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RBG-48271 10 CFR 50.90

January 12, 2024

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

Subject: Supplement to License Amendment Request to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b" and Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

River Bend Station, Unit 1
NRC Docket No. 50-458
Renewed Facility Operating License No. NPF-47

By the two letters dated February 27, 2023 (References 1 and 2), Entergy Operations, Inc. (Entergy) submitted two separate license amendment requests (LAR) for River Bend Station Unit 1 (RBS). The proposed amendment in Reference 1 would modify Technical Specifications (TS) requirements to permit the use of Risk-Informed Completion Times in accordance with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b" (ADAMS Accession No. ML18183A493). The proposed amendment in Reference 2 would modify the RBS licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors."

By letter dated October 6, 2023 (Reference 3), the Nuclear Regulatory Commission (NRC) notified Entergy of their intent to conduct a regulatory virtual audit with Entergy staff in support of the License Amendment Requests (LARs) in References 1 and 2. Following the audit, the NRC suggested that Entergy respond in the form of a supplement to selected audit questions, either for clarification, to add or remove detail, or to formally document Entergy's responses to the questions.

This letter is a supplement to both References 1 and 2 LARs. Attachments 1 and 2 to this letter provide a response to several of the audit questions posed by the NRC staff during the regulatory virtual audit. Attachment 1 addresses the TSTF-505 LAR and Attachment 2 addresses the 10 CFR 50.69 LAR. Attachment 3 provides markups of select entries to Enclosure 1, Table E1-1 of Reference 1 in support of TSTF-505 audit questions EEEB-01, EEEB-02, and EEEB-03. The unaffected portion of Enclosure 1, Table E1-1 of Reference 1 remains unchanged. Attachment 4 provides a revised markup of a select TS page from

Reference 1 in support of TSTF-505 audit question STSB-01. The revised markup of this select TS page supersedes what was provided in Reference 1 for this page and all other TS markups remain unchanged.

Entergy has reviewed the information supporting the No Significant Hazards Consideration Determination and the Environmental Consideration that was previously provided to the NRC in References 1 and 2. The additional information provided in this LAR supplement does not impact the conclusion that the proposed license amendments in both References 1 and 2 do not involve a significant hazards consideration. Additionally, the information does not impact the conclusion that there is no need for an environmental assessment to be prepared in support of the proposed amendments.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), a copy of this supplement, with attachments, is being provided to the designated State Officials.

This letter and its attachments do not contain any new commitments.

Should you have any questions or require additional information, please contact Mr. Randy Crawford, Regulatory Assurance Manager at 225-381-4177.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on January 12, 2024

Respectfully,

Philip Couture

Digitally signed by Philip Couture
Date: 2024.01.12 14:18:22 -06'00'

Phil Couture
PC/rd

Attachments:

1. Responses to NRC Audit Questions related to the TSTF-505 LAR
2. Responses to NRC Audit Questions related to the 10 CFR 50.69 LAR
3. Markup of Entries to TSTF-505 License Amendment Request Table E1-1, "In-scope TS/LCO Conditions to Corresponding PRA Functions"
4. Technical Specification Page Revised Markups

References:

1. Entergy Operations, Inc. letter to U.S. Nuclear Regulatory Commission, "License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," (ADAMS Accession No. ML23058A215), dated February 27, 2023.

2. Entergy Operations, Inc. letter to U.S. Nuclear Regulatory Commission, "Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," (ADAMS Accession No. ML23058A217), dated February 27, 2023.

3. U.S. Nuclear Regulatory Commission Letter to Entergy Operations, Inc., "River Bend Station, Unit 1 – Regulatory Audit Plan in Support of License Amendment to Adopt Risk-Informed Completion Times and Implement the Provisions of 10 CFR 50.69 (EPID L-2023-LLA-0037 and EPID L-20230LLA-0038)," (ADAMS Accession No. ML23278A240), dated October 6, 2023.

cc: NRC Regional Administrator – Region IV
NRC Project Manager – River Bend Station
NRC Senior Resident Inspector – River Bend Station
Louisiana Department of Environmental Quality

Attachment 1

RBG-48271

Responses to NRC Audit Questions
Related to the TSTF-505 LAR

(32 pages follow)

RESPONSES TO NRC AUDIT QUESTIONS

NOTE: The U.S. Nuclear Regulatory Commission (NRC) staff's audit questions are in italics throughout this attachment to distinguish from the Entergy Operations, Inc. (Entergy) responses.

Audit Question APLA-01 (TSTF-505) – In-Scope LCOs and Corresponding PRA Modeling:

The NRC's safety evaluation for NEI 06-09-A specifies that the LAR should provide a comparison of the TS functions to the PRA modeled functions to show that the PRA modeling is consistent with the licensing basis assumptions or to provide a basis when there is a difference. Table E1-1 of LAR Enclosure 1 identifies each Limiting Condition for Operation (LCO) in the TSs proposed for inclusion in the RICT program. The table also describes whether the systems and components covered by the LCO are modeled in the PRA and, if so, presents both the design success criteria and PRA success criteria. For certain LCOs, the table explains that the associated structures, systems, and components (SSCs) are not modeled in the PRAs but will be represented using a surrogate event that fails the function performed by the SSC. For some LCOs, the LAR did not provide an adequate description for the NRC staff to conclude that the PRA modeling will be sufficient.

a) *Regarding TS LCO 3.3.1.A/B, Table E1-1 states that, for reactor protection system (RPS) instrumentation that an RPS failure model is planned to be incorporated into the Electronic Risk Assessment Tool (ERAT) based on NUREG/CR-5500 (Volume 3) "Reliability Study: General Electric Reactor Protection System, 1984-1995," dated May 1999. It also states that the intent of the RBS ERAT RPS model is to be used as a surrogate for unmodeled RPS SSCs. However, it is stated later that the simplified RPS model provides a more conservative result than the NUREG/CR-5500 model when a Function channel is inoperable or bypassed. Clarity is needed to understand how the RBS ERAT RPS channel modeling is more conservative than NUREG/CR-5500 base probability if the base results in exactly matching the NUREG value. The NRC staff notes that in section 5 of NUREG/CR-5500, the overall failure probability appears to include operator manual scram, control rod system, and hydraulic control unit system (scram discharge volume and solenoid operated valves (SOVs)). NUREG/CR-5500 provides the following failure rates: (1) section 3.3 states a failure rate of 3.8E-06 for the channel and trip portion of RPS, and (2) section 5 states the mean RPS unavailability as 5.8E-06. Clarity is needed to understand how the RBS RPS model incorporates all of the necessary SSCs and operator actions to represent the as-built, as-operated plant for the associated proposed RICT LCOs.*

   The NRC staff notes that section 5 of NUREG/CR-5500 states that the failure probabilities used were based on U.S. General Electric commercial data from 1984 through 1995, and that the 2009 American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard supporting requirement DA-C1 lists NUREGs that contain failure data from recognized sources. One of those sources, NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," dated February 2007 (Reference 15), contains recent industry data including a 2020 update. It is unclear to the staff how Capability Category (CC)-II technical acceptability is met by using the NUREG/CR-5500 data for this application, or if the RBS RPS modeling is being implemented as a surrogate for RICT calculations.

i. *Provide justification that the RBS ERAT RPS model provides conservative results when compared to the NUREG/CR-5500 model. Include in this discussion of the failure probability values used from NUREG/CR-5500.*

ii. *Justify that all of the SSCs associated with the proposed RICT LCOs related to RPS fully represent the functionality of those LCOs. Include in this discussion the apparent disparity of SSCs mentioned in NUREG/CR-5500 to those discussed in Table E1-1 of the LAR, and how the NUREG/CR-5500 failure probability value provides an appropriate comparison for this application.*

iii. *Regarding the proposed RBS ERAT RPS model meeting the 2009 ASME/ANS PRA standard CC-II requirements and being used as a surrogate for RICT calculations:*

1) *Provide justification that the proposed RBS ERAT RPS model will meet the associated CC-II requirements. Include in this discussion how the use of NUREG/CR-5500 information meets CC-II requirements.*

2) *If the proposed RBS ERAT RPS cannot meet CC-II requirements, then provide justification that the use of the surrogate is either conservative or bounding when compared to a CC-II model.*

3) *If the proposed surrogate cannot be justified as either conservative or bounding, then propose a mechanism to ensure the RBS ERAT RPS model is conservative or bounding prior to any RICT calculation.*

b) *Regarding TS LCO 3.3.5.1.D Function 3.e and 3.3.5.3.D Function 4, Table E1-1 states that for the Suppression Pool Water Level (SPWL) – High channel, which is not modeled, that the Condensate Storage Tank (CST) Level Low channel, (which is modeled), will be used as a surrogate. The NRC staff notes that the stated function for this LCO is to align high pressure core spray (HPCS) and reactor core isolation cooling (RCIC) pumps suction from the CST to the suppression pool for continued operations of the HPCS/RCIC pumps. Switching from the CST on low level appears to address inventory control for the HPCS/RCIC pumps, whereas switching to the suppression pool for high level appears not to be related to inventory concerns for the pumps. It is unclear to the NRC staff how the suppression pool high level logic is related to the CST low level function and therefore how the surrogate is either bounding or conservative for this LCO.
Provide justification that the surrogate is related and bounds the suppression pool level high function.*

c) *Regarding TS LCO 3.3.5.1.F/G and 3.3.6.4.A, Table E1-1 states that, for automatic depression system (ADS) initiation logic and instrumentation functions not modeled, that the each of the ADS solenoid operated valves (SOVs) in the associated train will remain closed. However, it further states that the RBS ERAT model will be updated to incorporate power dependencies for each ADS steam relief valve (SRV) pilot valve to address train specific ADS SRV instrumentation or pilot valve outages. It is unclear to the NRC staff the impact of the proposed RBS ERAT ADS model update on the proposed surrogates for these LCOs.*

i. *Provide justification that the proposed RBS ERAT ADS model will meet the associated ASME/ANS PRA CC-II requirements.*

ii. *If the proposed RBS ERAT ADS cannot meet CC-II requirements, then provide justification that the use of the surrogate is either conservative or bounding when compared to a CC-II model.*

iii. *If the proposed surrogate cannot be justified as either conservative or bounding, then propose a mechanism to ensure the RBS ERAT ADS model is conservative or bounding prior to any RICT calculation.*

d) *Regarding TS LCO 3.3.6.1.A Functions 2.a, 2.b, and 2.c, Table E1-1 states that, for reactor vessel level – low-low, Level 2, drywell pressure – high, and containment purge isolation radiation – high channels not modeled, that the primary containment isolation valves (PCIVs) associated with the affected trip system that is modeled will be used as a surrogate. The NRC staff notes that the Table E1-1 entry for TS LCO 3.6.1.3.A, PCIVs, states that not all PCIVs are modeled and for the unmodeled PCIVs that a pre-existing containment failure will be used as a surrogate. It is unclear to the staff if all the trip system functions of this LCO associated PCIVs are modeled and if the LCO 3.6.1.3.A surrogate is required.*

i. *Confirm that all of the PCIVs associated with the LCO 3.3.6.1.A functions are modeled.*

ii. *For the PCIVs associated with LOC 3.3.6.1.A that are not modeled, identify the surrogate to be used for RICT calculations. Include in this discussion justification that the surrogate bounds the associated function.*

e) *Regarding TS LCO 3.3.6.4.A and 3.6.1.6.A, Table E1-1 states that, for relief and low-low (LLS) instrumentation trip system or LLS valve not modeled, that the SRV fail to reclose failure probability that is modeled will be doubled and used as a surrogate. The NRC staff notes that the LLS LCO function is to have the SRVs remain open longer to prevent multiple actuations and pressure loads. It is unclear to the staff how the doubling of the SRV fail to reclose probability is either conservative or bounding.*

i. *Provide justification that doubling of the surrogate's failure probability is conservative or bounds the LLS relief point function.*

ii. *If the doubling of the surrogate's failure probability is determined not to be bounding or conservative, then provide an updated surrogate that is bounding or conservative for this LCO.*

f) *Regarding TS LCO 3.3.8.1.A, Table E1-1 states that, for loss of power (LOP) instrumentation channels not modeled, that a single event per division that represents a loss of emergency buss undervoltage will be used as a surrogate. It is unclear to the NRC staff what constitutes the single event and therefore how it is either conservative or bounding.*

i. *Identify the single event in the PRA model that will used to represent all of the associated functions for TS LCO 3.3.8.1.A.*

ii. *Provide justification that the surrogate is conservative or bounds the LOP Instrumentation function.*

g) *Regarding TS LCO 3.6.1.2.C, Table E1-1 states that, for primary containment air locks not modeled, that a large pre-existing containment isolation failure that is modeled will be used as a surrogate. The NRC staff notes the associated Note 8 of Table E1-1 states that the failure probabilities of the surrogate will be increased. It is unclear to the staff how increasing the failure probability of the surrogate is either conservative or bounding.*

i. *Detail the intended increase in failure probability of the surrogate associated with this LCO.*

ii. *Provide justification that increasing the surrogate's failure probability is conservative or bounds the LCO function.*

iii. *If the increasing of the surrogate's failure probability is determined not to be bounding or conservative, then provide an updated surrogate that is bounding or conservative for this LCO.*

h) *Regarding TS LCO 3.6.1.3.A, Table E1-1 states that, for PCIVs not modeled, that a pre-existing large containment isolation failure that is modeled will be used as a surrogate. The NRC staff notes the associated Note 9 of Table E1-1 states that for the redundant, unisolated valve the respective failure probability will be added. It is unclear to the staff what the added failure probability consists of and if it is either conservative or bounding.*

i. *Detail the intended increase in failure probability of the surrogate associated with this LCO.*

ii. *Provide justification that increasing the surrogate's failure probability is conservative or bounds the LCO function.*

iii. *If the increasing of the surrogate's failure probability is determined not to be bounding or conservative, then provide an updated surrogate that is bounding or conservative for this LCO.*

i) *Regarding TS LCO 3.8.9.A, Table E1-1 states that Division I or II AC electrical power distribution subsystems are modeled in the PRA. However, the NRC staff notes that the Comments section states, regarding unmodeled distribution panels, to see Note 13. It is unclear to the NRC staff if the column entry for this LCO that states 'Yes' for SSCs being modeled in the PRA is accurate. The NRC staff notes the associated Note 13 of Table E1-1 states that for the unmodeled load centers, MCCs, or power panels the SSCs placed out of service are a best estimate surrogate. It is unclear to the staff how this constitutes a best estimate surrogate and if it is either conservative or bounding.*

i. *Confirm that all of the SSCs associated with TS LCO 3.8.9.A have a one to one relationship to the RBS PRA models used for this application.*

ii. *Clarify what is meant by 'best estimate surrogate'.*

iii. *Provide justification that the 'best estimate surrogate' is conservative or bounds the LCO function.*

iv. *If the 'best estimate surrogate' is determined not to be bounding or conservative, then provide an updated surrogate that is bounding or conservative for this LCO.*

Entergy Response:

a) i. The River Bend Station Unit 1 (RBS) RPS model is a simplified five event model with the data of each event derived from the cutset solution of the NUREG/CR-5500 model. Four of the events represent channel or subsystem failures, with one event for each of the subsystems, arranged in a one-of-two taken twice failure logic. The fifth event is a common cause failure leading to system failure. The simplified model excludes the mechanical failures which are not subject to TS 3.3.1.1 and are modeled in detail in the fault tree model. The following Figure 1 illustrates the RBS RPS electrical model logic:

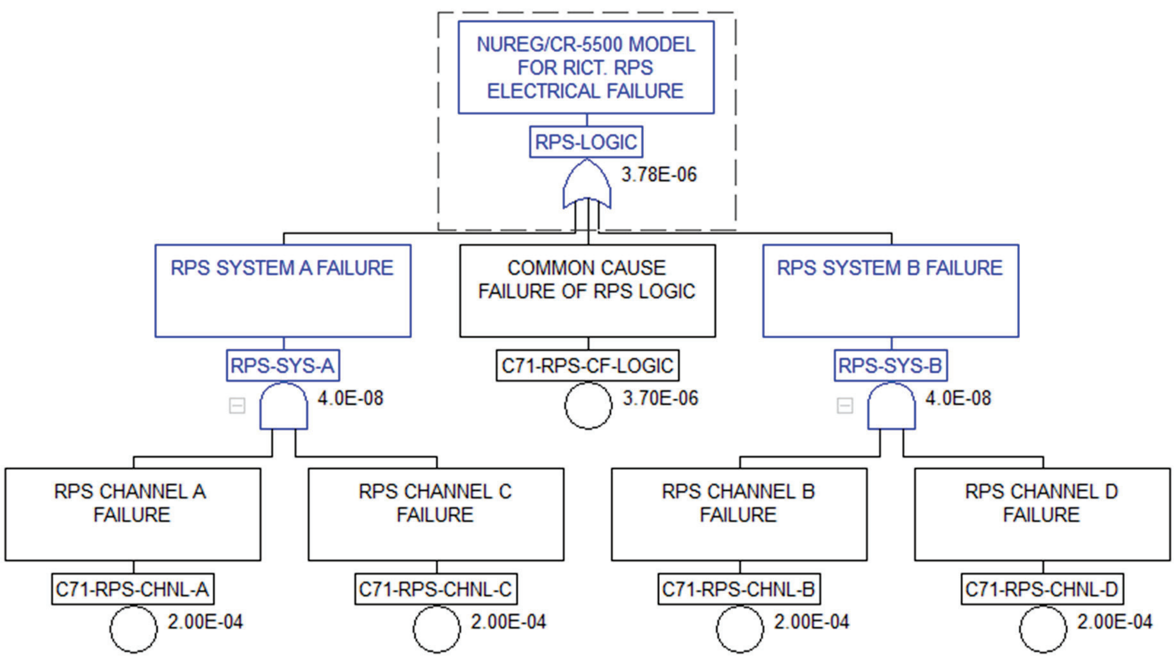

Figure 1: Simplified RPS Electrical Failure Logic with Gate Probabilities

The model is designed to be applied for Risk-Informed Completion Times (RICTs) as a total channel inoperable for any function within that channel being inoperable. When no channels are inoperable, quantification of the simplified RBS model results in the same numerical solution as the NUREG/CR-5500 model for the electrical component failures.

When channels or systems are inoperable, the simplified model provides conservative results. Table 1 below compares the solution between the NUREG/CR-5500 model and the RBS simplified RPS model:

Table 1: Comparison of RPS Model Results

Inoperable Case	Simplified RICT Model Result	NUREG/CR-5500 Model Result	Assessment
No Maintenance	3.78E-6	3.77E-6	RBS RPS surrogate model is equivalent
All A Channel K Relays (Div. I impact)	2.04E-4	5.07E-6	RBS RPS surrogate model is bounding
All A and B Channel K Relays (Div. I and II impact)	4.04E-4	6.37E-6	RBS RPS surrogate model is bounding
All A and C Channel K Relays (Div. I Unavailable)	1	9.97E-1	RBS RPS surrogate model is equivalent

Note that the RPS Model used in the RBS RICT model is consistent with the RPS model used in the Columbia Generating Station RICT Application, which was approved by the NRC (ML23013A081).

a) ii. The NUREG/CR-5500 model includes demand failures of high reactor pressure and low reactor water level (equivalent to RBS functions 3 and 4). These two automatic functions are appropriate to represent the full range of plant events, because for identified events, it has been shown that at least two automatic functions are challenged for the design basis accidents in the Updated Safety Analysis Report (USAR). Attachment 5 of the TSTF-505 License Amendment Request (LAR) identifies at least one diverse automatic trip function for each TS function for design accidents. Therefore, it is conservative to assume that there are only two RPS scram automatic functions challenged for any plant event. It is also appropriate for these two automatic functions to represent the reliability of other automatic functions, as there is little variation between the SSCs of each automatic function from a data standpoint. Therefore, the probability of an RPS scram failure as calculated by NUREG/CR-5500 is a conservative model for the whole range of automatic functions and their SSCs proposed in the RICT LCOs. Because the RBS RPS model takes an entire subsystem from service for RICT applications to quantify the RPS failure probability, it addresses all functions within that sub-system when the event is made unavailable (failed).

a) iii.1) The RBS RPS model meets the 2009 ASME/ANS Probabilistic Risk assessment (PRA) standard CC-II requirements.

Requirement SY-A7 CC I-II states:

> "DEVELOP detailed systems models, unless (a) sufficient system-level data are available to quantify the system failure probability, or (b) system failure is dominated by operator actions, and omitting the model does not mask contributions to the results of support systems or other dependent-failure modes. For case (a), USE a single data value only for systems with no equipment or human-action dependencies, and if data exist that sufficiently represent the unreliability or unavailability of the system and account for plant-specific factors that could influence unreliability and unavailability. Examples of systems that have sometimes not been modeled in detail include the scram system, the power-conversion system, instrument air, and the keep-fill systems. JUSTIFY the use of limited (i.e., reduced, or single data value) modeling."

The use of the NUREC/CR-5500 model as a point estimate meets the American Society of Mechanical Engineers (ASME) standard requirements, because the model can be sufficiently addressed by a point estimate, the model is not dependent on human actions, the data was system specific and used state of the art approaches still in practice today that are consistent with Category II and III ASME supporting requirements, and the results of the model remain conservative when compared to newer data. Also, the ASME standard cites RPS models as an example where a point estimate system model is acceptable. The simplified RBS RPS model is based on the point estimate approach but was refined to addressed inoperable channels.

Requirement DA-C1 for CC 1-III states:

> "OBTAIN generic parameter estimates from recognized sources. ENSURE that the parameter definitions and boundary conditions are consistent with those established in response to DA-A1 to DA-A4. (Example: some sources include the breaker within the pump boundary, whereas others do not.) DO NOT INCLUDE generic data for unavailability due to test, maintenance, and repair unless it can be established that the data is consistent with the test and maintenance philosophies for the subject plant.
>
> Examples of parameter estimates and associated sources include:
>
> (a) component failure rates and probabilities: NUREG/CR-4639 [2-7], NUREG/CR-4550 [2-3], NUREG-1715 [2-21], NUREG/CR-6928 [2-20]
>
> (b) common cause failures: NUREG/CR-5497 [2-8], NUREG/CR-6268 [2-9]
>
> (c) AC off-site power recovery: NUREG/CR-5496 [2-10], NUREG/CR-5032 [2-11]
>
> (d) component recovery
>
> See NUREG/CR-6823 [2-1] for a listing of additional data sources."

The NUREG/CR-5500 model uses data from 1984 to 1995, which predates data from NUREG/CR-6928 and its updates through 2020. A review of the NUREG/CR-6928 data for RPS systems indicates that the data comes from NUREG/CR-5500 for the period of 1986 through 1995. The primary difference is that the NUREG/CR-5500 Volume 3 data is for the Boiling Water Reactor (BWR) RPS systems, while the RPS data reported in NUREG/CR-6928 (2020 update) comes from the combined studies of NUREG/CR-5500 Volumes 2, 3, 10, and 11 (i.e., Westinghouse, General Electric (GE), Combustion Engineering (CE), Babcock & Wilcox (B&W)) that is not differentiated by reactor brand or type in the combined analysis. Therefore, it is recognized that the BWR RPS data from NUREG/CR-5500 Volume 3 remains the industry state of the art data for BWR RPS data. Some data types can be compared to newer data from NUREG/CR-6928 for 2006 through 2020 (taken from the EPIX/RADs system). However, these data types showed a small decrease in probabilities (approximately 30%) over the older data used in NUREG/CR-5500 Vol. 3. Therefore, using the older data remains acceptable with only a minor conservatism added to the solution value.

Because the data in NUREG/CR-5500 was specifically taken from RPS systems and has not been updated by a more current study, it remains the industry state of the art study and meets the intent of ASME DA-C1 as a generic data source.

a) iii 2) Not applicable based on the response to APLA Question 1 a) iii 1).

a) iii 3) Not applicable based on the response to APLA Question 1 a) iii 1).

b) Regarding LCO 3.3.5.1.D Function 3.e and 3.3.5.3.D Function 4, the function of the suppression pool water level - high is to provide a signal to transfer the suction source of HPCS or RCIC from the CST to the suppression pool. The suppression pool water level – high function is a diverse function to the CST level - low function, but the suppression pool water level – high function is not modeled in the PRA. Therefore, using the CST level - low level function is a bounding surrogate for the high suppression pool instrumentation function impact of LCOs. It is bounding because in the LCO condition, only one operable function channel is credited in the PRA to transfer the HPCS or RCIC suction, versus in the plant, the operable channel and diverse function remain available to transfer the HPCS or RCIC suction.

c) i. The division specific ADS valve solenoid pilot valves are used as surrogates for any inoperability of an ADS instrumentation division function. This is conservative, as the solenoid valve response is the output of the ADS instrumentation system. The Revision 6 PRA models included logic for each SRV failing to open with a common gate modeling loss of both divisions of DC power. For the Revision 6 Fire PRA, individual solenoid valves were added to each SRV top gate to address division specific fire impacts identified in the circuit analysis. However, the power supplies remained under a common gate for the depressurization function. It was recognized that if one division of ADS solenoids was inoperable, the remaining operable division was incorrectly crediting two divisions of DC power due to the common dc power gate structure. Therefore, the individual power supplies were linked to each solenoid valve in the model used for the RICT analysis as well as the Revision 7b model to be used in the implemented ERAT model. This modeling approach is explicit and meets ASME standard SY-B5 CC I-III which states:

> "**ACCOUNT explicitly for the modeled system's dependency** on support systems or interfacing systems in the modeling process. This may be accomplished in one of the following ways:

(a) for the fault tree linking approach by **modeling the dependencies as a link** to an appropriate event or gate in the support system fault tree

(b) for the linked event tree approach, by using event tree logic rules, or calculating a probability for each split fraction conditional on the scenario definition"

c) ii. Not appliable based on the response to APLA Question 1 c) i.

c) iii. Not appliable based on the response to APLA Question 1 c) i.

d) i. This LCO is associated with Primary and Secondary Containment Isolation Instrumentation, specifically functions for LCO 3.3.6.1 functions 2.a, 2.b, and 2.c and the use of the associated train PCIVs as surrogates for any channel inoperable. The approach used by RBS for the PCIV Instrumentation surrogates is consistent with the approach used in the Columbia Generating Station RICT LAR Application that was approved by the NRC.

Not all PCIVs associated with LCO 3.3.6.1. functions 2.a, 2.b, and 2.c are modeled. Only those penetrations that have been identified as Large Early Release Frequency (LERF) pathways are modeled. Penetration pathways that screened from LERF consideration include one or more of the following attributes:

1. The piping is less than 3 inches in diameter.
2. The penetrations that connect to the suppression pool below the liquid level are considered sealed.
3. The penetration has three or more normally closed check valves
4. The penetration connects to closed piping in the containment or drywell, (i.e., not directly open to drywell atmosphere, the RPV, or it's connecting piping.)
5. The penetration has closed piping in the auxiliary building and is not an interfacing system Loss of Coolant Accident (LOCA) pathway. Interfacing system LOCA pathways are treated uniquely as a direct bypass.
6. The penetration consists of two or more normally closed Motor Operated Valves (MOVs) or manual valves that do not change state during an event.

The remaining unscreened valves associated with LCO 3.3.6.1.A functions 2.a, 2.b, and 2.c instrumentation are modeled explicitly.

d) ii. The ERAT model is set up to fail all the modeled PCIVs as a surrogate (fail to close) for the associated inoperable instrumentation division. Therefore, only the redundant single valve associated with the available instrumentation train for each penetration is credited. Since all pathways are LERF pathways, failure of any valve to close that is associated with the available operable instrument train results in a large early release. This is very conservative, as each isolation valve receives multiple diverse isolation signals including manual. The LCO is for instrumentation and not for PCIVs; therefore, the approach is bounding. Unmodeled PCIVs are not LERF paths, and the likelihood of multiple failed paths screened by the size criterion that would be required to create a LERF path is considered insignificant.

e) i. The purpose of the LLS instrumentation is twofold: First, to reduce the cycling of SRVs thereby reducing the likelihood of a stuck open SRV, and second, to reduce the number of dynamic pressure-loads the suppression pool may experience. The first of the two functions impacts a failure mode modeled in the PRA, potential for SRV LOCA. The second is not a PRA function that is explicitly modeled but is applicable to containment integrity during an accident, and a failure probability for a surrogate can be conservatively developed to bound the impact.

For the LCO condition, one LLS subsystem inoperable will not render the LLS function inoperable and the probability of multiple cycles would not be significantly higher. For the LCO condition with two channels remaining, the failure probability of the remaining two-out-of-two logic can be estimated using NUREG/CR-6928 2020 update data for bistables (5.44E-4) and pressure transmitters (1.17E-4) producing a combined channel probability of 6.61E-4 and estimated subsystem failure probability of twice that at 1.32E-3.

The probability for one stuck open and two stuck open SRVs is proposed to be tripled as a conservative estimate. This is an increase of 1.77E-3 for one stuck open SRV and an increase of 3.38E-4 for two stuck open SRVs. The sum of this increase to the SRV failure probability is 2.11E-3. This is a bounding value compared to the 1.32E-3 LLS failure probability increase for the LCO condition.

For the impact of the SRVs cycling on dynamic loading of the suppression pool, the use of the pre-existing large containment failure is selected as a surrogate. In the absence of experiential data to support a value, a conservative value will be used. The normal value for this event is 2.7E-3; but, when used as a surrogate for other LCOs, its value is increased to 5.2E-2, an increase of 4.9E-2. Based on engineering judgement of the containment ruggedness against repetitive design loading and the estimated LLS subsystem failure probability of 1.32E-3, the 5.2E-2 number is considered conservative and bounding.

e) ii. Not applicable based on response to APLA Question 1 e) i.

f) i. There is a single LOP instrumentation event used for each emergency bus. Therefore, there are three events.

EGS-DGN-DN-SIG1A, Loss of auto signal to start and load standby diesel generator 1EGS*EG1A, 2.48E-5

EGS-DGN-DN-SIG1B, Loss of auto signal to start and load standby diesel generator 1EGS*EG1B, 2.48E-5

EGS-DGN-DN-SIG1C, Loss of auto signal to start and load standby (HPCS) diesel generator 1E22*S001, 2.48E-5

f) ii. The NUREG/CR-6928 data for relay failures is 2.48E-5. This value is applied to the LOP model as a conservative estimate. A fault tree model was created in response to this question to demonstrate that use of this single point estimate approach is conservative compared to a more detailed model. The model is based on the plant logic, NUREG/CR-6928 2020 data and a conservative Common Cause Failure (CCF) factor of 0.1. Figure 2, Division 1 base (all relays operable) model, is shown below.

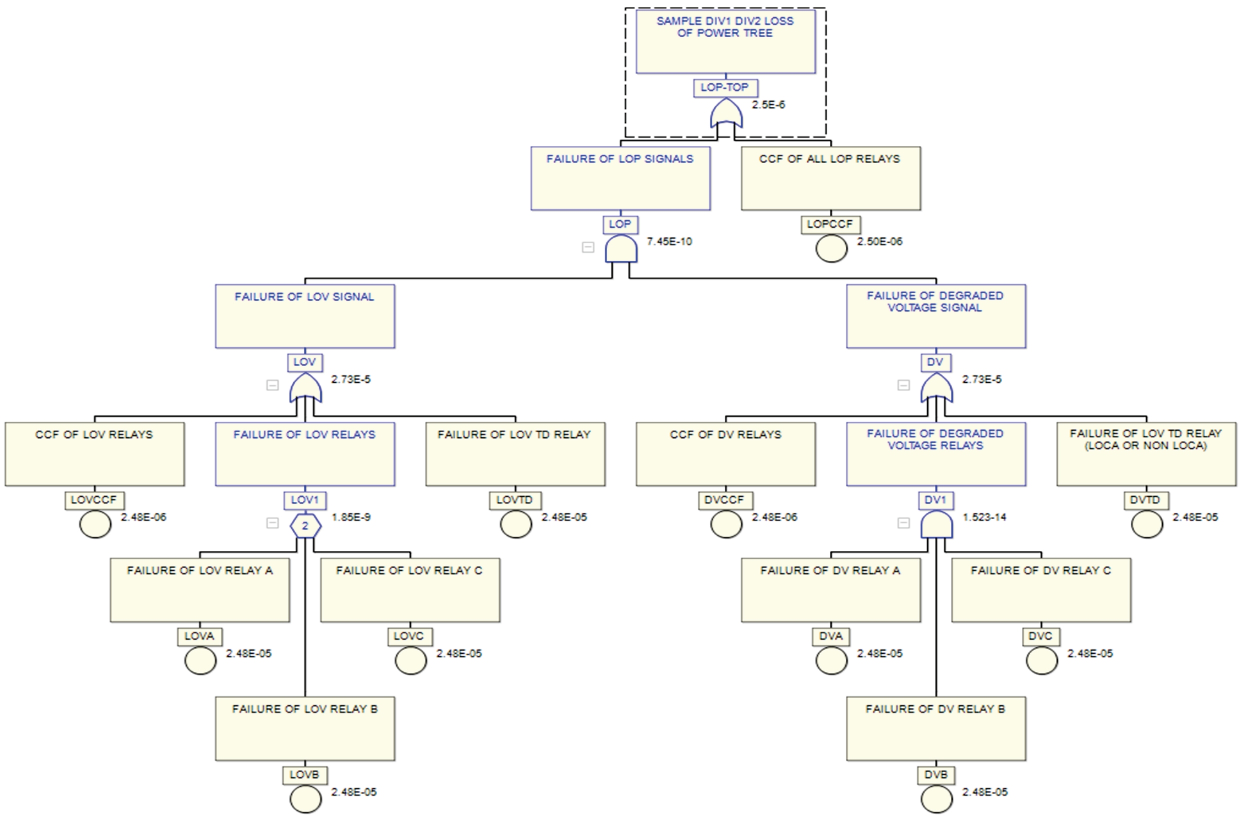

Figure 2: Division 1 Base (All Relays Operable) Model

Using the base model for Division 1, a top gate probability of 2.5E-6 was calculated as shown in the fault tree above. Note that the PRA models the initiating event, Loss of Offsite Power, for which both the Loss of Voltage (LOV) and Degraded Voltage (DV) function will be challenged.

The solution value for the Division 2 and 3 fault tree is similar.

For one or two relays inoperable, the probability of the tree will typically remain below 2.73E-5. When comparing this to the single point estimate event set to 1.0, the surrogate is bounding.

For the case where both the LOV and DV Time Delay (TD) relays are inoperable, the above tree calculates a value of 1.0. In this case, the sample model is equivalent to the single surrogate set to 1.0. Therefore, the use of a single surrogate event for RICT calculations is conservative and bounding for the range of possible components inoperable.

g) i. Event PRECONTFAIL is the modeled event for a pre-existing containment failure. The use of a large pre-existing containment isolation failure as a surrogate for unmodeled containment penetrations, including containment air locks, is consistent with the Columbia Generating Station (ML23013A081), Nine Mile Point Nuclear Station (ML21083A221), Clinton Power Station (ML21132A288), and LaSalle County Station (ML21162A069) RICT LARs that have been approved by the NRC. The preexisting large containment failure event probability was derived by the Pacific Northwest National Laboratory for the NRC (see EPRI Risk Impact Assessment of Extended Integrate Leak Rate Test Intervals, TR-1009325, plus the use of NUREG-1493). RBS is not an outlier in the use of this generic industry accepted data that addresses the operating experience-based probability of containment release pathways being larger than "small". Because the containment hatch doors have no dependencies, for the LCO condition, it is appropriate to increase the failure probability of the surrogate event in the ERAT program (versus setting to logical True) for the RICT calculation. This added probability represents the likelihood of failure of the redundant operable door.

A bounding individual door failure probability was derived by taking the square root of the pre-existing large isolation failure probability. The probability of the pre-existing large containment failure is 2.7E-3. First, the sum of all the failures making up the probability of this event can be applied to one penetration or airlock of interest for the RICT analysis. Second, to maximize the calculation of the individual barrier (isolation valve or airlock door) failure probability, there is no common cause contribution. Therefore, if the probability represents the logical AND of an inner and outer isolation failure, the square root of the failure probability (SQRT(2.7E-3) = 0.052) is the conservative failure probability that is applied to the PRECONTFAIL event, representing the operable airlock door.

g) ii. The value of 0.052 is conservative and bounding because it is based on the sum of probabilities of all containment leakage sources and applies the square root of that value to the individual remaining operable airlock door. Further, NUREG/CR-4220 predicts airlock unavailability is in the range of 5E-5, which would have a square root of 0.007 as an individual door unavailability. The proposed value is a factor of 7 higher than that value. Therefore, the proposed value is conservative and bounding.

g) iii. Not applicable based on response to APLA Question 1 g) ii.

h) i. The surrogate is the same as in response g) and will use the same value based on the same considerations. It should be understood, that using the surrogate creates an additional LERF conditional probability of 0.052 that is in addition to the normally calculated LERF conditional probability.

h) ii. Because the unmodeled paths are not LERF paths, PCIV inoperability is not expected to impact LERF. Therefore, adding a LERF conditional probability 0.052 that is in addition to the normally calculated conditional LERF is conservative and bounding.

h) iii. Not applicable based on response to APLA Question 1 h) ii.

i) i. The AC electrical power system is modeled in great detail and was marked as yes for being modeled. For the vast majority of AC switchgears, load centers, Motor Control Centers (MCCs), and panels in scope of the TS, there is a one-to-one relationship. However, there are a handful of MCCs or power panels identified in Note 13 of Table E1-1 that are addressed by the LCO condition, as described in the TS Basis document, that are not modeled explicitly. They are not modeled because at the time of building the PRA model these MCCs or panels were not identified as failing equipment credited in the PRA. During development of the LAR, these MCCs and panels were found in the TS Basis and needed to be addressed. The decision was made that if one of these MCCs or panels were to be inoperable, for the RICT calculation, the affected loads that are modeled in the PRA would be failed. The alternative would be to model each MCC or panel explicitly.

i) ii. "Best-estimate surrogate" refers to the specific loads of the unmodeled inoperable MCC or panel, as identified in OSP-0019, "Electrical Bus Outages," and the assumption that the loads are directly failed. It would have been more appropriate to simply call them "surrogates".

i) iii. The surrogate is conservative and bounding as the effected PRA loads are failed, regardless of what the actual function impacted is (e.g., monitoring circuit would not fail a load). The ERAT includes the unmodeled MCCs and panels as components that can be removed from service, and the component is mapped to fail the loads identified in OSP-0019 as affected.

i) iv. Not applicable based on response to APLA Question 1 i) iii.

Audit Question APLA-02 (TSTF-505) – Credit for FLEX Equipment and Actions:

NRC memorandum dated May 6, 2022[1] provides the NRC's staff updated assessment of identified challenges and strategies for incorporating Diverse and Flexible Mitigation Capability (FLEX) equipment into a PRA model in support of risk-informed decision making in accordance with the guidance of RG 1.200[2].

Section 3 of Enclosure 1 of the LAR states that a number of FLEX related sensitivities, which increased certain failure probabilities, demonstrated the impact on RICTs to be less than five percent. However, Section 3.2.9 of the RBS 10 CFR 50.69 application states that a sensitivity which removed FLEX credit impacted CDF by approximately twelve percent. The NRC staff notes that when assessing the uncertainties related to FLEX modeling the sensitivity is performed by removing FLEX credit. Given the twelve percent change associated with this uncertainty the staff notes it is possible for this source of uncertainty can significantly impact certain RICT calculations.

It is unclear exactly how the sensitivity was performed to assess FLEX's impact on the RICTs. Provide a more complete assessment and justification regarding how uncertainty in FLEX modeling could impact RICT calculations.

Entergy Response:

In order to evaluate the sensitivity of FLEX on the RICT durations, two RICT sensitivity cases were performed. These sensitivity cases were (1) Credit for FLEX Structures, Systems, and Components (SSC) basic events and (2) FLEX Human Error Probabilities (HEPs). Each of these sensitivity cases was run with seven RICT configurations that have both long and short RICT durations, including cases of two LCOs of major ECCS components coincidently entered.

The FLEX sensitivity study performed as part of the RBS internal events Rev. 7 PRA update effectively failed all FLEX equipment and did not credit any FLEX Human Failure Events (HFEs), which resulted in the 12% increase in Core Damage Frequency (CDF). This sensitivity is very

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[1] U.S. NRC memorandum, "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Risk Assessments," dated May 6, 2022 (ADAMS Accession No. ML22014A084).

[2] U.S. Nuclear Regulatory Commission, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," RG 1.200, Revision 3, December 2020 (ADAMS Accession No. ML20238B871).

conservative relative to the sensitivity performed to evaluate the adjustment of the failure rate for the FLEX events by a factor of 2, which is a more realistic approach to understanding how sensitive the selected cases are to crediting FLEX equipment and HFEs separately.

The RICT FLEX Sensitivity results, as provided in Table H.2 from Entergy Engineering Report PSA-RBS-08-11, are shown in the table below.

Table H.2: Results of FLEX Data Sensitivities.

OOS Configuration (LCO: SSCs)	Base RICT (days)	Sensitivity RICT (days)	% Change	Change (hours)
FLEX SSC Reliability Sensitivity				
3.7.1.F: SSWP 2A and SSWP 2D	14.41	13.96	-3.1%	10.7
3.5.1.C: RHS A & LPCI C	30.00	30.00	0.0%	0.0
3.5.1.C: LPCS & RHS B	30.00	30.00	0.0%	0.0
3.8.1.E: DIV II OSP & DG1	22.56	22.51	-0.2%	1.3
3.3.8.1.A: 4KV LOV DIV I & 3.8.1.C: DG3	30.00	30.00	0.0%	0.0
3.7.1.E: SSWP 2B & 3.8.1.C: DG3	20.45	19.56	-4.4%	21.5
FLEX Human Reliability Sensitivity				
3.7.1.F: SSWP 2A and SSWP 2D	14.41	14.07	-2.4%	8.2
3.5.1.C: RHS A & LPCI C	30.00	30.00	0.0%	0.0
3.5.1.C: LPCS & RHS B	30.00	30.00	0.0%	0.0
3.8.1.E: DIV II OSP & DG1	22.56	22.49	-0.3%	1.6
3.3.8.1.A: 4KV LOV DIV I & 3.8.1.C: DG3	30.00	30.00	0.0%	0.0
3.7.1.E: SSWP 2B & 3.8.1.C: DG3	20.45	19.69	-3.7%	18.2

Sensitivity 1

The FLEX equipment credited in the model are the portable diesel generators and diesel driven pumps. Each component has a failure to start and a failure to run basic event. The failure to start basic events failure rates were multiplied by a factor of 10. The failure rate of the failure to run events was doubled, which equals a multiplier of 1.71 based on an exposure time of 24 hours. The failure to start events were selected to have a higher multiplier than the run failures to address the effect of initial startup from an extended period of inactivity.

No changes greater than 5% in the RICT durations occurred as part of this sensitivity. The 5% threshold is used as a common indicator of risk sensitivity.

Sensitivity 2

All HEPs related to FLEX were multiplied by a factor of 2, including joint and independent HEPs.

No changes greater than 5% in the RICT durations occurred as part of this sensitivity.

Audit Question APLA-03 (TSTF-505) – Determination of Key Sources of Uncertainty and Sensitivity Results:

The NRC staff safety evaluation to NEI 06-09 specifies that the LAR should identify key assumptions and sources of uncertainty and to assess and disposition each as to their impact on the RMTS application.

NUREG-1855 "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, Main Report," (ADAMS Accession No. ML17062A466) presents guidance on the process of identifying, characterizing, and qualitative screening of model uncertainties.

Enclosure 9 of the LAR states that eleven internal events (including internal flooding) and eighteen fire key assumptions and uncertainties were identified. For certain sources of uncertainty sensitivity studies were conducted. However, none of these twenty-nine sources of uncertainty or sensitivity results were provided in the application. The NRC staff reviewed the uncertainty documents provided on this audit's electronic portal for the internal events, internal flooding, and fire PRA and found that further clarification is necessary regarding the review of these assumptions and sources of uncertainty for this application. It is unclear what additional analysis was performed and documented to determine if any source of uncertainty could adversely impact any RICT calculation. In light of these observations, provide the following information:

a) Provide details of how the RBS PRA sources of uncertainty were evaluated as a potential key source of uncertainty for this application. Include in this discussion any documentation of this process.

b) Provide the results of sensitivity studies that determined the impact on risk for each associated source of uncertainty. Include in this discussion justification that the sensitivity results demonstrate that the associated source of uncertainty does not adversely impact any RICT calculation.

Entergy Response:

a) The 29 sources of uncertainty were identified via a systematic evaluation of assumptions and generic sources of uncertainty documented in the base internal events (including internal flooding) and fire PRA model sources of uncertainty notebooks. These 29 sources of uncertainty were evaluated to determine if they were key sources of uncertainty for the TSTF-505 application. This evaluation consisted of a screening process using the following criteria:

- Criterion #1: Candidate uncertainties that are qualitatively shown to have a very small impact on total risk and would be expected to have a negligible impact on delta-CDF and delta LERF (particularly uncertainties that pertain to parts of the model that would not impact components that are in the RICT program, such as changes to non-support system initiating event frequencies, human error probabilities not related to RICT-eligible equipment, etc.).
- Criterion #2: Candidate uncertainties that are represented through conservative PRA modeling that would be expected to have a negligible or conservative impact on delta-risk RICT calculations.

- Criterion #3: Candidate uncertainties that were identified, but for which current industry-accepted approaches and data were used, are not considered as key sources of uncertainty. This is consistent with the ASME/ANS PRA Standard definition of a “source of modeling uncertainty” which states: “a source is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model”.
- Criterion #4: Candidate uncertainties that were examined via sensitivity studies to confirm that the impact on baseline CDF and LERF are negligibly small are not considered as key sources of uncertainty for the RICT program.

Nine of the 11 internal events sources of uncertainty and all 19 fire sources of uncertainty were screened based on these criteria. An example of the screening evaluation is shown below for one of the sources of uncertainty, as provided in Table 1 from Entergy report PSA-RBS-08-09 Revision 0:

Table 1: Potential Impacts of RBS Key Assumptions and Sources of Uncertainty (Internal Events/Internal Flood PRA)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
1	Credit for FPW injection under LOOP/SBO conditions	It is assumed that Fire Protection Water cannot be aligned for any station blackout sequences due to timing to complete the actions. If the service water valves needed for the alignment are in the desired position, less time would be required to complete the recovery action. No credit is taken for the type and timing of the DG failures which could result in the service water valves to be in the desired position to make the action feasible.	Modeling credit for FPW as an initial injection source in SBO conditions could potentially have a significant impact on the model due to the risk significance of low pressure injection.	Additional credit for FPW would reduce base case risk. Per Criterion #2, credit for FPW injection under LOOP/SBO conditions is not considered a key source of uncertainty because it is representative of conservative PRA modeling that may impact base risk, but would be expected to have a negligible impact on delta-risk RICT calculations. Note that a sensitivity (Reference 14) for this issue concluded that credit for FPW in SBO conditions would result in an overall 2.31% decrease in CDF.	This item does not represent a key source of uncertainty for the RICT calculations.

There were two sources of uncertainty that did not meet the screening criteria and required further evaluation. These two topics were 1) credit for containment airlock venting and 2) credit for recovery actions with limited procedural guidance. These sources of uncertainty required additional evaluation via sensitivity studies as described in the response to Question APLA-03 Part b.

b) As described in the response to Question APLA-03 Part a, additional evaluation via sensitivity studies was only required for two sources of uncertainty that did not meet the screening criteria. To evaluate the impact of these two sources of uncertainty on the TSTF-505 application, the potential impact of each source of uncertainty was evaluated and RICT-specific sensitivity cases were run to determine whether the source of uncertainty could significantly affect the calculated completion time.

<u>Credit for Containment Airlock Venting</u>

The operator action for containment airlock venting was determined to be a risk significant action, and the modeling to develop the HEP is uncertain due to concerns about the applied stress level. To evaluate the impact of this uncertainty, the independent HEP was multiplied by two and its joint probability HEPs were adjusted accordingly. The results of this sensitivity for relevant out of service (OOS) configurations are shown in Table 2 below.

Table 2: Results of Alternate Containment Venting HEP Sensitivities.

OOS Configuration (LCO: SSCs)	Base RICT (days)	(2X HEP) Sensitivity RICT (days)	% Change	Change (hours)
3.7.1.F: SSWP 2A and SSWP 2D	14.41	11.43	-20.7%	71.5
3.5.1.C: RHS A & LPCI C	30.00	30.00	0.0%	0.0
3.5.1.C: LPCS & RHS B	30.00	30.00	0.0%	0.0
3.8.1.E: OSP DIV II & DG1	22.56	22.51	-0.2%	1.3
3.3.8.1.A: 4KV LOV DIV I & 3.8.1.C: DG3	30.00	30.00	0.0%	0.0
3.7.1.E: SSWP 2B & 3.8.1.C: DG3	20.45	15.05	-26.4%	129.7

Two of the RICT configurations used in this study are sensitive to the uncertainty associated with the alternate containment venting HFE (TS 3.7.1.F and the combination of TS 3.7.1.E and TS 3.8.1.C). These two LCOs are sensitive to internal flood induced loss of service water when multiple divisions are impacted by the LCO. These two TS categories are representative of a class of LCO combinations involving two divisions with at least one system being standby service water. This would not be a typical planned maintenance configuration and would likely be a rare emergent plant condition.

If a configuration is entered (such as the two highlighted above) that elevate the impact of the uncertainty of the alternate containment venting actions, that action would also be elevated in importance in the configuration cutsets, as well as the redundant and diverse mitigation equipment and actions within those cutsets. Procedure ADM-0096, *Risk Management Program Implementation And On-Line Maintenance Risk Assessment*, directs the operations crew to identify risk management actions appropriate for the specific configuration that can be used to offset the impact of this uncertainty. The ERAT using the Phoenix Risk Monitor program will identify the important components, initiators, and human actions that arise for any specific plant configuration. Fire and flooding initiators can also be translated to the compartments to expand the scope of the risk management actions (RMAs). The importance measures are presented tabularly and graphically to support the RMA decision making. For instance, in the above shaded conditions, the configuration risk is dominated by certain initiators. RMAs for this configuration could include walkdowns or watches to minimize the occurrence of initiators, including potential flood and fire initiators (e.g., identify pipe leaks, remove transient combustibles). Important operator actions for the configuration would be reviewed by the operations staff. Diverse and redundant equipment would be verified operable.

The risk insights derived from the importance measures can be incorporated into the RMA procedure and/or into the (a)(4) protected equipment list per EN-OP-119, “Protected Divisions Postings.”

As an example, the potential RMAs for the highlighted configurations in Table 2 have been reviewed and identified for consideration. These RMAs (recoveries, initiators, and protected components) are examples only and are subject to change with data updates and refined analysis leading to model revisions.

When in the two highlighted configurations in Table 2, the ERAT identifies the most important human actions. The alternate containment venting action was identified as risk significant (by Fussell-Vesely importance) leading to the identification of the following RMA:

- Reinforce training on EOP-0005 Enclosure 21 venting of Primary Containment via the 171' Airlock inner door seal, including staging of the personnel protective equipment of EOP-0005 Step 3.6.7.15 (steam suit (preferred) or bunker gear and Self-Contained Breathing Apparatus (SCBA)).

The following is an example of an initiator-related RMA associated with the highlighted configurations:

- Inspect for leaks in flood areas identified as risk significant in the ERAT.

Initiator RMAs are based on the relative importance of the initiators. For the sensitivity case, Service Water (SW) piping breaks dominate internal events risk. Other initiators are less important to the sensitivity case.

RMAs will prioritize important equipment to return to service based on Fussell-Vesely importance. Since the uncertainty impact increases when two divisions are affected, the division with the inoperable SSC contributing most to risk will generally be prioritized to be returned to service unless the other SSC can be returned to service quickly.

RMAs will be developed to protect important redundant and diverse equipment based on risk achievement worth (RAW) importance. The ERAT lists the components with the highest RAW in the table labeled Protection Advice. Equipment in this list will be prioritized in RMA development that may include for example such measures as operability determinations, walkdowns, curtailment of work activities, and protective measures to prevent/mitigate flood and fire impacts.

Credit for Recovery Actions with Limited Procedural Guidance

There are 20 operator actions associated with internal flood source isolations and one operator action associated with recovering a mispositioned standby service water return-valve that have limited procedural guidance but are credited in the PRA model. The limited procedural guidance is due to the nature of the events requiring discovery of the cause and decision making for the mitigation strategy. For these events, the symptoms do not clearly define the nature of the event (e.g., break location, break size, isolation location) without localized discovery. Due to the limited guidance, the assigned HEPs may have embedded optimism and could potentially represent non-conservatism in the model. To evaluate this, the relevant HEPs were all increased to the 95^{th} percentile values simultaneously. The 95^{th} percentile HEPs represent the upper bound values and are suitable for this sensitivity analysis of the impact on RICT durations. The results of this sensitivity for relevant OOS configurations are shown in Table 3 below.

Table 3: Results of Limited Procedural Guidance HEP Sensitivities.

OOS Configuration (LCO: SSCs)	Base RICT (days)	(95th HEP) Sensitivity RICT (days)	% Change	Change (hours)
3.7.1.F: SSWP 2A and SSWP 2D	14.41	14.39	-0.11%	-0.4
3.5.1.C: RHS A & LPCI C	30.00	30.00	0.00%	0.0
3.5.1.C: LPCS & RHS B	30.00	30.00	0.00%	0.0
3.8.1.E: OSP DIV II & DG1	22.52	22.45	-0.29%	-1.6
3.3.8.1.A: 4KV LOV DIV I & 3.8.1.C: DG3	30.00	30.00	0.00%	0.0
3.7.1.E: SSWP 2B & 3.8.1.C: DG3	20.46	20.43	-0.18%	-0.9

Using a 5% change in RICT duration as a threshold for risk sensitivity to the source of uncertainty, these results show that no RICTs were sensitive to increasing the HEPs for the actions with limited procedural guidance.

<u>Audit Question APLA-04 (TSTF-505) – Digital Instrumentation and Control Modeling:</u>

Concerning the quality of the PRA model, NEI 06-09-A states that RG 1.174 and RG 1.200 define the quality of the PRA in terms of its scope, level of detail, and technical adequacy. The quality must be compatible with the safety implications of the proposed TS change and the role the PRA plays in justifying the change.

Regarding digital instrumentation and control (I&C), the NRC staff notes the lack of consensus industry guidance for modeling these systems in plant PRAs to be used to support risk-informed applications. In addition, known modeling challenges exist, such as the lack of industry data for digital I&C components, the difference between digital and analog system failure modes, and the complexities associated with modeling software failures including common-cause software failures. Also, though reliability data from vendor tests may be available, this source of data is not a substitute for in-the-field operational data. Given these challenges, the uncertainty associated with modeling a digital I&C system could impact the RICT program. Therefore, address the following:

a) Clarify whether digital I&C systems are credited in the PRA models that will be used in the RICT program.

b) If digital I&C systems are credited in the PRA models that will be used in the RICT program, provide justification that demonstrates the modeling uncertainty associated with crediting digital I&C systems has an no adverse impact on the RICT calculations

Entergy Response:

a) The only system credited in the RBS PRA with digital instrumentation and control is the Control Building chilled water system (system designator HVK). Digital controls were installed on the Control Building chillers, HVK-CHL1A/B/C/D, under EC31803 to improve system reliability and availability.

The control building chilled water system includes 4 100% capacity Chillers, two per Division, and is designed to remove heat generated within the control building to maintain required environmental conditions. It is a closed loop cooling system consisting of chillers, circulating pumps, piping, valves, compression tank, instrumentation and controls. It provides continuous flow of chilled water to the control building Air Conditioning Units (ACU's) during normal, shutdown, and design basis accident conditions.

EC31803 replaced aging control components with modern digital equipment which provides increased availability and flexibility. The replacement equipment mimics the function of the existing equipment.

b) Since the function of the compressor is to provide cooling for control building air, the failure modes modeled in the RBS PRA are unchanged (e.g., Failure to Start, Failure to Run). Specific failure modes associated with the digital controls of this component are beneath the level of detail of the RBS PRA. Digital controls for the chilled water compressor do not introduce any new interactions with other systems, unlike digital controls associated with, for example, Reactor Protection or Feedwater Level Control systems. There are no special failure modes associated with digital controls which impact the PRA or require modeling in the PRA, as for PRA system modeling purposes there is no difference here between analog and digital chiller controls.

The Control Building Chilled Water System consists of four 100% capacity chillers. Should the aligned chiller fail, the system automatically starts the selected Backup chiller in the opposite division. If the Backup chiller would also fail, operator manual action to align and start either of the other two remaining chillers is credited. The operator action to align and start the remaining chillers must be performed within four hours to maintain acceptable temperatures for 4160 volt Divisional switchgear. Should none of the chillers respond, operator action is required to open Switchgear room doors within 4 hours to provide adequate natural circulation cooling of the Switchgear rooms, based upon GOTHIC room temperature calculations.

The risk significance of the chillers is low, with maximum FV values of less than 1E-04 for Internal Event CDF and less than 4E-04 for Fire CDF and maximum RAW values of 1.03 for Internal Events CDF and less than 1.4 for Fire LERF.

The digital controls are expected to result in significant improvement in component reliability. Based on the low risk significance and available mitigating actions in event of a chiller failure, there are no additional modeling uncertainties associated with adoption of digital controls for the Control Building Chilled Water system and the system would have no adverse impact on RICT calculations.

Audit Question APLA-05 (TSTF-505) – Impact of Seasonal Variations:

The Tier 3 requirement of Regulatory Guide (RG) 1.177, Revision 2, "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications, dated January 2021, stipulates that a licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity.
Section 2.3.4 of NEI 06-09-A states, in part, that:

> *If the PRA model is constructed using data points or basic events that change as a result of time of year or time of cycle…, then the RICT calculation shall either 1) use the more conservative assumption at all time, or 2) be adjusted appropriately to reflect the current (e.g., seasonal or time of cycle) configuration for the feature as modeled in the PRA.*

Enclosure 8 of the LAR states that outside air temperatures on system requirements and severe weather conditions is addressed in the CRMP model. However, it does not appear to specify the modeling adjustments needed to account for seasonal variations and what kind of adjustments will be made. Therefore, address the following to clarify the treatment of seasonal and time of cycle variations:

a) Explain how the RICT calculations address changes in PRA data points, basic events, and SSC operability constraints as a result of extreme weather conditions, seasonal variations, or other environmental factors. Also, explain how these adjustments are made in the configuration risk management program (CRMP) model and how this approach is consistent with the guidance in NEI 06-09-A and its associated NRC final SE.

b) Describe the criteria used to determine when PRA adjustments due to extreme weather conditions, seasonal variations, other environmental factors, or time of cycle variations need to be made in the CRMP model and what mechanism initiates these changes.

Entergy Response:

a) The ERAT addresses both seasonal temperature variations and severe weather

conditions. There are three temperature ranges modeled in the ERAT model, each having an impact on the required number for cooling fans for the Normal Service Water (NSW) system to operate. This impacts the loss of NSW initiating event frequency and response model. Severe weather conditions are modeled in the ERAT as both increasing the loss of offsite power frequency and changing the offsite power recovery curve to use only the severe weather curve (higher failure probabilities of recovery at each time interval) versus a composite recovery curve. A no-weather curve is reserved for "clear-skies" conditions, and a composite curve is reserved for "weather-watch" conditions. This approach is consistent with the guidance in NEI 06-09-A, which states:

> "If the PRA model is constructed using data points or basic events that change as a result of time of year or time of cycle (examples include moderator temperature coefficient, summer versus winter alignments for HVAC, seasonal alignments for service water), then the RICT calculation shall either 1) use the more conservative assumption at all time, or 2) be adjusted appropriately to reflect the current (e.g., seasonal or time of cycle) configuration for the feature as modeled in the PRA. Otherwise, time-averaged data may be used in establishing the RICT.

> CRM tools should explicitly model external conditions, such as weather impacts, or a process to adequately address the impact of these external conditions exists."

In summary, RBS models both the impacts of time of cycle (seasonal variations) and external hazards (severe weather), and is consistent with the guidance in NEI 06-09-A.

b) The guidance for outdoor temperature and severe weather are provided in ADM-0096, "Risk Management Program Implementation and On-Line Maintenance Risk Assessment."

Before assessing a plant configuration for 10CFR50.65 (a)(4) or RICT, outdoor temperature daily high is determined, and the temperature threshold is set in the ERAT in the system alignment.

Similarly, for severe weather, if the plant is in a severe weather watch, or severe weather warning as determined by AOP-0029, "Severe Weather", this will determine how the weather conditions are changed in the ERAT prior to performing an evaluation for (a)(4) or RICT.

Work is scheduled, if possible, to avoid an Orange condition in the event that an unanticipated AOP-0029 entry occurs. For emergent work, the risk evaluations consider the weather condition, and a new RICT duration would be calculated for each configuration change, including changes in environmental conditions. The goal is to avoid an Orange condition in the configuration, implement RMAs, and have contingencies in place to minimize risk.

Audit Question APLC-01 (TSTF-505 and 50.69) – Seismic Risk Contribution Analysis:

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 Technical Specifications] 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 [Risk-Informed Completion Time] 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."

In Section 3 of Enclosure 4 to the LAR, the licensee provided its seismic risk contribution analysis. The licensee concluded that RBS is more robust than was credited in the GI-199 and provided the HCLPF of 0.3g and a composite uncertainty factor (β_c) of 0.5 as plant level fragility. The NRC staff noted that GI-199 shows HCLPF = 0.1g and β_c = 0.4 for RBS, which is consistent with an EPRI document dated March 11, 2014 (ML14080A589). The licensee provided two seismic re-evaluation documents to support its plant-level fragility (PLF), RBS-SA-11-00001, Revision 0, EC93084 and PSA-RBS-04-021, Revision 0, EC93084 on the portal for NRC staff review. The NRC staff reviewed the documents and identified the following questions:

1. *The calculation RBS-SA-11-00001, Revision 0, EC93084 discusses different approaches to estimating the PLF and provides several different sets of HCLPF and β_c estimates in Attachments 4, 7 and 8. These approaches include the separation of variables (SOV) method, the hybrid method, and a scaling method based on the SSE to GMRS ratio.*

   a) *The staff notes that the HCLPF and βc developed based on SOV in Attachment 4 are used to support the RBS RICT program. The staff understands that the median factors of safety and variabilities provided in Kennedy et al. (1980, 1984) are used in SOV estimation of the PLF in Attachment 4. To the best of staff's understanding, SOV method is used to determine the fragility of individual systems, structures, or components (SSCs) which is then used for a SMA or seismic PRA (SPRA). It is the SMA or SPRA that provides the PLF, which is a representation of the combined behavior of all the modeled SSCs. Therefore, the PLF depends on a plant-specific mix of SSCs, and based on the staff's understanding, the contribution is usually higher from components compared to structures for seismic CDF. To the best of staff's understanding, the SOV method has not been applied to directly determine the PLF and its application by the licensee is beyond the scope of its applicability. Provide justification for the first-of-a-kind use of SOV in estimating the PLF and plant-specific basis for selecting the parameters and their values for the SOV method in the licensee's calculation.*

   b) *For the SOV method used in Attachment 4, the licensee states that the various factors and estimate values are based on the available material. However, it does not specify the "available material" used for estimates for various factors used in SOV. Provide the sources of information for the median factors used in the SOV method in Attachment 4.*

   c) *In Attachment 7, the licensee also used the SOV method. Attachment 7 states that "[t]his white paper provides a basis for the HCLPF and fragility calculations performed in Reference 11." Reference 11 in the quoted statement refers to Attachment 4 discussed above. However, the staff notes that Attachment 7 uses different values for median factors than those used in Attachment 4. Justify the use of different values for median factors in the SOV method described in Attachments 4 and 7.*

   d) *In the SOV method of estimating an HCLPF in Attachment 7, the staff found an error in the calculation of HCLPF value. Correction of the error results in a calculated HCLPF of 0.20g. This HCLPF of 0.20g is lower than the HCLPF of 0.30g developed in Attachment 4 and used in the RICT. Justify the use of an unconservative value (HCLPF of 0.30g PGA) in the RICT program.*

   e) *Several different sets of HCLPF and βc estimates are also developed using the hybrid method in Attachments 4, 7 and 8. However, justification for the use of the minimum capacity of 0.8g ground peak spectral acceleration as the screening criteria for SSCs at RBS is not provided. Provide site-specific justification for the use of 0.8g ground peak spectral acceleration as the screening criteria.*

2. *In Attachment 8, the licensee calculated a scaling factor of 1.3 at 1 Hz based on an SSE to*

GMRS demand ratio. This method, which was approved by the staff for Waterford TSTF-505 LAR, can provide an alternative to SOV and hybrid methods discussed in Item #1 above, especially for determining the PLF for seismic CDF. If this method is adopted by the licensee to determine the PLF for seismic CDF for use in the RICT LAR, provide detailed steps involved in calculations and the resulting HCLPF value and selected β_c.

3. *As an alternative to items #1 and #2 above, the licensee may choose an option for a full-scope SMA to determine the PLF. The licensee may upgrade its reduced-scope SMA performed as part of the RBS IPEEE to a full-scope or focused-scope SMA consistent with appropriate NRC-endorsed guidance and leveraging to the extent possible, with justification, prior plant-specific walkdowns, such as those performed in response to post-Fukushima actions. If this approach is adopted by the licensee, please provide resulting reports, calculation notebooks, and conclusions for NRC staff review in a regulatory audit.*

4. *Re-evaluate and provide the seismic penalty based on updated HCLPF and βc values, if they are different from those provided in the LAR.*

Entergy Response:

1.a) Calculation RBS-SA-11-00001 Attachment 4 first developed the plant-level fragility (PLF) following the Hybrid and SOV methodologies using heuristic judgements for a site located in low seismicity region. Attachment 7 then attempted to independently justify the median factors and composite variability employed in Hybrid and SOV calculations in Attachment 4. As noted, RBS is a reduced scope seismic margins assessment (SMA) plant for Individual Plant Examination for External Events (IPEEE) purposes. Per Report NE-RA-93-009-M, "Seismic IPE Review River Bend Nuclear Station Unit -1" the IPEEE seismic walkdowns followed screening guidelines in Tables 2-3 and 2-4 of Electric Power Research Institute (EPRI) NP-6041 for screening of structures and equipment and established that RBS is seismically rugged and all components in the success path logic diagram (SPLD) are screened out with a minimum screening spectral acceleration of 0.80g. No High Confidence of Low Probability of Failure (HCLPF) calculations were performed as it was not necessary for the reduced scope IPEEE effort. No relay assessment, soil failure, or anchorage evaluations were necessary. As acknowledged in Attachment 7, a detailed SOV calculation is impossible to perform without a detailed response analysis and capacity re-assessment for plant systems. Since all components of the SPLD were seismically rugged, the reactor building was chosen as a representative surrogate for PLF. Further, with no detailed data available to perform an SOV evaluation, the range of median factors and variabilities prescribed in Kennedy et al. (References 6 and 7 of Attachment 7) for a reinforced concrete reactor building structure was used as a surrogate to verify the heuristic fragility developed in Attachment 4. Because this approach did not succeed in establishing a HCLPF of 0.30g, RBS explored another alternative to independently assess the PLF using the Hybrid Method documented in Attachment 8. In addition to IPEEE walkdowns, Attachment 8 also considered the experience based seismic capacity, design basis information for relay chatter, soil failure, and anchorage to justify the use of a 0.30g HCLPF for the seismic penalty. Attachments 7 and 8 are independent reviews of the methodology provided in

Attachment 4. The final seismic penalty calculation used for the LAR submittal was based on the hybrid method described in Attachment 8.

As result of the NRC audit meeting on October 12, 2023, Calculation RBS-SA-11-00001 was replaced by a new evaluation documented in Entergy Report PSA-RBS-04-02, Rev 0 for the RBS plant level and containment seismic fragilities. The new fragility evaluation provides a plant level HCLPF of 0.16g and a containment HCLPF of 0.30g with a βc of 0.40. The revised plant level and containment seismic fragilities are based on the scaling method rather than the hybrid method.

1.b) See the response to APLC Question 1.a).

1.c) See the response to APLC Question 1.a).

1.d) See the response to APLC Question 1.a). Attachment 7 provided an independent review of the SOV methodology using the range of median factors and variabilities in Kennedy et al (References 6 and 7 of Attachment 7). This was used as a surrogate fragility. However, the values cited in Attachment 7 were not used in the seismic penalty calculation. The math error identified in Attachment 7 is being addressed through the corrective action program.

1.e) See the response to APLC Question 1.a). The seismic walkdowns for SMA documented in Section 5 of Report NE-RA-93-009-M, "Seismic IPE Review River Bend Nuclear Station Unit -1" used the screening guidelines in Tables 2-3 and 2-4 of EPRI-6041-SL for the spectral accelerations less than 0.80g. Section 6 of the same report concludes that RBS is seismically rugged and all components in the SPLD are screened out with no outliers requiring further evaluation. Section 2.0 of Attachment 8 indicates that a seismic walkdown was performed to meet the requirements of IPEEE per NUREG-1407 and EPRI NP-6041-SL. RBS used the reduced-scope SMA approach. The reduced-scope process included the development of a structure, system, and component (SSC) list per Section 3 of EPRI-6041-SL and focused on seismic walkdowns to identify weak-link items that need strengthening. The intent of this process was essentially a confirmation against design-basis with no additional seismic margins work. However, the RBS IPEEE effort was performed with sufficient rigor to demonstrate a 0.80g peak spectral capacity in accordance with EPRI NP-6041-SL. Appendix A of EPRI-6041-SL provides the basis for seismic capacity screening guidelines for SSCs. The discussion of screening guidelines in Appendix A, are heavily based on the recommendations of the expert panel on the quantification of seismic margins in NUREG/CR-4334. Per the IPEEE pre-screening assessment, the major structures and SPLD equipment at RBS were screened in the first screening lane and assigned a spectral capacity of 0.80g. The screening caveats were validated during the walkdowns following Appendix A and Appendix F of EPRI NP-6041-SL to confirm a spectral capacity of at least 0.80g. See the response to APLC Question 1.a).

2. See the response to APLC Question 1.a).

3. The purpose of Attachment 8 was to sufficiently close the gap between a reduced-scope SMA and a focused-scope SMA by reviewing the walkdown results per Section 3.2.4.1 of NUREG-1407 (Sections 2.0 and 4.2 of Attachment 8), performing relay evaluation per Section 3.2.4.2 of NUREG-1407 (Section 4.3 of Attachment 8), evaluation of soil failures per Section 3.2.4.3 (Section 4.5 of Attachment 8) and reviewing design information to determine

whether a HCLPF of 0.30g could be supported. It was concluded that the 0.30g HCLPF calculation is consistent with the guidance provided in Section 3.2.4 of NUREG-1407, while utilizing the Ground Motion Response Spectrum (GMRS) as the anchor reference. As a result of the NRC audit meeting on October 12, 2023, Calculation RBS-SA-11-00001 was replaced by a new fragility evaluation using the scaling method. See the response to APLC Question 1.a).

4. The RBS seismic penalty calculation was updated to address the revised RBS plant level HCLPF of 0.16g, containment HCLPF of 0.30g, and βc of 0.40. The seismic CDF is 3.93E-06 and the seismic LERF is 5.34E-07. The seismic CDF is ~18% of the total CDF (including Full Power Internal Events (FPIE), Fire, and Seismic), and the seismic LERF is ~39% of the total LERF (including FPIE, Fire, and Seismic). Note that the plant level fragility (0.16g HCLPF) is considered to be bounding based on the conservatism in the approach; therefore, the actual seismic CDF contribution is expected to be significantly lower than what is calculated in the seismic penalty calculation.

Audit Question APLC-02 (TSTF-505 and 50.69) – Extreme Wind or Tornado Screening Criteria:

Section 2.3.1, Item 7, of NEI 06-09-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 NRC staffs SE for NEI 06-09 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."

The table in Enclosure 4 of the TSTF-505 LAR screens the Extreme Wind or Tornado as PS4, "mean CDF is < 1E-06 per year." However, the table did not provide the high wind CDF value to meet the criteria, except for tornado missile hazard. The licensee did provide structure design criteria for straight wind and tornado. It appears this meet the criterion C1 for "Event damage is < events for which plant is designed."

Provide additional justification to screen the extreme wind or tornado hazard using criterion "PS4".

Entergy Response:

We agree with the NRC staff that the Extreme Winds or Tornados could have been identified as meeting screening Criterion C1 in the initial screening. However, since there were some non-conformances identified, additional evaluation was performed, which showed that the risk of tornado missiles meets screening Criterion PS4. Therefore, it can be concluded that this hazard screens based on a combination of Criteria C1 and PS4.

Audit Question EEEB-01 (TSTF-505) – LCO 3.8.4, Conditions A & B & LCO 3.8.9, Conditions A & C:

GDC 17 requires, in part, that both offsite and onsite electrical power systems be provided. LCO 3.8.4, Conditions A and B and LCO 3.8.9, Conditions A and C are for exclusively the inoperability of Division I or II battery charger or subsystem(s), respectively. LCO 3.8.4, Condition C and LCO 3.8.9, Condition E are for just Division III subsystem(s).

USFAR R27, Sections 8.1.6.2 and 8.3.1.1.2.1 reveal that the onsite electrical system has three 4.16 kV physically and electrically independent standby buses with each serving a safety-related division (load group). UFSAR Page 8.3-45 shows that two out of three load groups can provide the minimum safety functions to shut down the unit and maintain it in a safe shutdown condition. UFSAR Section 8.3.2 indicates that Class 1E DC systems have the same single failure criteria as the AC safety-related divisions to they provide control power. Required Action (RA) for TS LCO 3.8.4.B.1 seeks restoration of a Division I and II subsystems to operable status, as do RAs for TS LCOs 3.8.9.A.1 and C.1 for their respective Division I and II AC and DC subsystems.

Please explain why the DSC in Table E1-1 for TS LCO 3.8.4, Conditions A & B & TS LCO 3.8.9, Conditions A & C are not exclusively for Division I and II components and subsystem(s) without any reference to Division III which has separate LCOs for inoperability of its subsystem(s).

Entergy Response:

The Design Success Criteria (DSC) was written in terms of the remaining minimum load groups available to meet the safety function. The DSC requires two of the three load groups, so Division III was included. This is consistent with the TS Basis. However, Division III is not a design basis backup for Division I and II power and the listed DSC in Table E1-1 may have been confusing. Additionally, some DSCs were found to be in error either due to crediting a non-safety backup or failing to credit the safety-backup as meeting the LCO RA (see responses to TSTF-505 EEEB Question 2 and TSTF-505 EEEB Question 3).

The DSCs will be clarified in terms of only Division I and II load groups as follows:

RBS TS	TS Description	Design Success Criteria
3.8.4.A	One required battery charger on Division I or II inoperable	With one required Division I OR Division II battery charger inoperable, the design success criterion is met by the operable Division I OR Division II battery charger.
3.8.4.B	Division I or II DC electrical Power subsystem inoperable for reasons other than Condition A	With one Division I OR II DC electrical power subsystem inoperable, the design criterion is met by the operable Division I or II DC electrical power subsystem.
3.8.9.A	One or more Division I or II AC electrical power distribution subsystems inoperable	With one or more Division I OR Division II AC electrical power distribution subsystems inoperable, the design criterion is met by the operable Division I OR Division II AC electrical power distribution subsystems.

RBS TS	TS Description	Design Success Criteria
3.8.9.C	One or more Division I or II DC electrical power distribution subsystems inoperable	With one or more Division I OR Division II DC electrical power distribution subsystems inoperable, the design criterion is met by the operable Division I OR Division II DC distribution subsystems.

Attachment 3 of this letter contains mark-ups of Table E1-1 pages. Note that these include changes in response to TSTF-505 EEEB-02 and TSTF-505 EEEB-03.

Audit Question EEEB-02 (TSTF-505) – LCO 3.8.7, Condition A:

GDC 17 requires, in part, that both offsite and onsite electrical power systems be provided. At River Bend Station, UFSAR Section 8.3.1.1.3.5 indicates the 120-V ac uninterruptible power supplies (UPS) provide ac power for security, control, and instrumentation systems for the non-safety-related and engineered safeguard systems. LCO 3.8.7, Condition A, is for one inverter inoperable in either Division I or II.

USFAR R27, Section 8.3.1.1.3.7 reveals each UPS has an inverter as an essential component. UPSs ENB-INV01A and ENB-INV01A1 are associated with Division I, and UPSs ENB-INV01B and ENB-INV01B1 are associated with Division II. Only one UPS per division is required to be in service at any given time to supply power to its respective distribution panel with the other UPS for that division de-energized and available as a backup. All UPSs and their associated distribution panels are completely independent (by division). Those panels associated with standby systems serve redundant safety-related equipment.

Please explain why the DSC in Table E1-1 for TS LCO 3.8.7, Condition A states that the design success criterion is met by the associated spare inverter for that division" since the Division I and II inverters serve redundant loads (only one inverter required per division).

Entergy Response:

The DSC for Table E1-1 was incorrect. It should have read:

> "With one division inoperable due to the inverter, the design success criterion is met by the operable division with an operable inverter."

Attachment 3 of this letter contains mark-ups of Table E1-1 pages. Note that these include changes in response to TSTF-505 EEEB-01 and TSTF-505 EEEB-03.

Audit Question EEEB-03 (TSTF-505) – Table E1-1 for TS LCO 3.8.4, Condition A:

GDC 17 requires, in part, that both offsite and onsite electrical power systems be provided. UFSAR Section 8.3.2.1.1 indicates there are three Class 1E safety related 125-Vdc systems, and each 125-Vdc system has one battery charger. UFSAR Section 8.3.2.1.1 also states that a *separate battery charger, powered by either a non-safety-related power source or a portable diesel generator, performs as the backup battery charger for the three safety divisions I, II, and III safety-related and three of the non-safety-related chargers. The backup charger's breaker is taken from storage and placed in position to feed the bus of battery charger removed from service. Operation of the backup charger is under strict administrative control in that credit is taken for this charger in mitigating consequences of an accident, when used as a substitute for a Division I or II safety-related charger.*

Please explain why in Table E1-1 for TS LCO 3.8.4, Condition A assumes credit of backup charger for a DSC, even though no Surveillance Requirements are specified in TS 3.8.4 for the backup charger.

Entergy Response:

Table E1-1 is in error regarding the crediting of the backup charger as an equivalent design basis backup. See the response to TSTF-505 EEEB-01 for the corrected DSC for LCO 3.8.4 Condition A.

Audit Question EICB-01 (TSTF-505) – Loss of Function statements:

The TSTF-505, Revision 2 excludes loss of function (LOF) conditions. Please clarify whether below functions includes LOF conditions:

LCO 3.3.5.3 Action B

LCO 3.3.5.3 Action D

LCO 3.3.6.3 Action B.2

LCO 3.3.6.3 Action C.2

LCO 3.3.6.4 Action A.1

Entergy Response:

RCIC Instrumentation

LCO 3.3.5.3 Action B

If one or more Function 1 channels of RCIC instrumentation become inoperable, both required actions (RAs) 3.3.5.3.B.1 and 3.3.5.3.B.2 apply, but a loss of function is only addressed by RA B.1. If sufficient inoperable channels result in a loss of initiation capability (i.e., loss of function) that persists for one hour or more, RA B.1 requires RCIC to be declared inoperable resulting in entry into TS 3.5.3. According to the TS Basis document for RA B.1: "In this situation (loss of automatic initiation capability), the 24 hour allowance *(or the proposed RICT)* of Required Action B.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour after discovery of loss of RCIC initiation capability." If sufficient channels are restored or tripped to restore the initiation capability (i.e., restoration of the lost function), RCIC would no longer be declared inoperable, and RA B.2 would continue to track the Completion Time from original time of discovery to address each remaining inoperable channel. Since RA B.2 does not address a loss of function, a note to prohibit RICT entry for a loss of function was not applied to RA B.2. This is consistent with the BWR/6 Template for RCIC instrumentation as documented in the Technical Specifications Task Force Improved Standard Technical Specifications Change Traveler, TSTF-505-A, Revision 2.

LCO 3.3.5.3 Action D

If one or more Function 3 or Function 4 channels of RCIC instrumentation become inoperable, RAs 3.3.5.3.D.1 and either 3.3.5.3.D.2.1 or 3.3.6.3.D.2.2 apply, but a loss of function is only addressed by RA D.1. If sufficient inoperable channels result in a loss of automatic suction swap (i.e., loss of function) that persists for one hour or more, RA D.1 requires RCIC to be declared inoperable resulting in entry into TS 3.5.3. According to the TS Basis document for RA D.1: "In this situation (loss of automatic suction swap), the 24 hour allowance *(or the proposed RICT)* of Required Action D.2.1 and D.2.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour of discovery of loss of RCIC initiation capability." If sufficient channels are restored or tripped to restore the automatic suction swap (i.e., restoration of the lost function), RCIC would no longer be declared inoperable, and RA D.2.1 would continue to track the Completion Time from original time of discovery to address each remaining inoperable channel. Since RA D.2.1 does not address a loss of function, a note to prohibit RICT entry for a loss of function was not applied to RA D.2.1. This is consistent with the BWR/6 Template for RCIC instrumentation as documented in the Technical Specifications Task Force Improved Standard Technical Specifications Change Traveler, TSTF-505-A, Revision 2.

Containment Unit Cooler (CUC) Instrumentation

LCO 3.3.6.3 Action B.2

If one or more Function 1, Function 2, or Function 3 channels of CUC instrumentation become inoperable, both required actions (RAs) 3.3.6.3.B.1 and 3.3.6.3.B.2 apply, but a loss of function is only addressed by RA B.1. If sufficient inoperable channels result in a loss of initiation capability (i.e., loss of function) that persists for one hour or more, RA B.1 requires the associated CUC subsystems to be declared inoperable resulting in entry into TS 3.6.1.7. According to the TS Basis document for RA B.1: "In this situation (loss of automatic initiation capability), the 24 hour allowance (*or the proposed RICT*) of Required Action B.2 is not appropriate and both Containment Unit Cooler subsystems, made inoperable by Containment Unit Cooler System instrumentation, must be declared inoperable within 1 hour after discovery of loss of Containment Unit Cooler System initiation capability for both trip systems." If sufficient channels are restored or tripped to restore the initiation capability (i.e., restoration of the lost function), CUCs would no longer be declared inoperable, and RA B.2 would continue to track the Completion Time from original time of discovery to address each remaining inoperable channel. Since RA B.2 does not address a loss of function, a note to prohibit RICT entry for a loss of function was not applied to RA B.2.

LCO 3.3.6.3 Action C.2

If a CUC instrumentation Function 4 channel becomes inoperable, both required actions (RAs) 3.3.6.3.C.1 and 3.3.6.3.C.2 apply, but loss of function is only addressed by RA C.1. If a loss of initiation capability (i.e., loss of function) occurs and persists for one hour or more, RA C.1 requires the associated CUCs to be declared inoperable resulting in entry in tor TS 3.6.1.7. According to the TS Basis document for RA C.1: "In this situation (loss of automatic initiation capability), the 24 hour allowance (*or the proposed RICT*) of Required Action C.2 is not appropriate and both of the associated Containment Unit Cooler subsystems must be declared inoperable within 1 hour after discovery of loss of Containment Unit Cooler System initiation capability for both trip systems." If sufficient channels are restored to operable to restore initiation capability (i.e., restoration of the lost function), CUCs would no longer be declared inoperable, and the RA C.2 would continue to track the Completion Time from the original time of discovery to address the remaining inoperable channel. Since RA C.2 does not address a loss of function, a note to prohibit RICT entry for a loss of function was not applied to RA C.2.

<u>Relief and LLS Instrumentation</u>

LCO 3.3.6.4 Action A.1

The LCO is for one trip system of reactor steam dome pressure instrumentation inoperable. With one trip system inoperable, there is no loss of automatic initiation capability. Therefore, a loss of function statement to preclude RICT is not required. Actions B.1 and B.2 address loss of initiation capability and no RICT is applied to these actions.

<u>Audit Question EICB-02 (TSTF-505) – Redundant and Diverse Instrumentation Table:</u>

Attachment 5 Table 1, please clarify the mapping relationship between the "Transient Accident" column and the "Redundant/Diverse Instrumentation" column. For example, for Function #3 Reactor Vessel Steam Dome Pressure-High, are all functions listed in the 4^{th} Column diversities for each individual transient in Column #3?

<u>Entergy Response:</u>

Yes, for RPS Function 3, "Reactor Vessel Steam Dome Pressure – High", all functions listed in the 4^{th} column (Redundant/Diverse Instrumentation) are diverse instruments for each transient listed in the 3^{rd} column. There are six accident event types identified in the USAR (listed in Columns 2 and 3) that could initially cause a reactor vessel steam dome pressure high signal. For these six events, five automatic signals (listed in Column 4) are expected to occur in response to any of these six events during the transient period. These five automatic signals include four diverse signals and the redundant operable channels (in italics) to the LCO condition of the function of interest.

RPS Function 3 formatting is somewhat different from some of the other RPS functions (to reduce the size of the table) in that the redundant/diverse instrumentation was not repeated for each transient accident listed. This may have caused some confusion. However, this formatting is consistent across all the tables in Attachment 5 where the redundant/diverse instrumentation is the same for multiple accident types.

<u>Audit Question STSB-01 (TSTF-505) – TS 3.8.1 Required Action A.3, Proposed RICT Placement:</u>

In Attachment 2 of the LAR, the proposed change to add risk informed completion times (RICTs) for River Bend TS required action 3.8.1.A.3 (Restore required offsite circuit to OPERABLE status) is:

72 hours

<u>OR</u>

*In accordance with the Risk Informed
Completion Time Program*

<u>AND</u>

*24 hours from discovery of two
divisions with no offsite power*

<u>OR</u>

*In accordance with the Risk Informed
Completion Time Program*

NRC staff recognizes that the licensee's proposed change is consistent with the NUREG-1434 TS markups in TSTF-505, Revision 2. However, it has been brought to staff's attention that some of the TSTF-505 markups contain errors, introducing potential for licensee actions to be less conservative than the original intent of the requirements. To modify completion times that include the phrase "from discovery," the RICT shall start at discovery instead of the time the TS Action statement is entered, or the normal "time zero". This requirement is not clear when the RICT statement is separated from the "from discovery" statement.

To provide clarity, revise the placement of the proposed completion time for TS required action 3.8.1.A.3 similar to the placement of the proposed completion time for required action 3.8.1.C.4 (Restore required DG to OPERABLE status), which inserts "or in accordance with the Risk Informed Completion Time Program" between "72 hours" and "from discovery."

<u>Entergy Response:</u>

The placement of the proposed completion times for TS required action 3.8.1.A.3 is revised to match the placement of the proposed completion time for required action 3.8.1.C.4. Attachment 4 of this letter contains a revised TS markup page.

Attachment 2

RBG-48271

Responses to NRC Audit Questions
Related to the 10 CFR 50.69 LAR

(13 pages follow)

RESPONSES TO NRC AUDIT QUESTIONS

NOTE: The U.S. Nuclear Regulatory Commission (NRC) staff's audit questions are in italics throughout this attachment to distinguish from the Entergy Operations, Inc. (Entergy) responses.

<u>Audit Question APLC-01 (TSTF-505 and 50.69) – Seismic Risk Contribution Analysis:</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 Technical Specifications] 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 [Risk-Informed Completion Time] 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."

In Section 3 of Enclosure 4 to the LAR, the licensee provided its seismic risk contribution analysis. The licensee concluded that RBS is more robust than was credited in the GI-199 and provided the HCLPF of 0.3g and a composite uncertainty factor (β_c) of 0.5 as plant level fragility. The NRC staff noted that GI-199 shows HCLPF = 0.1g and β_c = 0.4 for RBS, which is consistent with an EPRI document dated March 11, 2014 (ML14080A589). The licensee provided two seismic re-evaluation documents to support its plant-level fragility (PLF), RBS-SA-11-00001, Revision 0, EC93084 and PSA-RBS-04-021, Revision 0, EC93084 on the portal for NRC staff review. The NRC staff reviewed the documents and identified the following questions:

1. *The calculation RBS-SA-11-00001, Revision 0, EC93084 discusses different approaches to estimating the PLF and provides several different sets of HCLPF and β_c estimates in Attachments 4, 7 and 8. These approaches include the separation of variables (SOV) method, the hybrid method, and a scaling method based on the SSE to GMRS ratio.*

    a) *The staff notes that the HCLPF and βc developed based on SOV in Attachment 4 are used to support the RBS RICT program. The staff understands that the median factors of safety and variabilities provided in Kennedy et al. (1980, 1984) are used in SOV estimation of the PLF in Attachment 4. To the best of staff's understanding, SOV method is used to determine the fragility of individual systems, structures, or components (SSCs) which is then used for a SMA or seismic PRA (SPRA). It is the SMA or SPRA that provides the PLF, which is a representation of the combined behavior of all the modeled SSCs. Therefore, the PLF depends on a plant-specific mix of SSCs, and based on the staff's understanding, the contribution is usually higher from components compared to structures for seismic CDF. To the best of staff's understanding, the SOV method has not been applied to directly determine the PLF and its application by the licensee is beyond the scope of its applicability. Provide justification for the first-of-a-kind* *use of SOV in estimating the PLF and plant-specific basis for selecting the parameters and their values for the SOV method in the licensee's calculation.*

b) *For the SOV method used in Attachment 4, the licensee states that the various factors and estimate values are based on the available material. However, it does not specify the "available material" used for estimates for various factors used in SOV. Provide the sources of information for the median factors used in the SOV method in Attachment 4.*

c) *In Attachment 7, the licensee also used the SOV method. Attachment 7 states that "[t]his white paper provides a basis for the HCLPF and fragility calculations performed in Reference 11." Reference 11 in the quoted statement refers to Attachment 4 discussed above. However, the staff notes that Attachment 7 uses different values for median factors than those used in Attachment 4. Justify the use of different values for median factors in the SOV method described in Attachments 4 and 7.*

d) *In the SOV method of estimating an HCLPF in Attachment 7, the staff found an error in the calculation of HCLPF value. Correction of the error results in a calculated HCLPF of 0.20g. This HCLPF of 0.20g is lower than the HCLPF of 0.30g developed in Attachment 4 and used in the RICT. Justify the use of an unconservative value (HCLPF of 0.30g PGA) in the RICT program.*

e) *Several different sets of HCLPF and βc estimates are also developed using the hybrid method in Attachments 4, 7 and 8. However, justification for the use of the minimum capacity of 0.8g ground peak spectral acceleration as the screening criteria for SSCs at RBS is not provided. Provide site-specific justification for the use of 0.8g ground peak spectral acceleration as the screening criteria.*

2. *In Attachment 8, the licensee calculated a scaling factor of 1.3 at 1 Hz based on an SSE to GMRS demand ratio. This method, which was approved by the staff for Waterford TSTF-505 LAR, can provide an alternative to SOV and hybrid methods discussed in Item #1 above, especially for determining the PLF for seismic CDF. If this method is adopted by the licensee to determine the PLF for seismic CDF for use in the RICT LAR, provide detailed steps involved in calculations and the resulting HCLPF value and selected β_c.*

3. *As an alternative to items #1 and #2 above, the licensee may choose an option for a full-scope SMA to determine the PLF. The licensee may upgrade its reduced-scope SMA performed as part of the RBS IPEEE to a full-scope or focused-scope SMA consistent with appropriate NRC-endorsed guidance and leveraging to the extent possible, with justification, prior plant-specific walkdowns, such as those performed in response to post-Fukushima actions. If this approach is adopted by the licensee, please provide resulting reports, calculation notebooks, and conclusions for NRC staff review in a regulatory audit.*

4. *Re-evaluate and provide the seismic penalty based on updated HCLPF and βc values, if they are different from those provided in the LAR.*

Entergy Response:

1.a) Calculation RBS-SA-11-00001 Attachment 4 first developed the plant-level fragility (PLF) following the Hybrid and SOV methodologies using heuristic judgements for a site located in low seismicity region. Attachment 7 then attempted to independently justify the median factors and composite variability employed in Hybrid and SOV calculations in Attachment 4. As noted, RBS is a reduced scope seismic margins assessment (SMA) plant for Individual Plant Examination for External Events (IPEEE) purposes. Per Report NE-RA-93-009-M, "Seismic IPE Review River Bend Nuclear Station Unit -1" the IPEEE seismic walkdowns followed screening guidelines in Tables 2-3 and 2-4 of Electric Power Research Institute (EPRI) NP-6041 for screening of structures and equipment, and established that RBS is seismically rugged and all components in the success path logic diagram (SPLD) are screened out with a minimum screening spectral acceleration of 0.80g. No High Confidence of Low Probability of Failure (HCLPF) calculations were performed as it was not necessary for the reduced scope IPEEE effort. No relay assessment, soil failure, or anchorage evaluations were necessary. As acknowledged in Attachment 7, a detailed SOV calculation is impossible to perform without a detailed response analysis and capacity re-assessment for plant systems. Since all components of the SPLD were seismically rugged, the reactor building was chosen as a representative surrogate for PLF. Further, with no detailed data available to perform an SOV evaluation, the range of median factors and variabilities prescribed in Kennedy et al. (References 6 and 7 of Attachment 7) for a reinforced concrete reactor building structure was used as a surrogate to verify the heuristic fragility developed in Attachment 4. Because this approach did not succeed in establishing a HCLPF of 0.30g, RBS explored another alternative to independently assess the PLF using the Hybrid Method documented in Attachment 8. In addition to IPEEE walkdowns, Attachment 8 also considered the experience based seismic capacity, design basis information for relay chatter, soil failure, and anchorage to justify the use of a 0.30g HCLPF for the seismic penalty. Attachments 7 and 8 are independent reviews of the methodology provided in Attachment 4. The final seismic penalty calculation used for the LAR submittal was based on the hybrid method described in Attachment 8.

As result of the NRC audit meeting on October 12, 2023, Calculation RBS-SA-11-00001 was replaced by a new evaluation documented in Entergy Report PSA-RBS-04-02, Rev 0 for the RBS plant level and containment seismic fragilities. The new fragility evaluation provides a plant level HCLPF of 0.16g and a containment HCLPF of 0.30g with a βc of 0.40. The revised plant level and containment seismic fragilities are based on the scaling method rather than the hybrid method.

1.b) See the response to APLC Question 1.a).

1.c) See the response to APLC Question 1.a).

1.d) See the response to APLC Question 1.a). Attachment 7 provided an independent review of the SOV methodology using the range of median factors and variabilities in Kennedy et al (References 6 and 7 of Attachment 7). This was used as a surrogate fragility. However, the values cited in Attachment 7 were not used in the seismic penalty calculation. The math error identified in Attachment 7 is being addressed through the corrective action program.

1.e) See the response to APLC Question 1.a). The seismic walkdowns for SMA documented in Section 5 of Report NE-RA-93-009-M, "Seismic IPE Review River Bend Nuclear Station Unit -1" used the screening guidelines in Tables 2-3 and 2-4 of EPRI-6041-SL for the spectral accelerations less than 0.80g. Section 6 of the same report concludes that RBS is seismically rugged and all components in the SPLD are screened out with no outliers requiring further evaluation. Section 2.0 of Attachment 8 indicates that a seismic walkdown was performed to meet the requirements of IPEEE per NUREG-1407 and EPRI NP-6041-SL. RBS used the reduced-scope SMA approach. The reduced-scope process included the development of a structure, system, and component (SSC) list per Section 3 of EPRI-6041-SL and focused on seismic walkdowns to identify weak-link items that need strengthening. The intent of this process was essentially a confirmation against design-basis with no additional seismic margins work. However, the RBS IPEEE effort was performed with sufficient rigor to demonstrate a 0.80g peak spectral capacity in accordance with EPRI NP-6041-SL. Appendix A of EPRI-6041-SL provides the basis for seismic capacity screening guidelines for SSCs. The discussion of screening guidelines in Appendix A, are heavily based on the recommendations of the expert panel on the quantification of seismic margins in NUREG/CR-4334. Per the IPEEE pre-screening assessment, the major structures and SPLD equipment at RBS were screened in the first screening lane and assigned a spectral capacity of 0.80g. The screening caveats were validated during the walkdowns following Appendix A and Appendix F of EPRI NP-6041-SL to confirm a spectral capacity of at least 0.80g. See the response to APLC Question 1.a).

2. See the response to APLC Question 1.a).

3. The purpose of Attachment 8 was to sufficiently close the gap between a reduced-scope SMA and a focused-scope SMA by reviewing the walkdown results per Section 3.2.4.1 of NUREG-1407 (Sections 2.0 and 4.2 of Attachment 8), performing relay evaluation per Section 3.2.4.2 of NUREG-1407 (Section 4.3 of Attachment 8), evaluation of soil failures per Section 3.2.4.3 (Section 4.5 of Attachment 8) and reviewing design information to determine whether a HCLPF of 0.30g could be supported. It was concluded that the 0.30g HCLPF calculation is consistent with the guidance provided in Section 3.2.4 of NUREG-1407, while utilizing the Ground Motion Response Spectrum (GMRS) as the anchor reference. As a result of the NRC audit meeting on October 12, 2023, Calculation RBS-SA-11-00001 was replaced by a new fragility evaluation using the scaling method. See the response to APLC Question 1.a).

4. The RBS seismic penalty calculation was updated to address the revised RBS plant level HCLPF of 0.16g, containment HCLPF of 0.30g, and β_c of 0.40. The seismic CDF is 3.93E-06 and the seismic LERF is 5.34E-07. The seismic CDF is ~18% of the total CDF (including Full Power Internal Events (FPIE), Fire, and Seismic), and the seismic LERF is ~39% of the total LERF (including FPIE, Fire, and Seismic). Note that the plant level fragility (0.16g HCLPF) is considered to be bounding based on the conservatism in the

approach; therefore, the actual seismic CDF contribution is expected to be significantly lower than what is calculated in the seismic penalty calculation.

Audit Question APLC-02 (TSTF-505 and 50.69) – Extreme Wind or Tornado Screening Criteria:

Section 2.3.1, Item 7, of NEI 06-09-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 NRC staffs SE for NEI 06-09 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."

The table in Enclosure 4 of the TSTF-505 LAR screens the Extreme Wind or Tornado as PS4, "mean CDF is < 1E-06 per year." However, the table did not provide the high wind CDF value to meet the criteria, except for tornado missile hazard. The licensee did provide structure design criteria for straight wind and tornado. It appears this meet the criterion C1 for "Event damage is < events for which plant is designed."

Provide additional justification to screen the extreme wind or tornado hazard using criterion "PS4".

Entergy Response:

We agree with the NRC staff that the Extreme Winds or Tornados could have been identified as meeting screening Criterion C1 in the initial screening. However, since there were some non-conformances identified, additional evaluation was performed, which showed that the risk of tornado missiles meets screening Criterion PS4. Therefore, it can be concluded that this hazard screens based on a combination of Criteria C1 and PS4.

Audit Question APLA-01 (50.69) – Credit for FLEX Equipment and Actions:

NRC memorandum dated May 6, 2022[3] provides the NRC's staff updated assessment of identified challenges and strategies for incorporating Diverse and Flexible Mitigation Capability (FLEX) equipment into a PRA model in support of risk-informed decisionmaking in accordance with the guidance of RG 1.200[4].

Section 3.2.9 of the Enclosure of the LAR states that the sensitivity that removes FLEX credit impacts CDF by approximately twelve percent. Attachment 6 of the LAR, "Disposition of Key Assumptions / Sources of Uncertainty," appears to only address one specific FLEX operator action and not the entirety of the uncertainties related to FLEX.

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[3] U.S. NRC memorandum, "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Risk Assessments," dated May 6, 2022 (ADAMS Accession No. ML22014A084).

[4] U.S. Nuclear Regulatory Commission, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," RG 1.200, Revision 3, December 2020 (ADAMS Accession No. ML20238B871).

The NRC staff notes that a twelve percent change in risk could significantly impact SSC categorization classifications.

Provide an assessment on the impact on SSC categorizations regarding the uncertainties related to FLEX.

Entergy Response:

As described in Section 3.2.9 of the Enclosure of the LAR, the sensitivity analysis removed credit for FLEX from the internal events and internal flooding PRA models. In completely removing credit for FLEX, this sensitivity analysis is very conservative, where a more realistic sensitivity analysis would have instead doubled or tripled the failure probabilities. Completely removing all credit for FLEX would be expected to have an impact on the categorization results for some systems. However, the current model realistically reflects the benefit provided by FLEX and is appropriate to credit.

The modeling of the FLEX system at RBS is not a source of model uncertainty because FLEX is being modeled in accordance with industry standards and NRC guidance. As defined in Section 1-2.2 of the ASME/ANS standard, a source of model uncertainty *is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model*. For modeling FLEX, the consensus approach comprises the industry guidance in NEI 16-06 and the NRC's assessment of that guidance with respect to the acceptability of the PRA results in risk-informed activities. In particular, an NRC memorandum (ML22014A084), dated May 6, 2022, updated the 13 conclusions from a previous NRC memorandum (ML17031A269), dated May 30, 2017, which assessed the guidance in NEI 16-06 in relation to determining the acceptability of the PRA. Based in part on the guidance in Section 3.2.11 of PWROG-20037-NP, PRA Upgrade/Maintenance and Newly Developed Methods, Entergy determined that the RBS FLEX modeling did not constitute a model upgrade that would require a peer review. Although no peer review was conducted, an independent review was conducted to assess whether the RBS FLEX modeling satisfies the expectations stated in the May 2022 NRC memorandum (ML22014A084). Model change requests (MCRs) were initiated to track recommendations from the independent assessment. MCRs are the tracking mechanism for the PRA configuration control process, as described in Section 3.2.7 of the LAR, and this process is subject to Peer Review as described in Section 3.3 of the LAR.

The May 2022 NRC memorandum (ML22014A084) approved PWROG-18042 for modeling FLEX equipment. The RBS FLEX System Notebook describes the component boundaries to be consistent with those provided in Table 4-1 of PWROG-18043-P, which is equivalent to PWROG-18042. Reliability data for FLEX diesel generators (DGs) and pumps were developed in the RBS Data Analysis Notebook from the values reported in PWROG-18042-P with a Bayesian update for the FLEX DGs based on plant-specific failure data and run time. For the FLEX motor-driven pump (FLX-P1), a conservative multiplier was applied to the motor-driven pump data from NUREG/CR-6928, since PWROG-18042 did not include the data for motor-driven centrifugal pumps. Based on one of the independent review recommendations, an MCR is being used to track an increase of that multiplier from five (5) to ten (10) based on a review of changes from Revision 0 to Revision 1 of PWROG-18042.

The May 2022 NRC memorandum (ML22014A084) approved, with certain exceptions, EPRI 3002013018 for FLEX Human Reliability Analysis (HRA). FLEX related HFEs were modeled following the methods and assumptions established in EPRI 3002013018. To address additional recommendations from the independent review, MCRs were written to explicitly document that pre-initiator HRA modeling is consistent with both EPRI KBA 2021-001, Guidance for Pre-Initiator HRA for FLEX and Portable Equipment, and 2021-007, Guidance for Modeling Refueling of FLEX and Portable Equipment. Regarding the NRC exceptions:

1) Connecting and disconnecting trailers: The RBS procedure for Initial Assessment and FLEX Equipment Staging was reviewed and has not been revised since the feasibility study in any manner that would impact the ability to connect/disconnect FLEX equipment or that would impact the HRA analyses. This is also true for the RBS Station FLEX Program Document. The RBS HRA reflects the current status of RBS mitigating strategies for ELAP, as this analysis was performed as part of PRA Revision 7, which was approved in December 2022, with updated Operator interviews conducted in June 2021. Additionally, the Entergy FLEX Program Document Bases procedure requires that PRA be consulted for planned changes to FLEX strategies. The RBS PRA does not explicitly model HFEs for connecting and disconnecting FLEX trailers, but those actions are implicitly included in the larger action for transporting FLEX equipment, which is included in the HFEs to align staged FLEX equipment. Explicitly modeling connecting/disconnecting FLEX trailers would change the execution HEP minimally. Therefore, the inclusion of specific execution errors for connecting and disconnecting FLEX trailers is not expected to impact the PRA. A cutset review suggests that much of the alignment impact is related to high cognitive dependence with other HFEs, such that separately modeling the non-dependent execution of connecting and disconnecting FLEX trailers would further minimize the impact of these steps on CDF. The addition of this documentation bases to the PRA HRA notebook is being tracked by an MCR to be completed in the next model revision.
2) Refueling of FLEX equipment: The basis for screening events related to refueling of FLEX equipment at RBS includes having refueling procedures and maintenance staff dedicated to monitoring the diesel.
3) Load shedding recovery factor: Because a preliminary assessment indicated no impact to the HEP for the execution of DC deep load shedding (DCP-XHE-FO-DEEPLS), an MCR is tracking the planned replacement of the 0.5 recovery factor for self-checking with credit for procedure-directed recoveries involving a peer checker.

In summary, the RBS PRA modeling of FLEX is appropriate for support of risk-informed decision making and is not a source of model uncertainty.

Audit Question APLA-03 (50.69) – Determination of Key Sources of Uncertainty for the 10CFR50.69 Categorization Process and Sensitivity Results:

Sections 50.69(c)(1)(i) and 50.69(c)(1)(ii) of 10 CFR 50.69 require that a licensee's PRA be of sufficient quality and level of detail to support the SSC categorization process, and that all aspects of the integrated, systematic process used to characterize SSC importance must *reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience. The guidance in NEI 00-04 specifies that sensitivity studies be conducted for each PRA model to address uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask the importance of components. The guidance in NEI 00-04 states that additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.*

Section 3.2.8 of the LAR Enclosure describes the process used for reviewing the PRA assumptions and sources of uncertainty. The NRC staff reviewed the uncertainty documents provided on this audit's electronic portal for the internal events, internal flooding, and fire PRA and found that further clarification is necessary regarding the review of assumptions and sources of uncertainty for this application. It is unclear if additional analysis was performed and documented to determine if any source of uncertainty could adversely impact any SSC categorization. In light of these observations, provide the following information:

a) *Provide details of how the RBS PRA sources of uncertainty were evaluated as a potential key source of uncertainty for this application. In this response provide any documentation of this process.*

b) *Provide the results of sensitivity studies that determined the impact on risk for each associated source of uncertainty. Include in this discussion justification that the sensitivity results demonstrate that the associated source of uncertainty does not adversely impact any SSC categorization.*

Entergy Response:

a. Sources of uncertainty were identified via a systematic evaluation of assumptions and generic sources of uncertainty documented in the base internal events (including internal flooding) and fire PRA model sources of uncertainty notebooks. The identified sources of uncertainty were evaluated to determine if they were key sources of uncertainty for the 10 CFR 50.69 application. This evaluation consisted of a screening process using the following criteria:
    - Criterion #1: Candidate uncertainties associated with topics addressed by sensitivity as part of the 50.69 process in accordance with NEI 00-04 (Reference 1 of the LAR), such as HEPs and CCFs. The 50.69 process acknowledges these candidate uncertainties as potentially impacting 50.69 and account for those impacts.
    - Criterion #2: Candidate uncertainties that are qualitatively shown to have a very small impact on total risk and would be expected to have a negligible impact on RAW and F-V (particularly uncertainties that pertain to parts of the model that would not impact components that are in the 50.69 program, such as changes to non-support system initiating event frequencies, human error probabilities not related to 50.69-eligible equipment, etc.).
    - Criterion #3: Candidate uncertainties that were identified, but for which current industry-accepted approaches and data were used, are not considered as key sources of uncertainty. This is consistent with the ASME/ANS PRA Standard

definition of a "source of modeling uncertainty" which states: "a source is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model".

- Criterion #4: Candidate uncertainties that were examined via sensitivity studies to confirm that the impact on baseline CDF and LERF and/or the change in importance measures is negligibly small are not considered as key sources of uncertainty for the 50.69 program.

An example of the screening evaluation is shown below for one of the sources of uncertainty, as provided in Table 1 from Entergy report PSA-RBS-08-10 Revision 0 shown below:

Table 1: Potential Impacts of RBS Key Assumptions and Sources of Uncertainty (Internal Events/Internal Flood PRA)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of 50.69 Impact	Suggested Approach for 50.69 Process
10	ISLOCA IE Frequency Determination	Common cause failure of the MOVs (in ISLOCA pathways) were not modeled since the internal environment for the two valves in series is significantly different. That is, one valve has constant RCS pressure while the other has essentially no pressure. WCAP-17154 also suggests that for passive failures CCF not be modeled.	This is potentially a non-conservatism in the base model. The impact of not modeling the common-cause failure of MOVs in ISLOCA pathways will need to be assessed for specific risk applications.	An overall industry-accepted approach is used for modeling ISLOCA frequency. Per Criterion #3, this is not a key source of uncertainty for 50.69. Additionally, ISLOCA is evaluated during the defense in depth portion of the 50.69 process. If an SSC can initiate an ISLOCA event, or if it can provide a significant level of mitigation of an ISLOCA event, it should be categorized as HSS. The MOVs in ISLOCA pathways would be categorized as HSS based on this defense in depth evaluation.	This item does not represent a key source of uncertainty for 50.69.

There were two sources of uncertainty that did not meet the screening criteria and required further evaluation. These two topics were 1) credit for fire protection water (FPW) injection and 2) non-proceduralized recovery actions that could be credited to recover areas after flooding. These sources of uncertainty require additional evaluation via sensitivity studies as described in the response to Question APLA-03 Part b.

b. As described in the response to Question APLA-03 Part a, additional evaluation via sensitivity studies was only required for two sources of uncertainty that did not meet the screening criteria. Sensitivities to address these two sources of uncertainty will be incorporated into the 50.69 categorization process as additional sensitivity studies that were identified in the characterization of PRA adequacy (see NEI 00-04 Table 5-2).

One of the sensitivity studies assesses the potential impact of crediting FPW as an initial source of injection to prevent core damage under Loss of Offsite Power (LOSP)/Station Blackout (SBO) conditions. The PRA model does not credit fire protection water (FPW) as an injection source unless HPCS or RCIC has been successful initially. The values for RCIC pump failure to start, failure to run in the first hour, and RCIC unavailability due to test/maintenance were reduced by 50% to simulate the increased availability of FPW. This approach was selected since FPW availability would be similar in effect to increasing RCIC (or HPCS) availability/reliability for SBO and Extended Loss of AC

Power (ELAP) sequences. This approach was found to be more reflective of the intent of the sensitivity than modifying the FPW injection operator actions.

The other sensitivity involves the conservative assumption that flood water in the penetration areas of the Auxiliary Building 70-foot elevation will prevent FLEX actions. The PRA model does not credit non-proceduralized recovery actions that could recover these areas after flooding. Gates which include all flooding initiators that are assumed to prevent the subject valve manipulations were removed to negate the conservative assumption and instead assume that the flooding events do NOT prevent the alignment of the FLEX valves.

Since these sensitivity studies will be incorporated as required sensitivities for each system categorized, any impact will be reflected in the categorization results.

Audit Question APLC-03 (50.69) – 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 §50.69(c)(1)(i), as well as the risk analyses used to address external events.

The staff has previously requested and reviewed 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 adoption of 10 CFR 50.69. The supplement to the 10 CFR 50.69 by Calvert Cliffs Nuclear Power Plant LAR dated May 10, 2019 (ADAMS Accession No. ML19130A180), contained additional information related to the alternate seismic approach including incorporation by reference docketed information related to case study Plants A, C, and D; the supplement dated July 1, 2019 (ADAMS Accession No. ML19183A012), further clarified the information related to the alternate seismic approach (see response to RAI 4); the supplement dated July 19, 2019 (ADAMS Accession No. ML19200A216), provided responses to support the technical acceptability of the PRAs used for the Plant A, C, and D case studies as well as technical adequacy of certain details of the conduct of the case studies; the supplement dated August 15, 2019 (ADAMS Accession No. ML19217A143) clarified a response in the July 19, 2019 supplement. The supplement dated July 19, 2019, included modifications to the content of the EPRI report. In addition, the licensee removed several paragraphs related to its previous seismic submittals, categorization team evaluations, and IDP's decision process from a typical Section 3.2.3.

Since the above-mentioned information was requested and reviewed by the staff for Calvert Cliffs Nuclear Power Plant's LAR for adoption of 10 CFR 50.69, the staff is unable to use it for the licensee's docket unless it is incorporated in the licensee's LAR. The above-mentioned information is necessary for the staff to make its regulatory finding on the licensee's proposed alternate seismic approach. The information is neither included in the LAR nor is it available in the EPRI report supporting the licensee's proposed approach.

a) *Provide the above-mentioned information to support the staff's regulatory finding on the alternate seismic approach by either incorporating the information by reference the identified supplements or responding to the RAIs in the identified supplements.*

b) *If differences exist between the licensee's proposed alternate seismic approach and the information in the supplements stated above, identify such differences and either incorporate them in the licensee's proposed approach or justify their exclusion.*

c) *The licensee is required to re-evaluate seismic risk to be low compared to total plant risk, due to changes of HCLPF and β_c values in APLC Question 01.*

Entergy Response:

a) Additional references to be added to the LAR regarding the alternate seismic approach are listed below. The plant specific test case information from the RBS LAR Reference 4 that Entergy is using from other licensees and being incorporated by reference into this application is described in Case Study A (References 5, 6, and 7), Case Study C (References 8 and 9), and Case Study D (References 10, 11, 12, 13, and 14).

1. Exelon letter to NRC, Calvert Cliffs Nuclear Power Plant, Units 1 and 2, "Revised submittal to Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors," dated May 10, 2019 (ADAMS Accession No. ML19130A180).

2. Exelon letter to NRC, 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,'" dated July 1, 2019 (ADAMS Accession No. ML19183A012).

3. Exelon letter to NRC, 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,'" dated July 19, 2019 (ADAMS Accession No. ML19200A216).

4. Exelon letter to NRC, 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," dated August 5, 2019 (ADAMS Accession No. ML19217A143).

5. Peach Bottom Atomic Power Station Seismic Probabilistic Risk Assessment Report, "Response to NRC Request Regarding Recommendation 2.1 of the Near Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," August 28, 2018 (RS-18-098) (ADAMS Accession No. ML18240A065).

6. Peach Bottom Atomic Power Station, Units 2 and 3 – Staff Review of Seismic Probabilistic Risk Assessment, "Associated with Reevaluated Seismic Hazard Implementation of the Near-Term Task Force Recommendation 2.1: Seismic," (EPID NO. L-2018-JLD-0010), June 10, 2019 (ADAMS Accession No. ML19053A469).

7. Peach Bottom Atomic Power Station, Units 2 and 3 – Correction Regarding Staff Review of Seismic Probabilistic Risk Assessment, "Reevaluated Seismic Hazard Implementation of the Near-Term Task Force Recommendation 2.1: Seismic," (EPID NO. L-2018-JLD-0010), October 8, 2019 (ADAMS Accession No. ML19248C756).

8. Plant C Nuclear Plant, Units 1 and 2, License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process, June 22, 2017 (ADAMS Accession No. ML17173A875).

9. Plant C Nuclear Plant, Units 1 and 2, "Issuance of Amendments Regarding Application of Seismic Probabilistic Risk Assessment Into the Previously Approved 10 CFR 50.69 Categorization Process (EPID L-2017-LLA-0248)," August 10, 2018 (ADAMS Accession No. ML18180A062).

10. Seismic Probabilistic Risk Assessment for Plant D Nuclear Plant, Units 1 and 2, "Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the NTTF Review of Insights from the Fukushima Dai-ichi Accident," June 30, 2017 (ADAMS Accession No. ML1718A485).

11. Plant D Nuclear Plant, Units 1 and 2, Seismic Probabilistic Risk Assessment Supplemental Information, April 10, 2018 (ADAMS Accession No. ML18100A966).

12. Plant D Nuclear Plant, Units 1 and 2 – Staff Review of Seismic Probabilistic Risk Assessment Associated With Reevaluated Seismic Hazard Implementation, of the NTTF Recommendation 2.1: Seismic (CAC NOS. MF9879 AND MF9880; EPID L-2017-JLD-0044) July 10, 2018 (ADAMS Accession No. ML18115A138).

13. Plant D Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," November 29, 2018 (ADAMS Accession No. ML18334A363).

14. Plant D Nuclear Plant, Units 1 And 2 - Issuance of Amendment Nos. 134 And 38 Regarding, Adoption of 10 CFR 50.69, "Risk-Informed Categorization and Treatment Of Structures, Systems, and Components For Nuclear Power Plants" (EPID L-2018-LLA-0493) April 30, 2020 (ADAMS Accession No. ML20076A194).

b) There are no differences identified.

c) As described in the response to TSTF-505 APLC-01, the plant HCLPF and βc values were re-assessed and a re-evaluation of the seismic risk was performed. The RBS seismic penalty calculation was updated to address the revised RBS plant HCLPF of 0.16g, containment HCLPF of 0.30g, and βc of 0.40. The updated seismic CDF is 3.93E-06 and the updated seismic LERF is 5.34E-07. The seismic CDF is ~18% of the total CDF (including FPIE, Fire, and Seismic), and the seismic LERF is ~39% of the total LERF (including FPIE, Fire, and Seismic). Note that the plant level fragility (0.16g HCLPF) is considered to be bounding based on the conservatism in the approach; therefore, the actual seismic CDF contribution is expected to be significantly lower than what is calculated in the seismic penalty calculation.

Audit Question APLC-04 (50.69) – External Flooding:

Paragraph 50.69(b)(2)(ii) of 10 CFR requires that the quality and level of detail of the systematic processes that evaluate the plant for external events during operation is adequate for the categorization of SSCs.

In Enclosure 4 of the LAR, Table E4-1 provides the External Hazard Screening. In the external flooding section, it states "External flooding events will cause no flooding damage to RBS safety-related structures, systems and components." The licensee didn't provide a list of SSCs, such as exterior doors, that are credited for this screening and must categorize as HSS based on NEI 00-04.

Provide a list of the specific exterior doors that will be assigned HSS since they are credited for screening the external flood hazard (in accordance with Figure 5-6 in NEI 00-04).

Entergy Response:

RBS does not currently have plans to categorize doors. If categorized, the following list of doors will be assigned High Safety Significance (HSS) since they are credited for screening the external flood hazard (in accordance with Figure 5-6 in NEI 00-04):

DG-098-H01	JRB-D01HTCH	AB-098-05
DG-098-H02	AB-098-03	FB-098-04
DG-098-01	AB-098-04	AB-098-06
DG-098-02	CB-098-17	CB-098-01
DG-098-03	FB-095-01	DG-098-11
DG-098-H03	SP-098-01	

Attachment 3

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Markup of Entries to TSTF-505 License Amendment Request Table E1-1, "In-scope TS/LCO Conditions to Corresponding PRA Functions"

(4 pages follow)

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

RBS TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.4	**Electrical Power Systems: DC Sources - Operating**						
3.8.4.A	One required battery charger on Division I or II inoperable	Two primary battery chargers.	Yes	Provide DC power to Division I or II loads during normal operation and abnormal operation	With one required Division I OR Division II battery charger inoperable ~~and associated DC bus is aligned to backup charger~~, the design success criterion is met by ~~any two of three Division I, II, or III 125 VDC power divisions~~ the operable Division I OR Division II battery charger.	With one required battery charger inoperable and associated DC bus is aligned to backup charger, the PRA success criterion is met by any one of two Division I or II 125 VDC power divisions. Division III DC electrical power only required if on AC crosstie or if HPCS is success path.	SSCs are modeled consistent with the TS scope and so can be directly included in the ERAT for the RICT program.

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

RBS TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.4.B	Division I or II DC electrical power subsystem inoperable for reasons other than Condition A	The Division I and II 125 VDC busses, breakers, instrumentation, and supports	Yes	Provide DC power to Division I or II loads during normal and abnormal operation	With one Division I OR Division II DC electrical power subsystem inoperable, the design success criterion is met by the ~~remaining divisions of electrical power (Division I or II, AND Division III).~~ operable Division I OR Division II DC electrical power subsystem.	With Division I or II DC electrical power subsystem inoperable, the PRA success criterion is met by one of two Division I or II DC electrical power subsystems. Division III DC electrical power only required if on AC crosstie or if HPCS is success path.	SSCs are modeled consistent with the TS scope and so can be directly included in the ERAT for the RICT program.
3.8.7	**Electrical Power Systems: Inverters - Operating**						
3.8.7.A	Division I or Division II inverter inoperable	Division I and Division II inverters (Two operating, two spare)	Yes	Provide AC instrument power to vital buses	With one division inoperable due to the inverter, the design success criterion is met by the operable division with an operable inverter. ~~associated spare inverter for that division. One inverter in each division meets the design success criteria.~~	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the ERAT for the RICT program.

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

RBS TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.9	**Electrical Power Systems: Distribution Systems – Operating**						
3.8.9.A	One or more Division I or II AC electrical power distribution subsystems inoperable	Division I and II AC electrical distribution subsystems consisting of 4.16kV buses, 480V load centers and distribution panels	Yes	Provide AC power to the associated ESF loads	With one or more Division I OR Division II AC electrical power distribution subsystems inoperable, the design success criterion is met by the ~~remaining divisions of AC electrical power distribution subsystems (Division I or II, and Division III)~~ operable Division I OR Division II AC electrical power distribution subsystems.	With Division I or II AC electrical power distribution subsystems inoperable, the design success criterion is met by the operable Division I or II electrical power distribution subsystem.	SSCs are modeled consistent with the TS scope and so can be directly included in the ERAT for the RICT program. (For unmodeled distribution panels, see Note 13)
3.8.9.B	One or more Division I or II AC vital bus distribution subsystems inoperable	Division I and II 120V AC vital bus distribution subsystems	Yes	Provide Vital AC power to the required divisional loads	With one division of Vital AC inoperable, the remaining division AC vital bus electrical power distribution subsystem is capable of supporting minimum safety functions.	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the ERAT for the RICT program.

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

RBS TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.9.C	One or more Division I or II DC electrical power distribution subsystems inoperable	Division I and II 125V DC bus distribution subsystems	Yes	Provide DC power to the associated ESF loads	With one or more Division I OR Division II DC electrical power distribution subsystems inoperable, the design success criterion is met by the ~~remaining divisions of DC electrical power distribution subsystems (Division I or II, AND Division III)~~ operable Division I OR Division II DC distribution subsystems.	With Division I or II DC electrical power distribution subsystems inoperable, the PRA success criterion is met by the operable Division I or II DC electrical power distribution subsystems. Division III DC electrical power only required if on AC crosstie or if HPCS is success path.	SSCs are modeled consistent with the TS scope and so can be directly included in the ERAT for the RICT program. (For unmodeled distribution panels, see Note 13)

Attachment 4

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Technical Specification Page Revised Markups

(1 page follows)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	-------------------NOTE------------------ Verification is only required if 22 kV onsite circuit is supplying Division III safety related bus E22-S004 from normal power transformer STX-XNS1C. --	
	A.2 Verify E22-S004 is aligned to transfer to the preferred station transformer powered by the OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND	
	A.3 Restore required offsite circuit to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program AND 24 hours from discovery of two divisions with no offsite power or in accordance with the Risk Informed Completion Time Program
B. Automatic transfer function not OPERABLE	B.1 Restore Division III power source to the preferred station service transformers	12 hours

(continued)