T-1f, T-1g, T-1h, T-1k, T-1m	PAUs based on SLICE data query using System Code & Segregation Codes.
GS_DIV1_UNL	Turbine Gland Seal Steam (Division 1)	A-2k, CB-1f, CB-2, CB-3a, CB-4, R-1i, R-1p, R-1t, T-1e, T-1f, T-1g, T-1h, T-1j, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
GS_DIV2_UNL	Turbine Gland Seal Steam (Division 2)	CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, R-1t, T-1e, T-1g, T-1h, T-1j, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
GS_UNL	Turbine Gland Seal Steam (All Division)	A-2k, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, R-1i, R-1p, R-1t, T-1e, T-1f, T-1g, T-1h, T-1j, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
IA_DIV1_UNL	Instrument Air (Division 1)	A-1b, A-2k, A-2m, A-2n, A-6, C-1, C-2, CB-1f, CB-2, CB-3a, CB-4, R-1i, R-1o, R-1p, R-1q, R-1r, R-1t, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
IA_DIV2_UNL	Instrument Air (Division 2)	A-1a, A-1b, A-3d, A-3f, A-3g, C-2, CB-2, F-1p, R-1q	PAUs based on SLICE data query using System Code & Segregation Codes.
IP_DIV1_UNL	Misc Instrumentation & Control Power (Division 1)	CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, CB-7, R-1i, R-1p, R-1t, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
IP_DIV2_UNL	Misc Instrumentation & Control Power (Division 2)	CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, R-1i, R-1p, R-1t, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
MC_DIV1_UNL	Main Condenser (Division 1)	A-1b, A-2d, A-2k, A-2n, CB-2, CB-3a, CB-4, F-1m, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1p, R-1q, R-1r, R-1t, T-1a, T-1f, T-1h, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
MC_DIV2_UNL	Main Condenser (Division 2)	A-3d, A-3f, C-2, CB-2, CB-3a, CB-4, R-1a, R-1c, R-1h, R-1i, R-1o, R-1p, R-1q, R-1r, R-1t, SY-1b, T-1a, T-1f, T-1h, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
MC_UNL	Main Condenser (All Divisions)	A-1b, A-2d, A-2k, A-2n, A-3d, A-3f, C-2, CB-2, CB-3a, CB-4, F-1m, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1p, R-1q, R-1r, R-1t, SY-1b, T-1a, T-1f, T-1h, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
MPT_UNL	Main Power Transformers	A-2k, CB-1f, CB-2, CB-3a, CB-4, CB-7, R-1c, R-1i, R-1p, R-1t, T-1a, T-1f, T-1h, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.

TABLE APLA-07-A.1

UNL COMPONENTS IN THE CLINTON FPRA

UNL COMPONENT ID	DESCRIPTION	PAUS	COMMENT
MS_DIV1_UNL	Main Steam (Division 1)	A-2k, A-2n, C-1, C-2, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, CB-7, R-1c, R-1i, R-1p, R-1t, T-1a, T-1e, T-1f, T-1g, T-1h, T-1k, T-1m	PAUs based on SLICE data query using System Code & Segregation Codes.
MS_DIV2_UNL	Main Steam (Division 2)	A-3d, A-3f, CB-2, CB-3a, CB-4, CB-7, R-1i, R-1p, R-1q, R-1t, T-1a, T-1e, T-1f, T-1g, T-1h, T-1k, T-1m	PAUs based on SLICE data query using System Code & Segregation Codes.
MS_SHUTOFF_UNL	Main Steam Shutoff Valves (Division 2)	A-1a, A-1b, A-2b, A-2f, A-3a, A-3d, A-3f, C-1, C-2, CB-1g, CB-2, CB-3a, CB-4, CB-6b, R-1i, R-1p, R-1t, T-1a, T-1e, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
MS_UNL	Main Steam (All Divisions)	A-2k, A-2n, A-3d, A-3f, C-1, C-2, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, CB-7, R-1c, R-1i, R-1p, R-1q, R-1t, T-1a, T-1e, T-1f, T-1g, T-1h, T-1k, T-1m	PAUs based on SLICE data query using System Code & Segregation Codes.
OG_DIV1_UNL	Off Gas (Division 1)	A-2k, CB-1f, CB-2, CB-3a, CB-4, R-1c, R-1h, R-1i, R-1p, R-1t, T-1f, T-1g, T-1h, T-1j, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
OG_DIV2_UNL	Off Gas (Division 2)	A-3d, A-3f, CB-2, R-1c, R-1h, R-1i, R-1p, R-1q, R-1t, T-1f, T-1g, T-1h, T-1j, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
OSP_UNL	Offsite Power	A-1a, A-1b, A-3d, A-3f, CB-2, CB-3a, CB-4, CST-1, R-1c, R-1h, R-1i, R-1p, R-1r, R-1t, SY-1b, T-1f, T-1h, TY-1b, Y-1	PAUs based on SLICE data query using System Codes & Segregation Codes. Added SY-1a based on equipment locations.
RAT_SVC_UNL	RAT Static VAR Compensator (SVC)	TY-1a	Assumed to be limited to RAT PAU.
RD_DIV1_UNL	CRD (Division 1)	A-1a, A-1b, A-2d, A-2k, A-2n, A-6, C-1, C-2, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, CB-7, F-1m, R-1c, R-1i, R-1p, R-1t, T-1a, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
RD_DIV2_UNL	CRD (Division 2)	A-1a, A-1b, A-2k, A-3d, A-3f, C-2, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, R-1c, R-1i, R-1p, R-1t, T-1a, T-1f, T-1h, T-1k	PAUs based on SLICE data query using System Code & Segregation Codes.
RH_DIV2_UNL	RHR (Division 2)	A-1a, A-1b, A-1e, A-2b, A-2k, A-3a, A-3b, A-3c, A-3d, A-3f, A-6, C-1, C-2, CB-1c, CB-1d, CB-1e, CB-1f, CB-1g, CB-2, CB-3a, CB-4, CB-6b, R-1c, R-1i, R-1p, R-1t, T-1a	PAUs based on SLICE data query using System Code & Segregation Codes. PAU A-2b is excluded from this UNL. Cable 1RH57D (RHR Div 2 Cable) is routed through a Junction Box (1JB619) in this PAU, but this Junction Box is not associated with any PRA equipment.

TABLE APLA-07-A.1

UNL COMPONENTS IN THE CLINTON FPRA

UNL COMPONENT ID	DESCRIPTION	PAUS	COMMENT
RM_DIV1_UNL	Radiation Monitoring (Division 1)	A-1a, A-1b, A-2d, A-2k, A-2m, A-2n, A-3f, C-1, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1g, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6b, CB-6d, CB-7, D-10, F-1m, F-1p, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1p, R-1q, R-1r, R-1t, SY-1b, T-1a, T-1c, T-1e, T-1f, T-1g, T-1h, T-1k, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
RM_DIV2_UNL	Radiation Monitoring (Division 2)	A-1b, A-2k, A-2m, A-3d, A-3e, A-3f, A-3g, C-1, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, F-1a, F-1m, F-1p, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1p, R-1q, R-1t, T-1a, T-1e, T-1f, T-1g, T-1h, T-1k, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
RP_DIV1_UNL	Reactor Protection (Division 1)	A-1b, A-2k, A-2n, C-1, C-2, CB-1c, CB-1d, CB-1e, CB-1f, CB-1g, CB-1i.1, CB-1i.2, CB-3a, CB-3f, CB-4, CB-5a, CB-5c, CB-6b, CB-6d, CB-7	PAUs based on SLICE data query using System Code & Segregation Codes.
RP_DIV2_UNL	Reactor Protection (Division 2)	A-1a, A-1b, A-3d, A-3f, C-1, C-2, CB-1c, CB-1d, CB-1e, CB-1f, CB-1g, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-3e, CB-4, CB-5c, CB-6b, CB-6d, CB-7	PAUs based on SLICE data query using System Code & Segregation Codes.
RP_DIV4_UNL	Reactor Protection (Division 4)	A-3f, C-2, CB-3a, CB-3b, CB-4, CB-7	PAUs based on SLICE data query using System Code & Segregation Codes.
SA_DIV1_UNL	Service Air (Division 1)	A-1a, A-1b, A-2k, A-2n, C-1, C-2, CB-1e, CB-1f, CB-2, CB-3a, CB-4, R-1i, R-1o, R-1p, R-1q, R-1r, R-1s, R-1t, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes. PAUs A-3d and A-3f removed from the list. Only cable 1SA12E is located in these PAUs, but this cable is associated with control of solenoid valve 1SA036, which is not a PRA component. Therefore, this UNL is excluded from these PAU.
SA_DIV2_UNL	Service Air (Division 2)	A-3d, A-3f, C-2, CB-2, CB-3a, CB-4, R-1i, R-1p, R-1q, R-1t, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
SA_UNL	Service Air (All Divisions)	A-1a, A-1b, A-2k, A-2n, A-3d, A-3f, C-1, C-2, CB-1e, CB-1f, CB-2, CB-3a, CB-4, R-1i, R-1o, R-1p, R-1q, R-1r, R-1s, R-1t, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.
SC_DIV1_UNL	SLC (Division 1)	A-2k, A-2n, C-1, C-2, CB-1f, CB-1g, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6b, CB-6d	PAUs based on SLICE data query using System Code & Segregation Codes.
SC_DIV2_UNL	SLC (Division 2)	A-1a, A-1b, A-3d, A-3f, C-1, C-2, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-5c, CB-6d	PAUs based on SLICE data query using System Code & Segregation Codes.

ATTACHMENT
Response to Request for Additional Information

TABLE APLA-07-A.1

UNL COMPONENTS IN THE CLINTON FPRA

UNL COMPONENT ID	DESCRIPTION	PAUS	COMMENT
SX_DIV1_UNL	Shutdown Service Water (Division 1)	A-1a, A-1b, A-2a, A-2b, A-2c, A-2d, A-2k, A-2m, A-2n, A-6, C-2, CB-1a, CB-1c, CB-1e, CB-1f, CB-1g, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, D-2, D-5a, F-1m, F-1p, M-1, M-2c, R-1p, R-1t, RCT-1, T-1h, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
SX_DIV2_UNL	Shutdown Service Water (Division 2)	A-1a, A-1b, A-2e, A-3a, A-3b, A-3d, A-3e, A-3f, A-3g, C-2, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-5c, CB-6d, D-6a, F-1a, F-1m, F-1p, M-2b, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
TW_UNL	Treated Water	SY-1b	PAUs based on SLICE data query using System Code & Segregation Codes.
UAT_UNL	Unit Auxiliary Transformers	A-3d, A-3f, CB-2, CB-3a, R-1i, R-1p, R-1t, T-1a, T-1f, T-1h, Y-1	Based on 1AP01* & 1AP02* cable routings.
VQ_DIV1_UNL	Drywell Purge (Division 1)	A-1b, A-2d, A-2k, A-2m, A-2n, C-1, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, D-5a, F-1m, F-1p	PAUs based on SLICE data query using System Code & Segregation Codes.
VQ_DIV2_UNL	Drywell Purge (Division 2)	A-3f, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, D-6a	PAUs based on SLICE data query using System Code & Segregation Codes.
VQ_UNL	Drywell Purge (All Divisions)	A-1b, A-2d, A-2k, A-2m, A-2n, A-3f, C-1, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, D-5a, D-6a, F-1m, F-1p	PAUs based on SLICE data query using System Code & Segregation Codes.
VR_DIV1_UNL	Containment Building HVAC (Division 1)	A-1a, A-1b, A-2d, A-2k, A-2m, A-2n, C-1, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, D-10, F-1m, F-1p, R-1c, R-1i, R-1p, R-1t, T-1a	PAUs based on SLICE data query using System Code & Segregation Codes.
VR_DIV2_UNL	Containment Building HVAC (Division 2)	A-3d, A-3f, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-2, CB-3a, CB-4, CB-5c, CB-6d, D-10	PAUs based on SLICE data query using System Code & Segregation Codes.
VR_UNL	Containment Building HVAC	A-1a, A-1b, A-2d, A-2k, A-2m, A-2n, A-3d, A-3f, C-1, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, CB-5c, CB-6d, D-10, F-1m, F-1p, R-1c, R-1i, R-1p, R-1t, T-1a	PAUs based on SLICE data query using System Code & Segregation Codes.
WO_DIV1_UNL	Chilled Water (Division 1)	A-1a, A-1b, A-2d, A-2k, A-2n, C-1, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, D-10, F-1a, F-1m, F-1p, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1p, R-1q, R-1r, R-1s, T-1a, T-1e, T-1f, T-1g, T-1h, T-1k, T-1m	PAUs based on SLICE data query using System Code & Segregation Codes.

TABLE APLA-07-A.1

UNL COMPONENTS IN THE CLINTON FPRA

UNL COMPONENT ID	DESCRIPTION	PAUS	COMMENT
WO_DIV2_UNL	Chilled Water (Division 2)	A-1a, A-1b, A-3d, A-3f, C-1, C-2, CB-1b, CB-1c, CB-1d, CB-1e, CB-1f, CB-1i.1, CB-1i.2, CB-2, CB-3a, CB-4, D-10, F-1a, F-1m, R-1a, R-1c, R-1h, R-1i, R-1m, R-1o, R-1p, R-1q, R-1r, R-1s, R-1t, T-1a, T-1e, T-1f, T-1g, T-1h, T-1k, T-1m	PAUs based on SLICE data query using System Code & Segregation Codes.
WS_DIV1_UNL	Service Water (Division 1)	A-2k, A-2m, CB-1f, CB-2, CB-3a, CB-4, F-1m, F-1p, M-2c, R-1m, R-1o, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
WS_DIV2_UNL	Service Water (Division 2)	A-3d, A-3e, A-3f, CB-1e, CB-1f, CB-2, CB-3a, CB-4, D-10, F-1a, F-1m, F-1p, M-2c, R-1i, R-1p, R-1q, R-1t, T-1e, T-1f, T-1h, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
WS_UNL	Service Water (All Divisions)	A-2k, A-2m, A-3d, A-3e, A-3f, CB-1e, CB-1f, CB-2, CB-3a, CB-4, D-10, F-1a, F-1m, F-1p, M-2c, R-1i, R-1m, R-1o, R-1p, R-1q, R-1t, T-1e, T-1f, T-1h, Y-1	PAUs based on SLICE data query using System Code & Segregation Codes.
WT_UNL	TBCCW	CB-2, CB-3a, CB-4, R-1c, R-1i, R-1p, R-1t, T-1a, T-1f, T-1h	PAUs based on SLICE data query using System Code & Segregation Codes.

TABLE APLA-07-A.2

NO UNL RICT CALCULATION SENSITIVITY RESULTS FOR FPRA

(FPIE & SEISMIC RESULTS REMAIN UNCHANGED FOR SENSITIVITY ANALYSIS)

			BASE RESULTS (CL-LAR-010, Rev. 0)				APLA-08 SENSITIVITY RESULTS (UNL FAILURES REMOVED)		
CASE	**DESCRIPTION**	**CDF vs. LERF**	**DELTA FPIE CDF**	**DELTA FIRE CDF**	**DELTA SEISMIC CDF**	**RICT ESTIMATE (DAYS) (30 DAY MAXIMUM)**	**DELTA FIRE CDF**	**RICT ESTIMATE (DAYS) (30 DAY MAXIMUM)**	**RICT % CHANGE**
3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	CDF	1.35E-05	1.12E-04	6.40E-06	27.7	8.87E-05	30.0	8.37%
3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	CDF	2.01E-05	2.59E-04	6.40E-06	12.8	2.33E-04	14.1	9.85%
3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable.	CDF	5.86E-05	5.00E-04	6.40E-06	6.5	3.39E-04	9.0	40.06%
3.5.1.B	High Pressure Core Spray (HPCS) System inoperable.	CDF	1.35E-05	1.12E-04	6.40E-06	27.7	8.87E-05	30.0	8.37%
3.5.1.F	One required ADS valve inoperable AND One low pressure ECCS injection/spray subsystem inoperable.	CDF	5.86E-05	5.03E-04	6.40E-06	6.4	3.41E-04	9.0	39.82%
3.7.1.B	Division 1 or 2 SX subsystem inoperable.	CDF	1.68E-05	3.03E-04	6.40E-06	11.2	2.88E-04	11.7	4.80%
3.8.1.B	One required DG inoperable.	CDF	1.14E-05	2.68E-04	6.40E-06	12.8	2.68E-04	12.8	0.03%
3.8.1.D	One offsite circuit inoperable AND One required DG inoperable.	CDF	2.99E-05	3.33E-04	6.40E-06	9.9	3.37E-04	9.8	-1.10%
3.8.4.A	One battery charger on Division 1 or 2 inoperable.	CDF	1.80E-05	3.68E-04	6.40E-06	9.3	3.38E-04	10.1	8.16%
3.8.9.B	One or more Division 1 or 2 uninterruptible AC bus distribution subsystems inoperable.	CDF	1.84E-04	7.26E-04	6.40E-06	4.0	5.62E-04	4.9	21.90%

b) Fire PRA HRA

The uncertainty associated with the FPRA HRA is evaluated as part of the parametric uncertainty evaluation which is documented in Appendix A of CL-PRA-021.12, Uncertainty & Sensitivity Analysis Notebook. For the independent and dependent HEP values, a Lognormal error distribution was used for the parametric uncertainty evaluation.

With regards to the potential impact on the TSTF-505 RICT program, DRA/APLA RAI 04 discusses use of the parametric mean (instead of the point-estimate) and the potential impact on the risk-informed applications.

DRA/APLB RAI 01 - Reduced Transient Heat Release Rates (HRRs)

The key factors used to justify using transient fire reduced HRRs below those prescribed in NUREG/CR-6850 are discussed in the June 21, 2012, letter from Joseph Giitter, U.S. NRC, to Biff Bradley, NEI, "Recent Fire PRA Methods Review Panel Decisions and Electrical Power Research Institute (EPRI) 1022993, Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires," (ADAMS Package Accession No. ML12172A406).

If any reduced transient HRRs below the bounding 98% HRR of 317 kW from NUREG/CR-6850 were used, discuss the key factors used to justify the reduced HRRs. Include in this discussion:

a) Identification of the fire areas where a reduced transient fire HRR is credited and what reduced HRR value was applied.

b) A description for each location where a reduced HRR is credited, and a description of the administrative controls that justify the reduced HRR including how location-specific attributes and considerations are addressed. Include a discussion of the required controls for ignition sources in these locations and the types and quantities of combustible materials needed to perform maintenance. Also, include discussion of the personnel traffic that would be expected through each location.

c) The results of a review of records related to compliance with the transient combustible and hot work controls.

Response

a) For the Cable Spreading Rooms (CSRs) (Fire Zones CB-2 and CB-4), three potential transient fires were assessed: 60kW, 145kW, and 317kW. For each transient fire size, a detailed fire modeling calculation was developed that analyzed the transient fire growth and spread up to the specified heat release rate (e.g., one calculation analyzed a fire growing to 60kW, another calculation up to 145kW, and a final calculation up to 317kW). The results of the three transient fire sizes were averaged and the average probability of target damage (i.e., product of Severity Factor and Non-Suppression Probability) was used in the FPRA model.

The Cable Spreading Rooms are located on the 781' elevation of the Control Building and are relatively small rooms with typically two stacks of cable trays along the walls (with cable trays typically arranged from floor to ceiling with 1.5' separation).

The bounding heat release rate for transient fires provided in NUREG/CR-6850 was used in locations where a large amount of combustibles could be stored. These include most general locations in the plant such as large equipment areas, general floor areas, and storage areas. Many locations in the plant are small rooms where walking / storage room is very limited (e.g., narrow corridors, cable spreading rooms). A smaller transient would be expected in these types of areas. The use of a lower heat release rate is considered applicable based on a detailed review of the NUREG/CR-6850 transient fire test.

The transient fire test used as the basis of the NUREG/CR-6850 HRR includes:

- The Nowlen Test with and without acetone
- The Van Volkinburg Test which were performed in conditions that are not consistent with the spatial characteristics of areas of concern in the FPRA
- The Chavez test which are similar to the Nowlen test with acetone
- The Lee test which are similar to the Nowlen test with acetone
- The Cline test which were not considered in NUREG/CR-6850

Based on the test characterizations, the Van Volkinburg and Cline test are not considered applicable.

The remaining tests are grouped as follows:

- Transient fires without wood or acetone: HRR 12 – 60 KW
- Transient fires with acetone: HRR 32 - 145 KW
- Transient fires with wood: HRR 186 – 327 KW

Given the above, the 60kW and 145kW transient fires are considered representative for transients in small areas that lack the space for storage and multiple pieces of equipment to perform maintenance on.

b) Procedure OP-AA-201-009, Control of Transient Combustible Material, provides the administrative controls for handling and limiting the use of combustible materials inside critical buildings and / or the Power Block. Transient Combustible Permits (TCPs) are requested and reviewed by the Fire Marshall and procedure provides specific guidelines for storing and transporting combustible materials. Attachment 5 of the procedure provides Clinton-specific information (e.g., list of Transient Combustible Free Zones (TCFZs)). Per the procedure, combustible material is limited to those materials required to support work activity and all waste resulting from the work activity are immediately removed from the area following completion of the task.

Table APLB-01-1 summarizes the TCPs created for the Cable Spreading Rooms from 2016 to current. The typical types of combustible materials used in the Cable Spreading Rooms include metal scaffolding, misc. cables for cable replacement activities, computers, tools, signage, and other misc. equipment need for routine maintenance activities. Table APLB-01-1 also summarizes the amount of combustible load introduced with each TCP (in units of BTU). As shown in the table, the TCPs requested for the Cable Spreading Rooms introduced negligible combustible load to the areas when compared to the standard cable loading (BTU) of the areas (i.e., largest permit increase combustible loading by ~1.3% for 20 days). Therefore, given the typical types of combustible material used in these areas for normal work activities, as well as the small combustible load associated with these materials, use of three potential transient fires (i.e., 60kW, 145kW, and 317kW) is appropriate.

In terms of personnel traffic expected in the Cable Spreading Rooms, these rooms can be used as a pathway to the Auxiliary Electric Equipment Room (AEER), but the AEER is also accessible through the Auxiliary Building, which is the preferred pathway for plant personnel. The Clinton FPRA utilizes an "Average" transient influence factor for the "occupancy" category for these areas.

c) Action Request (AR) Reports from 2016 to current were reviewed to identify any compliance issues related to transient combustible controls. The keyword searches and the results of the searches are summarized below:

- "housekeeping" (685 records, 11 records reviewed in detail, none related to transient combustible controls in Cable Spreading Rooms)
- "combustible" (53 records, 4 records reviewed in detail, none related to transient combustible controls in Cable Spreading Rooms)
- "violation" (60 records, none related to transient combustible controls)

Based on the review of the AR Reports, there have been no compliance issues or violations associated with the use of transient combustibles in the Cable Spreading Rooms since 2016. Therefore, use of the lower HRRs is appropriate for these areas.

TABLE APLB-01-1

TRANSIENT COMBUSTIBLE PERMITS FOR CLINTON POWER STATION

(2016 – CURRENT)

FIRE ZONE CB-2: DIVISION 2 CABLE SPREADING ROOM **BASE CABLE LOADING = 456773790 BTU**					
TCP #	**COMB. LOAD**	**UNITS**	**%INCREASE**	**DURATION**	**TYPE OF COMBUSTIBLE**
TCP-90	30000	BTU	0.01%	01/22/16 - 06/02/16	Pearl-weave fencing material for 3 scaffolds used for control cable mod
TCP-360	150500	BTU	0.03%	04/24/17 - 04/28/17	Blue matting, tools, signage, FME covers, cart & tub for VX coil replacement
TCP-366	1063100	BTU	0.23%	04/24/17 - 04/28/17	Computer (unit)
TCP-900	4637144	BTU	1.02%	09/26/19 - 10/10/19	Refueling Outage C1R19: computer (unit), cart with 10lb plastic wheels, UPS cabinet (plastic), misc. cabling
TCP-906	5887400	BTU	1.29%	10/05/19 - 10/25/19	Computer (unit), misc. cabling, cart, & paper
FIRE ZONE CB-4: DIVISION 1 CABLE SPREADING ROOM **BASE CABLE LOADING = 275457088 BTU**					
TCP #	**COMB. LOAD**	**UNITS**	**%INCREASE**	**DURATION**	**TYPE OF COMBUSTIBLE**
TCP-91	10000	BTU	< 0.01%	01/22/16 - 06/02/16	Pearl-weave fencing material for scaffold used for control cable mod
TCP-468	167000	BTU	0.06%	07/18/17 - 07/28/17	cart, cooler, tub, matting, tools, signage, parts & bucket for cooler replacement

<u>DRA/APLB RAI 02 – Treatment of Sensitive Electronics</u>

FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics" (ADAMS Accession No. ML13322A085) provides supplemental guidance for application of the damage criteria provided in Sections 8.5.1.2 and H.2 of NUREG/CR-6850, Volume 2, for solid-state and sensitive electronics.

a) Describe the treatment of sensitive electronics for the FPRA and explain whether it is consistent with the guidance in FAQ 13-0004, including the caveats about configurations that can invalidate the approach (i.e., sensitive electronics mounted on the surface of cabinets and the presence of louver or vents).

b) If the approach cannot be justified to be consistent with FAQ 13-0004, then justify that the treatment of sensitive electronics has no impact on the RICT calculations.

c) As an alternative to item b above, add an implementation item to replace the current approach with an acceptable approach prior to the implementation of the RICT program. Include a description of the replacement method along with justification that it is consistent with NRC accepted guidance.

Response

a) The Clinton FPRA treats sensitive electronics as contained within the electrical cabinets such that the cabinet walls, top, front, and back doors shield the components from radiant energy of an exposure fire and the thermoset failure criteria is retained for these components, which is consistent to FAQ 13-0004. The caveats (e.g., surface-mounted electronics) specified in FAQ 13-0004 are the only aspects of the FAQ that are not explicitly evaluated.

Although the caveats in FAQ 13-0004 are not explicitly evaluated, sensitive electronics are treated as fire-induced targets within the specified Zone of Influence (ZOI) of the various ignition sources that are analyzed with detailed fire modeling. Even though the sensitive electronics will fail sooner than all other targets in the ZOI, creating a separate fire scenario where the targets are limited to sensitive electronics (and the original ignition source) would have a negligible impact on the overall fire risk due to the functions associated with the sensitive electronics (e.g., a single logic channel failed for a given system).

Therefore, the gap would be cases where the exposed sensitive electronics are at a substantial distance away from the ignition source such that the ZOI of the ignition source would be insufficient (i.e., the ZOI of the ignition source using the sensitive electronics damage criteria extends beyond the ZOI using the thermoset criteria). For the Clinton FPRA, if an ignition source has the potential to grow beyond the analyzed ZOI, then a "beyond ZOI" scenario was developed and a full room burnout was assumed (i.e., if the specified horizontal ZOI is 10 ft, but the fire has the potential to grow beyond that distance, a new scenario was developed and all targets within the PAU are assumed to be failed). Therefore, the "beyond ZOI" scenarios capture the potential gap where the sensitive electronics are outside the ZOI of the ignition source.

Additionally, specific cabinets housing sensitive electronics were identified for further evaluation. Based on the sampling results presented in Table APLB-02-1, the caveats presented in FAQ 13-0004 did not pose a challenge to the general treatment of sensitive electronics within the Fire PRA for one or more reasons:

1. No impact to a PRA credited function. Sensitive electronics were identified apart from the Fire PRA and many were subsequently determined to be associated with non-credited functions such as turbine-generator protection or room cooling not required within the PRA mission time.

2. No impact to the cabinet housing the sensitive electronics. The nearby ignition sources are non-propagating, or the cabinet / functional failure was included in the applicable scenarios. Transient ignition sources were not generally postulated next to fixed ignition sources and would add an insignificant frequency contribution to the existing fire scenario(s) for the cabinet housing the sensitive electronics.

TABLE APLB-02-1

SAMPLING RESULTS FOR SENSITIVIE ELECTRONICS IDENTIFICATION

FIRE ZONE	CABINET HOUSING SENSITIVE ELECTRONICS	DISPOSITION
CB-3A (AEER)	1PL90JA	1PL90JA was failed in the 'orange' (beyond ZOI) scenarios. There are no credited PRA functions impacted by this panel.
A-2N (Div. 1 SWGR Room)	1PL65JA	1PL65JA, which houses a temperature transmitter for room cooling, was included as a target for nearby cabinet fire scenarios based on the ZOI using the thermoset cable damage criteria. Also, this cabinet was included in the "orange" (beyond ZOI) scenarios. However, switchgear room cooling is not required in the PRA and therefore there are no credited PRA functions impacted by this panel.
CB-6A (MCR)	1H13-P614 1H13-P632 1H13-P642 1H13-P678 1H13-P826	These panels are located in the MCR and their associated 'orange' (beyond ZOI) scenario is modeled as a MCR abandonment scenario. Additionally, several of these panels were included as targets for nearby cabinet fire scenarios based on the ZOI using the thermoset cable damage criteria.

In summary, the treatment of sensitive electronics in the Clinton FPRA is appropriate and suitable for risk-informed applications.

b) N/A – See responses to part (a).

c) N/A – See responses to part (a).

<u>DRA/APLB RAI 03 - Minimum Joint Human Error Probability</u>

NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines- Final Report," (ADAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple human failure events (HFEs) in HRAs.

NUREG-1921 refers to Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," (ADAMS Accession No. ML051160213), which recommends that joint HEP values should not be below 1E-5. Table 4-4 of 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. Therefore, the guidance in NUREG-1921 allows for assigning joint HEPs that are less than 1E-5, but only through assigning proper levels of dependency.

The LAR does not provide this information and does not explain what minimum joint HEP value is currently assumed in the FPRA. Also, even if the assumed minimum joint HEP values are shown to have no impact on the current FPRA risk estimates, it is not clear to the NRC staff how it will be ensured that the impact remains minimal for future PRA model revisions. In light of these observations:

a) Explain what minimum joint HEP value was assumed in the FPRA.

b) If a minimum joint HEP value less than 1E-05 was used in the FPRA, then provide a description of the sensitivity study that was performed and the quantitative results that justify that the minimum joint HEP value has no impact on the RICT application.

c) If, in response part (b), if it cannot be justified that the minimum joint HEP value has no impact on the application, then provide the following:

   i. Confirm that each joint HEP value used in the FPRA below 1E-5 includes its own justification that demonstrates the inapplicability of the NUREG-1792 lower value guideline (i.e., using such criteria as the dependency factors identified in NUREG-1921 to assess level of dependence). Provide an estimate of the number of these joint HEP values below 1.0E-5, discuss the range of values, and provide at least two different examples where this justification is applied.

   ii. If joint HEP values used in the FPRA below 1E-5 cannot be justified, add an implementation item to set these joint HEPs to 1E-5 in the FPRA prior to the implementation of the RICT program.

Response

a) The HRA Dependency Analysis methodology used for the Clinton FPRA has been evaluated as part of the April 2018 Peer Review [7] 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 FPRA 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] acknowledges that the floor value of 1E-05 stated in NUREG-1792 [3] 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.

   The FPRA HRA Dependency Analysis is documented in Section 5.1 of CL-PRA-021.09, Fire HRA Notebook.

b) A sensitivity analysis was performed where all JHEPs nominally less than 1E-05 were escalated to 1E-05. Table APLB-03-1 summarizes the results of this sensitivity on a sample of TSTF-505 RICT calculations. The sample TS cases were selected because they represent all cases documented in CL-LAR-010 [6] that produced a RICT estimate less than 30 days.

As shown in Table APLB-03-1, the delta Fire CDF values are essentially unchanged from the base Fire PRA model (which uses the 1E-06 / 5E-07 JHEP floor values). Since the delta values are essentially unchanged, the RICT estimates (in days) are also essentially unchanged.

Therefore, the JHEP floor values used in the FPRA 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.

c) N/A – See response to part (b) for details.

TABLE APLB-03-1

JHEP FLOOR VALUE RICT CALCULATION SENSITIVITY RESULTS FOR FPRA

(FPIE & SEISMIC RESULTS REMAIN UNCHANGED FOR SENSITIVITY ANALYSIS)

			BASE RESULTS (CL-LAR-010, Rev. 0)				APLB-04 SENSITIVITY RESULTS (MINIMUM JHEP = 1E-05)		
CASE	**DESCRIPTION**	**CDF vs. LERF**	**DELTA FPIE CDF**	**DELTA FIRE CDF**	**DELTA SEISMIC CDF**	**RICT ESTIMATE (DAYS) (30 DAY MAXIMUM)**	**DELTA FIRE CDF**	**RICT ESTIMATE (DAYS) (30 DAY MAXIMUM)**	**RICT % CHANGE**
3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	CDF	1.35E-05	1.12E-04	6.40E-06	27.7	1.12E-04	27.6	-0.22%
3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	CDF	2.01E-05	2.59E-04	6.40E-06	12.8	2.59E-04	12.8	-0.01%
3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable.	CDF	5.86E-05	5.00E-04	6.40E-06	6.5	5.01E-04	6.5	-0.02%
3.5.1.B	High Pressure Core Spray (HPCS) System inoperable.	CDF	1.35E-05	1.12E-04	6.40E-06	27.7	1.12E-04	27.6	-0.22%
3.5.1.F	One required ADS valve inoperable AND One low pressure ECCS injection/spray subsystem inoperable.	CDF	5.86E-05	5.03E-04	6.40E-06	6.4	5.03E-04	6.4	-0.02%
3.7.1.B	Division 1 or 2 SX subsystem inoperable.	CDF	1.68E-05	3.03E-04	6.40E-06	11.2	3.03E-04	11.2	-0.02%
3.8.1.B	One required DG inoperable.	CDF	1.14E-05	2.68E-04	6.40E-06	12.8	2.68E-04	12.8	0.00%
3.8.1.D	One offsite circuit inoperable AND One required DG inoperable.	CDF	2.99E-05	3.33E-04	6.40E-06	9.9	3.33E-04	9.9	-0.01%
3.8.4.A	One battery charger on Division 1 or 2 inoperable.	CDF	1.80E-05	3.68E-04	6.40E-06	9.3	3.68E-04	9.3	-0.02%
3.8.9.B	One or more Division 1 or 2 uninterruptible AC bus distribution subsystems inoperable.	CDF	1.84E-04	7.26E-04	6.40E-06	4.0	7.26E-04	4.0	-0.02%

<u>DRA/APLB RAI 04 – Well-Sealed MCC Cabinets</u>

Guidance in Frequently Asked Question 08-0042 from Supplement 1 of NUREG/CR-6850 applies to electrical cabinets below 440 V. With respect to Bin 15 as discussed in Chapter 6, 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 Electrical Panels Greater than 440 V" (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.

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

b) If well-sealed cabinets less than 440 V are included in the Bin 15 count of ignition sources, provide justification for using this approach as this is contrary to the guidance.

<u>Response</u>

a) The Clinton FPRA assumes that all MCC cabinets <u>are not</u> well-sealed (i.e., all fires originating from MCC cabinets are assumed to damage external targets). Therefore, FAQ 14-0009 does not apply to the Clinton FPRA. In addition, all MCCs at Clinton are 480V.

   This treatment is not explicitly discussed in the Fire PRA Notebooks since the FAQ is not used and MCCs are treated like a vented electrical panel.

b) The Clinton FPRA assumes that all cabinets <u>are not</u> well-sealed (i.e., all fires originating from the electrical cabinets are assumed to damage external targets). Therefore, electrical cabinets are apportioned Bin 15 counts using the appropriate counting methodology.

<u>DRA/APLC RAI 01 – Determination of Seismic LERF "Penalty"</u>

Section 2.3.1, Item 7, of NEI 06-09, Revision 0-A (ADAMS Accession No. ML12286A322), states that the "impact of other external events risk shall be addressed in the [Risk Managed TS] RMTS program," and explains that one method to do this is by "performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated RICT." The NRC staff's safety evaluation for NEI 06-09 (ADAMS Accession No. ML071200238) states that "[w]here [probabilistic risk assessment] PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT."

A seismic PRA (SPRA) model is not available for CPS, Unit 1, and the seismic hazard cannot be screened out for the RICT application. Details of the approach for determining the seismic "penalty" are provided in Section 3 of Enclosure 4 to the LAR which states that the seismic Large Early Release Frequency (SLERF) estimate is conservatively used in the RICT process and that the proposed SLERF penalty was determined by multiplying the estimated seismic Core Damage Frequency (SCDF) by an estimated average seismic conditional large early release probability (CLERP). The LAR explains that the average seismic CLERP (SCLERP) was determined using: (1) an estimation of the breakdown of SCDF by accident sequence type using fragility information from industry seismic PRAs, and (2) the CLERP for each accident type based on the current CPS internal events PRA model. The percent contribution of different accident types to SCDF was then multiplied by the internal events PRA CLERP to produce a sequence weighted average SCLERP. The LAR indicates that, in general, for a given plant, the SPRA does not necessarily produce the same distribution of accident scenarios as the internal events PRA. The LAR explains that containment isolation failures were evaluated to ensure that random and seismically-induced failures leading directly to LERF are reasonably addressed in the estimation of the average SCLERP.

The LAR presents a graphical approach in Figure E4-1, "Contribution of SCDF By Accident Sequence Type," for determining the contribution of different accident types to SCDF based on fragility information from cited industry seismic PRAs. The graphic resembles an event tree in which the top events and end states are used to define the accident types of interest and the branch point probabilities are used to calculate the percent contribution of each accident type to the total SCDF. The branch point probabilities appear to be determined by the seismic failure of components relevant to the branch points based on "review of industry fragility information." The details involved in assigning a failure probability to each branch point in the graphic based on "review of industry fragility information" is not clear. Given this lack of clarity, address the following:

a) Regarding Figure E4-1, describe how the branch point probabilities were determined. Discuss the information used from the cited industry seismic PRA reports, the basis for the selection of representative SSCs for developing the branch point probabilities and the basis for the applicability of that information to CPS. The discussion should provide the range of seismic capacities collected from industry seismic PRA reports and the basis to support the licensee's selections.

b) Explain how the contribution of different scenarios and the average seismic CLERP developed for CPS compares to the seismic CLERP for plants most like CPS when the seismic CLERP is directly computed from the SCDF and SLERF for those plants.

c) Justify that the seismic LERF "penalties" provided in the submittal to support RICT calculations for CPS are conservative. The justification should include an explanation of the use of the internal events SCLERP values given that the internal events results only reflect random failures and do not capture the seismic failures of SSCs important for containment performance. The justification should also include a discussion of the impact of the estimate seismic "penalties" on the calculations for the proposed RICTs. If the approach to estimating seismic LERF cannot be justified as conservative for this application, then provide, with justification, the conservative SLERF "penalties" for use in RICT calculations.

Response

Response to Question APLC 1a

The question is correct that Figure E4-1 of the LAR is a graphical presentation of the approach used to apportion the calculated SCDF into contributions of different accident sequence type so that SCLERPs can be assigned consistent with each accident sequence type. The entries shown at each nodal branch in the Figure E4-1 graphic are not probability values but rather the median seismic acceleration failure value in units of peak ground acceleration (PGA) used to calculate the failure probabilities of the branch. A convolution calculation (i.e., multiplication of hazard function and fragility function) is performed using a spreadsheet to produce the accident sequence type frequencies to determine the percentage breakdown of the SCDF. The convolution calculation is typical of such simple calculations and applies the same approaches as the NRC used in the GI-199 risk assessment [9]. The attributes of the convolution calculation for the SCDF apportionment shown in Figure E4-1 of the LAR are summarized below:

- CPS mean PGA seismic hazard curve used (as shown in Table E4-1 of Enclosure 4 of the LAR)
- Ten hazard intervals (from 0.05g to >3g PGA) are used in the SCDF accident sequence contribution convolution
- Geometric mean approach is used to define the PGA representative magnitude for each hazard interval (same approach as that discussed in the SCDF convolution discussion in Enclosure 4 of the LAR)
- Nodal probabilities for the seismic capacities (also in units of g PGA) assigned to each node of Figure E4-1 are calculated using the typical double lognormal fragility probability model used in most SPRAs (as discussed in the Seismic Failure Probabilities section of Enclosure 4 of the LAR).
- Convolution calculation of accident type frequencies includes preceding nodal complement (up-branch) probabilities where applicable. SCDF accident type contribution percentage is then determined by the individual accident type seismic-induced frequency divided by the sum of all the accident type frequencies in Figure E4-1.

The definition of the nodes in Figure E4-1 is discussed immediately following Figure E4-1 in Enclosure 4 of the LAR. Those discussions describe the type of SSCs applicable to a given node and that the lower range of seismic capacity (based on a review of industry studies) is conservatively selected for each node. For example, the offsite power seismic capacity used in the S-LOOP node of Figure E4-1 is selected as Am=0g PGA (i.e., seismic-induced LOOP is assumed to occur 100% of the time for any size earthquake).

The range of seismic capacities applicable for a given node are based on Exelon review of numerous industry studies. The studies reviewed are:

- UCID-20571 [10]
- EPRI 2013 SPRA Implementation Guide [11]
- NTTF 2.1 Seismic SPRA submittals ([12] thru [25])

The approach to the RICT seismic penalty SLERF estimation is to provide a conservative estimation of the breakdown of SCDF by accident sequence type and then apply seismically-biased CLERP estimates to each accident sequence type to arrive at a weighted-average SCLERP. The goal of this SCDF breakdown was to tend toward a conservative apportionment that gives more weight to accident types with higher SCLERP values. Given that the CPS SCDF estimate is based on a convolution of the CPS seismic hazard curve with a limiting plant HCLPF (based on the IPEEE SMA), and the fact that CPS does not have a detailed plant-specific seismic PRA to assist in estimating the spectrum of SPRA accident sequence types, the breakdown of SCDF by accident sequence type must consider representative fragility information from available industry sources. The need for the use of representative seismic fragility information is acknowledged in multiple industry and NRC guidelines (e.g., App. H of [11] and Section 2.4.2 of Volume 2 of [26]). Over 150 seismic capacity data points were compiled as part of this review and used to define the ranges of seismic capacities related to the functions addressed in Figure E4-1.

The lower ends of the capacity ranges were then selected to assign the seismic capacities to the nodes in Figure E4-1. Although it is true that the specifics of site geology and specifics of structures and equipment configurations define the precision of seismic fragility estimates, selection of the low end of capacities from review of industry studies is judged reasonable for the purposes of this SCDF apportionment and from a general sense on the conservative side given the newer vintage of the CPS plant in comparison to the majority of plants reviewed. In addition, the Clinton SSE (0.26g PGA) is at the higher end of most plants; other than Diablo Canyon in the high seismicity area of Southern California, CPS has the highest SSE of all the NTTF 2.1 seismic SPRAs reviewed.

<u>Response to Question APLC 1b</u>

The Clinton plant is a General Electric (GE) BWR-6 NSSS design with a GE Mark III containment design. The Mark III plant design is significantly different compared to that of BWR Mark I and Mark II designs. In general terms, the Mark I and Mark II designs have the potential for higher CLERP values than do Mark III designs due to differences in design. A Mark III containment design has a large PWR-style containment structure that surrounds the drywell and suppression pool structures and most key safety systems are located outside the containment. This is unlike the Mark I and Mark II designs where the reactor building housing many key safety systems encloses the drywell and suppression pool. The volume of the Mark III containment is significantly larger than that of Mark I and II primary containments. Certain Mark I and Mark II severe accident phenomena resulting in modeled high magnitude and potentially early releases (such as corium attack of drywell shell and release up through the refueling floor) do not apply to the Mark III containment design.

Only eight Mark III plants exist in the world (Clinton, Cofrentes, Grand Gulf, Kuosheng 1 & 2, Leibstadt, Perry and River Bend) and no publicly available SPRA exists for a Mark III plant. An at-power SPRA for the Grand Gulf, River Bend or Perry plants does not exist as part of past programs such as RMIEP, NUREG-1150, TAP-A45 studies, NTTF 2.1 Seismic, etc. Exelon also contacted the Leibstadt (KKL) plant in Switzerland to determine if a publicly available version of an at-power KKL SPRA exists for their plant and learned that one does not. As such, a specific comparison to results from a Mark III BWR SPRA cannot be provided.

<u>Response to Question APLC 1c</u>

The estimation of a seismic LERF contribution for the CPS RICT program is a "conservative" analysis that uses an estimated averaged seismic conditional large early release probability (SCLERP) to determine a seismic LERF that is then conservatively used in RICT assessments. A truly "bounding" estimate for seismic LERF would require assuming SCLERP = 1.0 (i.e., all core damage scenarios lead directly to LERF), which is neither reasonable nor useful; as such, the CPS SLERF "penalty" is calculated and applied with approaches that involve conservatisms. This is explained below according to the following three key aspects of the process:

- SCDF used in SLERF penalty estimation (not necessarily conservative)
- Averaged SCLERP estimation (conservativisms)
- Use of SLERF penalty in RICT calculations (conservativisms)

SCDF Used in SLERF Penalty Estimation

The SCDF value itself that is used as an input to the SLERF penalty estimation is not necessarily conservative. The SCDF value is a nominal estimate determined from a mathematical convolution of the Clinton plant PGA-based seismic hazard curve and the Clinton PGA-based plant level fragility. This convolution estimation approach is a common analysis in approximating an SCDF for use in risk-informed decision making (e.g., it is commonly used in RICT seismic penalty calculations; the NRC used this approach in the GI-199 risk assessment in [9] in absence of a current full-scope SPRA.

Averaged SCLERP Estimation

The calculation of the averaged SCLERP (a conditional probability multiplied with SCDF to estimate SLERF) involves the following two main calculational aspects and each involves conservative aspects:

- SCDF breakdown by accident type
- Seismically biased SCLERP per SCDF accident type

Figure E4-1 and associated discussions in Attachment 4 of the LAR describe the process to apportion the convolved SCDF estimate into different accident types. The goal of this SCDF breakdown was to tend toward a conservative apportionment that gives more weight to accident types with higher SCLERP values. Key conservatisms in the SCDF breakdown aspect of the averaged SCLERP estimation are:

- Seismic transients excluded: Seismic-induced loss of offsite power is assumed to occur with a probability of 1.0 for the entire seismic hazard curve. This is accomplished by conservatively assuming a median seismic capacity of 0g PGA for offsite power. This results in conservatively removing non-LOOP seismic transients from the assumed spectrum of SCDF accident types used in the averaged SCLERP calculation. Non-LOOP seismic transients would have the lowest SCLERP values of any of the Figure E4-1 SCDF accident types.

- Low seismic capacities: Seismic capacities used to model the seismic induced failures at each node of the SCDF apportionment structure shown in Figure E4-1 are selected to be at the low end of the range of seismic capacities (as described in the response to question APLC 1a). This results in assigning a larger percentage of the SCDF to accident types with higher SCLERPs.

Full Power Internal Events (FPIE)-based CLERPs are not used in the seismic penalty calculation. As discussed in Attachment 4 of the LAR, the CPS FPIE PRA logic model is used as a vehicle for calculating seismically biased CLERPs. SCLERP is assumed to be 1.0 for the direct-to-LERF SCDF accident type. Key conservatisms in the other accident sequence SCLERPs used in the seismic penalty calculation are as follows:

- S-LOOP Assumes SBO and no RPV Injection at t=0: The SCLERP for the S-LOOP SCDF accident type is modeled by quantifying with the FPIE-based logic model a Station Blackout (SBO) with no RPV injection at t=0 and no recovery of electric power or injection to the RPV or containment. This results in assuming the worst-case scenario for the S-LOOP accident type. S-LOOP scenarios with EDGs operating and steam-driven RPV injection operating initially (which have a much lower CLERP value) are excluded in this SCLERP assignment. The resulting LERF cutsets for this case are overwhelmingly post-core damage accident progression phenomena (not random equipment failures). As such, the SCLERP for this accident type is primarily determined by the likelihood of various post-core damage accident progression phenomena without any system or operator mitigation.

- S-LOOP-LOCA Disregards Low Pressure Benefit: The seismic LOOP with coincident LOCA accident type (S-LOOP-LOCA) uses the higher SCLERP of the S-LOOP above. The low pressure reactor condition associated with a LOCA reduces the probability of certain core damage progression phenomena (e.g., direct containment heating; high pressure blowdown overwhelming vapor suppression) and an S-LOOP with coincident LOCA would exhibit a slightly lower SCLERP than an S-LOOP alone. This is a minor effect.

- S-ATWS Assumes Worst Case Location and No Mitigation: The majority of unmitigated (i.e., no reactivity control) ATWS scenarios in the CPS internal events PRA result in non-LERF releases. Like the S-LOOP scenario, the seismically-biased S-ATWS SCLERP assumes no RPV injection at t=0 and no system or operator mitigations. The SCLERP assigned to this case is driven by the conditional probability (during an unmitigated isolation ATWS) of the containment failure location (i.e., below the suppression pool water line) that bypasses radionuclide scrubbing in the suppression pool.

Use of SLERF Penalty in RICT Calculations

The final aspect is the use of the SLERF penalty in the RICT process. The calculated total annual seismic LERF penalty documented in Enclosure 4 of the LAR (1.6E-6/yr) is conservatively treated as ΔSLERF and added to the internal events and fire ΔLERF for each RICT calculation, regardless of the LCO or plant configuration.

The total ΔCDF and ΔLERF associated with various RICTs (based on the RICT ICDP and ILERP limits) are listed in Table APLC-01-1 below. In comparison, the Clinton Seismic Penalty Factors are ΔCDF = 6.4E-6/yr and ΔLERF = 1.6E-6/yr.

TABLE APLC-01-1

TOTAL RICT ΔCDF AND ΔLERF

(BASED ON RICT ICDP AND ILERP LIMITS)

RICT (DAYS)	ΔCDF (/YR)	ΔLERF (/YR)
1	3.7E-03	3.7E-04
5	7.3E-04	7.3E-05
10	3.7E-04	3.7E-05
20	1.8E-04	1.8E-05
30	1.2E-04	1.2E-05

The penalty factors are a small percentage of the total ΔCDF and ΔLERF associated with RICTs, especially for the shorter timeframe RICTs. Increasing the penalty factor would have a very small impact on the RICTs for risk significant configurations (i.e., short RICTs). For example, Table APLC-01-2 and Figure APLC-01-1 below show the impact of an assumed doubling of the Clinton SCDF and SLERF penalty factors submitted in the LAR.

TABLE APLC-01-2

SEISMIC PENALTY SENSITIVITY ON RICT CALCULATIONS

RICT (DAYS)	2X SCDF PENALTY		2X SLERF PENALTY	
	RICT (DAYS)	% RICT DECREASE	RICT (DAYS)	% RICT DECREASE
1	0.998	0.2%	0.996	0.4%
5	4.96	0.9%	4.89	2.1%
10	9.8	1.7%	9.6	4.2%
20	19.3	3.4%	18.4	8.1%
30	28.5*	5.0%	26.5*	11.6%

Note to Table APLC-01-2:

* Most conservative result, assumes initial RICT calculated to exactly 30.0 days (i.e., not beyond the backstop).

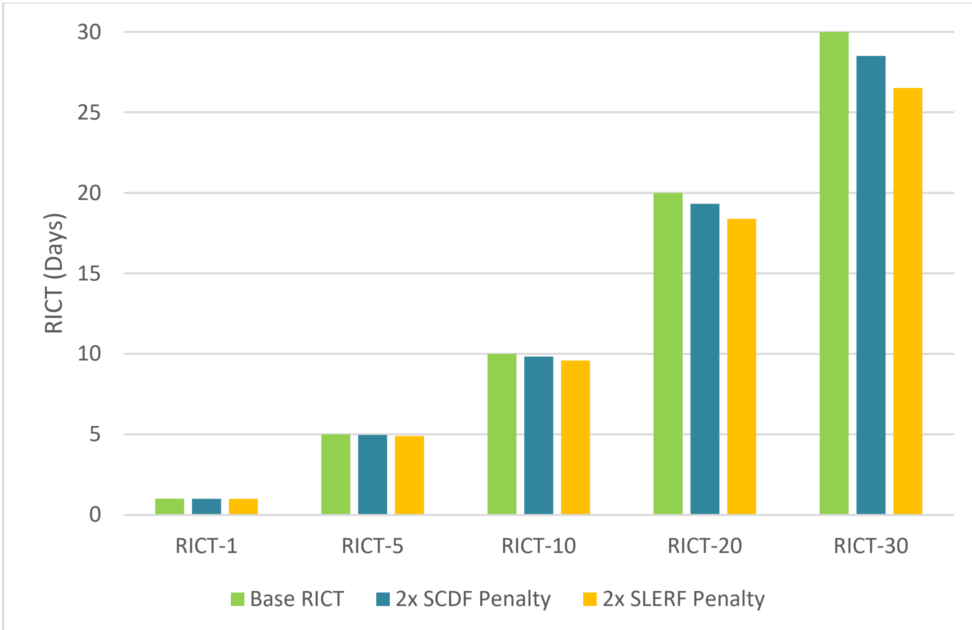

FIGURE APLC-01-1

IMPACT ON RICT OF DOUBLING CLINTON SCDF AND SLERF PENALTY FACTORS

DRA/APLC RAI 02 – External Hazard Screening

Section 2.3.1, "Configuration Risk Management Process & Application of Technical Specifications," Item 7, of NEI 06-09, Revision 0-A, states that the "impact of other external events risk shall be addressed in the RMTS program," and explains that one method to do this is by documenting prior to the RMTS program that external events that are not modeled in the PRA are not significant contributors to configuration risk. The SE for NEI 06-09 (ADAMS Accession No. ML071200238) states that "[o]ther external events are also treated quantitatively, unless it is demonstrated that these risk sources are insignificant contributors to configuration-specific risk."

LAR Enclosure 4, Section 6 provides an evaluation of other external hazards for screening in the RICT program and Section 7 concludes that no additional external hazards other than seismic events need to be added to the existing PRA model. The LAR also states that hazards are evaluated for plant configurations allowed under the RICT program. LAR Enclosure 4, Table E4-7, indicates that criterion "C1" (event damage potential is less than events for which plant is designed) was used to screen the snow hazard. LAR Enclosure 4, Table E4-7, also indicates that criterion "C1" was used to screen the sand or dust storm hazard. The LAR states that "per the IPEEE, snow storm need not be considered per the guidance contained in NUREG 1407" and "per the IPEEE, sand or dust storm need not be considered per the guidance contained in NUREG 1407," but does not explain why it does not need to be considered since NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" is IPEEE guidance and not specifically written for this application. It is unclear to the NRC staff that updated hazard information, since the IPEEE, has been considered in the screening analysis, and therefore, that the assumptions that resulted in the screening based on criterion C1 will continue to remain valid during the proposed RICTs.

LAR Enclosure 4, Table E4-7, indicates that criterion "C4" (event is included in the definition of another event) was used to screen the ice cover hazard. It is unclear to the staff that the primary effect of potential accumulation of frozen water on a lake would be to cause a loss of offsite power.

Justify the screening of the following hazards and discuss how the assumptions that resulted in the screening will continue to remain valid during the proposed RICTs.

(1) snow hazard
(2) sand or dust storm hazard
(3) ice cover hazard

Response

Snow Hazard

The CPS USAR Section 2.4.2.3 discusses snow and ice loads and accounts for the latest hazard information. The hazard is slow to develop and can be identified via monitoring and managed via normal plant processes. Therefore, in addition to Criterion C1 listed in Enclosure 4, Table E4-7 of the LAR, Criterion C5, "Event develops slowly, allowing adequate time to eliminate or mitigate the threat," also applies.

Risk management actions, such as those described in the station's adverse weather procedures, will ensure that the assumptions that resulted in the screening will continue to remain valid during the proposed RICTs.

Sand or Dust Storm Hazard

The Clinton plant is not located in an area impacted by sand or dust storms. More common wind-borne dirt can occur but poses no significant risk given the robust structures and protective features of the plant. Therefore, in addition to Criterion 1 listed in Enclosure 4, Table E4-7 of the LAR, the applicable screening Criterion C3 (Event cannot occur close enough to the plant to affect it) also applies.

Ice Cover Hazard

Per USAR Section 2.4.7, the submerged location of the ultimate heat sink pond suction prevents ice formation or ice jams from affecting the performance of the ultimate heat sink. From Enclosure 4, Table E4-7 of the LAR, the applicable screening criterion is Criterion C1, "Event damage potential is < events for which plant is designed."

DRA/APLC RAI 03 – Alternate Seismic Approach

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires, for license amendment, a description of measures taken to assure the level of detail of the systematic processes that evaluate the plant. This includes the internal events at power PRA required by 10 CFR 50.69(c)(1)(i) as well as the risk analyses used to address external events.

The proposed alternative seismic approach for Tier 1 plants is based on insights from EPRI 3002017583, which describes the risk insights derived from four case studies. These case studies compare the High Safety Significance (HSS) SSCs determined based on a seismic PRA (SPRA) against the HSS SSCs determined from other PRAs used for categorization. All four case studies included a full-power internal events (FPIE) PRA but only two included fire PRA information. Sections 3.3 through 3.5 of the EPRI report provide general information about the peer reviews conducted for the PRAs used in the case studies. However, the level of information provided is insufficient to determine whether the PRAs used in the case studies have been performed in a technically acceptable manner.

The staff has previously used additional supporting information to support its decision on the technical acceptability of the PRAs used in the case studies as well as details of the conduct of the case studies. This information is included in the supplements to the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, LAR for the adoption of 10 CFR 50.69. The supplement dated July 1, 2019 (ADAMS Accession No. ML19183A012) clarified the alternate seismic approach (see response to RAI 4). The supplement dated July 19, 2019 (ADAMS Accession No. ML19200A216), supported the technical acceptability of the PRAs used for Plants A, C, and D as well as the technical adequacy of how the case studies were conducted; and included modifications to the content of the EPRI report. The supplement dated August 5, 2019 (ADAMS Accession No. ML19217A143), clarified a response in the July 19, 2019 supplement. The above-mentioned information is necessary for the staff to make its regulatory finding on the licensee's proposed alternative seismic approach and has not been provided by the licensee.

1. Provide an explanation of the differences between EPRI Report 3002017583 and EPRI Report 3002012988 which was used to support the staff's review of Calvert Cliffs 10 CFR 50.69 LAR wherein an alternative seismic approach was proposed that is similar to that proposed by this licensee. The explanation should include justification for why the staff can rely on its review of EPRI Report 3002012988 and does not need to review EPRI Report 3002017583 separately.

2. Provide the above-mentioned information to support the staff's regulatory finding on the alternate seismic approach. This information can be provided by incorporating by reference the identified supplements, except any Calvert Cliffs specific information, as part of the licensee's LAR or responding to the RAIs in the identified supplements.

3. Identify differences (if any) between the licensee's proposed alternative seismic approach and the alternative seismic approach previously approved by the staff (ADAMS Accession No. ML19330D909). Incorporate the differences in the licensee's proposed approach or justify their exclusion.

Response

1. The technical criteria in EPRI Report 3002017583 is unchanged from EPRI Report 3002012988. The Product Description at the beginning of EPRI Report 3002017583 states the following:

   > *"This Technical Update incorporates updates submitted to the NRC in an RAI submittal for the Calvert Cliffs 50.69 LAR into the previous version of this report, EPRI 3002012988. Aside from those updates, the technical criteria in this report remains unchanged."*

   Exelon provided the seismic alternative markups to Report 3002012988 in Attachment 2 of its July 19, 2019 RAI response submittal (ML19200A216).

   In addition, EPRI Report 3002017583 incorporated a few minor editorial changes including the following:

   a) Figure 1-2 was edited to include EPRI 3002017583 in the list of §50.69 supplemental guidance documents.

   b) Figure 2-2, Low Seismic Hazard Site: Typical SSE to GMRS Comparison – replaced graph with correct graph.

2. Additional references to be added to the LAR are listed below.

   a) Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors," July 1, 2019 (ML19183A012).

   b) Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors," July 19, 2019 (ML19200A216).

   c) Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Revised Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors, letter dated July 19, 2019" August 5, 2019 (ML19217A143).

3. There are no differences identified.

DRA/APLC RAI 04 – External Hazard Screening

NEI 00-04, Rev. 0, Section 5, "Component Safety Significance Assessment" states, "If the plant does not have an external hazards PRA, then it is likely to have an external hazards screening evaluation that was performed to support the requirements of the IPEEE." NEI 00-04, Rev. 0, also states in Section 3.3.2, "Other Risk Information (including other PRAs and screening methods)," that the characterization of the adequacy of risk information should include "a basis for why the other risk information adequately reflects the as-built, as-operated plant."

Attachments 4 and 5 of the LAR provide the screening results of other external hazards for 10 CFR 50.69 categorization and Section 3.2.4 concludes that all external hazards, except for seismic, were screened for applicability to CPS. Attachment 4 of the LAR indicates that criterion "C1" (event damage potential is less than events for which plant is designed) was used to screen the snow hazard and the sand or dust storm hazard. The LAR states that "per the IPEEE, snow storm need not be considered per the guidance contained in NUREG 1407" and "per the IPEEE, sand or dust storm need not be considered per the guidance contained in NUREG 1407," but does not include the basis for why these evaluations adequately reflect the as-built, as-operated plant. It is unclear to the NRC staff that updated hazard information, since the IPEEE, has been considered in the screening analysis, and therefore, that the assumptions that resulted in the screening based on criterion C1 are appropriate for screening these external hazards from 10 CFR 50.69 categorization.

Attachment 4 to the LAR indicates that criterion "C4" (event is included in the definition of another event) was used to screen the ice cover hazard. It is unclear to the staff that the primary effect of potential accumulation of frozen water on a lake would be to cause a loss of offsite power.

Address the following hazards and explain the basis for determining that the screening analysis adequately reflects the as-built, as-operated plant. Also, discuss how the assumptions that resulted in the screening will continue to remain valid during the implementation of 10 CFR 50.69.

(1) snow hazard
(2) sand or dust storm hazard
(3) ice cover hazard

<u>Response</u>

<u>Snow Hazard</u>

The CPS USAR Section 2.4.2.3 discusses design snow and ice loads and accounts for the latest hazard information. The hazard is slow to develop and can be identified via monitoring and managed via normal plant processes. Therefore, in addition to Criterion C1, Criterion C5, "Event develops slowly, allowing adequate time to eliminate or mitigate the threat," also applies.

<u>Sand or Dust Storm Hazard</u>

The Clinton plant is not located in an area impacted by sand or dust storms. More common wind-borne dirt can occur but poses no significant risk given the robust structures and protective features of the plant. Therefore, in addition to Criterion 1, screening Criterion C3 (Event cannot occur close enough to the plant to affect it) also applies.

Ice Cover Hazard

Per USAR Section 2.4.7, the submerged location of the ultimate heat sink pond suction prevents ice formation or ice jams from affecting the performance of the ultimate heat sink. From Attachment 5, the applicable screening criterion is Criterion C1, "Event damage potential is < events for which plant is designed."

As discussed in NEI 00-04 Section 12, scheduled periodic reviews (e.g., once per two fuel cycles) will be conducted to evaluate new insights from available risk information changes, design changes, operational changes, and SSC performance. If it is determined that there are changes that have affected the risk information or other elements of the categorization process such that the categorization results are "more than minimally" affected, then the risk information and the categorization process will be updated. Therefore, the assumptions that resulted in the screening will continue to remain valid during the implementation of 10 CFR 50.69.

DEX/EEOB RAI 01

According to TS 3.3.8.1, Action A.1, the current TS requires the licensee to place any inoperable channel relating to Loss of Voltage or Degraded Voltage Function in the trip position within one hour. According to Action A.2, the current TS requires the licensee to restore any channel relating to Degraded Voltage Function within 7 days. The licensee has proposed to extend the completion times as per the RICT program.

Page E1-11 of the LAR, for TS 3.3.8.1.A, states that the individual instrument channels for "Loss of Power Instrumentation" are not modeled. Therefore, a surrogate relay is chosen that fails the Diesel Generator (DG) start mode or undervoltage relay. Clarify whether the "Inoperability of One or more channels" is modeled as failure of the autostart mode of the associated DG, and that the manual start mode of DG is still considered available in the RICT model.

Response

The "Inoperability of One or more channels" is modeled as a failure of the autostart mode of the associated DG. The manual start mode of the associated DG is still considered available in the RICT model.

DEX/EEOB RAI 02

According to TS 3.3.8.1, Action A.1, the current TS requires the licensee to place any inoperable channel relating to Loss of Voltage or Degraded Voltage Function in the trip position within one hour. The RICT program would allow this completion time (CT) to be extended up to a maximum of 30 days. The associated NOTE states that the RICT program is "Not applicable when trip capability is not maintained."

Explain the impact on the trip logic and trip capability of each function, if the inoperable channel is not placed in the trip position during the CT allowed by RICT program.

Response

The Division 1 and 2 emergency bus Loss of Voltage Function is monitored by two undervoltage relays on the emergency bus and two undervoltage relays on each of the two offsite power sources. This results in a total of six relay/channels for the Loss of Voltage Function for each Division. Each of these relays is an inverse time delay relay. The outputs of these relays are arranged in a two-out-of-two taken three times logic configuration to start the respective Divisional Diesel Generator. Any relay/channel failing will result in the respective Diesel Generator not starting.

The Division 3 emergency bus Loss of Voltage Function is monitored by four undervoltage relays whose outputs are arranged in a one-out-of-two taken twice logic configuration. The output of this logic inputs to a time delay relay. A failure of a channel associated with Function 2.a will not result in a loss of the function. However, a failure of Function 2.b will result in the Division 3 Diesel Generator not starting.

Each Division 1, Division 2, and Division 3 emergency bus Degraded Voltage Function is monitored by two undervoltage relays for each emergency bus whose outputs are arranged in a two-out-of-two logic configuration. The output of this logic inputs to a time delay relay on each emergency bus. Any relay/channel failing will result in the respective Diesel Generator not starting.

DEX/EEOB RAI 03

In the LAR, Enclosure 12, "Risk Management Action (RMA) Examples," the licensee stated that multiple example RMAs may be considered during a RICT program entry to reduce the risk impact and ensure adequate defense-in-depth.

Provide a list of RMAs that are likely to be considered during the implementation of RICT programs relating to the following TS Conditions and Required Actions:

(a) TS 3.8.1, Condition C, "Two offsite circuits inoperable," Action C.2.

(b) TS 3.8.4, Condition B "One battery on Division 1 or 2 inoperable," Action B.1.

(c) TS 3.8.4, Condition C "Division 1 or 2 DC electrical power subsystem inoperable for reasons other than Condition A or B," Action C.1.

(d) TS 3.8.7, Condition A "Division 1 or 2 inverter inoperable," Action A.1.

(e) TS 3.8.9, Condition A, "One or more Division 1 or 2 AC electrical power distribution subsystems inoperable," Action A.1.

(f) TS 3.8.9, Condition B, "One or more Division 1 or 2 uninterruptible AC bus distribution subsystems inoperable," Action B.1.

(g) TS 3.8.9, Condition C, "One or more Division 1 or 2 DC electrical distribution subsystems inoperable," Action C.1.

Response

Risk Management Actions both for regular and common cause considerations are developed for the specific configuration following the steps outlined in Enclosure 12 of the LAR.

a) TS 3.8.1, Condition C, "Two offsite circuits inoperable," Action C.2

Similar to the single offsite source inoperable sample RMAs in the LAR, candidate RMAs for Action C.2 could include:

1. Actions to increase risk awareness and control
    - Briefing of the on-shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a loss of offsite power and station blackout including power supply crossties
    - Notification of the TSO of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred
    - Proactive implementation of RMAs during times of high grid stress conditions prior to reaching the RMAT, such as during high demand conditions
2. Actions to reduce the duration of maintenance activities
    - For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability
    - Confirmation of parts availability prior to entry into a preplanned RICT
    - Walkdown of work prior to execution
3. Actions to minimize the magnitude of the risk increase
    - Evaluation of weather conditions for threats to the remaining capability of the offsite power sources
    - Deferral of elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers
    - Maximizing the remaining capability of the offsite sources, including switchyard and transformers
    - Deferral of planned maintenance or testing that affects the reliability of DGs and their associated support equipment; treat these as protected equipment
    - Implementation of 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected offsite sources

b) TS 3.8.4, Condition B "One battery on Division 1 or 2 inoperable," Action B.1

Candidate RMAs for Action B.1 could include:

1. Actions to increase risk awareness and control
   - Briefing of the on-shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a loss of a battery and station blackout
   - Briefing of the on-shift operations crew concerning the impact the inoperable battery has on the potential response to plant events such as reduced control systems
   - Prior to removal from service if in a Planned RICT, the actions in the associated loss of battery procedure would be reviewed and implemented
   - Minimize activities that could trip the unit
2. Actions to reduce the duration of maintenance activities
   - For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability
   - Confirmation of parts availability prior to entry into a preplanned RICT
   - Walkdown of work prior to execution
3. Actions to minimize the magnitude of the risk increase
   - Deferral of elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers
   - Protection of the remaining batteries and buses
   - Remove nonessential loads from battery to extend time voltage will remain above minimum required level
   - Implementation of any 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected battery

c) TS 3.8.4, Condition C "Division 1 or 2 DC electrical power subsystem inoperable for reasons other than Condition A or B," Action C.1

Candidate RMAs for a DC electrical distribution subsystem could include:

1. Actions to increase risk awareness and control
   - Brief shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a loss of DC power and station blackout
   - Brief shift operations crew concerning the impact the DC division has on the potential response to plant events such as reduced control systems
2. Actions to reduce the duration of maintenance activities
   - Pre-stage materials for work activity
   - Walkdown of work prior to execution
3. Actions to minimize the magnitude of the risk increase
   - Protect remaining DC power subsystems, per OP-AA-108-117
   - Prohibit any elective maintenance on all DC distribution subsystems
   - Take the required actions per procedures for loss of DC safety related bus
   - Prohibit trip sensitive activities and activities that could result in a plant transient
   - Implementation of any 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected DC electric power subsystem

d) TS 3.8.7, Condition A "Division 1 or 2 inverter inoperable," Action A.1

Candidate RMAs for a Division 1 or 2 inverter could include:

1. Actions to increase risk awareness and control
   - Brief Shift crews on required actions per associated emergency operating procedures for loss of an inverter
2. Actions to reduce the duration of maintenance activities
   - Pre-stage materials for work activity
   - Walkdown of work prior to execution
3. Actions to minimize the magnitude of the risk increase

- Prohibit planned maintenance on associated EDG
- Prohibit planned maintenance on other RPS or ECCS actuation channels
- Implementation of any 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected inverter
- Protect remaining power supply to uninterruptible AC bus as directed by OP-AA-108-117

e) TS 3.8.9, Condition A, "One or more Division 1 or 2 AC electrical power distribution subsystems inoperable," Action A.1

Candidate RMAs for a Division 1 or 2 AC electrical distribution subsystem could include:

1. Actions to increase risk awareness and control
   - Brief shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a loss of AC distribution
2. Actions to reduce the duration of maintenance activities
   - Pre-stage material for work activity and ensure parts availability for any contingent work
3. Actions to minimize the magnitude of the risk increase
   - Protect remaining AC power distribution subsystems, as directed by OP-AA-108-117
   - Prohibit any elective maintenance on ALL safety related AC and DC distribution subsystems
   - Take the required actions per procedure for loss of AC distribution subsystem
   - Prohibit trip sensitive activities and activities that could result in a plant transient
   - Implementation of any 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected AC distribution sub system
   - Minimize activities on equipment powered by remaining AC power buses

f) TS 3.8.9, Condition B, "One or more Division 1 or 2 uninterruptible AC bus distribution subsystems inoperable," Action B.1

Candidate RMAs for a Division 1 or 2 AC uninterruptible AC bus distribution subsystems could include:

1. Actions to increase risk awareness and control

    - Brief shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a loss of uninterruptible AC bus

    - Brief shift operations crew concerning the impact the uninterruptible AC bus has on the potential response to plant events such as reduced control systems or system initiation

2. Actions to reduce the duration of maintenance activities

    - Pre-stage materials for work activity

    - Walkdown of work prior to execution

3. Actions to minimize the magnitude of the risk increase

    - Protect remaining uninterruptible AC subsystems, per OP-AA-108-117

    - Minimize activities on equipment powered by remaining uninterruptible AC bus

    - Prohibit any elective maintenance on ALL vital AC and DC distribution subsystems

    - Take the required actions per emergency operating procedures for loss of uninterruptible AC bus distribution subsystem

    - Prohibit trip sensitive activities and activities that could result in a plant transient

    - Implementation of any 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected uninterruptible AC distribution sub system

g) TS 3.8.9, Condition C, "One or more Division 1 or 2 DC electrical distribution subsystems inoperable," Action C.1

Candidate RMAs for a DC electrical distribution subsystem could include:

1. Actions to increase risk awareness and control

    - Brief shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a Loss of DC power and station blackout

- Brief shift operations crew concerning the impact the DC division has on the potential response to plant events such as reduced control systems

2. Actions to reduce the duration of maintenance activities

    - Pre-stage materials for work activity

3. Actions to minimize the magnitude of the risk increase

    - Protect remaining AC and DC power distribution subsystems, per OP-AA-108-117
    - Prohibit any elective maintenance on ALL vital AC and DC distribution subsystems
    - Take the required actions per emergency operating procedures for loss of a DC electrical distribution subsystem
    - Prohibit trip sensitive activities and activities that could result in a plant transient
    - Implementation of any 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected DC distribution sub system

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The regulation under 10 CFR 50.36(c)(2) requires that TSs contain LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCO can be met. Typically, the TSs require restoration of equipment in a timeframe commensurate with its safety significance, along with other engineering considerations. The regulation under 10 CFR 50.36(b) requires that TSs be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto.

In determining whether the proposed TS remedial actions should be granted, the Commission will apply the "reasonable assurance" standards of 10 CFR 50.40(a) and 50.57(a)(3). The regulation at 10 CFR 50.40(a) states that in determining whether to grant the licensing request, the Commission will be guided by, among other things, consideration about whether "the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other TS, or the proposals, in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in Part 20 of this chapter, and that the health and safety of the public will not be endangered."

TSTF-505, Revision 2 (ADAMS Accession No. ML18183A493) does not allow TS Conditions in the RICT program that represent a loss of a specified safety function or inoperability of all trains of a system required to be deemed OPERABLE. Throughout the TS markup pages in Attachment 2, the licensee proposed the following note for certain TS LCO Conditions to prohibit entering a RICT during a loss of function condition: "Not applicable if loss of function." However, the proposed change to TS LCO 3.7.6 (Main Turbine Bypass System) Condition A (Requirements of LCO not met) appears to include a TS loss of function and has no such note. Enclosure 1 to the LAR, Table E1-1, indicates that the design basis success criteria for TS 3.7.6.A requires all six turbine bypass valves; this indicates that the loss of even one valve is a loss of function. Accordingly, provide technical justification that the proposed RICT for TS 3.7.6.A does not include a TS loss of function condition and address any discrepancies within LAR Table E1-1 for LCO 3.7.6.

Response

The purpose of LCO 3.7.6 is to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization. This TS function is accomplished by either: (1) the Main Turbine Bypass System, or (2) by limiting reactor power and implementing designated thermal limits in accordance with the cycle-dependent Core Operating Limits Report (COLR) and modifications to the MCPR limits and LHGR limits.

The cycle-specific analyses as documented in the COLR evaluates different combinations of turbine bypass valves (TBVs) out-of-service (OOS). The COLR for the current operating cycle supports full power operation with one TBVOOS, but also considers different conditions including all BPVs OOS. As a result, the LCO function of limiting peak pressure in the main steam lines and maintaining reactor pressure within acceptable limits continues to be met under the conditions analyzed in the cycle-specific analyses, which includes all BPVs OOS. As long as the correct COLR thermal limit set is selected, the analyzed design function of the TBVs will be met regardless of how many are in/out of service. Therefore, inoperability of multiple TBVs does not result in a loss of the function.

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