Southern Nuclear

Cheryl A. Gayheart
Regulatory Affairs Director

3535 Colonnade Parkway
Birmingham, AL 35243
205 992 5316 tel
205 992 7601 fax
cagayhea@southernco.com

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

Joseph M. Farley Nuclear Plant – Units 1 and 2
Response to Request for Additional Information Regarding Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"

Ladies and Gentlemen:

By letter dated June 18, 2020, Southern Nuclear Operating Company (SNC) submitted an application to modify the Joseph M. Farley Nuclear Plant (Farley) 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 email dated January 12, 2021, the Nuclear Regulatory Commission (NRC) staff issued a request for additional information (RAI). Enclosure 1 to this letter provides the SNC response to the NRC staff's RAI. Enclosure 2 provides referenced Electric Power Research Institute (EPRI) Report 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization".

The conclusions of the No Significant Hazards Consideration and Environmental Consideration contained in the original application have been reviewed and are unaffected by this response.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 2nd day of March 2021.

Respectfully submitted,

Cheryl Gayheart
Director, Regulatory Affairs
Southern Nuclear Operating Company

CAG/RMJ

Enclosures: 1. SNC Response to NRC RAI
2. EPRI Report 3002017583

cc: Regional Administrator, Region II
NRR Project Manager – Farley
Senior Resident Inspector – Farley
Director, Alabama Office of Radiation Control
RTYPE: CFA04.054

Joseph M. Farley Nuclear Plant – Units 1 and 2
Response to Request for Additional Information Regarding Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"

Enclosure 1

SNC Response to NRC RAI

NRC RAI 1:

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

In the LAR, the licensee proposes to address seismic hazard risk using the alternative seismic approach for seismic Tier-1 plants described in Electric Power Research Institute (EPRI) Report 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization" (https://www.epri.com/research/products/000000003002017583) and other qualitative considerations. The NRC staff understands that EPRI 3002017583 is an updated version of EPRI 3002012988 that was reviewed in conjunction with its review of the Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2, LAR for adoption of 10 CFR 50.69 (precedent) dated November 28, 2018 (ADAMS Accession No. ML 18333A022). The staff has not reviewed or endorsed EPRI 3002012988 as a topical report for generic use. As such, each licensee needs to perform a plant-specific review for applicability of the Tier-1 alternative seismic approach. The NRC staff reviewed and approved CCNPP's alternative seismic approach, which was based on information for Tier-1 plants included in EPRI 3002012988 and information provided in the supplements to the CCNPP LAR. Information in the supplements to the CCNPP LAR (ADAMS Accession Nos. ML19130A180, ML19200A216, ML19217A143, and ML19183A012) that was used to support the staff's review and approval of that approach is included in the staff's safety evaluation for the CCNPP LAR (ADAMS Accession No. ML19330D909).

The NRC staff notes that the licensee's proposed alternative seismic approach is similar to that reviewed and approved in the CCNPP safety evaluation. However, the licensee's proposed approach is based on information for Tier-1 plants as described in EPRI 3002017583 instead of EPRI 3002012988.

Further, the staff notes that EPRI 3002017583 does not contain all the information in the supplements to the CCNPP LAR that supported the use of EPRI's alternative seismic approach for Tier-1 plants in the CCNPP plant-specific safety evaluation. Therefore, the licensee is requested to address the following:

(a) The licensee cited EPRI report 3002017583 as applicable to their submittal, please submit EPRI report 3002017583 on the docket.

(b) Identify and describe the differences between EPRI 3002017583 and EPRI 3002012988.

(c) Explain whether EPRI 3002017583 includes all the information from the CCNPP LAR supplements that was used to support the staff's review of the alternative seismic approach for Tier-1 plants described in EPRI 3002012988. If any information from the CCNPP LAR supplements are not included in EPRI 3002017583, justify such exclusion for the licensee's proposed alternative seismic approach or indicate where it is addressed in the licensee's application for Farley.

(d) Based on the responses to items (b) and (c), justify why a separate staff review of EPRI 3002017583 for the licensee's proposed alternative seismic approach is not warranted.

(e) Identify and justify any differences between the licensee's proposed alternative seismic approach and the NRC staff approval of the precedent documented in the CCNPP safety evaluation, including any Farley-specific considerations.

SNC Response to NRC RAI 1:

(a) The requested EPRI document is provided as Enclosure 2.

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

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

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

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

1. Figure 1-2 was edited to include EPRI 3002017583 in the list of §50.69 supplemental guidance documents.
2. Figure 2-2, Low Seismic Hazard Site: Typical SSE to GMRS Comparison – replaced graph with correct graph.

(c) EPRI 3002017583 has incorporated all the information and follow up actions from the CCNPP LAR supplements that was agreed upon by the NRC staff's review of the alternative seismic approach for Tier -1 plants. Therefore, Attachment 2 of ML19200A216 are applicable to Farley since it is using the updated EPRI Document 3002017583, Alternative Approaches for Addressing Seismic Risk in 10CFR 50.69 Risk-Informed Categorization. Farley's applicability to other CCNPP LAR supplements and attachments are addressed below:

Table RAI-01-1. Farley Applicability to CCNPP LAR supplements.

Item	Applicable to Farley	Incorporated into EPRI 3002017583	Basis
ML19130A180 CCNPP Supplement 05/10/2019			
Attachment 1	x		The revisions to the CCNPP LAR, expect areas that were specific to CCNPP, were included throughout Section 3 of the Farley LAR.
ML19183A012 CCNPP Supplement 07/01/2019			
RAI 4 a.	x		The two paragraphs cited are in Section 3.2.3 of the Farley LAR. The clarification discussed in the CCNPP response would apply to Farley.
RAI 4 b.	x		Farley has a peer reviewed Seismic PRA that meets the guidance in RG 1.200. All F&Os were closed through the Appendix X process.
RAI 5	x		SSCs credited for screening of external hazards will be evaluated according to the flow chart in NEI 00-04, Figure 5-6. See SNC response to RAI 1(e).
RAI 6	x		In accordance with NEI 00-04 and existing SNC fleet procedures, Interfacing functions/SSC will not be categorized and will not be subject to alternative treatment until categorization of all the systems that are supported is complete unless the function/SSC is initially categorized as HSS
RAI 8 a.	x		For plant Farley, FLEX methodology, equipment, and associated operator actions are discussed in RAI 2.
RAI 8 b.	x		For plant Farley, FLEX methodology, equipment, and associated operator actions are discussed in RAI 2.
ML19200A216 CCNPP Supplement 07/19/2019			
RAI 1 a.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 1 b.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583

Table RAI-01-1. Farley Applicability to CCNPP LAR supplements.

Item	Applicable to Farley	Incorporated into EPRI 3002017583	Basis
RAI 1 c.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 1 d.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 1 e.			Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583.
RAI 1 f.	x		Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583.
RAI 2 a. i.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 2 a. ii.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 2 b. i.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 2 b. ii.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 3 a. i.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583

Table RAI-01-1. Farley Applicability to CCNPP LAR supplements.

Item	Applicable to Farley	Incorporated into EPRI 3002017583	Basis
RAI 3 a. ii.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 3 b.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 3 c. i.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 3 c. ii.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 3 c. iii.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 3 c. iv.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 3 d. i.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583
RAI 3 d. ii.	x	x	Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583

Table RAI-01-1. Farley Applicability to CCNPP LAR supplements.

Item	Applicable to Farley	Incorporated into EPRI 3002017583	Basis
RAI 7 a.	x		The PRA key assumptions and sources of uncertainty are determined consistent with the definitions in RG 1.200. The Disposition of Key Assumptions/ Sources of Uncertainty are discussed in Attachment 6 of the June 18, 2020 Farley application.
RAI 7 b.			Not applicable to Farley. No new key assumptions and sources of uncertainty have been identified for the application.
RAI 7 c.	x		The Disposition of Key Assumptions/ Sources of Uncertainty are discussed in Attachment 6 of the June 18, 2020 Farley application.
ML19217A143 CCNPP Supplement 08/05/2019			
RAI 3 c. iv. Revised	x	x	This supplement revised part of the response to RAI 3 c. iv. Farley will be using EPRI 3002017583 and this response addresses the clarification requested by the NRC for the case studies within EPRI 3002017583.

(d) A separate staff review of EPRI 3002017583 for the Farley proposed alternative seismic approach is not warranted based on the following:

i. Other than the incorporation of updates submitted to the NRC in an RAI submittal for the CCNPP 50.69 LAR, the technical criteria from EPRI 3002012988 is unchanged, as described in response 1b;

ii. SNC confirmation that EPRI 3002017583 has incorporated the agreed upon information and follow up actions from the CCNPP LAR supplements from NRC staff's review of the alternative seismic approach for Tier 1 as described in response 1c; and

iii. Review of CCNPP LAR supplements for their applicability to Farley in response 1c.

(e) In review of the CCNPP SE, two differences were identified from the proposed alternative seismic approach documented in the Farley LAR. As discussed below, both of these differences will be incorporated into the categorization process.

1. In the section "Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach" of the CCNPP SE, the configuration control program for CCNPP had been updated to include a checklist of configuration activities to recognize those systems that have been categorized in accordance with 10 CFR 50.69, to ensure that any physical change to the plant or change to plant documents is evaluated prior to implementing those changes. This checklist is the same as what is included in Section 3.5 of the Farley LAR except for "Review of impact to seismic loading and SSE seismic requirements, as well as the method of combining seismic components." This checklist item will also be included in the SNC configuration control program.

2. Section 3.5.3.2 of the CCNPP SE discusses categorization assessment of other external hazard, "… NEI 00-04 requires that, as part of the external hazard screening, an evaluation be conducted to determine if there are components that participate in screened scenarios and whose failure would result in an unscreened scenario and that such SSCs are required to be high safety-significant in the categorization process." The evaluation is not mentioned in the Farley LAR. This evaluation will also be included in the SNC categorization assessment of other external hazard risk. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered high safety-significant (HSS).

NRC RAI 2:

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

Section 3.3 of the LAR mentions the PRA modeling of FLEX equipment and FLEX operator actions. More information is needed for the NRC staff to determine the acceptability of incorporation of FLEX equipment into the PRA models. Please provide the following information for the internal events and internal flooding PRAs, as appropriate:

(a) A discussion detailing the extent of incorporation, i.e., summarizing the supplemental equipment and compensatory actions, including FLEX strategies, that have been credited quantitatively for each of the PRA models used to support this application.

(b) A discussion detailing the methodology used to assess the failure probabilities of any modeled equipment credited in the licensee's mitigating strategies (i.e., FLEX). The discussion should include a justification of the rationale for parameter values, and how the uncertainties associated with the parameter values are considered in the categorization process in accordance with ASME/ANS RA-Sa–2009, as endorsed by RG 1.200 (e.g., supporting requirements for HLR-DA-D).

(c) A discussion detailing the methodology used to assess operator actions related to FLEX equipment and the licensee personnel that perform these actions. The discussion should include:

   i. A summary of how the licensee evaluated the impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)–(j) of supporting requirement HR-G3 of ASME/ANS RA-Sa–2009, as endorsed by RG 1.200.

   ii. Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and whether the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of ASME/ANS RA-Sa–2009, as endorsed by RG 1.200.

   iii. If the licensee's procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, a discussion detailing the technical basis for probability of failure to initiate mitigating strategies.

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

i. Provide an evaluation of the model changes associated with incorporating mitigating strategies, which demonstrates that none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences,

OR

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

(e) OR, as an alternative to Parts (a), (b), (c), and (d), above: Remove credit for FLEX equipment in the PRA used to support this LAR, and provide updated risk results (i.e., LAR Attachment 2) that does not credit FLEX equipment and actions.

SNC Response to NRC RAI 2

(a) As part of the plant modifications associated with FLEX implementation, certain portable FLEX equipment is stored in the FLEX storage dome and are installed in the plant. The plant procedures have also been revised to include these plant modifications. The FLEX operator actions, and equipment are credited in the Farley Internal Events, Internal Flooding, and Fire PRA models, which are used in 10 CFR 50.69 categorization. A summary of credited FLEX strategies, supplemental equipment, and compensatory actions associated with extended loss of offsite power conditions are shown in Table RAI-02-1.

Table RAI-02-1. Credited FLEX Strategy, Equipment, and Actions in PRA

Credited Action	Flex Strategy – Modeled	Supplemental Equipment	Compensatory Actions
Stage portable battery powered lighting for use in MCR after load shed	x	x	x
Minimize battery load to extend operations of the DC and Vital AC busses	x		x
Open doors for ventilation of the battery and DC equipment rooms	x		x
Manually control the TDAFWP			x
Opening of the MCR access doors for sufficient ventilation	x		x
Stage and connect 600V Flex DG to DC bus battery chargers	x	x	x
Deploy portable fans for switchgear rooms for proper ventilation	x	x	x
Stage and connect SG Flex pump in the event the TDAFWP fails	x	x	x
Transfer makeup water from RMWST to the CST using the SG Flex pump before CST inventory is exhausted.		x	x
Install portable fans in MCR to maintain an acceptable temperature	x	x	x

(b) Neither plant-specific data nor generic industry parameter estimates are available for the portable FLEX diesel generators, pumps, and fans; therefore, use of the Farley Bayesian updated data for diesel generators and diesel HVAC fans are used for FLEX diesel generators and fans. Use of generic industry parameter estimates from NUREG/CR-6928 are used for the FLEX pumps.

The portable FLEX diesel generators, pumps, and fans are not like other installed plant equipment, thus an additional factor of two is applied to the unreliability failure probabilities. This escalation is a reasonable approximation of the unreliability of portable FLEX equipment according to the guidance in NEI 12-06 until industry data is published.

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

A sensitivity analysis was performed to assess the impact FLEX equipment probability has on CDF and LERF. The FLEX equipment failure probabilities were increased by a factor of five. A factor of 3 means that the resulting sensitivity is larger than the base case 95th percentile. Table RAI-02-2 summarizes the results from the sensitivity. The sensitivity shows there is a small change to CDF and LERF, and the dominant failures of FLEX strategies in the PRA model are HRA related. Since there is no significant impact on CDF or LERF from the parametric uncertainty, there is no significant impact on the 50.69 program risk assessment.

Table RAI-02-2. FLEX Equipment Failure Rate Sensitivity

	Unit 1 Delta	Unit 2 Delta
Internal Events		
CDF	7.0E-08	2.3E-08
LERF	1.4E-10	4.5E-11
Internal Flood		
CDF	0.0E+00	0.0E+00
LERF	0.0E+00	0.0E+00
Fire		
CDF	1.5E-06	1.5E-08
LERF	3.3E-09	0.0E+00
Total Delta CDF	1.5E-06	3.8E-08
Total Delta LERF	3.4E-09	4.5E-11

(c)

i. The impacts of the performance shaping factors on the HEPs for the operator actions associated with the FLEX modeling were evaluated in HRA Post-Initiators & Dependency Analysis and are listed in Table RAI-02-3. The aggregate HEP for the FLEX/ELAP actions is approximately 1.0E-1, which is consistent with the NEI 16-06 (Reference F.13) screening probabilities for FLEX strategies and considered to represent a reasonable approximation of the overall failure probability of a complex mitigation strategy employing portable equipment in potentially challenging conditions.

Table RAI-02-3. FLEX Credited Actions

Credited Action	PSF Impact Basis	PSF Impact
Stage portable battery powered lighting for use in MCR after load shed	• For a case in which the determination that AC power will not be recovered within 4 hours of plant trip has been made by the procedurally directed time frame from plant trip, the operators would potentially be finished with MCR light de-energization with plenty of time to spare from plant trip. This limited amount of time required to finish the MCR light de-energization would allow them to completely re-perform the MCR light de-energization action, if necessary (based on the slowest recorded validation time). With one operator per unit, this is not a high workload task. • The operators would be working in SBO conditions in which emergency and/or portable lighting would be required.	Negative
Minimize battery load to extend operations of the DC and Vital AC busses	• For a case in which the determination that AC power will not be recovered within 4 hours of plant trip has been made by the procedurally directed time frame from plant trip, the operators would potentially be finished with load within minimal time from plant trip. This would allow them to completely re-perform the load shed action, if necessary (based on the slowest recorded validation time). With one SO per unit, this is not a high workload task. • The SOs would be working in SBO conditions in which emergency and/or portable lighting would be required.	Negative

Table RAI-02-3. FLEX Credited Actions

Credited Action	PSF Impact Basis	PSF Impact
Manually control the TDAFWP	• For this case, the plant response is not as expected in that there is an extended SBO, but the FLEX strategies are available to mitigate these conditions and the crews have trained to perform them. While the ELAP scenario is undesirable, load shed actions would have been completed successfully or be in progress, SG level would initially be maintained by TDAFW, and adequate time would be available for the SOs to perform the action. There is no indication that the plant is heading toward core damage. • The crew would have adequate time to begin the process of establishing local control of the TDAFW pump, though there are actions to which the responsible SOs would be assigned prior to taking local control of the TDAFW pump. However, those tasks are expected to be complete with adequate time to spare. This does not correlate to conditions in which the crew would be at the limit of what they could achieve in the time that is available. "Low" workload is assigned. • Due to loss of power, Auxiliary Building elevators and HVAC will be unavailable. Flashlights and headlamps will be available for personnel. Due to heat in the area around the TDAFW pump, the operator may move to a cooler area and make periodic checks on the pump. Continuous monitoring is not required. "Negative" PSFs are considered to always be applicable.	Negative
Stage and connect 600V Flex DG to DC bus battery chargers Opening of the MCR access doors for sufficient ventilation	• For this case, the plant response is not as expected in that there is an extended SBO, but the FLEX strategies are available to mitigate these conditions and the crews have trained to perform them. Once ELAP is declared, this is the expected response. While the ELAP scenario is undesirable, the load shed action would have been completed successfully. The TDAFW failure is an additional complication, but the time available for the alignment of the 600V FLEX D/G is still the same and the response plan for the generator crew would remain the same. If a serious problem occurred with the pump deployment task that required the crew to help with that task, it would cause a disruption, but this is not assumed to happen. • There is an abundant amount of recovery time available for this action even with debris removal required. The workload is not high. • The lineup includes outdoor activity and for SBO cases, adverse weather is likely (adverse conditions assumed to exist).	Moderate

Table RAI-02-3. FLEX Credited Actions

Credited Action	PSF Impact Basis	PSF Impact
Deploy portable fans for switchgear rooms for proper ventilation Stage and connect 600V Flex DG to DC bus battery chargers	• For this case, the plant response is not as expected in that there is an extended SBO, but the FLEX strategies are available to mitigate these conditions and the crews have trained to perform them. Once ELAP conditions are identified, this is the expected response. While the ELAP scenario is undesirable, the load shed action would have been completed successfully, SG level would initially be being maintained by TDAFW, and adequate time (many hours) would be available for the alignment of the portable fans for DC Switchgear and Battery Charger Room cooling. However, for the PRA scenarios, TDAFW pump fails as early as 4 hours from plant trip, which may be before or during the time when the crew is staging and connecting the portable fans. The TDAFW failure is an additional complication, but the time available for the alignment of the fans is still the same and the response plan for the crew would remain the same. If a serious problem occurred with the pump deployment task that required the crew to help with that task, it would cause a disruption, but this is not assumed to happen. • There are several hours available to perform the initial work completing the initiation of the fans after 600V FLEX DG start was described as not challenging by plant staff. • Other than obtaining the ductwork from the FLEX trailer and transport vehicle, the work is performed indoors. If, for some reason, the work is completed near the end of the allow time window, temperatures would be around 120 degrees, which correlates to a "negative" condition. In addition, the work would be performed in emergency lighting conditions, which are not optimal.	Moderate

Table RAI-02-3. FLEX Credited Actions

Credited Action	PSF Impact Basis	PSF Impact
Stage and connect SG Flex pump in the event the TDAFWP fails Deploy portable fans for switchgear rooms for proper ventilation	• For this case, the plant response is not as expected in that there is an extended SBO, but the FLEX strategies are available to mitigate these conditions and the crews have trained to perform them. Once the FLEX strategies are initiated, this is the expected response. While the ELAP scenario is undesirable, the load shed action would have been completed successfully, SG level would initially be being maintained by TDAFW, and adequate time (many hours) would be available for the alignment of the SG FLEX Pump to help maintain the plant in a stable condition. However, for the PRA scenarios, TDAFW pump fails as early as 4 hours from plant trip, which may be before or during the time when the crew is staging and connecting the SG FLEX Pump. Loss of the only available SG makeup pump would put pressure on the crew members because unless they succeed, core damage would occur unless AC power is recovered. There is substantial time available to complete the SG FLEX Pump alignment, but the crew members would not have a clear indication of how much time is available to them for the alignment (though it is known for the PRA evaluation). This is considered to be a condition that is not an expected plant response.	High
Initiate SG Flex Pump makeup after TDAFW Pump failure stage and connect SG Flex pump in the event the TDAFWP fails	• For this case, the plant response is not as expected in that there is an extended SBO, but the FLEX strategies are available to mitigate these conditions and the crews have trained to perform them. While the ELAP scenario is undesirable, load shed actions would have been completed successfully, SG level would have been maintained by TDAFW for 4 hours, and adequate time (several hours) would be available for the initiation of the SG FLEX Pump to help maintain the plant in a stable condition after completion of the SG FLEX pump staging/connection action. For these reasons, this is not considered to be a high stress action, particularly because of the long period of time over which the action occurs and the significant time margin that is available to the crew. • The crew would have several hours to initiate SG FLEX pump makeup, which does not correlate to conditions in which the crew would be at the limit of what they could achieve in the time that is available. "Low" workload is assigned. • Work in ELAP conditions may require the use of portable lighting, working in difficult positions in the main steam valve room, and would likely be performed and in poor weather conditions related to the loss of offsite power. "Negative" PSFs are considered to always be applicable.	Moderate

ii. Farley's assessment for pre-initiator human failure maintenance events is not possible to cause an event since FLEX equipment is not aligned for operation but stored in FLEX dome during normal operation. Miscalibration of FLEX equipment or errors during maintenance of FLEX equipment do not need to be separately modeled since any miscalibration which fail FLEX equipment when demanded is already accounted for FLEX equipment failure probabilities. These probabilities are estimated based on number of demands and run hours and number of failures. Miscalibration of equipment often causes auto operation signals for normal components. However, FLEX equipment is manually started and aligned only when they are deployed, therefore miscalibration error is not applicable to FLEX equipment.

iii. The modeling used to represent the failure to initiate mitigating strategies was accomplished by using a screening HEP consisting of the operators failing to identify ELAP conditions and enter FLEX strategies. This modeling and quantification is based on the IDHEAS Delay Implementation event tree from EPRI Report 3002013018, "Human Reliability Analysis (HRA) for Diverse and Flexible Mitigation Strategies (FLEX) and Use of Portable Equipment".

(d)

i. See response of 2(d)ii.

ii. A focus-scope peer review was performed in December 2019 on FLEX modeling for Plant Farley Internal Events, Internal Flood, and Fire. The scope consisted of a review of 116 relevant SRs contained in Sections 2, 3, and 4 of the ASME/ANS PRA Standard. There were no finding level F&Os to be resolved. All SRs were met at Capability Category II or higher.

(e) Not Applicable

NRC RAI 3:

RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," provides risk acceptance criteria. in terms of the change in risk in combination with either total core damage frequency or large early release frequency.

RG 1.174 and Section 6.4 of NUREG 1855, Revision 1, for a Capability Category II risk evaluation, indicate that the mean values of the risk metrics (total and incremental values) need to be compared with risk acceptance guidelines. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on the PRA input parameters and model uncertainties explicitly reelected in the PRA models. In general, the point estimate CDF and LERF obtained by quantification of the cutset probabilities using mean values for each basic event probability does not produce a true mean of the CDF/LERF. Under certain circumstances, a formal propagation of uncertainty may not be required if it can be demonstrated that the state-of-knowledge correlation (SOKC) is unimportant (i.e., the risk results are well below the acceptance guidelines).

Section 8 of NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline,” requires a cumulative sensitivity study to evaluate the potential impact on CDF and LERF based on a postulated reduction in reliability due to the special treatment of selected SSCs. The guidance states that the results of this study should be compared to the risk acceptance guidelines of RG 1.174 as a measure of acceptability.

LAR Attachment 2 presents estimates of the total CDF and LERF based on the internal events (including flooding) and fire risk. NRC staff notes that for FNP, the total CDF of 8.4E-05 per year begins to approach the RG 1.174, Revision 3, threshold of 1E-04 per year for total CDF without considering the risk increase due to SOKC.

Please address the following:

(a) Demonstrate that FNP's total CDF and LERF mean values meet the RG 1.174 risk acceptance guidelines.

(b) As an alternative to Part (a), provide justification that the FNP risk values represent an acceptable level of risk to public safety.

(c) Clarify, with regards to the NEI 00-04 Section 8 sensitivity study, that the FNP calculation will use the mean risk values of each PRA modeled hazard group. Include in this discussion what steps FNP will perform in the case the sensitivity results exceed the RG 1.174 acceptance guidelines.

(d) Alternatively to Part (c), propose a mechanism that ensures the NEI 00-04 Section 8 cumulative sensitivity study results is in conformance with the RG 1.174 risk acceptance guidelines when the internal events, internal flooding, and fire PRA mean values are used in the study.

SNC Response to NRC RAI 3

(a) Tables RAI-03-1 and RAI-03-2 demonstrates the FNP total CDF and LERF mean values meet RG 1.174 risk acceptance guidelines.

RAI-03-1. FNP CDF and LERF Mean Values

Unit	Hazard	CDF/LERF	Mean Value
1	IE	CDF	2.9E-06
	IE	LERF	1.9E-08
	IF	CDF	4.2E-06
	IF	LERF	1.8E-08
	F	CDF	7.7E-05
	F	LERF	2.7E-06
	S	CDF	8.9E-07
	S	LERF	7.4E-08
2	IE	CDF	3.0E-06
	IE	LERF	2.0E-08
	IF	CDF	4.0E-06
	IF	LERF	1.7E-08
	F	CDF	7.7E-05
	F	LERF	5.2E-06
	S	CDF	8.9E-07
	S	LERF	7.6E-08

Note: IE-Internal Events, IF-Internal Flood, F-Fire, S-Seismic

RAI-03-2. Total FNP CDF and LERF Mean Values

	Unit 1 Total	Unit 2 Total
CDF	8.4E-05	8.5E-05
LERF	2.8E-06	5.3E-06

(b) Not applicable.

(c) The FNP calculation to meet the requirements of the sensitivity study described in NEI 00-04, Section 8, will use mean values for comparison to the RG 1.174 acceptance guidelines. If the sensitivity results exceed the RG 1.174 guidelines, FNP will follow the NEI 00-04 guidance by reducing the FV and RAW threshold values used for categorization. The NEI 00-04 HSS thresholds will be lowered until such time as the RG 1.174 values can again be met. If previously categorized systems are impacted, the existing SSC categorization and treatment process will be followed.

(d) Not applicable.

Joseph M. Farley Nuclear Plant – Units 1 and 2
Response to Request for Additional Information Regarding Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"

Enclosure 2

EPRI Report 3002017583

ELECTRIC POWER RESEARCH INSTITUTE

Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization

3002017583

Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization

3002017583

Technical Update, February 2020

EPRI Project Manager
J. Richards

All or a portion of the requirements of the EPRI Nuclear Quality Assurance Program apply to this product.

YES NO

ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1338 ▪ PO Box 10412, Palo Alto, California 94303-0813 ▪ USA
800.313.3774 ▪ 650.855.2121 ▪ askepri@epri.com ▪ www.epri.com

THE FOLLOWING ORGANIZATION, UNDER CONTRACT TO EPRI, PREPARED THIS REPORT:

Simpson Gumpertz & Heger Inc

This is an EPRI Technical Update report. A Technical Update report is intended as an informal report of continuing research, a meeting, or a topical study. It is not a final EPRI technical report.

NOTE

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Simpson Gumpertz & Heger Inc.
4695 MacArthur Court, Suite 500
Newport Beach, CA 92660

Principal Investigator
G. Hardy

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Stephen Kimbrough, Duke Energy

Greg Krueger, Nuclear Energy Institute

Allen Moldenhauer, Dominion Energy

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Charles Young, JENSEN HUGHES

Joe Vasquez, Dominion Energy

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PRODUCT DESCRIPTION

The U. S. Nuclear Regulatory Commission (NRC) amended its regulations to provide an alternative approach for establishing the requirements for treatment of structures, systems, and components (SSCs) for nuclear power reactors using a risk-informed method of categorizing SSCs according to their safety significance. The NRC's 10 CFR 50.69 process allows a plant to categorize the safety significance of its SSCs using a robust categorization process defined in Nuclear Energy Institute (NEI) 00-04, *10 CFR 50.69 SSC Categorization Guideline,* as endorsed by the NRC in Regulatory Guide 1.201. The risk-informed categorization process helps focus attention on SSCs that are the most important to plant safety while allowing increased operational flexibility for SSCs that are less important to plant safety.

Background

Seismic risks are one of the screening criteria evaluated in the categorization process specified in NEI 00-04. Seismic risks can be evaluated using a seismic probabilistic risk assessment (Seismic PRA or SPRA) or a seismic margin assessment (SMA) if an SPRA is not available. Alternatively, they can be screened out if the seismic core damage frequency (SCDF) and seismic large early release frequency (SLERF) are very small compared to the full power internal events (FPIE) PRA core damage frequency and large early-release frequency.

Some plants do not have an acceptable SPRA or SMA and cannot screen out of seismic considerations. Therefore, cost-effective alternatives for accounting for the insights of seismic risks in the 50.69 categorization process must be considered.

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

Objectives

- To develop alternative approaches for plants to provide the necessary seismic risk insights within the 50.69 categorization process.

Approach

Trial 50.69 categorization evaluations are performed at four plants with SPRAs and high seismic hazards compared to their seismic design bases to determine the seismic-related categorization insights. Those insights are compared with categorization insights at the same plants using their FPIE PRAs and fire PRAs if available to determine the degree to which the seismic insights produce unique categorization insights.

The results of the trial cases are used to develop a risk-informed graded approach based on the degree to which the seismic hazard exceeds the seismic design basis ground motions and the degree to which unique seismic categorization insights are likely.

The treatment of potentially seismically correlated failures in SPRAs and identification of seismic interactions can lead to unique 50.69 categorization insights. Therefore, a process is developed to identify the plant conditions that would be treated as seismically correlated failures or interaction failures in an SPRA if one were available. For those conditions, a sensitivity study is recommended using the FPIE PRA to determine the impact of treating such seismic failures as common -cause events. Using this process, the necessary seismic risk insights can be identified for the 50.69 categorization process.

Results

Detailed analyses of seismic risks show very few insights to the 50.69 categorization results that uniquely identify SSCs as high-safety-significant. The primary unique categorization insights that would result from treatment of seismic-correlated failures in SPRAs can be derived using a process described in the report. A three-tiered, graded evaluation process is developed for considering seismic risk insights in the 50.69 categorization process.

Keywords

10 CFR 50.69
Risk-informed categorization
Seismic risk

Deliverable Number: 3002017583

Product Type: Technical Report

Product Title: Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization

PRIMARY AUDIENCE: Individuals developing License Amendment Requests to implement 10 CFR 50.69

SECONDARY AUDIENCE: Individuals involved in performing categorization using the 10 CFR 50.69 process

KEY RESEARCH QUESTION

Can an alternative process be used to provide the necessary seismic risk insights to perform 10 CFR 50.69 categorization without requiring development of a new seismic probabilistic risk assessment (SPRA) for low and moderate seismic hazard sites?

RESEARCH OVERVIEW

Sensitivity studies were performed at four nuclear plants to investigate the degree to which seismic risk insights uniquely contributed to the 10 CFR 50.69 categorization process. These studies were performed at plants with high seismic hazards relative to their design basis and that had developed new SPRAs. The results showed that few if any seismic risk insights uniquely caused an SSC to be classified as high-safety-significant (HSS) under 50.69 criteria and that the unique seismic conditions were generally associated with correlated seismic fragilities. Given these results, a three-tiered graded approach (low, medium, and high) was developed for considering seismic risks in the 10 CFR 50.69 process. In addition, a sensitivity study process was developed to derive seismic-correlated insights using an internal events probabilistic risk assessment (PRA) and common-cause techniques.

KEY FINDINGS

- At high-seismic-hazard plants—where the ground motion response spectrum significantly exceeds the safe shutdown earthquake—very few SSCs would be categorized as HSS solely for seismic risk reasons (Section 3).
- At high-seismic-hazard plants, the assumption of correlated seismic equipment failures can lead to unique 50.69 categorization insights (Section 3).
- Risk insights from internal events PRAs, combined with the insights from a fire PRA (if available) and other non-PRA aspects of the 50.69 categorization process, adequately identify the HSS structures, systems, and components necessary to address seismic risks, with the exception of a few unique seismic challenges.
- A graded approach based on the degree to which the ground motion response spectrum (GMRS) exceeds the safe shutdown earthquake (SSE) can be used to appropriately focus resources on the plants where seismic risk insights are more likely to uniquely contribute to the 50.69 categorization results (Section 2).
- A process is defined to identify conditions where seismically correlated component failures and seismic interactions would be modeled in an SPRA with a sensitivity study to determine the categorization impacts of those conditions using an internal events PRA and common-cause analysis techniques (Section 2.3.1 and Appendices A and B).

ELECTRIC POWER RESEARCH INSTITUTE

EXECUTIVE SUMMARY

WHY THIS MATTERS

Some plants do not have the tools identified in Nuclear Energy Institute (NEI) 00-04 to consider seismic insights in the categorization process. This report provides a graded, cost-effective, alternative process to appropriately consider seismic insights without requiring development of new SPRAs at low and moderate seismic hazard sites. The technical basis for the methods are specific to the 50.69 categorization process and are directly applicable to that process. Any plant with a low or medium seismic hazard would be eligible to apply this process.

HOW TO APPLY RESULTS

Plants should compare their GMRS with their SSE and determine if they fall in the low, medium, or high seismic hazard tier. Low-seismic-hazard plants should follow the process outlined in Section 2.2, medium-seismic-hazard plants should follow the process in Section 2.3, and high-seismic-hazard plants should follow the process in Section 2.4.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- Periodic workshops on 50.69-related topics are being held. Contact Pat O'Regan for additional information.

EPRI CONTACTS: John Richards, Technical Executive, jrichards@epri.com and Pat O'Regan, Technical Executive, poregan@epri.com

PROGRAM: Risk and Safety Management, P41.07.01

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Electric Power Research Institute
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

ACRONYMS AND ABBREVIATIONS

AC	alternating current
ACUBE	Advanced Cutset Upper Bound Estimator
AFW	auxiliary feedwater
AFWPH	auxiliary feedwater pumphouse
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOV	air-operated valve
ASME	American Society of Mechanical Engineers
BDD	binary decision diagram
BPVC	Boiler and Pressure Vessel Code
BWR	boiling water reactor
CAFTA	Computer Aided Fault Tree Analysis System
CB	circuit breaker
CCDP	conditional core damage probability
CCF	common cause failure
CCW	component cooling water
CDF	core damage frequency
CIV	containment isolation valve
CLERF	conditional large, early release probability
CR	control rod
CRM	configuration risk management
CST	condensate storage tank
DG	diesel generator
DID	defense in depth
EDG	emergency diesel generator
ELAP	extended loss of alternating current power
EPRI	Electric Power Research Institute
F&O	fact and observation
FIVE	Fire-Induced Vulnerability Evaluation

FLEX	diverse and flexible mitigation strategies
F-MCUB	factor-minimum cutset upper bound
FPIE	full power internal events
FPRA	fire probabilistic risk assessment
FTREX	Fault Tree Reliability Evaluation eXpert
F-V	Fussell-Vesely
GE	General Electric
GIP	generic implementation procedure
GMRS	ground motion response spectrum
HCLPF	high confidence of low probability of failure
HCU	hydraulic control unit
HHSI	high head safety injection
HPCI	high pressure coolant injection
HSS	high safety significant
HVAC	heating, ventilation, and air condition
HX	heat exchanger
IDP	integrated decision-making panel
IE	internal events
IEEE	The Institute of Electrical and Electronics Engineers
IFV	Integrated Fussell-Vesely
IPEEE	Individual Plant Examination of External Events
IRAW	Integrated Risk Achievement Worth
ISRS	in-structure response spectra
LERF	large early release frequency
LHSI	low head safety injection
LOCA	loss of coolant accident
LOOP	loss of off-site power
LSS	low safety significant
LUHS	loss of normal access to the ultimate heat sink
MCC	motor control center
MCR	main control room
MOV	motor-operated valve
MSA	mitigation strategy assessment
MWe	mega-watt, electric
MWt	mega-watt, thermal
NEI	Nuclear Energy Institute

NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NSCW	nuclear service cooling water
NSSS	nuclear steam supply system
NTTF	Near Term Task Force
OOS	out of service
PGA	peak ground acceleration
PRA	probabilistic risk assessment
PSHA	probabilistic seismic hazard analysis
PWR	pressurized water reactor
RAW	risk achievement worth
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
RG	Regulatory Guide
RISC	risk informed safety class
SBO	station blackout
SCDF	seismic-induced core damage frequency
SEL	seismic equipment list
SF	severity factor
SFP	spent fuel pool
SGIG	safety grade instrument gas
SI	safety injection
SLERF	seismic-induced large early release frequency
SLOCA	seismic-induced loss of coolant accident
SMA	seismic margin assessment
SPCL	success path component list
SPID	screening, prioritization and implementation details
SPRA	seismic probabilistic risk assessment
SQUG	Seismic Qualification Utility Group
SSC	structures, systems and components
SSD	safe shutdown
SSE	safe shutdown earthquake
SSEL	safe shutdown equipment list

UHRS	uniform hazard response spectra
USI	Unresolved Safety Issue
VAC	volts alternating current
VDC	volts direct current

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1
BACKGROUND

The U. S. Nuclear Regulatory Commission (NRC) amended its regulations to provide an alternative approach for establishing the requirements for treatment of structures, systems and components (SSCs) for nuclear power reactors using a risk-informed method of categorizing SSCs according to their safety significance. The NRC's 10 CFR 50.69 process [1] allows a plant to categorize SSCs using a robust categorization process defined in NEI 00-04, *10 CFR 50.69 SSC Categorization Guideline* [2], as endorsed by NRC in Regulatory Guide 1.201 [3]. The risk-informed categorization process helps focus attention on SSCs that are the most important to plant safety while allowing increased operational flexibility for SSCs that are less important to plant safety.

One of the criteria evaluated in the categorization process specified in NEI 00-04 is seismic risks, which can be evaluated using a seismic probabilistic risk assessment (Seismic PRA or SPRA), or a seismic margin assessment (SMA) if an SPRA is not available, or screened out if the seismic core damage frequency (SCDF) and seismic large early release frequency (SLERF) are very small compared to the full power internal events (FPIE) PRA CDF and LERF.

There are a number of plants that do not have an SPRA or SMA available to assess seismic risk in the categorization process and cannot screen out of seismic considerations by demonstrating very low seismic risks compared to FPIE risks, therefore a need exists to consider alternatives for considering the insights of seismic risks in the 50.69 categorization process. This report develops alternate approaches for plants to provide the necessary seismic risk insights within the 50.69 categorization process.

1.1 Seismic Evaluations at Nuclear Power Plants

Nuclear power plants are built to withstand environmental hazards, including earthquakes. The nuclear power plant regulatory process requires that seismic activity be taken into account as part of the design, operation and maintenance of the nuclear fleet. Safety-significant structures, systems, and components (SSC) are designed to withstand the effects of earthquakes and to maintain the capability to perform their intended safety functions. Several codes and standards govern aspects that directly affect the seismic margins inherent in the nuclear plants along with the estimation of the seismic risks, including standards from the American National Standards Institute (ANSI), American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), American Concrete Institute, and the Institute of Electrical and Electronics Engineers (IEEE).

Historically, when significant new seismic hazard information or new seismic capacity information became available, an assessment of this new data and models was undertaken to assess the impacts of this new data/methods. Several such major seismic reassessments have taken place in the United States that have impacted the majority of the nuclear plants in the fleet.

1. Unresolved Safety Issue A-46 (Generic Letter 87-02) [4]) – Operability of safety related equipment subjected to earthquakes
2. Individual Plant Examination of External Events (IPEEE) For Severe Accident Vulnerabilities – 10 CFR 50.54(f), (Generic Letter No. 88-20, Supplement 4), [5] – Seismicity of sites and beyond design basis evaluation
3. NRC Fukushima 50.54(f) letter [6] – Post Fukushima seismic reviews

The insights and conclusions from these seismic programs provide a good calibration for the proposed categorization of seismic risk/margin insights within the 50.69 categorization process.

1.1.1 Unresolved Safety Issue A-46

In December 1980, the U.S. Nuclear Regulatory Commission (NRC) initiated Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Nuclear Plants," [4] to address concerns that seismic qualification of equipment in older nuclear power plants might not be meeting expectations of newer seismic qualification criteria. The purpose of the USI A-46 program was to verify the seismic adequacy of essential equipment in operating plants not qualified in accordance with more recent criteria (that is IEEE 344-1975 [7]). In 1982, the Seismic Qualification Utility Group (SQUG) was formed to develop a practical approach for seismic qualification of equipment in operating plants. The approach developed by SQUG used experience data from equipment in power plants and industrial facilities that experienced actual earthquakes as the primary basis for evaluating the seismic ruggedness and functionality of essential equipment in nuclear power plants. A generic implementation procedure (GIP) [8] was developed that included the evaluation of active electrical and mechanical equipment, relay performance, tanks, heat exchangers cable raceways, and identification and resolution of possible seismic spatial systems interactions. Emphasis was placed on anchorage of equipment (a key insight that contributed to a significant number of earthquake equipment failures) and seismic walkdowns (a key tool to validating the installed condition of plant equipment and confirming characteristics of seismically rugged equipment).

A significant finding of the earthquake experience research is that conventional electrical and mechanical equipment included in the scope of the GIP will withstand earthquakes significantly higher than the design basis earthquakes for eastern U.S. nuclear plants, provided a set of key conditions are met. The guidelines in the GIP provide a systematic, controlled, and well-documented method of applying the lessons learned from review of earthquake experience data. The GIP screens out those types of conventional equipment that have been shown to be insensitive to earthquake motions expected in eastern U.S. plants and focuses on actual equipment and installation vulnerabilities identified in strong motion earthquakes as well as prior qualification test experience. Modifications were typically made for the safe shutdown equipment that did not meet the GIP criteria.

1.1.2 Individual Plant Examination of External Events

On June 28, 1991, the NRC issued Supplement 4 to Generic Letter (GL) 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" [4]. This supplement to GL 88-20, referred to as the IPEEE program, requested that each licensee identify and report to the NRC all plant-specific vulnerabilities to severe accidents caused by external events. The IPEEE program included the following four supporting objectives:

- Develop an appreciation of severe accident behavior
- Understand the most likely severe accident sequences that could occur at the licensee's plant under full-power operating conditions
- Gain a qualitative understanding of the overall likelihood of core damage and fission product releases
- Reduce, if necessary, the overall likelihood of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents

Seismic loading was one of the key elements of the IPEEE program. The IPEEE program resulted in a comprehensive seismic risk/margin assessment for the U.S. NPP fleet and, as such, represents a valuable resource for risk- informed applications such as 10 CFR 50.69.

The seismic IPEEE review results for 110 units are summarized in the EPRI 1000895 [9][1]. Of the 75 submittals reviewed, 28 submittals (41 units) used seismic probabilistic risk assessment (PRA) methodology; 45 submittals (65 units) performed seismic margin assessments (SMAs); and two submittals (four units) used site- specific seismic programs for IPEEE submittals.

Almost all licensees reported in their IPEEE submittals that no plant vulnerabilities were identified with respect to seismic risk (the use of the term "vulnerability" varied widely among the IPEEE submittals). However, most licensees did report at least some seismic "anomalies," "outliers," or other concerns. In the few submittals that did identify a seismic vulnerability, the findings were comparable to those identified as outliers or anomalies in other IPEEE submittals. Seventy percent of the plants proposed improvements as a result of their seismic IPEEE analyses.

[1] NRC performed a comparable review of IPEEE results in NUREG-1742, Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program. [39]

1.1.3 Post Fukushima Seismic Reviews

Following the accident at the Fukushima Daiichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the NRC established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations. The NTTF was also tasked with determining if the agency should make additional improvements to its regulatory system. The NRC issued an information request [6] associated with a seismic assessment on March 12, 2012, including recommendations 2.1 and 2.3 which required seismic evaluations and seismic walkdowns respectively. In addition, one other seismic program that came out of the post Fukushima requirements consisted of an assessment of the new mitigation strategies put into place under NTTF Recommendation 4 [6] by each nuclear plant. Each of these three post Fukushima seismic-related programs are briefly summarized below.

1.1.3.1 NTTF Recommendation 2.3 Walkdowns

The NTTF 2.3 program consisted of a relatively near term walkdown review of a sample of the safety related equipment in each U.S. nuclear plant to assess the seismic adequacy. The NRC requested that a walkdown review be conducted to address the plant specific vulnerabilities and to verify the seismic adequacy of the plant to the design basis level. EPRI 1025286 [10] provided technical guidance in 2012 for performing walkdowns to address the NTTF 2.3 request.

Lessons learned from these NTTF 2.3 walkdown reviews consisted of the following:

- The vast majority of the equipment and systems reviewed were demonstrated to be seismically adequate and in compliance with the design basis.
- A relatively minor number of issues were noted from these walkdown reviews:
  - Some anchorage conditions were identified that required actions to restore to the original condition
  - Some seismic interaction issues were noted
  - Some degraded equipment/hardware were noted (missing parts, corrosion, leaks, etc.)

Any issues identified as part of the NTTF 2.3 walkdowns were addressed by the licensees under their corrective action programs.

1.1.3.2 NTTF Recommendation 2.1 Seismic Evaluations

NTTF 2.1 was the longer term more detailed assessment of the implications of new seismic hazards on plant risk. The requested seismic information associated with recommendation 2.1 consisted of:

- Updated site-specific seismic hazards at operating nuclear power plants (NPPs)
- A seismic risk evaluation (SMA or seismic probabilistic risk assessment (SPRA)), as applicable, using the updated seismic hazard
- An assessment of the spent fuel pool (SFP) using the updated seismic hazard

The NRC requested each U.S. nuclear plant to provide information about the current hazard and potential risk posed by seismic events using a graded screening/evaluation approach. Depending on the comparison between the re- evaluated seismic hazard and the current seismic design basis, plants were requested to perform increasing levels of reevaluations. EPRI 1025287 [11] (known as the SPID) documents the methods undertaken by the U.S. nuclear industry to respond to the NTTF 2.1 request.

All U.S. nuclear plants performed a detailed reevaluation of the seismic hazard using modern Probabilistic Seismic Hazard Analysis (PSHA) criteria. These new seismic hazards were reviewed and approved by the NRC and formed the basis for the remainder of the NTTF 2.1 seismic evaluations. A significant number of U.S. plants completed (or are in the process of completing) SPRAs to address the NTTF 2.1 requirements. Four of the plants with new SPRAs have performed sensitivity studies documented in Section 3 to determine if there are any unique seismic insights that contribute to the 10 CFR 50.69 categorization process.

1.1.3.3 Mitigation Strategy Assessment

The third program that provided seismic insights for the U.S. NPP fleet consisted of a mitigation strategy assessment (MSA) conducted by all U.S. nuclear plants associated with the beyond seismic design basis evaluations of new mitigation equipment procured following the Fukushima event. The U.S. nuclear power industry initiated a program to add new capabilities and equipment to each plant. This initiative is referred to as FLEX [12] and includes the incorporation of strategies to safely respond to an assumed extended loss of alternating current (AC) power (ELAP) with a loss of normal access to the ultimate heat sink (LUHS) from an unspecified event. The NRC developed a recent regulation (NRC Draft Rule – Mitigation of Beyond Design Basis Events [13]) which required a beyond design basis review for these new FLEX mitigation systems. Relative to earthquakes, this rule required the assessment of the impact on the mitigation systems to the newly re-evaluated seismic hazards.

NEI has documented a detailed approach to demonstrate the seismic adequacy of mitigation strategy systems in NEI 12-06 Appendix H [12]. The seismic methods and criteria for evaluating MSA seismic adequacy incudes a graded approach. Five separate paths have increased requirements as a function of the degree that the latest seismic hazard exceeds the seismic design basis at the nuclear plant. In addition, these paths also take into account the degree/quality of existing seismic risk/margin evaluations that exist for the plant. The requirements and detail of these paths appropriately increase as the potential risk associated with the beyond design basis seismic event is deemed to potentially increase based on screening criteria agreed to by both the NEI/EPRI team as well as the NRC.

The MSA evaluations require demonstration that the FLEX strategies developed, implemented and maintained in accordance with NRC Order EA-12-049 [14] can be implemented considering the impacts of the reevaluated seismic hazard. The seismic insights from the seismic MSA assessments completed to date included many of the same insights observed from previous seismic programs:

- Anchorage of the equipment is important in the seismic event
- Seismic walkdowns by trained engineers are critical to identifying key issues that can best be identified in the field (for example, seismic interactions and vulnerabilities in operator pathways)
- Operator actions following the seismic event can address some seismic anomalies that occur during the earthquake (for example, resetting of relays, clearing operator pathways of smaller fallen objects)

1.1.4 Insights from Past Seismic Programs and Studies

A collective set of insights can be gained from the major seismic programs described above. While these programs varied in terms of their vintage and their required scope, the seismic insights have been quite consistent.

- Most SSCs have an inherent degree of seismic ruggedness
  - Earthquake experience data from large historical earthquakes have shown that the majority of equipment and structure types existing at nuclear power plants perform very well.
  - Mechanical equipment (pumps, valves, compressors, diesel generators) have a significant amount of seismic margin due to their being designed for operating loads in addition to the seismic loads.
  - Shake table test data demonstrate even higher levels of seismic capacity for safety related SSCs.
  - Distributed systems (HVAC ducting, cable trays, welded steel piping) perform very well in earthquakes and tests and have high inherent ruggedness
- A limited set of failure modes and seismic risk contributors exist
  - Anchorage – Anchorage is one of the key failure modes that results from earthquakes. For the safety-related nuclear plant equipment and systems, applicable design codes and standards require that seismic margin be designed into the anchorage. As such, for the moderate hazard sites, anchorage is not expected to contribute to the seismic risk until the earthquake reaches several times the design level.
  - Brittle Failure modes – Examples of a brittle failure mode could include configurations such as ceramic materials in electrical equipment or of cast iron anchorage. Past seismic programs such as the USI A-46 and the IPEEE seismic programs have identified these brittle failure modes and where these were identified as issues, modifications were typically conducted.

- Relay chatter – relay chatter for certain types of relays can occur at moderate earthquake levels. The most problematic relays (referred to as "bad actors") were reviewed and addressed already as part of the USI A-46 program. Since relay chatter does not represent a failure of the relay (the relay functions normally following the earthquake), operator actions can be undertaken to address relay chatter effects during the earthquake.
- Seismic interactions – insights from past earthquakes have shown that seismic interactions (block walls falling, lights falling, impact from cabinet deflections, seismic-induced flooding, etc.) can happen at moderate earthquake levels. The most appropriate method to identify these interactions is based on a trained team of engineers performing a walkdown. These walkdowns were integral parts of all three of the seismic programs summarized in Sections 1.1.1 through 1.1.3 above.
- Seismic correlation – Insights from past earthquakes have shown that the seismic damage to similar equipment/systems which are subjected to the same earthquake motions can be correlated. As such, if one cabinet fails during an earthquake, there is a reasonable chance that a similar cabinet in the same area could also fail during the earthquake[2]. The seismic correlation insights from past studies[2] are unique to seismic risk studies and are accounted for in an SPRA.

These collective insights have been integrated into the proposed alternate approaches for addressing the seismic risk in the 10 CFR 50.69 risk-informed categorization approach recommended in this report.

1.2 10 CFR 50.69 Categorization Process

NEI 00-04 [2] as endorsed in Regulatory Guide 1.201 [3] is one acceptable method for conducting a risk-informed categorization of structures, systems and components (SSCs) that provides evidence and confidence that SSCs will be categorized in a robust and integrated process consistent with 10 CFR 50.69(c)(1)(iv) [1]. The categorization process is performed for entire systems, one or more systems at a time, to ensure that all functions (which are primarily a system-level attribute) for a given component within a given system are appropriately considered.

The process described in NEI 00-04 and presented in Figure 1-1 [16] contains a number of key elements which are summarized below. These elements are used to arrive at a preliminary component categorization (that is, High Safety Significant (HSS) or Low Safety Significant (LSS)).

1. Full power internal events PRA
2. Internal and external hazards
3. Seven qualitative criteria in Section 9.2 of NEI 00 04
4. Defense-in-depth assessment
5. Passive categorization methodology

[2] See discussion in Appendix A

The analyses that can be used to address the hazards in items 1 and 2 above include:

- Internal Event Risk Analysis: Full power internal events PRA, including internal flooding.
- Internal Fire Events: EPRI Fire Induced Vulnerability Evaluation (FIVE) [15] screening process or Fire PRA.
- Seismic Events: Success Path Component List[3] (SPCL) from an IPEEE seismic margin analysis, SPRA or screening if the SPRA CDF is a small fraction of the internal events CDF (that is, <1%).
- Other External Events: (for example, tornados, external floods): External [hazard] PRA model and / or IPEEE screening process.
- Low Power and Shutdown Risks: Qualitative defense-in-depth (DID) shutdown model for shutdown configuration risk management (CRM) based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management."

[3] The term SPCL is used interchangeably in many seismic IPEEE documents with Safe Shutdown Equipment List (SSEL).

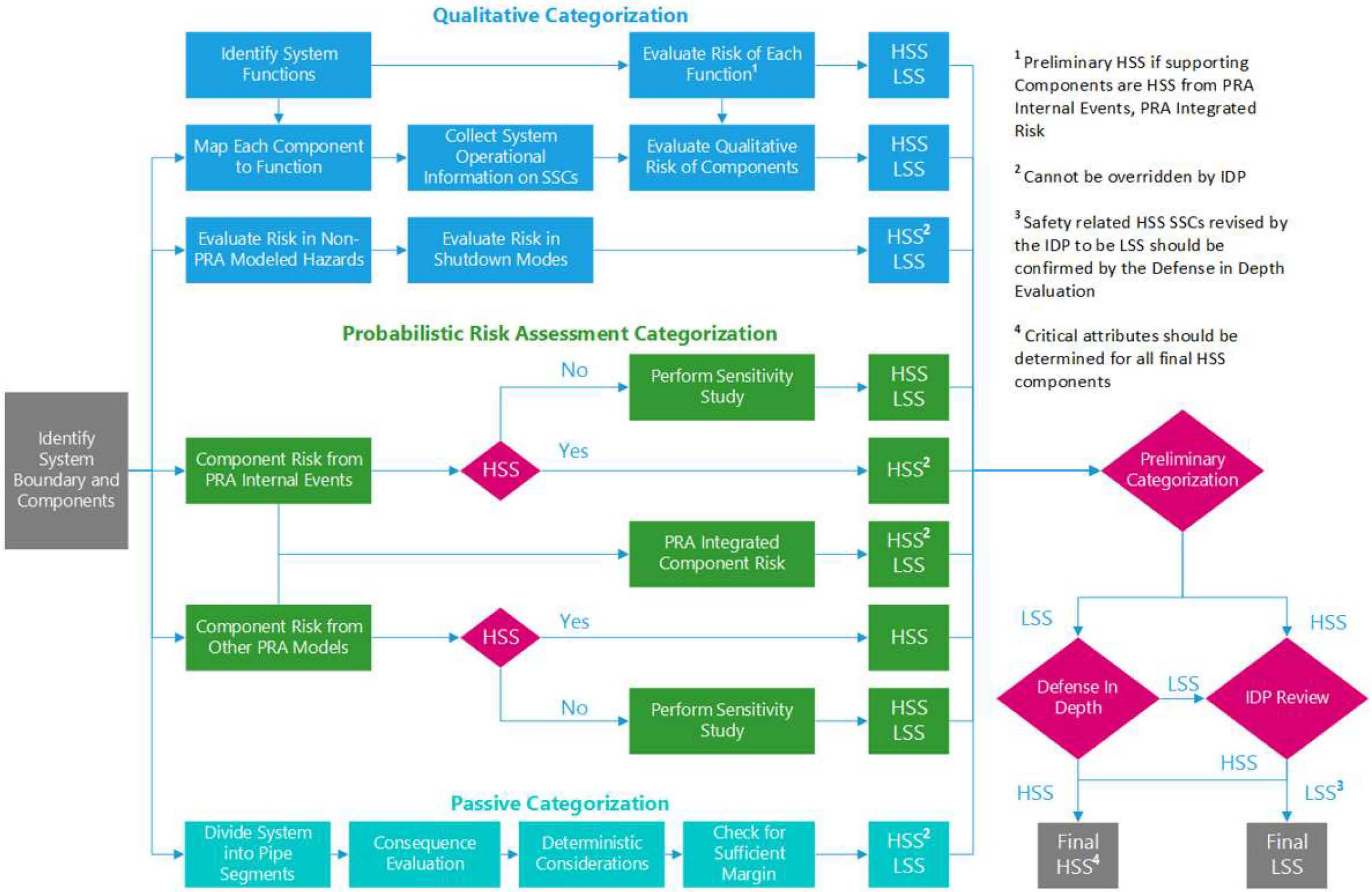


Figure 1-1
Categorization Process Overview (from NEI 16-09)

With respect to the seven qualitative criteria contain in Section 9.2 of NEI 00-04, the purpose of these considerations is to determine whether these functions/SSCs are not implicitly depended upon to maintain safe shutdown capability, prevention of core damage and maintenance of containment integrity. Specifically, consideration is given to whether:

1. Failure of the active function/SSC will not directly cause an initiating event that was originally screened out of the PRA based on anticipated low frequency of occurrence.
2. Failure of the active function/SSC will not cause a loss of reactor coolant pressure boundary integrity resulting in leakage beyond normal makeup capability.
3. Failure of the active function/SSC will not adversely affect the defense-in-depth remaining to perform the function.
4. The active function/SSC is not called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means for the successful performance of operator actions required to mitigate an accident or transient.
5. The active function/SSC is not called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means of achieving actions for assuring long term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities.
6. Failure of the active function/SSC will not prevent the plant from reaching or maintaining safe shutdown conditions; and the active function/SSC is not significant to safety during mode changes or shutdown.
7. Failure of the active function/SSC that acts as a barrier to fission product release during plant operation or during severe accidents would not result in the implementation of off-site radiological protective actions.

As discussed in Sections 6 and 9 of NEI 00-04 [2], in cases where the component is safety-related and found to be of low risk significance, it is appropriate to confirm that defense-in-depth is preserved. This includes consideration of the events mitigated, the functions performed, the other systems that support those functions and the complement of other plant capabilities that can be relied upon to prevent core damage and large, early release. Specific criteria are provided for assessing core damage defense-in-depth, including preventing core damage and limiting the frequency of the events being mitigated (Section 6.1), and containment defense-in-depth, including containment bypass, containment isolation, early hydrogen burns and long-term containment integrity (Section 6.2).

Per NEI 00-04, Defense-in-Depth is maintained if:

1. Reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of consequences of an offsite release.
2. There is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design.
3. System redundancy, independence, and diversity are preserved commensurate with the expected frequency of challenges, consequences of failure of the system, and associated uncertainties in determining these parameters.
4. Potential for common cause failures is taken into account in the risk analysis categorization.
5. The overall redundancy and diversity among the plant's systems and barriers is sufficient to ensure that no significant increase in risk would occur.

Finally, pressure boundary components (that is passive components and the passive function of active components) are evaluated using a consequence assessment approach where the component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment. Deterministic considerations (for example, DID, safety margins) are then also applied to determine the final safety significance from a passive perspective. Component supports are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model.

As can be clearly seen, the determination of safety significance through the various elements identified above provides a robust and integrated categorization of SSCs. The results of these elements are used as inputs to arrive at a preliminary component categorization (that is, High Safety Significant (HSS) or Low Safety Significant (LSS)) that is then presented to the Integrated Decision-Making Panel (IDP), a multi-discipline panel of experts that reviews the results of the initial categorization and finalizes the categorization of the SSCs/functions. Note: the term "preliminary HSS or LSS" is synonymous with the NEI 00-04 term "candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 1-1 below. Consistent with NEI 00-04, the categorization of a component or function will only be "preliminary" until it has been confirmed by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final Risk Informed Safety Class (RISC) category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04 Section 10.2. The IDP may always elect to change a preliminary LSS component or function to HSS, however the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 [2] and endorsed by RG 1.201 [3]. Table 1-1 summarizes these IDP limitations in NEI 00-04. The steps of the process are performed at either the function level, component level, or both. This is also summarized in the Table 1-1. A component is assigned its final RISC category upon approval by the IDP.

As a final note relative to the purpose of this report, the NEI 00-04 section on Integrated Risk Assessment includes the following.

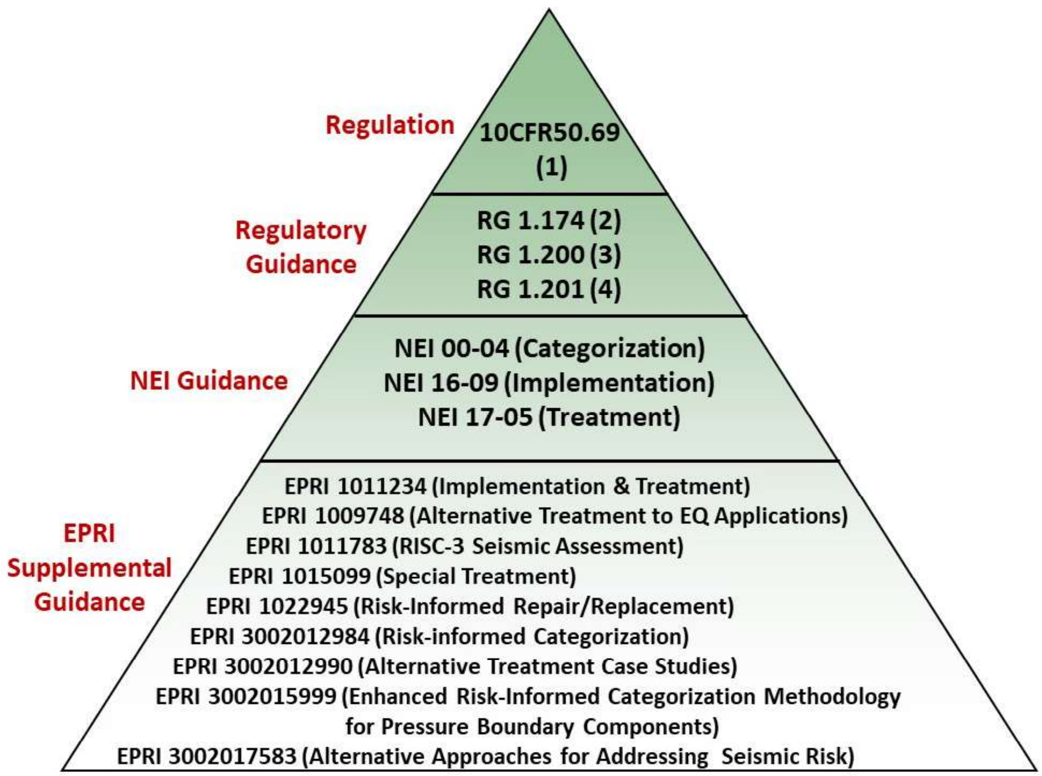
> Each risk contributor is initially evaluated separately in order to avoid reliance on a combined result that may mask the results of individual risk contributors. The potential masking is due to the significant differences in the methods, assumptions, conservatisms and uncertainties associated with the risk evaluation of each. In general, the quantification of risks due to external events and non-power operations tend to contain more conservatisms than internal events, at-power risks. As a result, performing the categorization simply on the basis of a mathematically combined total CDF/LERF would lead to inappropriate conclusions. However, it is desirable in a risk-informed process to understand safety significance from an overall perspective, especially for SSCs that were found to be safety-significant due to one or more of these risk contributors.

Table 1-1
Integrated Decision-Making Panel Changes from Preliminary High Safety Significant to Low Safety Significant

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Risk (PRA Modeled)	Internal Events Base Case – Section 5.1	Component	Not Allowed	Yes
	Fire, Seismic and Other External Events Base Case		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment – Section 5.6		Not Allowed	Yes
Risk (Non-modeled)	Fire, Seismic and Other External Hazards –	Component	Not Allowed	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

1.3 Relationship to the Rule and Other Guidance Documents

Figure 1-2 illustrates how this report relates to the 50.69 Rule [1] and other guidance documents. Requirements for implementing risk-informed categorization and treatment of SSCs are described in 10 CFR 50.69 [1], the adoption of which is optional by each licensee. The rule provides requirements for both phases of implementation; categorization and the resulting treatment allowances.



Notes:
(1) "50.69 Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"
(2) "An approach for using probabilistic risk assessment in risk-informed decisions on plant-specific changes to the licensing basis"
(3) "An approach for determining the technical adequacy of probabilistic risk assessment results for risk-informed activities"
(4) "Guidance for categorizing structures, systems and components in nuclear power plants according to their safety significance"

Figure 1-2
Relationship With the 10 CFR 50.69 Rule and Other Guidance Documents

2 PROPOSED APPROACH

The current approaches in NEI 00-04 [2] for considering seismic risks in the categorization process include the following options.

- An SPRA can be used with specified risk ranking and sensitivity studies to determine seismic related HSS SSCs; it is expected that an SPRA used for this purpose would meet RG 1.200 [17]
- An IPEEE Seismic Margins Assessment (SMA) equipment list can be used where all of the SSCs on the equipment list are designated HSS
- If the SPRA CDF is a small fraction of the internal events CDF (that is, <1%), then safety significance of SSCs considered in the SPRA can be considered LSS from a seismic perspective.

There are a number of plants that do not fit into any of these options. These are typically plants with moderate seismic hazards that did not use an SMA for their IPEEE response and were not required to perform an SPRA to respond to the NRC's Fukushima 50.54(f) letter [6]. This situation prompted a review for alternatives that could provide the appropriate seismic related insights to the categorization process.

A series of test cases were evaluated at sites with high seismic hazards and RG 1.200 compliant SPRAs to determine the types of seismic insights that would contribute to 50.69 categorization decisions. These test cases, described in Section 3, led to the development of the graded approach for categorization of seismic inputs described in this section.

2.1 Overview of Approaches

A graded approach is recommended that supports the 50.69 categorization process. The key premise of the approach is that most seismic related SSCs that would be categorized as HSS in accordance with NEI 00-04 would also be categorized as HSS for other reasons (that is internal events PRA insights, other external hazard risk insights, Defense-in-Depth considerations). Therefore, the goal of the graded approach is to identify SSCs that may be categorized as HSS based solely on seismic risk insights.

A second key premise is that the degree to which the plant seismic hazard, represented by the ground motion response spectrum (GMRS), exceeds the plant seismic design basis, represented by the Safe Shutdown Earthquake (SSE), influences the likelihood that unique seismic-related HSS SSCs will be identified. The 50.69 categorization process uses the F-V and RAW importance measures to determine relative ranking of the PRA SSCs. Since these are relative risk measures, even at a plant with low seismic hazards, there will always be a distribution of relative importance measures from high to low. However, at higher seismic hazard plants, the chances of identifying an unusual seismic induced condition that would cause SSCs to be HSS is greater. For example, the likelihood that nearby block walls will collapse and prevent important SSCs from performing their required functions becomes greater as the seismic hazard increases. In addition, the available seismic margin even in seismically designed SSCs, decreases as the seismic hazard increases, leading to greater likelihood of seismically induced failures and greater challenges to plant systems. Therefore, 50.69 seismic categorization test cases performed using plants with high seismic hazards relative to their seismic design basis would be more likely to identify unique conditions that would lead to the identification of HSS SSCs for unique seismic related reasons. Thus, plants with relatively high seismic hazard were chosen as test cases for the graded approach.

Three tiers are recommended within this graded approach. The primary measure for determining the appropriate tier for a plant is similar to the grading process used in EPRI 1025287 [11] for the Fukushima 50.54(f) letter responses and in NEI 12-06 [12] for the Mitigation Strategy Assessment. This measure is a comparison of the site-specific ground motion response spectrum (GMRS) to the site plant design basis (typically the Safe Shutdown Earthquake (SSE)) over the frequency range of 1.0 to 10 Hz. At sites where the GMRS/SSE ratio is low, there is a lower chance that seismic unique insights would contribute to HSS categorization. At sites where the GMRS/SSE ratio is higher, there are higher chances that seismic unique insights would contribute to HSS categorization.

The three recommended tiers are the following.

- Tier 1: Plants where the GMRS peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0 Hz and 10 Hz. Examples are shown in Figures 2-1 and 2-2. At these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are not expected.
- Tier 2: Plants where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3. At these sites, the unique seismic categorization insights are expected to be limited.
- Tier 3: Plants where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is high enough that the NRC required the plant to perform an SPRA to respond to the Fukushima 50.54(f) letter [6]. The NRC used a variety of site-specific quantitative and qualitative considerations in making these decisions [for example, 40, 41] and it represents the best available assessment of when an SPRA should be employed in risk-informed evaluations. The plants in this category are listed in Table 2-1[4]. Note that several plants planning to shutdown applied for, and received extensions of their SPRA due dates. Those plants are not included in Table 2-1.

[4] Note that several plants planning to shutdown applied for, and received extensions of their SPRA due dates from the NRC. Those plants are not included in Table 2-1. If those shutdown decisions change, they could be treated consistent with Tier 3 plants.

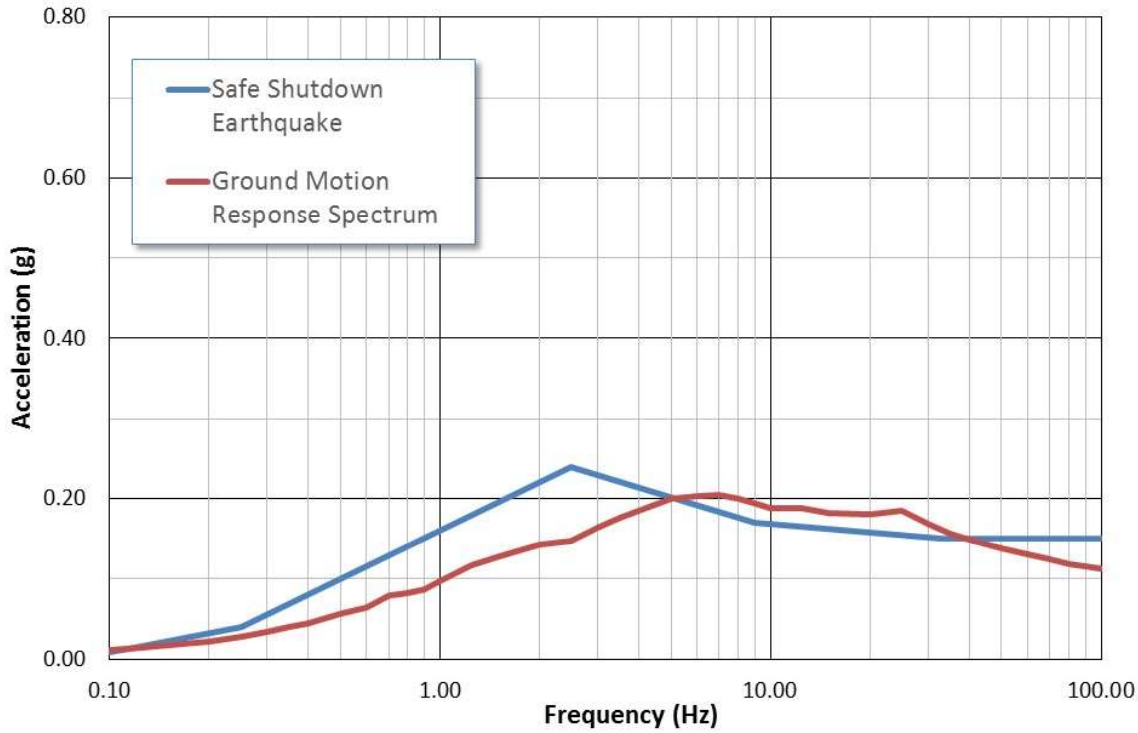


Figure 2-1
Low Seismic Hazard Site: Low GMRS Peak Acceleration

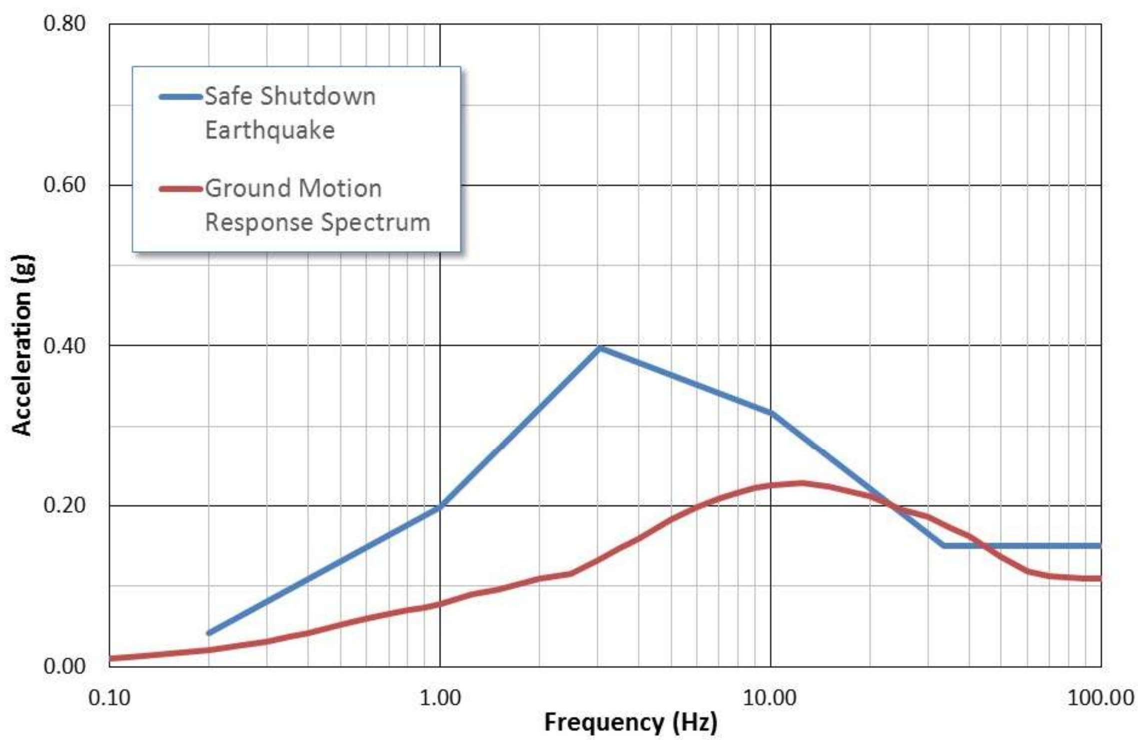


Figure 2-2
Low Seismic Hazard Site: Typical SSE to GMRS Comparison

Table 2-1
Plants Performing Seismic Probabilistic Risk Assessments for the Fukushima 50.54(f) Letter

Beaver Valley	DC Cook	Oconee	Sequoyah
Browns Ferry	Diablo Canyon	Palisades	VC Summer
Callaway	Dresden	Peach Bottom	Vogtle
Columbia	North Anna	Robinson	Watts Bar

The criteria for considering seismic risk insights in the 50.69 categorization process for each Tier is described below.

2.2 Tier 1 – Low Seismic Hazard / High Seismic Margin Sites

2.2.1 Description of the Approach

For Tier 1 plants, the GMRS is either very low or similar to the SSE such that unique seismic categorization insights are expected to be minimal. Since little to no unique seismic insights, would be anticipated for such sites, the approach is to rely on the 50.69 categorization process using the full power internal events (FPIE) and other risk evaluations along with the Defense-in-Depth, passive evaluations, and Integrated Decision-making Panel (IDP) assessment of the qualitative criteria. This process is expected to adequately identify the safety-significant functions and SSCs required for those functions.

2.2.2 Technical Basis for Approach

The test cases described in Section 3 showed that even for plants with high seismic ground motions compared to their design basis, there would be very few if any SSCs designated HSS for seismic unique reasons. At the low seismic hazard sites in Tier 1, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low.

2.2.2.1 Integral Assessment

In addition, since the seismic hazards for these sites are low, the seismic CDF would also be expected to be low. This is important because the NEI 00-04 [2] categorization process includes an Integral Assessment that weights the importance from each risk contributor (for example, internal events, fire, SPRAs) by the fraction of the total core damage frequency contributed by that contributor. The risk from an external hazard only drives a component to a required HSS determination if the results of the integral assessment meet the importance measure criteria for HSS. The integral assessment uses the following equations.

$$IFV_i = \frac{\sum_j \left(FV_{i,j} \times CDF_j\right)}{\sum_j CDF_j}$$

Equation 2-1
Integrated Fussell-Vesely Importance

where,

IFV_i = Integrated F-V Importance of Component i over all CDF Contributors

$FV_{i,j}$ = F-V Importance of Component i for CDF Contributor j

CDF_j = CDF of Contributor j

$$IRAW_i = 1 + \frac{\sum_j \left(RAW_{i,j} - 1\right) \times CDF_j}{\sum_j CDF_j}$$

Equation 2-2
Integrated Risk Achievement Worth Importance

where,

$IRAW_i$ = Integrated Risk Achievement Worth of Component i over all CDF Contributors

$RAW_{i,j}$ = Risk Achievement Worth of Component i for CDF Contributor j

CDF_j = CDF of Contributor j

Using these equations, if the seismic CDF is low, then the SSC importance measures from the SPRA are weighted lower than other risk contributors and the resulting seismic inputs to the categorization process are weighted lower. This would further reduce the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS

2.2.2.2 Relays and Contactors

The categorization of relays and contactors warrants some additional discussion of how they are addressed in the 50.69 categorization process. Relays and contactors are often considered to be seismically sensitive items and their performance during earthquakes can be important contributors to seismic risks.

FPIE PRAs do not typically model relays explicitly. They are included implicitly as part of modeled assemblies such as Control Panels and Motor Control Centers. For example, in backup power systems, the diesel generator control panel, including the relays inside the panel, would be considered part of the backup power system. If the control panel is determined to be HSS in the FPIE PRA, which is very likely since backup power is important to response of the plant to accidents, then the components in the control panel would be considered HSS by default as part of the HSS function. If a subsequent detailed categorization evaluation of the components in the control panel was performed in accordance with NEI 00-04 [2], then the function of the components in the control panel would be evaluated based on their contribution to successful performance of the control panel function. Therefore, relays that contribute to successful performance of the control panel function, or where intermittent chatter could prevent successful performance, in supporting backup power, would be categorized as HSS.

2.2.3 Summary/Conclusion/Recommendation

For Tier 1 plants, the GMRS is either very low or similar to the SSE such that unique seismic categorization insights are expected to be minimal. At the low seismic hazard sites in Tier 1, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low. Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the FPIE PRA and other risk evaluations along with the Defense-in-Depth and qualitative assessment by the Integrated Decision-making Panel (IDP) are expected to adequately identify the safety-significant functions and SSCs required for those functions and no additional seismic reviews are necessary for 50.69 categorization.

2.3 Tier 2 – Moderate Seismic Hazard / Moderate Seismic Margin Sites

2.3.1 Description of the Approach

For Tier 2 plants, the GMRS to SSE comparison is higher than Tier 1 plants but not high enough to be treated as Tier 3 plants. In Tier 2, there may be a limited number of unique seismic insights appropriate for consideration in determining HSS SSCs. These insights would be most likely attributed to the possibility of seismically correlated failures or seismic interaction related failures, not identified in the FPIE or Fire PRAs. Therefore, a special sensitivity study is recommended using a Common Cause approach in the FPIE PRA to account for similar categorization insights. These seismic insights would be considered with the other categorization insights by the Integrated Decision-making Panel (IDP) for the final HSS determinations.

The seismic insights from the four seismic categorization test cases are described in Section 3. One of the key seismic insights is the importance of considering seismic correlation effects on the plant risk. The correlated seismic response of SSCs that may occur in a seismic event is not captured in the internal event PRA or the fire PRA. As such, for 50.69 purposes, for Tier 2 plants should consider these seismic correlation insights when performing system categorization. Through correlated impacts, seismic events can fail both trains of SSCs in a two-train system depending on the seismic capacity and the seismic demand of the SSCs. Current SPRAs typically assign SSCs to be fully correlated when SSCs have the following same or similar conditions:

- The same seismic capacity based on similar governing failure modes in the same equipment (for example, anchor bolt tensile failure, functional failure based on testing, or bearing failure)
- The same seismic demand based on the location of the equipment (for example, same building, elevation), and similar orientation if the failure is dependent on the earthquake direction.

These seismic failures, referred to as correlated failures, are similar in impact to common cause failures (CCFs) that are typically modeled in an internal events PRA. The importance of the CCF basic events in the internal events PRA may be used to assess the importance of the seismic correlated failures. However, the probabilities of internal event CCF basic events are generally lower than the conditional probabilities of seismic correlated failures at higher ground motion levels. In addition, the internal events PRA may not have a CCF failure mode and basic event for some of the SSCs that may be subject to correlated seismic failures. For example, seismic failure of two tanks may be modeled as correlated failures in an SPRA, but common cause failure of tanks in the internal events PRA is typically screened out due to the very low probability of random occurrence.

Another key insight from the four seismic categorization test cases is that seismic interactions (for example, seismic induced falling, deflections and flooding that affect nearby safety related components) are unique to seismic risk studies and can result in seismic unique insights potentially leading to HSS SSCs.

To better assess the importance of SSCs to seismic event response at Tier 2 plants, the internal events PRA may be used with some modifications and augmented by focused seismic walkdowns to obtain an indication of the importance of SSCs for mitigating seismic events. The process is depicted in Figure 2-3 and the steps are summarized. Note that this process is performed on a system basis, as the 50.69 categorization process is performed for a given system.

1. Identify the set of SSCs within the system to be categorized.
2. Group the SSCs within the system into the classes of equipment and distributed systems used for SPRAs. This format of grouping allows for an efficient assessment from a seismic perspective. Industry documents such as the EPRI 3002000709 [18] and EPRI NP-6041-SL [19] identify the list of these classes. For example, separately group all manual valves, all check valves, all MOVs, all AOVs, all pumps, etc. This will make it easier to screen SSCs in the next step, as well as to see which SSCs already have CCF basic events modeled in the FPIE PRA.
3. Refine the list of SSCs in the system being categorized based on a series of screens to minimize the number of SSCs required to be evaluated as part of this correlation sensitivity study. Note that any/all of these screens can be incorporated into the process and that any order of implementing these three screens is acceptable. These screening decisions may be plant specific and likely will involve cost/benefit decisions in terms of how best to complete the sensitivity study.
    a. Screen out inherently rugged components. NEI 12-06 Appendix H [12] provides the following list of inherently rugged components.
        i. Strainers and small line mounted tanks
        ii. Welded and bolted piping
        iii. Manual valves, check valves, and rupture disks
        iv. Power operated valves (MOVs and AOVs) not required to change state

b. Screen out SSCs that are not used in safety functions that support mitigation of core damage or containment performance. This will likely already be identified as part of the function definition within the 50.69 categorization process. An example would be a chiller system that maintains the temperature of water in a tank but is not part of the safety function of the tank (for example, provides core inventory). The SSCs in the chiller subsystem can be screened.

c. Screen out from further evaluation the SSCs that have already been identified as being HSS from either the FPIE PRA or HSS from the Integral Assessment. For this screening step, the categorization based on the FPIE PRA and the Integral Assessment would need to have already been completed.

4. Those SSCs screened out in Steps 3a, 3b, or 3c can be removed from the sensitivity study. In addition, SSCs screened out below in Step 5c, 6, or 9 can be removed from further seismic consideration.

5. For system components not screened out in Step 3, perform a seismic walkdown focused on the three activities listed below. The purpose of this step is to identify SSCs that could experience seismic correlated failures or could be subject to seismic interactions that would lead to failure of more than one SSC within the system being categorized. The following elements contribute to identifying these conditions.

   a. Assess if the subject SSCs would likely experience correlated failures during a seismic event. Seismic correlation walkdown reviews are performed as part of an SPRA and the guidance associated with performing that correlation walkdown is documented in Appendix A.

   b. Assess potential seismic interactions to identify conditions that would be treated as seismic correlated failures and therefore, should be evaluated as common cause failures. Guidance for this seismic interaction walkdown review is also contained in Appendix A.

   c. Screen out SSCs that are determined to be sufficiently rugged such that they would not be significant contributors to seismic risk in an SPRA. This screen focuses on the SSC seismic capacity associated with functional failures and anchorage. The screen can also be applied to identified seismic interactions provided the seismic capacity of the interacting item (for example, block wall) has a seismic capacity that meets the screening level. Appendix B contains a description of the approach recommended for this screening.

6. SSCs that are determined through the walkdown to be of high seismic capacity and not included in seismically correlated groups or correlated interaction groups can be screened out from further seismic considerations since their non-seismic failure modes are already addressed in the FPIE PRA and fire PRA.

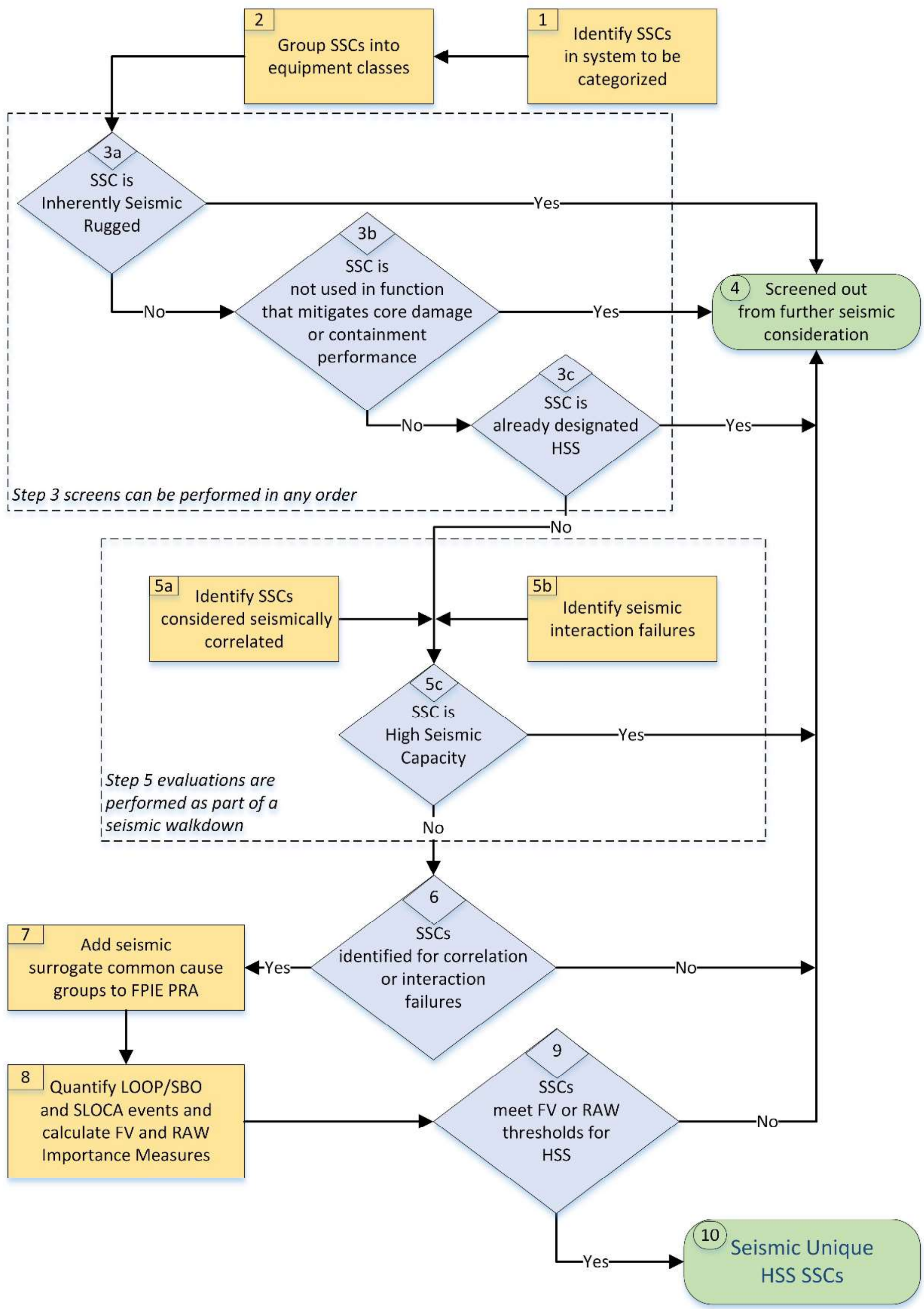


Figure 2-3
Seismic Correlated Failure Assessment

7. Add new seismic surrogate events to the FPIE PRA logic model for the potential seismically correlated conditions identified in the previous steps. New surrogates should be added to the PRA under the appropriate areas in the logic model. For example, a new surrogate basic event that models seismic correlated failure of two tanks would be added to the PRA logic model under the gates that model the individual tank failures. Seismic interaction surrogate events should be added to the model such that they fail the SSCs affected by the interaction. For example, a seismic interaction surrogate event that models a block wall falling onto two nearby pumps should be added to the PRA logic model under the gates that model the pumps.

   The probability for the new seismic surrogate basic events should be set to a value equivalent to a "typical" total seismic hazard exceedance frequency above which SPRAs would typically model loss of offsite power and for which correlated failures may be likely. The recommended value is 1.0E-04, but other appropriately justified values may be used.

8. Quantify the FPIE PRA model for LOOP and small LOCA events using the modified model with surrogate events, and calculate importance measures for the seismic surrogate events. Since the majority of seismic risk in many SPRAs are the LOOP and small LOCA accident sequences, these events represent are the most appropriate events for performing this correlation study. The process is as follows.

   a. The recommended event frequency for the LOOP initiator is 1.0 and for the small LOCA initiator is 1.0E-02. The LOOP frequency value of 1.0 is recommended since the probability of the surrogate events (from Step 7) is the total seismic hazard exceedance frequency above which SPRAs would typically model loss of offsite power. The basis for the small LOCA frequency of 1.0E-02 is that seismically-induced small LOCAs require failures that SPRAs show typically occur at much lower frequency than seismically-induced LOOP. Other appropriately justified values small LOCA frequency may be used. The majority of seismic risk is from the LOOP/SBO and small LOCA accident sequences.

   b. Set the frequency for all initiators other than LOOP and small LOCA to 0. Note that many FPIE PRAs have multiple LOOP initiating events (for example, grid centered, switchyard centered, etc.). Only one of these needs to be set to 1.0 in step 8a, above, while all the rest can be set to 0.

   c. Since a seismic event that causes a small LOCA is also assumed to cause a LOOP, update the PRA model to account for this. This is typically done by setting a conditional LOOP probability to 1.0, but can be done in any appropriate manner.

   d. Many FPIE PRA models credit restoration of offsite power in the LOOP/SBO accident sequences. This credit should not be taken in this process since recovery of offsite power after a seismic event is not generally credited in a seismic event.

9. For each seismic surrogate event, compare the results to the F-V and RAW HSS criteria for common cause components in the FPIE PRA from NEI 00-04 (that is, F-V > 0.005 or RAW > 20). For seismic surrogate events, if the F-V or RAW criteria are met, all SSCs modeled by that surrogate event should be considered HSS.

10. Since this process is a pseudo-deterministic evaluation process rather than a full risk informed process, these seismically correlated group HSS designations should be treated similar to HSS designations using the IPEEE SMA SSEL and in general, not be subject to reconsideration by the Integrated Decision-making Panel (IDP). However, SSCs which are HSS solely due to surrogate events representing seismic induced interactions (for example, block walls impacting equipment) may be downgraded to LSS by the IDP with appropriate justification. In addition, the IDP may direct further engineering evaluation to refine any of the seismic evaluation insights.

2.3.2 Technical Basis for Approach

The test cases described in Section 3 showed that even for plants with high seismic ground motions compared to their design basis, there are very few if any SSCs that would be designated HSS for seismic unique reasons and the technical basis for the Tier 1 approach in Section 2.2.2 generally apply for the Tier 2 plants.

The test cases did identify a small number of instances where unique seismic insights associated with seismically correlated failures led to unique HSS SSCs. While these unique HSS SSCs would be unusual for moderate hazard plants, it is prudent to perform additional evaluations to identify the conditions where these correlated failures may occur, and determine their impact in the 50.69 categorization process.

For a system being categorized under 50.69, the process described in Section 2.3.1 identifies the conditions that would be treated as seismically correlated fragilities or interaction consequences if an SPRA were being performed. It screens out SSCs that would either be very low contributors to seismic risk or would not be potential candidates for HSS, or were already categorized as HSS. These SSCs do not require additional seismic evaluations for considerations in the 50.69 categorization process. After that initial work, the evaluation follows a thought process similar to that in a SPRA, using a seismic walkdown to identify correlated conditions and potential seismic interaction and screen low seismic contributors.

The resulting unscreened SSCs are candidates for seismically correlated conditions that should be evaluated for their impacts on 50.69 categorization. In an FPIE PRA, common cause evaluations behave similarly to seismically correlated conditions in that they model the impacts of multiple SSC failures; therefore, the common cause methods can be employed to determine the necessary insights for 50.69 categorization.

The recommended methods and evaluation parameters in Section 2.3.1 for adding the surrogate seismic common cause events into the FPIE PRA and performing the sensitivity study serve to identify to necessary categorization insights. A common cause failure probability of 1.0E-4 is used to align with the total seismic hazard exceedance frequency above which SPRAs would typically model loss of offsite power and for which correlated failures may be likely.

Seismically induced LOOP and small LOCA events represent the majority of seismic risk in many SPRAs and are the most appropriate events for performing this correlation study. The process assumes a LOOP occurs (frequency set to 1.0) and uses a small LOCA frequency of 1.0E-02 because seismically-induced small LOCAs require failures that SPRAs show typically occur at much lower frequency than seismically-induced LOOP. This combination of events typically encompasses the majority of seismic risk.

The importance measures derived from this sensitivity study can then be used to identify the appropriate SSCs that should be HSS due to seismically correlated failures or seismic interaction related failures.

2.3.3 Summary

For Tier 2 plants, the GMRS to SSE comparison is higher than Tier 1 plants but not high enough to be treated as Tier 3 plants. In Tier 2, there may be a limited number of unique seismic insights, most likely attributed to the possibility of seismically correlated failures, appropriate for consideration in determining HSS SSCs. The special sensitivity study recommended using common cause failures, similar to the approach taken in a FPIE PRA, can identify the appropriate seismic insights to be considered with the other categorization insights by the Integrated Decision-making Panel (IDP) for the final HSS determinations.

2.4 Tier 3 – High Seismic Hazard / Low Seismic Margin Sites

For Tier 3 plants, the GMRS to SSE comparison is high enough that no alternate approach for seismic inputs to the 50.69 categorization process is proposed and the existing processes identified in NEI 00-04 for considering seismic risk in the categorization process should be used. The four seismic categorization test cases described in Section 3 suggest that even for Tier 3 plants, only a limited number of seismic unique insights are likely to contribute to the categorization process. However, in Tier 3 the potential GMRS seismic demands exceed the seismic design bases by high enough ratios that it is difficult to have high confidence that unique seismic conditions would not affect the set of SSCs captured in the categorization process.

Therefore, the available methods in NEI 00-04 can be used to provide seismic inputs to the categorization process. These methods include the use of an SPRA or an SMA as described in Section 5.3 of NEI 00-04 [2].

3 SEISMIC PRA INSIGHTS AND TRIAL CATEGORIZATION STUDIES CONDUCTED ON HIGH SEISMIC HAZARD SITES

Trial 50.69 categorization evaluations were performed at four plants with relatively high seismic hazards using the SPRAs available for these plants. These plants were characterized as having and high seismic hazards, based on the ground motion response spectrum (GMRS), compared to their seismic design bases, defined by the safe shutdown earthquake (SSE). These trial characterizations were undertaken to determine the seismic related categorization insights. Those seismic insights are compared with categorization insights at the same plants using their FPIE PRAs and fire PRAs if available to determine the degree to which the seismic insights produce unique categorization insights. The trial studies were not performed to implement actual SSC 50.69 categorization and did not include the NEI 00-04 criteria for the IDP. The trials were performed to derive seismic related categorization insights in support of this report.

3.1 Introduction to Trial Categorization Studies

Sensitivity studies were performed at four sites with newly developed SPRAs. The selected sites represent a distribution of different reactor types, containment types, and nuclear steam supply system (NSSS) types as shown in Table 3-1.

Table 3-1
Trial Plant Summary

Pilot Plant	MWe (Unit 1 / Unit 2)	NSSS Type	Containment Type
Plant A	1180 / 1180	GE / BWR 4	Mark I (Steel Drywel and Wetwell)
Plant B	1000 / 1000	Westinghouse / 3-loop	Large Dry, Subatmospheric
Plant C	1150 / 1150	Westinghouse / 4-loop	Large Dry
Plant D	1000 / 1000	Westinghouse / 4-loop	Wet, Ice Condenser

As described in Section 2.1, the degree to which the plant seismic hazard (GMRS) exceeds the seismic design basis (SSE) is expected to influence the likelihood that unique seismic-related HSS SSCs will be identified. At higher seismic hazard plants, the chances of identifying an unusual seismic induced condition that would cause SSCs to be HSS is greater. Therefore, 50.69 seismic categorization test cases performed using plants with high GMRS relative to their SSE would be more likely to identify unique conditions that would lead to the identification of HSS SSCs for unique seismic related reasons.

The GMRS to SSE comparisons at the four trial plants are shown in Figures 3-1 through 3-4. In each case, the GMRS exceeds the SSE by a significant margin.

Each trial is described in the following sections.

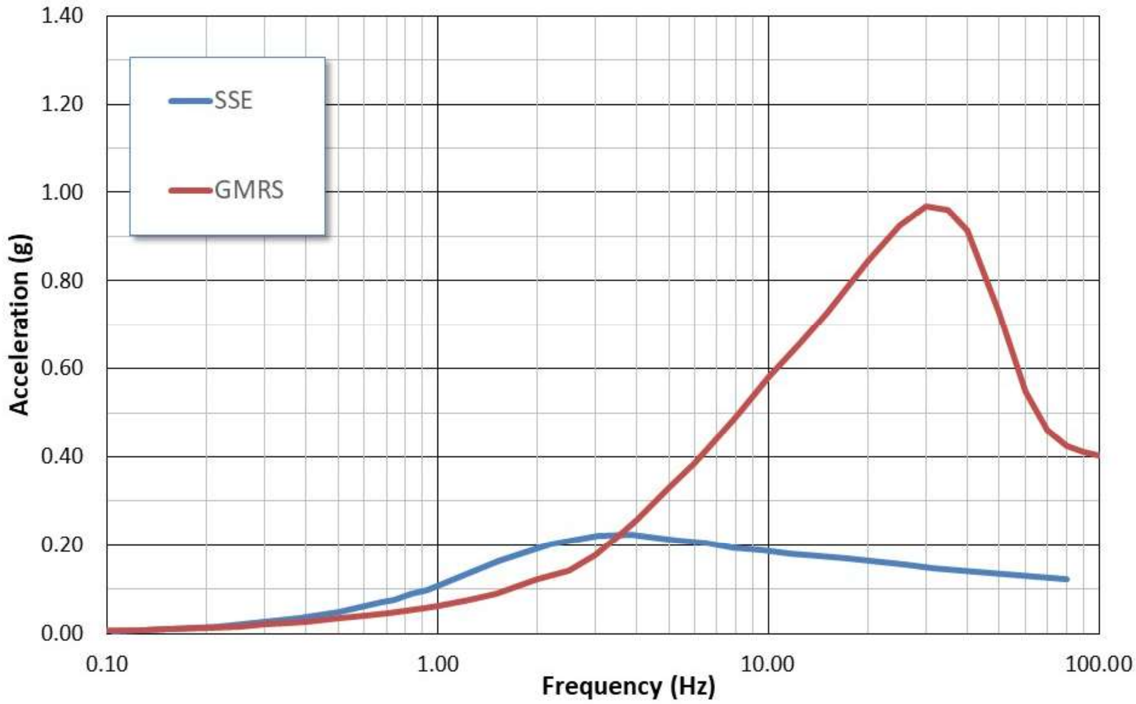


Figure 3-1
Plant A Ground Motion Response Spectrum to Safe Shutdown Earthquake Comparison

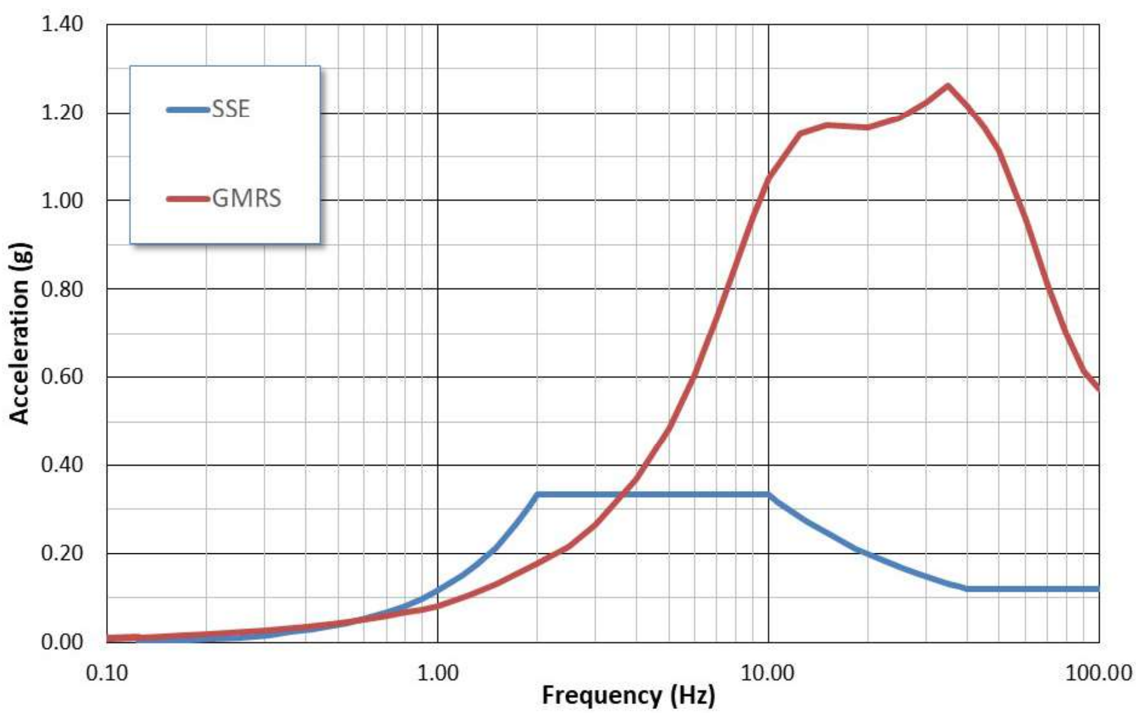


Figure 3-2
Plant B Ground Motion Response Spectrum to Safe Shutdown Earthquake Comparison

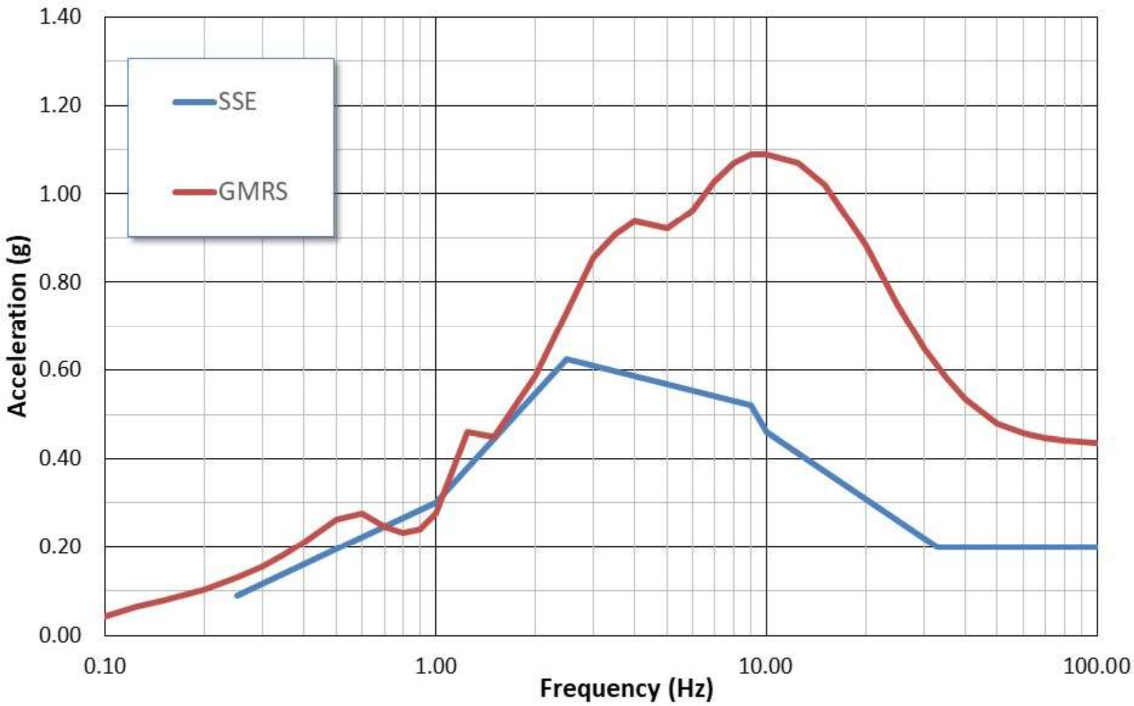


Figure 3-3
Plant C Ground Motion Response Spectrum to Safe Shutdown Earthquake Comparison

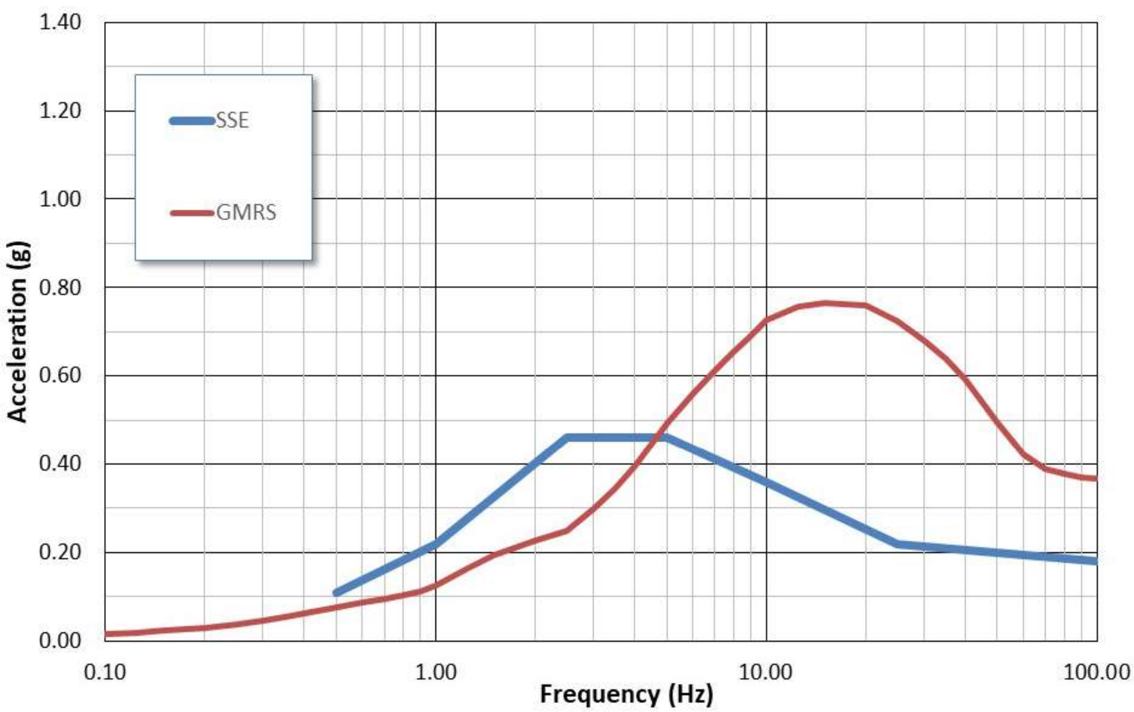


Figure 3-4
Plant D Ground Motion Response Spectrum to Safe Shutdown Earthquake Comparison

3.2 Plant A Trial Categorization Evaluation

3.2.1 Introduction

Plant A is a two-unit plant. Each unit has a GE BWR 4 Reactor and Mark I Containment. The initial plant design was 3458 MWt and 1180 MWe per unit. Plant A has implemented various power uprates (Leading Edge Flow Meter, Extended Power Uprate and Measurement Uncertainty Recapture) yielding current ratings of 4016 MWt and 1366 MWe, per unit.

The PRA models used in this study are described below.

3.2.2 Seismic PRA High Safety Significant Evaluation

3.2.2.1 Background

The risk-informed categorization of SSCs in nuclear power plant applications requires the use of an appropriately detailed PRA of sound technical quality. Plant A has a preliminary SPRA that is of sufficient quality to support this study. It was Peer Reviewed in March 2017 and is scheduled to be submitted to the NRC in 2018 in response to the Fukushima 50.54f letter [6]. This section will describe how the quantitative insights from the SPRA model are developed for this study.

3.2.2.2 Description of Model Used for Analysis

The SPRA models used for this analysis provided seismic insights into the as built, as operated plant. The risk significance of each identified component was examined using the SPRA. This evaluation consisted of examining the results from each unit for both core damage frequency (CDF) and large early release frequency (LERF). The single top SPRA model was created and quantified using the EPRI R&R Workstation software suite [20], with truncation values ranging from 1E-06/yr to 1E-10/yr for CDF and truncation values ranging from 5E-8/yr to 1E-12/yr for LERF. The seismic PRA uses a discretized set of seismic intervals to represent the full seismic hazard, evaluates the seismic CDF and LERF for each seismic interval, and integrates the results over the set of seismic intervals. Due to the special circumstances within seismic modeling (that is, over-counting caused by numerous high failure probability events), the EPRI Advanced Cutset Upper Bound Estimator (ACUBE) code [21], which uses the Binary Decision Diagram (BDD) algorithm, is used for more accurate quantification of the SPRA model.

The risk importances are calculated using cutset results (as typical in an R&R Workstation environment) and the ACUBE software to determine the individual basic event risk importance values. The seismic CDF (SCDF) Fussell-Vesely (F-V) values for SSC fragilities are calculated by integrating the various seismic interval basic event risk importances to determine the risk importance for a given SSC. The SCDF F-V values are based on a weighted sum of the individual SSC F-V values calculated for the individual seismic hazard intervals. In other words, the total F-V of an SSC fragility is the weighted sum of the associated eight SSC fragility basic events (one per hazard interval). Similarly, the SCDF risk achievement worth (RAW) risk importance measures are integrated for all eight basic events. This is typically performed manually in an Excel spreadsheet using the ACUBE output importance information. The seismic LERF (SLERF) F-V and RAW values are calculated using the same method discussed for SCDF.

Similar to Internal Events, the component importances were evaluated using the methodology described in NEI 00-04 [2] and the importance calculation criteria identified in Table 5-1 of NEI 00-04. The determination of High Safety Significant (HSS) or Low Safety Significant (LSS) from the SPRA is described in Section 3.2.5.

3.2.3 Full Power Internal Events PRA High Safety Significant Evaluation

3.2.3.1 Background

The Plant A full power internal events (FPIE) PRA model was Peer Reviewed in November 2010. It models severe accident scenarios resulting from internal initiating events occurring at full power operation as required to support the 50.69 categorization process. The results of the Internal Events quantification are used as one of the inputs in the 50.69 categorization process. This section describes how the quantitative insights from the Plant A Internal Events model are developed and used for categorization.

3.2.3.2 Description of Model Used for Analysis

The FPIE PRA models used for this analysis were 50.69 application specific models, adjusted to provide better insight into the as built as operated plant.

The risk significance of each identified component was examined using the FPIE PRA. This evaluation consisted of examining the results from both units for both CDF and LERF. While quantifying the FPIE PRA model using the EPRI PRAQuant Software [22], truncation values for CDF and LERF were both 5E-12. The component importances were evaluated using the methodology described in NEI 00-04 [2] and the importance calculation criteria identified in Table 5-1 [2]. The Determination of HSS or LSS from the Internal Events PRA is described in Section 3.2.5.

3.2.3.3 Identification of Risk Criteria and Analysis of Component Importances

NEI 00-04 [2] provides the following guidance.

> "An essential element of the SSC categorization process is a plant-specific full power internal events PRA, which should satisfy the accepted standards for PRA technical adequacy, reflect the as-built and as-operated plant, and quantify core damage frequency (CDF) and large early release frequency (LERF) for power operations due to internal events. Assessments of other hazards and modes of plant operation should be reviewed to ensure that the results and/or insights are applicable to the as-built, as-operated plant. PRAs provide an integrated means to assess relative significance. In cases where applicable quantitative analyses are not available, the categorization process will generally identify more SSCs as safety-significant than in cases where broader scope PRAs are available."

For the purposes of this analysis a full system mapping was not needed as only specific components, identified as HSS by the SPRA were evaluated. In practice a complete system mapping would be performed for the system being evaluated.

Per the NEI 00-04 process, a number of PRA risk significance evaluations are performed to determine the risk significance of PRA-modeled components. In order to perform the PRA risk significance evaluations a mapping of system components to PRA basic events (for example, pump fails to start, pump fails to run) would be performed. Typical events that are included in the mapping are events representing independent component unreliability, unavailability and common cause failures. Some components may not be explicitly modeled in the PRA however, surrogate events, such as operator actions may be used to provide insights into a components significance.

Basic event component mapping was performed consistent with the 50.69 categorization process. Each component was evaluated using two PRA importance metrics, Fussell-Vesely (F-V) and Risk Achievement Worth (RAW). Table 3-2 [2] lists the high-level criteria for risk significance of PRA modeled components. Common cause failure (CCF) contribution is included in the evaluation using a separate metric as defined in the table. Per the NEI 00-04 [2] guidance, for each PRA risk significance evaluation discussed below, the PRA F-V values for each basic event are added together for a given component; the maximum value of RAW for each basic event is selected for a given component; and the maximum value of common cause RAW for each basic event is selected for a given component. If a component exceeds any of the HSS criteria, the component is preliminarily categorized as HSS. The evaluation was performed in accordance with Section 5 of NEI 00-04.

Table 3-2
PRA Risk Criteria per NEI 00-04

PRA Ranking	NEI 00-04 Criteria
HSS	Sum of F-V for all basic events modeling the component of interest, including common cause events >0.005
HSS	Maximum of component basic event RAW values >2
HSS	Maximum of applicable common cause basic events RAW values >20
LSS	Modeled SSCs that do not meet any of the HSS criteria

3.2.4 Fire PRA High Safety Significant Evaluation

The Plant A Fire PRA was used to perform the 50.69 categorization process evaluation. The model was Peer Reviewed in November 2011 and the F&O Finding Closure Review was conducted in November 2016. This section describes how the quantitative insights from the fire model are developed and used for categorization.

3.2.4.1 Description of Model Used for Analysis

The Fire Models used for this analysis were application-specific models adjusted to provide better insight into the as built, as operated plant.

The risk significance of each identified component was examined using the Fire PRA, using the results from both units for both CDF and LERF. While quantifying the Fire PRA model using the EPRI PRAQuant Software [22], truncation values for CDF and LERF were both 5E-12. Similar to the FPIE PRA, the component importance measures were evaluated using the methodology described in NEI 00-04 and the importance calculation criteria identified in Table 3-2. The determination of HSS or LSS from the Fire PRA is described in Section 3.2.5.

3.2.4.2 Identification of Risk Criteria and Analysis of Component Importances

The same criteria and methodology as described in Section 3.2.3.3 (FPIE PRA) was also used to determine a component's safety significance with respect to the Fire PRA.

3.2.5 Comparison of Seismic PRA Results to Other PRA Results for High Safety Significant Structures, Systems, and Components

Each seismic failure event (fragility group) in the SPRA that exceeded the HSS thresholds was evaluated to determine if the same HSS determination would be made from either the FPIE PRA, FPRA or both. The HSS numerical criteria from Table 3-2 were applied to all PRAs used in this study. Table 3-5 provides the results of this comparison.

3.2.5.1 Explicit Modeling

As described in earlier sections some components can be explicitly represented in the PRA by using basic events that represent either unreliability or unavailability of the SSC. For the purposes of this study, some of the seismic failure event identified as being HSS from the SPRA were explicitly modeled in either the Internal Events PRA, Fire PRA or both. These fragilities are shown in Table 3-3.

Table 3-3
Seismic Fragilities Addressed by Explicit Modeling

Fragility Group	Description
OSP	Offsite Power
S-NRBY2 -	Nearby Hydroelectric Plant (Offsite Power Source)
S-ACPA1-	120 VAC Bus 00Y03
S-DCBS10-	250 VDC Bus 30D11
S-DCBS2-	125 VDC Buses/MCCs 0(A-D)D13
S-DCBS4-	DC Panel 20D24, 30D21
S-DCBT1-	DC Batteries 2(A-D)D01, 3(A-D)D01
SCRAM	RPV Internals (Scram)
BOC	Break Outside Containment
SML	Seismic Induced Medium LOCA
S-PCI2	Primary Containment Isolation (Inboard and Outboard MSIVs)

3.2.5.2 Implicit Modeling

As described in earlier sections some components can be implicitly represented in the PRA by using basic events such as super components or operator actions as examples. For the purposes of this study, some of the seismic failure events (fragility groups) identified as being HSS from the SPRA were implicitly modeled in either the FPIE PRA, the Fire PRA or both. The methodology for choosing the implicit modeling for each of the seismic failure events of interest is summarized in Table 3-4.

Table 3-4
Seismic Fragilities Addressed by Implicit Modeling

Seismic Fragility Group	Description of Fragility	Implicit Modeling
S-CC014	Correlated Relay Group 14 (All 4KV Buses Unrecoverable)	This is a correlated relay group. Some of the relays within this group are explicitly modeled but some are not. However, the relay's supported 4 kV busses are modeled in the FPIE PRA and Fire PRA and are HSS. The failure of the 4 kV busses was used as a surrogate to bound the relay categorization.
S-CC023	Correlated Relay Group 23 (All 4KV Buses Unrecoverable)	Similar to S-CC014, this fragility is a correlated relay group. Some of the relays within this fragility are explicitly modeled but some are not. However, the relay's supported 4 kV busses are modeled in the FPIE PRA and Fire PRA and are HSS. The failure of the 4 kV busses was used a surrogate to bound the relays categorization.
S-DGTK2	E1-E4 EDG Fuel Oil Day Tank 0(A-D)T40	The fuel oil tanks are not specifically modeled in the FPIE or Fire PRA models. However, they support the EDGs which are HSS in both the FPIE and Fire PRA models and would be implicitly mapped to the failure of the EDGs. Additionally, both the EDGs and EDG fuel oil tanks would be modeled in the same function during the categorization process, therefore since the EDGs are HSS in the FPIE PRA model, so would the fuel oil tanks.
S-CC127A	Correlated Relay Group 127 and 157 (EDG A and C-Recoverable)	Failure of this group leads to the failure of the DIV 1 and DIV 3 EDGs. For this reason, the relays were implicitly mapped to the failure of their impacted EDG.
S-CC187	Relay Chatter Event 187 (EDG B-Recoverable via operator action)	Failure of this group leads to failure of the DIV 2 EDG. For this reason, the relays were implicitly mapped to the failure of their impacted EDG.
S-CC191	Relay Chatter Event 191 (EDG D-recoverable via operator action)	Failure of this group leads to failure of the DIV 4 EDG. For this reason, the relays were implicitly mapped to the failure of their impacted EDG.
S-CC211	Relay Chatter Event 211 (HPCI - recoverable via operator action)	The relays associated with this group are not explicitly modeled; however, failure of the components lead to failure of the HPCI Pump. For that reason, the components are considered HSS by implicitly mapping them to the failure of the HPCI Pump.

Table 3-4 (continued)
Seismic Fragilities Addressed by Implicit Modeling

Seismic Fragility Group	Description of Fragility	Implicit Modeling
S-CC229	Relay Chatter Event 229 (RCIC inboard isolation valve)	The relays associated with this group are not explicitly modeled; however, failure of the components lead to failure of the RCIC Inboard Isolation Valve. For that reason, the components are considered HSS by implicitly mapping them to the failure of the RCIC Inboard Isolation Valve.
S-CC230	Relay Chatter Event 230 (RCIC - recoverable via operator action)	The relays associated with this group are not explicitly modeled however, failure of the components lead to failure of the RCIC Pump. For that reason, the components are considered HSS by implicitly mapping them to the failure of the RCIC Pump.
S-CNCT1	Condensate Storage Tank (CST) 20T010 and 30T010	Failures of the condensate storage tanks are modeled in the FPIE and Fire PRAs however, the failure probability is so low that the events truncate out of the cutsets. The tanks can also be implicitly represented by operator actions. The operator action to refill the CST was used to gain an insight into the importance of the CST as this action did not truncate out of the cutsets. This action was determined to be HSS.
S-SSGTK1	SGIG Nitrogen Tank	Similar to the S-CNCT1, the SGIG tank is explicitly modeled in the FPIE PRA but is LSS due to the low probability of failure. The tank can also be implicitly represented by operator actions. The operator action to align the tank was used to gain an insight into the importance of the SGIG tank as this action did not truncate out of the cutsets. The dependent operator actions were also used to determine the safety significance of the SGIG tank.
S-CEPA1	Panel 20C003, 20C004C, 30C003, 30C004C, 00C29(A-D)	The panels are not modeled explicitly in the FPIE PRA models but the panels support numerous components which are modeled as basic events in the PRA model. Many of the basic events modeling the supported components are HSS. For this reason it can be determined that the panels associated with this fragility group are also HSS.

3.2.6 Analysis and Conclusions

Examination of the Plant A SPRA, FPIE PRA and Fire PRA information and results in Table 3-5 shows that all SSCs or correlated fragility groups that are HSS in the SPRA are also HSS in the FPIE PRA or Fire PRA or both. This is shown graphically in Figure 3-5. There are 23 fragility groups in the SPRA that meet the F-V or RAW criteria for HSS. Of those 23 groups, 22 would also be identified as HSS in the FPIE PRA, 17 would be HSS in the FPRA, 12 would be HSS from implicit modeling, and five would be HSS from passive categorization considerations.

Every SSC that is HSS due to seismic considerations would also be HSS for other considerations and there are no HSS insights uniquely due to seismic.

The Plant A HSS SPRA SSCs are bounded by the HSS SSCs in the FPIE PRA or Fire PRA or both. No exceptions or outliers were identified.

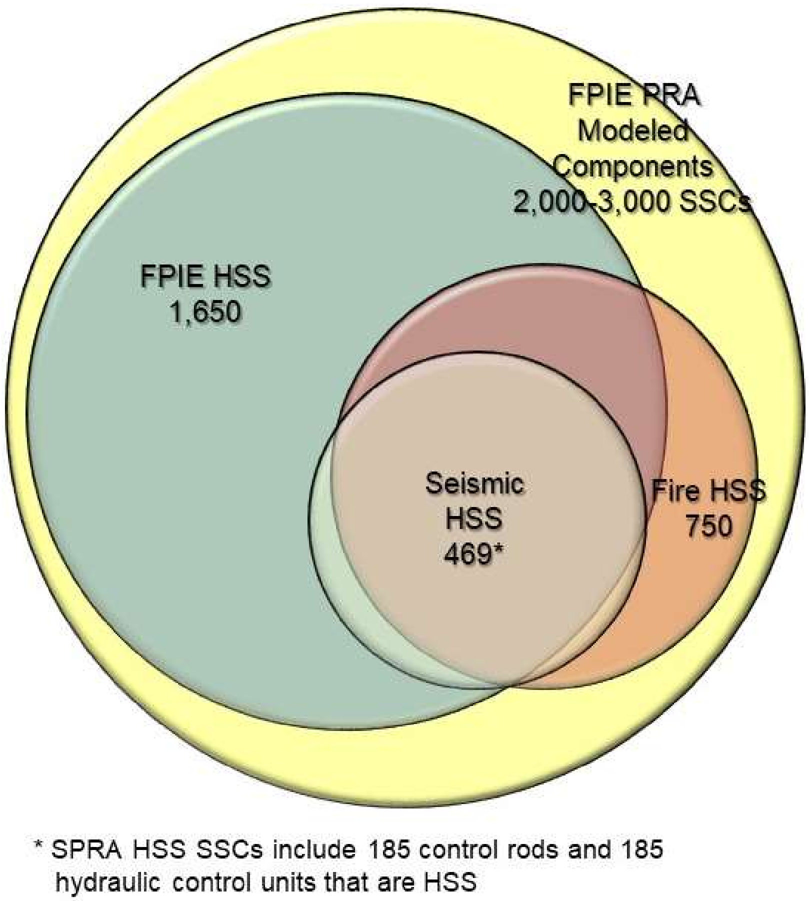


Figure 3-5
High Safety Significant Structures, Systems, and Components for Plant A

Table 3-5
Sensitivity Study Results for Plant A

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
Emergency Power	OSP	Offsite Power	Generic Fragility (for example, ceramic insulators in plant switchyard)		Anchorage	✓	✓	✓				
	S-CC014-	Correlated Relay Group 14 (All 4KV Buses Unrecoverable)	Contact Device CRL SI-Overcurrent	Relays in Bay 8, B	Functional	✓	✓	✓	✓			
	S-CC023-	Correlated Relay Group 23 (All 4KV Buses, Recoverable)	Contact Device CRLs	Relays in: Bay 2, Bay 4 Bay 5	Functional	✓	✓	✓	✓			
	S-NRBY2-	Nearby Hydroelectric Plant (OSP)	Generic Fragility (for example, ceramic insulators in Nearby Dam switchyard)		Anchorage	✓	✓					

Table 3 5 (continued)
Sensitivity Study Results for Plant A

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
Emergency Power	S-DGTK2-	E1-E4 EDG Fuel Oil Day Tank 0(A-D)T40	EDG Fuel Oil Day Tanks	0AT040 0BT040 0CT040	Anchorage	✓	✓	✓	✓			The fuel oil tanks are not specifically modeled in the FPIE and Fire PRA models. However, they support the EDGs which are HSS in both the FPIE and FPRA models. Both the EDGs and EDG fuel oil tanks would be modeled in the same function during the categorization process and therefore the EDGs being HSS in the PRA model would drive the fuel oil tanks HSS.
	S-ACPA1-	120 VAC Bus 00Y03	120 VAC Bus	00Y03	Anchorage	✓		✓				
	S-CC127A-	Correlated Relay Group 127 and 157 (EDG A and C-Recoverable)	Contact Device CRLs	0-52B-132-AG12 0-52B-132-CG12	Functional	✓	✓	✓	✓			
	S-CC187-	Relay Chatter Event 187 (EDG B-Recoverable)	Contact Device CRL	0-52B-132-BG12	Functional	✓	✓	✓	✓			
	S-CC191-	Relay Chatter Event 191 (EDG D-Recoverable)	Contact Device CRL	0-52B-132-DG12	Functional	✓	✓	✓	✓			

Table 3 5 (continued)
Sensitivity Study Results for Plant A

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
DC Power System	S-DCBS10-	250 VDC Bus 30D11	250V DC HPCI BUS	30D11	Anchorage	✓	✓	✓				
DC Power Systems	S-DCBS2-	125 VDC Buses/MCCs 0(A-D)D13	125 VDC Buses/MCCs	0AD013, 0BD013, 0CD013, 0DD013	Functional	✓	✓	✓				
	S-DCBT1-	DC Batteries 2(A-D)D01, 3(A-D)D01	125 VDC Battery	3AD01	Anchorage	✓	✓	✓				Most DC buses are individually HSS and CCF of all are HSS in both FPIE and FPRA (EBS13ALLCWI0).
	S-DCBS4-	DC Panel 20D24, 30D21	125 VDC Panel	20D24	Anchorage	✓	✓					
Safety Injection	S-CC211-	Relay Chatter Event 211 (HPCI - recoverable)	Contact Device CRL	2-23A-K036	Functional	✓	✓	✓	✓			
	S-CC229-	Relay Chatter Event 229 (RCIC inboard isolation valve - Unre-coverable)	Contact Device CRL	2-13A-K012	Functional	✓	✓		✓			
	S-CC230-	Relay Chatter Event 230 (RCIC - recoverable)	Contact Device CRL	2-13A-K022	Functional	✓	✓	✓	✓			

Table 3 5 (continued)
Sensitivity Study Results for Plant A

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
Reactor Protection System	SCRAM	RPV Internals (Scram)	RPV core shroud leg		Anchorage	✓	✓					
Onsite Water Sources	S-CNCT1-	Condensate Storage Tank 20T010, 30T010	Condensate Storage Tanks	20T010, 30T010	Anchorage	✓	✓		✓	✓		HSS due to operator action to align to the CST
Piping	BOC	Break Outside Containment	RCS piping outside containment (for example, Main Steam Lines)		Anchorage	✓	✓	✓		✓		THE BOC Initiating Events are HSS but a review of the components involved has also been performed to identify any HSS individually
Piping	SML	Seismic Induced Medium LOCA	RCS piping inside containment (for example, between 2" and 6" diameter)		Anchorage	✓	✓			✓		Medium LOCA is HSS in the Internal Events Model
Safety Grade Instrument Gas	S-SGTK1-	SGIG Nitrogen Tank	CAD Liquid Nitrogen Storage Tank	00T116	Anchorage	✓	✓	✓	✓	✓		SGIG Tank is not HSS Explicitly but it is HSS due to implicit modeling in operator actions (AHU--CADDXD2, AHU--CADDXI3)

Table 3 5 (continued)
Sensitivity Study Results for Plant A

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
MISC Panels	S-CEPA1-	Panel 20C003, 20C004C, 30C003, 30C004C, 00C29(A-D)	Panels	00C29A, 00C29B, 00C29C, 00C29D	Anchorage	✓	✓	✓	✓			The Panels are not modeled explicitly in the PRA models but the panels support numerous components which are modeled as basic events in the PRA model. Many of the basic events modeling the supported components are HSS.
Contain-ment	S-PCI2	Primary Containment Isolation (Inboard and Outboard MSIVs)	Outboard MSIVs	AO-2-01-086A AO-2-01-086B AO-2-01-086C AO-2-01-086D AO-3-01-086A AO-3-01-086B AO-3-01-086C AO-3-01-086D	Functional	✓	✓	✓		✓		Mapped to FPH--MSTDXI2. MSIVS Fail to Remain Open During Transient. Also HSS due to %VMSL
							22	17	12	5		
Totals						**23**	**23 Seismic Fragility Groups classified as HSS via overlapping 50.69 criteria**				**0**	

3.3 Plant B Trial Categorization Evaluation

This section documents the sensitivity performed using the Plant B SPRA.

3.3.1 Introduction

Plant B is a two-unit Westinghouse PWR (three loop) site with a large, dry sub- atmospheric containments and approximately 1,000 MWe each. They entered commercial operation in 1978 (Unit 1) and 1980 (Unit 2). Water from an adjacent lake is used to cool the main condensers. Emergency core cooling is accomplished by three High Head Safety Injection (HHSI) pumps and two Low Head Safety Injection (LHSI) pumps. The HHSI pumps also provide normal Charging and Reactor Coolant Pump (RCP) seal injection during non-accident conditions. There are two Emergency Diesel Generators (EDGs) that power the two emergency buses if offsite power is lost. There are four 120 VAC vital buses (two per emergency bus) powered by either the batteries/inverters (four total) or directly from the emergency buses through transformers. Three Auxiliary Feedwater (AFW) pumps (two motor-driven and one turbine-driven) provide steam generator cooling if the main Feedwater pumps are unavailable. The ultimate heat sink is from the 22.5 million-gallon Service Water reservoir, where four pumps provide SW flow to both units via two headers. Two auxiliary SW pumps that take lake water suction are also available to provide SW to the station.

3.3.1.1 PRA Models

The FPIE PRA and SPRA models contain logic for quantifying CDF and LERF for each unit. For this sensitivity, the results are from the Unit 1 PRA models only. The results for Unit 2 would be similar given that both units are nearly identical.

The Plant B SPRA model was developed from the latest FPIE PRA. That is, seismic fragility groups that model seismic failure of the SSCs were added to the appropriate locations in the fault trees. The accident sequences that model the various seismic-induced initiating events (LOCAs, SBO, etc.) were developed from the FPIE PRA accident sequences. The SPRA has top gates (Ux-CDF-SEISMIC and Ux-LERF-SEISMIC) which are quantified using the EPRI PRAQuant code. The cutsets are then processed using ACUBE to obtain the final seismic CDF and LERF as well as the importance data.

The FPIE PRA underwent a full scope peer review in 2014. The FPIE model has been revised to address all F&Os that impact model logic. The remaining F&Os are for documentation improvements and are not expected to impact the results. These were reviewed as part of the development of the SPRA to verify no impact on the SPRA results.

The SPRA model used for this sensitivity was peer reviewed in July 2017. The SPRA model is in the process of being revised to address the F&Os. The results of this sensitivity are not expected to be impacted by these F&Os.

The following three upgrades have been incorporated into the FPIE and SPRA models. The first upgrade is credit for FLEX in the Station Blackout (SBO) sequences has been added to both the FPIE and SPRA models. The FLEX mitigating actions modeled are load shedding the batteries and aligning the FLEX 120VAC generators to power the vital AC buses and maintain instrumentation. Also, the FLEX mitigating action to align the FLEX RCS Injection Pump for

RCS makeup from the RCP seal leakage is modeled in the SBO sequences. This upgrade was peer reviewed in July 2017 as part of the SPRA peer review. The review team determined that the FLEX modeling was appropriate but identified that the uncertainties relating to Human Reliability Analysis of the FLEX actions and the failure data of the FLEX SSCs should be investigated further. Sensitivities will be included in the final SPRA submittal. The team also recommended improvements in the documentation of the FLEX modeling. The results of the SPRA are not expected to be impacted by these F&Os.

The second upgrade involves using a convolution approach to modeling recovery of offsite power in the SBO sequences. This is only applicable to the FPIE model since the SPRA SBO sequences do not credit offsite power recovery.

The third upgrade involves the replacement of the Reactor Coolant Pump (RCP) seals with the low leakage Flowserve N-9000 seals. The final RCP seal was replaced in March 2018. Therefore, the SPRA credits the Flowserve RCP seal LCOA model (that is not the Westinghouse seal LOCA model). This is upgrade has not been peer reviewed at this time but will be scheduled in 2018. The Flowserve seal LOCA model used in the Plant B FPIE and SPRA model is essentially the same as the logic used in a nearly identical plant FPIE PRA model that had a focused peer review of the upgrade in 2016. Since the PRA models for both plants are developed and maintained by the utility PRA group using the same procedures and methods, the RCP seal LOCA models are nearly identical. The F&Os from the nearly identical plant focused scope peer review of the RCP seal LOCA model were reviewed and verified to not impact the results of the Plant B SPRA.

3.3.2 Seismic PRA High Safety Significant Evaluation

This section contains a summary of the SPRA Model and identification of HSS SSCs from the SPRA.

3.3.2.1 Description of Seismic PRA Model

The SPRA is integrated into the FPIE PRA by adding seismic failure gates under the appropriate portions of the logic model that model the other failures of the SSCs (for example, pump failing to start). Seismic failures of SSCs are modeled using fragility groups, which represent failure of groups of SSCs, typically both (multiple) trains if the SSCs are assumed to be correlated. Most of the SSCs are assumed to be correlated given similar design, location, and configuration. For example, both Low Head Safety Injection (LHSI) pumps, 1-SI-P-1A &1B, are assumed to be correlated because they are both located in the Safeguards building, are of similar design and are installed in the same configuration. Therefore, seismic failure of these pumps is modeled using fragility group SEIS-SI-P-1AB, which is placed in the logic model under the gates for both LHSI pumps.

The seismic hazard curve is divided into eight intervals and is modeled by eight seismic initiating basic events %G01 through %G08. Each fragility group is therefore modeled by eight seismic failure basic events representing the probability of failure for each of the eight seismic intervals of the seismic hazard curve. The seismic failure basic events in the LHSI pump fragility group are SEIS-SI-P-1AB-C-%G01 through SEIS-SI-P-1AB-C-%G08.

The SPRA model is quantified using EPRI PRAQuant to generate the cutsets, which are then processed with ACUBE [21] that uses the Binary Decision Diagram (BDD) to obtain a more accurate solution that reduces the overestimation that occurs when basic event probabilities are high. The model was quantified using truncation limits of 1.0E-09 and 1.0E-10 for SCDF and SLERF, respectively. ACUBE allows processing a subset of cutsets using the BDD since computer memory typically limits the number of cutsets that can be processed. The remaining cutsets are then processed by the factored-minimum cutset upper bound (F-MCUB) routine. ACUBE combines these to obtain the final SCDF and SLERF as well as their importances. For Plant B, 2,000 cutsets were processed for the SCDF importances and 1,500 cutsets were processed for the SLERF importance.

ACUBE generates Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) importance data for each basic event or fragility group in the cutsets. Both F-V and RAW are used to obtain the importance of the fragility groups. The same approach used by Plant A was used for Plant B in combining the importances of each of the eight seismic failure basic events to obtain the importance of the fragility group.

3.3.2.2 Identification of HSS SSCs from the SPRA

Fragility groups are considered High Safety Significant (HSS) if the group F-V is greater than 0.005 or if the group RAW is greater than 2.0 for CDF or LERF. Thus, if a group has a SCDF (SLERF) F-V or RAW that meet these HSS thresholds, then the SSCs in the group are considered HSS. In the example of the LHSI pump fragility group, SEIS-SI-P-1AB, both 1-SI-P-1A and 1-SI-P-1B pumps would be considered HSS if the fragility group SCDF (SLERF) F-V or RAW meet the HSS threshold.

The SPRA also models non-seismic failures (for example, failure to start, run) of SSCs that can impact mitigating functions. Only two SSCs, the diesel-driven fire pump (1-FP-P-2) and the FLEX 120VAC generator (0-BDB-GEN-1A) have F-V importance greater than 0.005. The remaining basic events that model the non-seismic failures have F-Vs and RAWs that are less than the thresholds for HSS.

The results from the SCDF and SLERF quantification and importances show that 36 fragility groups and two non-seismic failure basic events are considered HSS. Table 3-7 lists these fragility groups and basic events. The SSCs that are modeled by these fragility groups are also listed in the table. There are over 200 SSCs within these fragility groups that are considered HSS for seismic risk.

3.3.3 Full Power Internal Events PRA High Safety Significant Evaluation

In the FPIE model, failure of SSCs are modeled for the different failure modes, such as pumps, fans, compressors, etc. failing to start, failing to run, failure to load, and out of service for test or maintenance. Additional failure modes may be modeled depending on the component. Common cause failures of the components are also included in the FPIE to account for possible design, maintenance and latent defects that could be common between similar components within the trains.

These SSC failures are modeled as "basic events" in the FPIE model. There are over 2800 basic events in the Plant B unit 1 FPIE PRA that model over 1300 SSCs. The FPIE PRA is quantified using the EPRI PRAQuant software [22] to obtain the CDF and LERF cutsets. The truncations selected for these meet the PRA Standard for acceptable truncation. For CDF, a truncation of 1.0E-12 was used, while a truncation of 1.0E-13 was used for LERF. The importances are obtained from the cutsets directly (that is ACUBE is not used). A component is considered HSS if the sum of the CDF (LERF) F-Vs of the failure modes for the component is greater than 5.0E-03, or if any failure mode CDF (LERF) RAW is greater than 2.0. A common cause failure basic event is considered HSS if the CDF (LERF) RAW is greater than 20. Of the 1300 SSCs, approximately 380 are HSS. The 50.69 categorization for Plant B using the FPIE importances has not been completed at this time. Therefore, the importance data is taken from the latest PRA input to the Maintenance Rule risk ranking and the latest FPIE PRA quantification results.

3.3.4 Fire PRA High Safety Significant Evaluation

Plant B does not have a Fire PRA at this time. So, the HSS comparison is only with the FPIE PRA results.

3.3.5 Comparison of Seismic PRA results to other PRA results for High Safety Significant Evaluation Structures, Systems, and Components

Table 3-7 contains the SPRA fragility groups that are HSS along with the SSCs that make up those fragility groups. The table also shows whether the corresponding FPIE basic events are HSS. The mapping of the seismic fragility groups to the corresponding basic events in the FPIE generally fell into two groups. Many of the seismic fragility groups model SSCs that are explicitly modeled in the FPIE PRA. Other fragility groups model passive SSCs or SSCs that are not directly modeled in the FPIE but the SSC functions are explicitly modeled. The following sections provide more details of how the fragility groups are mapped to the basic events in the FPIE PRA.

3.3.5.1 Explicitly Modeled SSCs

Most of the SSCs modeled by the fragility groups are explicitly modeled in the FPIE PRA. Fragility groups that model mechanical SSCs such as pumps, fans, EDGs, MOVs, and AOVs typically are modeled in the FPIE PRA. For example, the SEIS-SI-P-1AB fragility group models seismic failure of the Low Head Safety Injections pumps 1-SI-P-1A and 1B. The FPIE PRA has basic events that model failure to start, failure to run, and out of service for test and maintenance, which are modeled by basic events 1SI-PSB--FS-1A, 1SI-PSB--FR-1A, and 1SI-PSB--TM-1A, respectively for the A pump. The FPIE PRA also includes common cause failure of these pumps using basic events 1SI-PSB22FS-1A+B and 1SI-PSB22FR-1A+B for failure to start and run, respectively. Therefore, the mapping of the SSCs modeled in the seismic fragility groups for these types of SSCs is relatively straightforward.

3.3.5.2 Implicitly Modeled SSCs

Some of the seismic fragility groups model seismic failure of SSCs that are not explicitly modeled in the FPIE PRA. Table 3-6 contains details of how these fragility groups are mapped to corresponding basic events in the FPIE.

Table 3-6
Plant B Passive or Implicitly Modeled SSCs

Scope	Description
Buildings	Building failures are not typically modeled in the FPIE PRA given their relatively low probability of random failure. The SPRA models building failures as failing the SSCs within the building. Therefore, in the comparison with the FPIE PRA, the seismic fragility groups that model building failures were mapped to basic events in the FPIE PRA that model failure of the SSCs within the building (typically the CCF of the SSCs). For example, seismic fragility group SEIS-BLDG-AFWPH models failure of the Auxiliary Feedwater Pumphouse. This fragility group is mapped to the CCF basic event 1FW-PSB33FR-ALL-AFW, which models common cause failure of the AFW pumps inside the pumphouse.
Electrical Panels such as Main Control Room (MCR) Panels, FLEX distribution panels, Vital Bus panels	Failures of MCR panels are typically not modeled in the FPIE PRA because of their relatively low probability of random failure. The SPRA models failure of the panels as failing Operator actions that rely on the panels for indications and control of mitigating functions. Therefore, the seismic failure of the MCR panels' fragility group was mapped to an HEP in the FPIE PRA. For the FLEX distribution panels where the FLEX 120VAC generators are connected to power the vital buses, the seismic fragility group for the FLEX panels is mapped to the vital bus basic events in the FPIE PRA.
Containment penetrations such as electrical and mechanical penetrations, fuel transfer tube, and containment hatches	Containment penetrations except for containment isolation valves, are typically not modeled in the FPIE given their relatively low probability of random failure. The SPRA models failure of the reactor containment building, which includes electrical and mechanical penetrations, the hatches, and fuel transfer tube. Failure of these result in direct LERF and therefore are mapped to the LERF-83 plant damage state in the FPIE, which models direct LERF caused by containment bypass.
Relays	The FPIE PRA does model some relays for impacts on the functions of actuation systems (for example, Safety Injection, Containment Depressurization, etc.). The SPRA models relay chatter which impacts specific SSC functions due to spurious actuations (for example, starting/stopping of pumps, opening/closing of valves). Therefore, the seismic fragility groups that model relay chatter are mapped to the basic events of the corresponding SSC functions that are impacted in the FPIE PRA. The relays could also have been categorized in this sensitivity based on whether the cabinet the relays are located in are HSS.
Piping	Piping failure is modeled in the FPIE PRA as part of the internal flooding portion of the model as well as failure of the RCS piping resulting in the various size LOCAs. The SPRA models piping failures of the RCS with seismic fragility groups for the various size LOCAs. Therefore, these groups are mapped to the corresponding LOCA basic events in the FPIE PRA.

3.3.5.3 Correlation of SSCs and Common Cause Failures

Nearly all of the seismic fragility groups in the SPRA model correlated failures of the SSCs in the group. That is, given the common design, location, installation, and orientation of the SSCs, it is expected that both train's SSCs will fail given the same ground motion during a seismic event. This is similar to the modeling of common cause failures (CCF) in the FPIE PRA, where for multiple SSCs that have the same design and are maintained using the same maintenance processes, there is a probability that both components (for example, pumps) in the trains could fail due to common cause. However, some SSCs have such low failure probabilities in the FPIE PRA that common cause failures are not typically modeled. Tanks, heat exchangers, and electrical SSCs such as switchgear and motor control centers, are some examples of SSCs that may not have common cause failures modeled. In this sensitivity, the fragility groups that model correlated failures but do not have CCF basic events modeled in the FPIE were flagged in Table 3-7 with a check mark in the Correlation Review column. The six fragility groups identified here are all electrical SSCs such as switchgear, breakers, and electrical panels.

3.3.6 Analysis and Conclusions

As shown in Table 3-7, all SSCs modeled by the seismic fragility groups that are HSS in the SPRA are also HSS in the FPIE PRA. The 38 HSS seismic fragility groups in the SPRA, which model over 200 SSCs, are also HSS in the FPIE PRA. And the two non-seismic failure basic events that are HSS in the SPRA are also HSS in the FPIE PRA.

This sensitivity shows that all of the SSCs that are HSS in the SPRA are also HSS in the FPIE PRA.

Table 3-7
Sensitivity Study Results for Plant B

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
Containment Integrity	SEIS-BLDG-RC	Reactor Containment	1-BLD-BLD-RC-BLDG	1-BLD-BLD-RC-BLDG, 1-CE-EH-1, 1-CE-PH-1, 1-FH-TB-1, 1-PE-EP*, 1-PEN-PN*	Structural	✓	✓	N/A	✓			Containment is not modeled in the FPIE PRA; but failure of the containment building would result in failure of RCS piping, etc., which are considered HSS;
	SEIS-CV-TV-150ABCD	Containment Vacuum CIVs	1-CV-TV-150A/B/C/D	1-CV-TV-150A/B/C/D 1-CV-SOV-150A/B/C/D	Functional	✓	✓	N/A				
	SEIS-RS-P-1AB-RLY	Inside Recirc Spray Pump Relays	1-RS-3-1RSIA01-RELAY	1-RS-3-1RSIA01-RELAY 1-RS-3-1RSIB01-RELAY	Functional	✓	✓	N/A	✓			Relay chatter - Not explicitly modeled in the FPIE PRA; Chatter fails the function of the RS pumps.
	SEIS-RS-P-2AB-RLYSS	Outside Recirc Spray Pump Relays	1-RS-3A-1RSOA01-RELAY	1-RS-3A-1RSOA01-RELAY 1-RS-3A-1RSOB01-RELAY	Functional	✓	✓	N/A	✓			Relay chatter - Not explicitly modeled in the FPIE PRA; Chatter fails the function of the RS pumps.

Table 3 7 (continued)
Sensitivity Study Results for Plant B

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
Core Cooling and Inventory Control	SEIS-BLDG-AB-LOWER	Auxiliary Building Lower Floors	Building	1-BLD-BLD-CT-BLDG	Shear Wall Failure	✓	✓	N/A	✓			Failure of the Aux Bldg is not modeled in the FPIE PRA; Failure of the lower floors of the Auxiliary building is assumed to result in direct core damage.
	SEIS-CC-E-1AB	Component Cooling Heat Exchangers	1-CC-E-1A	1-CC-E-1A 1-CC-E-1B	Anchorage	✓	✓	N/A	✓			Failure of the CCW HXs results in failure of the SW piping to the HXs, which causes a flood in the Auxiliary building.
	SEIS-CH-P-1ABC-RLY	Charging / High Head SI Pump Relays	1-CH-86-1CHCC09-RELAY	Various relays in CH pump circuits	Functional	✓	✓	N/A	✓			Relay chatter - Not explicitly modeled in the FPIE PRA; Chatter fails the function of the CH pumps.
	SEIS-MOV-QSPH-RSHX	SW Supply Header to Recirc Spray HX Isol Valves	1-SW-MOV-101A/B/C/D	1-SW-MOV-101A/B/C/D 1-SW-MOV-103A/B/C/D 1-SW-MOV-104A/B/C/D 1-SW-MOV-105A/B/C/D	Functional	✓	✓	N/A				

Table 3 7 (continued)
Sensitivity Study Results for Plant B

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
Core Cooling and Inventory Control	SEIS-QS-TK-1	Refueling Water Storage Tank	1-QS-TK-1	1-QS-TK-1	Tank Overturning	✓	✓	N/A				
	SEIS-SI-P-1AB-RLY	Low Head SI Pump Lockout Relays	1-SI-86-1SILA01-RELAY	1-SI-86-1SILA01-RELAY 1-SI-86-1SILB01-RELAY	Functional	✓	✓	N/A	✓			Relay chatter - Not explicitly modeled in the FPIE PRA; Chatter fails the function of the SI pumps.
Criticality	SEIS-RC-CNTRL-RODS	Control Rods/Rx Internals	1-RCS-CRDM-xxx	1-RC-FA* 1-RC-LRI* 1-RC-URI* 1-RCS-CRDM-xxx	Failure of Fuel Hold Down Spring	✓	✓	N/A				
Secondary Cooling	1FP-DDP--TM-2	Not a fragility group; This is the Diesel-Driven Fire Pump being OOS for Maintenance	Fire pump out of service for maintenance	1-FP-P-2	N/A	✓	✓	N/A				This SSC is included because it is a non-seismic failure that has a F-V greater than 0.005 in the SPRA. This basic event is also HSS in the FPIE PRA.

Table 3 7 (continued)
Sensitivity Study Results for Plant B

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
Secondary Cooling	SEIS-BLDG-AFW	Auxiliary Feedwater (AFW) Pumphouse	1-BLD-BLD-AFPH-BLDG	1-BLD-BLD-AFPH-BLDG 1-BLD-BLD-AFWT-BLDG	Structural	✓	✓	N/A	✓			AFW pumphouse and AFW pipe tunnel are not modeled in the FPIE PRA; but failure of the pumphouse or pipe tunnel would fail the AFW pumps.
	SEIS-BLDG-MSVH	Main Steam Valve House	1-BLD-BLD-MSVH-BLDG	1-BLD-BLD-MSVH-BLDG	Structural	✓	✓	N/A	✓			MSVH failure is not in the FPIE model; but failure of the MSVH would fail the SG PORVs, TDAFW pumps steam supply, and direct LERF; These failures would be HSS in the FPIE PRA.
	SEIS-CN-TK-1	Emergency Condensate Storage Tank	1-CN-TK-1	1-CN-TK-1	Anchorage	✓	✓	N/A				
	SEIS-FW-P-2	Turbine-Driven AFW Pump	1-FW-P-2	1-FW-GOV-2 1-FW-P-2 1-FW-TK-2 1-MS-TV-115	Functional	✓	✓	N/A				
	SEIS-FW-P-3AB-RLY	Motor-Driven AFW Pump Lockout Relays	1-FW-86-1FWEA01-RELAY	1-FW-86-1FWEA01-RELAY 1-FW-86-1FWEA01-RELAY	Functional	✓	✓	N/A	✓			Relay chatter - Not explicitly modeled in the FPIE PRA; Chatter fails the function of the AFW pumps.

Table 3 7 (continued)
Sensitivity Study Results for Plant B

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
RCS Integrity	SEIS-MLOCA	Medium (2"-6") RCS Piping	1-BLD-PIPE-RCS-MLOCA	1-BLD-PIPE-RCS-MLOCA	Generic	✓	✓	N/A				Failure of medium size RCS piping is modeled in the FPIE PRA and is HSS.
	SEIS-SLOCA	Small (1"-2") RCS Piping	1-BLD-PIPE-RCS-SLOCA	1-BLD-PIPE-RCS-SLOCA	Generic	✓	✓	N/A				Failure of small size RCS piping is modeled in the FPIE PRA and is HSS.
RCS Integrity	SEIS-SSLOCA	Small-Small (>1") RCS Piping	1-BLD-PIPE-RCS-SSLOCA	1-BLD-PIPE-RCS-SSLOCA Various CH, SI and SS AOVs on small lines	Generic	✓	✓	N/A				Failure of very small size RCS piping is modeled in the FPIE PRA and is HSS.
AC Power	0BDBEDG--FR-1A-FLEX	Not a fragility group; FLEX 120VAC Generator	FLEX 120VAC Generator Fails to Run	0-BDB-GEN-1A	N/A	✓	✓	N/A				This SSC is included because it is a non-seismic failure that has a F-V greater than 0.005 in the SPRA. This basic event is also HSS in the FPIE PRA.
	SEIS-BDB-DB-123	FLEX Distribution Panels	1-BDB-DB-1-PANEL	1-BDB-DB-1-PANEL 1-BDB-DB-2-PANEL 1-BDB-DB-3-PANEL	Seismic Interaction	✓	✓	N/A	✓			FLEX panels not modeled in the FPIE PRA; Mapped to vital bus basic event since the FLEX panels power the vital buses during a SBO.
	SEIS-EDG-HJ-NR-RLY	EDG Relays - Non-Recoverable	1-EG-3AX-1EGSH10-RELAY	Various relays in EDG circuits	Functional	✓	✓	N/A	✓			Relay chatter - Not explicitly modeled in the FPIE PRA; Chatter fails the function of the EDGs.

Table 3 7 (continued)
Sensitivity Study Results for Plant B

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
AC Power	SEIS-EDG-HJ-RLY	EDG Relays	1-EG-53-1EGSJ05-K2-RELAY	Various relays in EDG circuits	Functional	✓	✓	N/A	✓			Relay chatter - Not explicitly modeled in the FPIE PRA; Chatter fails the function of the EDGs.
	SEIS-EE-BKR-HJ2-RLY	EDG Output Breaker Lockout Relays	1-EG-86-1EGPH01-RELAY	Various relays in EDG output breaker circuits	Functional	✓	✓	N/A	✓			Relay chatter - Not explicitly modeled in the FPIE PRA; Chatter fails the function of the EDG output breakers.
	SEIS-EE-BKR-HJ8-RLY	480V Bus Supply Breaker Lockout Relays	1-EE-64-1EJSH01-RELAY	Various relays in 480V bus supply breaker circuits	Functional	✓	✓	N/A	✓			Relay chatter - Not explicitly modeled in the FPIE PRA; Chatter fails the power to the 480V emergency buses.
	SEIS-EG-B-1234	EDG Batteries	1-EG-B-03C	1-EG-B-01A 1-EG-B-03C	Combined Structural / Function	✓	✓	N/A	✓			EDG batteries are not explicitly modeled in FPIE, but would fail the EDG, which is HSS in the FPIE PRA.
	SEIS-EG-P-HAB-JAB	EDG Fuel Oil Transfer Pumps	1-EG-LS-1JA	EDG day tank level switches, fuel oil transfer pumps	Functional	✓	✓	N/A				
	SEIS-EP-CB-4ABCD	120 VAC Vital Bus Distribution Panels	1-EP-CB-04A	1-EP-CB-04A - 04D	Anchorage	✓	✓	N/A				
	SEIS-EP-SS-1H-1J	480V Emergency Buses	1-EE-SS-1H	1-EE-SS-1H/1J 1-EE-ST-1H/1J	Functional	✓	✓	N/A				

Table 3 7 (continued)
Sensitivity Study Results for Plant B

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
AC Power	SEIS-EP-SW-1H-1J	4160V Emergency Buses	1-EE-SW-1J	1-EE-SW-1H/1J Associated breakers in switchgear	Combined Structural / Function	✓	✓	N/A				
AC Power	SEIS-LOOP	Seismic-Induced Loss of Offsite Power	LOOP	LOOP	Generic	✓	✓	N/A				
AC Power	SEIS-VB-INV-1234	120VAC Vital Bus Inverters	1-EE-TRAN-79A	1-VB-INV-01 - 04 Regulating transformers Hand switches	Functional	✓	✓	N/A				
DC Power	SEIS-BY-B-1-24	Station Batteries 1-II/1-IV	1-BY-B-1-IV	1-BY-B-1-II 1-BY-B-1-IV	Structural failure of rack	✓	✓	N/A				
DC Power	SEIS-EP-CB-12ABCD	DC Distribution Panels	1-EP-CB-12A	1-EP-CB-12A - 12D	Functional	✓	✓	N/A				
Control Room Panels	SEIS-EI-CB-MCR-PNL	Main Control Room Panels	1-EI-CB-03	1-EI-CB-01 - 07 1-EI-CB-21 1-EP-CB-80C & D	Functional	✓	✓	N/A	✓			Loss of function of MCR panels are not modeled in FPIE PRA, but failure would impact HEPs, many of which are HSS in the FPIE PRA.

Table 3 7 (continued)
Sensitivity Study Results for Plant B

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive Cat.		
Service Water	SEIS-BLDG-SWVH	Service Water Valve House	1-BLD-BLD-SWVH-BLDG	1-BLD-BLD-SWVH-BLDG	Structural	✓	✓	N/A	✓			SWVH is not modeled in the FPIE PRA; but failure of the Valve house would result in failure of SW system, which is HSS in the FPIE PRA.
	SEIS-SW-P-1AB-RLY	Service Water Pump Lockout Relays	1-SW-86-1SWEA01-RELAY	1-SW-86-1SWEA01-RELAY 1-SW-86-1SWEB01-RELAY	Functional	✓	✓	N/A	✓			Relay chatter - Not explicitly modeled in the FPIE PRA; Chatter fails the function of the Sw pumps.
					Totals	**38**	**38**		**19**	**0**	**0**	
							38 Seismic Fragility Groups classified as HSS via overlapping 50.69 criteria					

3.4 Plant C Trial Categorization Evaluation

This section documents the sensitivity performed using the Plant C SPRA.

3.4.1 Introduction

3.4.1.1 Plant Overview

Plant C is a two-Unit Westinghouse PWR (four loop) site with each unit operating at 1150 MWe. They entered commercial operation in 1987 (Unit 1) and 1989 (Unit 2). Emergency core cooling is accomplished via two centrifugal charging pumps, two safety injection pumps, and two residual heat removal pumps. There are two Emergency Diesel Generators (EDGs) that power the two emergency buses if offsite power is lost. There are four 120 VAC vital buses (two per emergency bus) powered by either the batteries/inverters or directly from the emergency buses through transformers. Three Auxiliary Feedwater (AFW) pumps (two motor-driven and one turbine-driven) provide steam generator cooling if the main Feedwater pumps are unavailable. The ultimate heat sink is four mechanical draft cooling towers, where four pumps provide nuclear service cooling water (NSCW) to safety and auxiliary non-safety components. These NSCW pumps also remove the decay heat from the reactor when the plant is offline.

3.4.1.2 PRA Models

The FPIE PRA, Fire PRA, and SPRA models contain logic for quantifying CDF and LERF for each unit. For this sensitivity, the results are from the Unit 1 PRA models only.

The SPRA model was developed by modifying the Full Power Internal Events (FPIE) PRA model to incorporate specific aspects of seismic analysis that are different from the FPIE. The logic model appropriately includes seismic-caused initiating events and other failures including seismic-induced SSC failures, non-seismic-induced unreliability and unavailability failure modes (based on the FPIE model), and human errors. The SPRA has top gates which are quantified using the EPRI FRANX software [23]. The cutsets are then processed using ACUBE [21] to obtain the final seismic CDF and LERF as well as the importance data.

The FPIE PRA underwent a full scope peer review in 2009. The FPIE model has been revised to resolve all F&Os received during the peer review. These were reviewed as part of the development of the SPRA to verify no impact on the SPRA results. The Fire PRA underwent a full scope peer review in 2012. The fire model has been revised to resolve all F&Os received during the peer review.

The SPRA model used for this sensitivity was peer reviewed in November 2014. The SPRA model has been revised to address the F&Os.

3.4.2 Seismic PRA High Safety Significant Evaluation

This section contains a summary of the SPRA Model and identification of HSS SSCs from the SPRA.

3.4.2.1 Description of Seismic PRA Model

The Plant C SPRA model was developed by starting with the internal events at power PRA and adapting the model in accordance with guidance in the SPID [11] and PRA Standard [24], including adding seismic fragility-related basic events to the appropriate portions of the internal events PRA, eliminating some parts of the internal events model that do not apply or that were screened-out, and adjusting the internal events PRA model human reliability analysis to account for response during and following a seismic event. The model is developed using the EPRI CAFTA software suite [25]. This model does not credit non-permanently installed FLEX equipment, but does include low leakage reactor coolant pump (RCP) seals. Both random and seismic-induced failures of modeled SSCs are included.

In the SPRA model, fully correlated components were assigned to correlated component groups so that all components in the group are modeled with the same basic event, such that if one fails, all fail at the seismic magnitude for each hazard bin. The model assumes fully correlated response of same or very similar equipment in the same structure, elevation, and orientation. Correlated component groups were developed for all redundant components in the model that met these correlation criteria. For correlated groups where there was a significant difference in fragilities, then the higher capacity was used to assign a higher correlated fragility to both components, but the lower capacity component was also assigned a unique seismic capacity that only failed that component. Thus, the lower capacity component could fail by itself, but was guaranteed to fail if the higher capacity component was failed.

The seismic hazard was modeled using 14 discrete hazard intervals (or bins) based on increasing peak ground acceleration. Each bin is treated as a seismic initiator and the SCDF (and SLERF) results are summed over all the bins to obtain the total SCDF (and SLERF). Bin-specific SSC fragilities are used in the accident sequences for each bin.

For the SPRA, the following approach was used to quantify the seismic plant response model and determine seismic CDF and LERF. The EPRI FRANX software [23] was used to discretize the seismic hazard into the 14 seismic initiators, and quantify to produce cutsets and estimate the mean SCDF. The EPRI ACUBE [21] code was then used to calculate the exact probability on the entire set of SCDF/SLERF cutsets. This does not require the typical min cut upper bound approximation which can be excessively conservative when using high-probability events. Additional details can be found in the following sections, along with descriptions of sensitivity studies, uncertainty estimations and a more complete description on the insights from top contributors to SCDF/SLERF.

The Plant C SPRA approach to determining the importance measures is to calculate the F-V and RAW measures for a component for each seismic acceleration interval, and then develop overall seismic importance values (for F-V and RAW) using the following weighted process to combine the importance values over all seismic acceleration intervals. For a component/basic event, the F-V and RAW are calculated by ACUBE 2.0 for each of the 14 seismic acceleration intervals, resulting in 14 F-V and RAW importance values by interval. The interval F-V values are weighted based on the seismic acceleration interval CDF divided by the total seismic CDF, and summed together for each seismically failed fragility group to obtain the total F-V from the seismic failure. The RAW values are weighted and summed similarly to the F-V importance values. A similar process and weighting is used for LERF importance measures.

3.4.2.2 Identification of HSS SSCs from the SPRA

Components are considered High Safety Significant (HSS) if the group F-V is greater than 0.005 or if the group RAW is greater than 2.0 for CDF or LERF. Therefore, if a group has a SCDF (SLERF) F-V or RAW that meet these HSS thresholds, then the SSCs in the group are considered HSS.

The SPRA also models non-seismic failures (for example, failure to start, run) of SSCs that can impact mitigating functions.

The results from the SCDF and SLERF quantification and importance show that 36 fragility groups and two non-seismic failure basic events are considered HSS. Table 3-9 lists these fragility groups and basic events. The SSCs that are modeled by these fragility groups are also listed in the table.

3.4.3 Full Power Internal Events PRA High Safety Significant Evaluation

In the FPIE model, failure of SSCs are modeled for the different failure modes, such as pumps, fans, compressors, etc. failing to start, failing to run, failure to load, and out of service for test or maintenance. Additional failure modes may be modeled depending on the component. Common cause failures of the components are also included in the FPIE to account for possible design, maintenance and latent defects that could be common between similar components within the trains.

These SSC failures are modeled as “basic events” in the FPIE model. There are over 6000 basic events in the Unit 1 FPIE PRA that model over 1500 SSCs. The FPIE PRA is quantified using the EPRI PRAQuant software [22] to obtain the CDF and LERF cutsets. The truncations selected for these meet the PRA Standard for acceptable truncation. For CDF, a truncation of 1.0E-13 was used, while a truncation of 1.0E-15 was used for LERF. The importances are obtained from the cutsets directly (that is ACUBE is not used). A component is considered HSS if the sum of the CDF (LERF) F-Vs of the failure modes for the component is greater than 5.0E-03, or if any failure mode CDF (LERF) RAW is greater than 2.0.

3.4.4 Fire PRA High Safety Significant Evaluation

The Fire PRA uses the FPIE model accident sequences. Fire scenarios were postulated and the equipment and cable failures were propagated through the appropriate accident sequences during the quantification process. The FRANX software [23] and the FTREX [26] quantification engine were used to quantify each fire scenario. FRANX software quantifies a CCDP or a conditional large, early release probability (CLERP) using the Minimal Cutset Upper Bound calculation. The CCDP or CLERP is combined with the product of the fire scenario ignition frequency, NSP and severity factor (SF) to calculate a CDF or LERF.

The FRANX function to create a "one top" model was used. The one top model was quantified using the EPRI PRAQUANT software and the FTREX quantification engine to obtain a single cutset file which is used for fire risk results, importance measures, and uncertainty evaluation. The FRANX software interacts with EPRI's CAFTA software suite, which is utilized by the FPIE PRA model. FTREX is the quantification software used for Units 1 and 2 Fire PRA, consistent with the FPIE model. The truncation values for CDF and LERF were 1E-12 and 1E-12 respectively. Similar to internal events the component importances were evaluated using the methodology described in Section 3.2.3. The determination of HSS or LSS from the Fire PRA can be found in Table 3-1 of this report.

3.4.5 Comparison of Seismic PRA Results to Other PRA Results for High Safety Significant Structures, Systems, and Components

Table 3-9 contains the SPRA fragility groups that are HSS along with the SSCs that make up those fragility groups. The table also shows whether the corresponding FPIE and Fire basic events are HSS. The mapping of the seismic fragility groups to the corresponding basic events in the FPIE and Fire generally fell into two groups. Many of the seismic fragility groups model SSCs that are explicitly modeled in the FPIE and Fire PRA. Whereas, other fragility groups model passive SSCs or SSCs that are not directly modeled in the FPIE but the SSC functions are explicitly modeled. The following sections provide more details of how the fragility groups are mapped to the basic events in the FPIE PRA.

3.4.5.1 Explicitly Modeled SSCs

Most of the SSCs modeled by the fragility groups are explicitly modeled in the FPIE PRA and Fire PRA. Fragility groups that model mechanical SSCs such as pumps, fans, EDGs, MOVs, and AOVs typically are modeled in the FPIE PRA. So the mapping of the SSCs modeled in the seismic fragility groups for these types of SSCs is relatively straightforward.

3.4.5.2 Implicitly Modeled SSCs

Some of the seismic fragility groups model seismic failure of SSCs that are not explicitly modeled in the FPIE PRA. Table 3-8 contains details of how these fragility groups are mapped to corresponding basic events in the FPIE.

Table 3-8
Plant C Passive or Implicitly Modeled SSCs

Scope	Description
Buildings	Building failures are not typically modeled in the FPIE PRA given their relatively low probability of random failure. The SPRA models building failures as failing key SSCs within the building or as leading directly to core melt or large early release. Therefore, in the comparison with the FPIE PRA or Fire PRA, the seismic fragility groups that model building failures were mapped to basic events in the FPIE and Fire PRA that model failure of the SSCs within the building. See Section 3.6.6 for additional discussion of categorization of Civil Structures.
Relays	The SPRA models relay chatter which impacts specific SSC functions due to spurious actuations (for example, starting/stopping of pumps, opening/closing of valves). Therefore, the seismic fragility groups that model relay chatter are mapped to the basic events of the corresponding SSC functions that are impacted in the FPIE and Fire PRA.
Piping	Piping failure is modeled in the FPIE as part of the internal flooding portion of the model as well as failure of the RCS piping resulting in the various size LOCAs. The SPRA models piping failures of the RCS with seismic fragility groups for the various size LOCAs. Therefore, these groups are mapped to the corresponding LOCA basic events in the FPIE PRA.

3.4.6 Analysis and Conclusions

As shown in Table 3-9, most SSCs modeled by the seismic fragility groups that are HSS in the SPRA are also HSS in the FPIE and/or the Fire PRA. The 28 seismic fragility groups in the SPRA model over 63 SSCs, of which 23 are also HSS in the FPIE and/or Fire PRA. Eight have non-seismic failure mechanisms (marked as Random Failure in Table 3-9) that are HSS in the SPRA and are also HSS in the FPIE and/or the Fire PRA.

There are five seismic fragility groups that are HSS in the SPRA but not for the other considered risk categories (FPIE PRA, Fire PRA, Implicit Modeling, Passive Categorization). These five fragility groups represent correlated seismic failures or seismic induced internal flooding failures. These insights contributed to the creation of the approach described in Section 2.3.1 to account for the possibility of seismically correlated failures or seismic interaction related failures.

Table 3-9
Sensitivity Study Results for Plant C

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive		
	S_1ACBS-120PN-CB180	120 VAC Panel CB 180	120 VAC Vital Panel	1ACBSQ3VI1 - 1ACBSQ3VI4	Functional (After)	✓	✓	✓				
	S_1ACIV-120-CB180	AC Inverter CB180	Vital AC Inverter	1ACIVY3IA1 - 11807Y3ID4	Functional (After)	✓	✓	✓				
	S_1ACSD-SEQ	SFTY Features Sequencer	SF Sequencer Board	1ACSDU3001 - 1ACSDU3002	Functional (After)	✓	✓	✓				
	S_1DCBC-CB180	Battery Charger CB180	Battery Charger	1AFPMP4002 - 1AFPMP4003	Functional (After)	✓	✓					
Emergency Power AC/DC	S_1DCBS-MCC-AB	125 VDC MCC 1AD1M and 1BD1M	125 VDC MCC	1AFPMP4001	Functional (After)	✓	✓					
	S_1DCBS-MCC-ALL	All 125 VDC MCC	125 VDC MCC	RL1AFW1512 9	Functional (After)	✓	✓					
	S_1DCBS-PN-CB180-1E	125 VDC 1E Distr. Panel - CB180	125 VDC Distr. Panel	1CCTKT4001 - 1CCTKT4002	Functional (After)	✓	✓	✓				
	S_1DCBS-SGR-CB180	125 VDC Switchgear CB180	125 VDC Switchgear	1DCBCB3CAA - 1DCBCB3CDB	Functional (After)	✓	✓	✓				
	S_1DCBY-CB180	125 VDC Battery CB180	125 VDC Battery	1DCBCS3DCA - 1DCBCS3DCB	Functional (After)	✓	✓	✓				

Table 3-9 (continued)
Sensitivity Study Results for Plant C

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive		
	N/A[5]	N/A[5]	Incoming 1AA02 FDR BKR	1AA0205	Random Failure	✓	✓	✓				
	N/A[5]	N/A[5]	Incoming 1BA03 FDR BKR	1BA0301	Random Failure	✓	✓	✓				
Emergency Power AC/DC	N/A[5]	N/A[5]	125 VDC Battery 1DD1B	1DCBCS3DD1	Random Failure	✓	✓					
	N/A[5]	N/A[5]	Reactor Trip Breaker 'A', Breaker 'B'	1RTA, 1RTB	Random Failure	✓		✓				
	N/A[5]	N/A[5]	Breaker to A Train NSCW Fan #1, #2, #3, #4	11ACDCS3AB B	Random Failure	✓		✓				

[5] This entry is not a Seismic Fragility Group. It is random failure of the SSC to function (i.e. start, run, or other PRA functional failure)

Table 3-9 (continued)
Sensitivity Study Results for Plant C

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive		
Auxiliary Feedwater	S_1AFPM-MDP	Both AFW MDP	AFW Motor Driven Pump	1DCBSS3DCA - 1DCBSS3DCC	Functional (After)	✓	✓	✓				
	S_1AFPM-TDP	AFW TDP	AFW TURB Driven Pump	1DCBSQ3DA1 - 1DCBSQ3DD1	Functional (After)	✓		✓				
	S_1AFW-AOV-RLY	Relay for AFW Pump Turb Trip & Throttle VLV	Relay for AFW Pump Turb Trip & Throttle VLV	1DCBSS3DSA - 1DCBSS3DSD	Functional (During)	✓	✓	✓	✓			
	N/A[5]	N/A[5]	AFW, TDAFW Pump, Disch, Isolation	11302U4015	Random Failure	✓	✓					
Component Cooling Water	S_1CCTK-4	CCW Surge Tank	CCW Surge Tank	1DCBYB3BYA - 1DCBYB3BYA	Anchorage	✓					✓	Correlated Failure drives the SSC to HSS
Emergency Diesel Generator	S_1DG	Diesel Generator	Diesel Generator	1DGG4001 - 1DGG4002	Functional (After)	✓	✓	✓				
	S_1DGDM-VENT	DG Vent Damper for Fans 1-4	DG AIR Supply Damper for Fans	1DGDM12050 - 1DGDM12054	Functional (After)	✓		✓				
	S_1DGFN-FAN	DG BLDG ESF Supply Fan	DG BLDG ESF Supply Fan	1DGFNB7002000 - 1DGFNB7004000	Functional (After)	✓					✓	Correlated Failure drives the SSC to HSS

Table 3-9 (continued)
Sensitivity Study Results for Plant C

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive		
Contain-ment Heat Removal	S_1FC-ACU-FLD	Anchorage Failure of ACU with NSCW FLD	CTB AUX Cooling Unit	1ACUA7002000	Anchorage	✓					✓	Flooding causes LUHS drives the SSC to HSS. Considered as a flooding interaction in the Correlation Review.
Nuclear Service Cooling Water	S_1SWFN-NSCW-FANS	NSCW Tower Fans	fan-NUC SERV Cool Tower	1NSCWW4001F01 - 1NSCW4002F04	Anchorage	✓	✓					
Auxiliary Component Cooling Water	S_1XCTK-4	ACCW Surge Tank	ACCW Surge Tank	1XCTKT4001	Anchorage	✓					✓	Correlated Failure drives the SSC to HSS
Essential Chilled Water	S_CB-CHLR-NSCW-FLOOD	Seismic Failure of CB ESF Chillers Cause NSCW Flood on CB 260	CB ESF Chiller	1CHLRC7001000 - 1CHLRC7002000	Anchorage	✓					✓	Correlated anchorage failure of two trains of ESF Chillers leads to flooding such that LUHS drives the SSC to HSS
Residual Heat Removal	N/A[6]	N/A[6]	RCS to RHR Pump B Suction MOV	1HV8702A	Random Failure	✓		✓				
Pressurizer	1RCPOPV0455A-U, 456-U	Pressurizer PORVs	Pressurizer PORVs	1PORV0455,1PORV456	Random Failure	✓		✓				

[6] This entry is not a Seismic Fragility Group. It is random failure of the SSC to function (i.e. start, run, or other PRA functional failure)

Table 3-9 (continued)
Sensitivity Study Results for Plant C

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire PRA	Implicit Modeling	Passive		
Structures			Category 1 Civil Structures	Containment, Auxiliary Building, Control Building, Emergency Diesel Generator Building, AFW Pump House, Nuclear Safety Cooling Water Towers	Structural	✓						HSS due to RAW criteria in the SPRA. See Section 3.6.6 for additional discussion of categorization of Civil Structures.
					Totals	**29**	**17**	**17**	**1**	**0**	**5**	
							23 Seismic Fragility Groups classified as HSS via overlapping 50.69 criteria					

3.5 Plant D Trial Categorization Evaluation

This section documents the sensitivity performed using the Plant D SPRA.

3.5.1 Introduction

Plant D is a two unit, Westinghouse PWR (four loop) site with sub-atmospheric (ice condenser) containments with each unit operating at approximately 1,150 MWe. Emergency core cooling is accomplished for each unit by two High Head Safety Injection pumps, two Intermediate Head Safety Injection pumps and two Low Head Safety Injection pumps. The high head pumps also provide normal Charging and RCP seal injection during non-accident conditions. There are four Emergency Diesel Generators (EDGs) that power the emergency buses if offsite power is lost. There are four 120 VAC vital buses for each unit powered by either the batteries/inverters or directly from the emergency buses through transformers. Three Auxiliary Feedwater (AFW) pumps (two motor-driven and one turbine-driven) provide steam generator cooling if the main Feedwater pumps are unavailable. The ultimate heat sink is from the river, where eight pumps provide service water flow to both units via two headers.

The plant's FPIE PRA and SPRA models contain logic for quantifying CDF and LERF for each unit. For this sensitivity, the results are from the Unit 1 PRA models only. The results for Unit 2 would be similar given that both units are nearly identical. This plant does not have a fire PRA but instead utilized the Appendix R Safe Shutdown (SSD) list of SSCs to classify components as HSS.

3.5.2 Seismic PRA High Safety Significant Evaluation

Seismic failures of SSCs are modeled using fragility groups, which represent failure of groups of SSCs, typically both (multiple) trains if the SSCs are assumed to be correlated. Most of the SSCs are assumed to be correlated given similar design, location, and configuration.

The seismic hazard curve is divided into eight intervals and is modeled by eight seismic initiating basic events %G01 through %G08. Each fragility group is therefore modeled by eight seismic failure basic events representing the probability of failure for each of the eight seismic intervals of the seismic hazard curve.

The SPRA model is quantified using EPRI FRANX [23] to generate the cutsets, which are then processed with ACUBE [21] that uses the Binary Decision Diagram (BDD) to obtain a more accurate solution that reduces the overestimation that occurs when basic event probabilities are high.

ACUBE generates Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) importance data for each basic event in the cutsets. Both F-V and RAW are used to obtain the importance of the fragility groups.

Fragility groups are considered High Safety Significant (HSS) if the group F-V is greater than 0.005 or if the group RAW is greater than 2.0 for CDF or LERF. Thus, if a group has a SCDF (SLERF) F-V or RAW that meet these HSS thresholds, then the SSCs in the group are considered HSS. Table 3-11 lists these fragility groups and basic events. The SSCs that are modeled by these fragility groups are also listed in the table.

3.5.3 Full Power Internal Events PRA High Safety Significant Evaluation

In the FPIE model, failure of SSCs are modeled for the different failure modes, such as pumps, fans, compressors, etc. failing to start, failing to run, failure to load, and out of service for test or maintenance. Additional failure modes may be modeled depending on the component. Common cause failures of the components are also included in the FPIE to account for possible design, maintenance and latent defects that could be common between similar components within the trains.

The FPIE PRA is quantified using the EPRI PRAQuant [22] software to obtain the CDF and LERF cutsets. The importances are obtained from the cutsets directly (that is ACUBE is not used). A component is considered HSS if the CDF (LERF) F-V of the failure mode for the component is greater than 5.0E-03, or if the CDF (LERF) RAW is greater than 2.0. A common cause failure basic event is considered HSS if the CDF (LERF) RAW is greater than 20.

3.5.4 Fire PRA High Safety Significant Evaluation

Plant D does not have a Fire PRA at this time. So, the HSS comparison is only with the FPIE PRA results.

3.5.5 Comparison of SPRA results to the FPIE PRA results for HSS SSCs

The mapping of the seismic fragility groups to the corresponding basic events in the FPIE generally fell into two groups. Many of the seismic fragility groups model SSCs that are explicitly modeled in the FPIE PRA. Whereas, other fragility groups model passive SSCs or SSCs that are not directly modeled in the FPIE but the SSC functions are explicitly modeled. The following sections provide more details of how the fragility groups are mapped to the basic events in the FPIE PRA.

3.5.5.1 Explicitly Modeled SSCs

Most of the SSCs modeled by the fragility groups are explicitly modeled in the FPIE PRA. Fragility groups that model mechanical SSCs such as pumps, fans, EDGs, MOVs, and AOVs typically are modeled in the FPIE PRA. Therefore, the mapping of the SSCs modeled in the seismic fragility groups for these types of SSCs is relatively straightforward.

3.5.5.2 Implicitly Modeled SSCs

Some of the seismic fragility groups model seismic failure of SSCs that are not explicitly modeled in the FPIE PRA. Table 3-10 contains details of how these fragility groups are mapped to corresponding basic events in the FPIE.

Table 3-10
Plant D Passive or Implicitly Modeled SSCs

Scope	Description
Electrical Panels such as Main Control Room (MCR) Panels	Failures of MCR panels are typically not modeled in the FPIE PRA because of their relatively low probability of random failure. The SPRA models failure of the panels as failing Operator actions that rely on the panels for indications and control of mitigating functions. Therefore, the seismic failure of the MCR panels' fragility group was mapped to an HEP in the FPIE PRA.
Containment penetrations such as electrical and mechanical penetrations, fuel transfer tube, and containment hatches	Containment penetrations except for containment isolation valves, are typically not modeled in the FPIE given their relatively low probability of random failure. The SPRA model includes failure of the containment penetrations by modeling a fragility group for containment penetrations, which includes electrical and mechanical penetrations, hatches, and the fuel transfer tube. Failure of these SSCs are modeled to result in direct LERF due to containment bypass.
Relays	The FPIE PRA does model some relays for impacts on the functions of actuation systems (for example, Safety Injection, Containment Depressurization). The SPRA models relay chatter which impacts specific SSC functions due to spurious actuations (for example, starting/stopping of pumps, opening/closing of valves). Therefore, the seismic fragility groups that model relay chatter are mapped to the basic events of the corresponding SSC functions that are impacted in the FPIE PRA.
Piping	Piping failure is modeled in the FPIE as part of the internal flooding portion of the model as well as failure of the RCS piping resulting in the various size LOCAs. The SPRA models piping failures of the RCS with seismic fragility groups for the various size LOCAs. Therefore, these groups are mapped to the corresponding LOCA basic events in the FPIE PRA.

3.5.5.3 Seismic Fragility Groups and Common Cause Failure

Nearly all of the seismic fragility groups in the SPRA model correlated failures of the SSCs they represent. That is, given the common design, location, installation, orientation, and function of the SSCs, it is expected that both train's SSCs will fail given the same ground motion during a seismic event. Therefore, the seismic fragility groups model common cause failure (CCF) of the SSCs during seismic events. In the mapping of the seismic fragility groups to the corresponding basic events in the FPIE PRA, the basic events that model failure of the individual SSC (that is not the CCF basic event) were selected if their F-V or RAW importances indicated they were HSS by themselves. However, if they were not HSS by themselves, then the fragility group was mapped to the CCF basic event.

3.5.6 Analysis and Conclusions

As shown in Table 3-11, half of the SSCs modeled by the seismic fragility groups that are HSS in the SPRA are also HSS in the FPIE and/or the Fire PRA. The results for Plant D include 21 individual breakers in low and medium voltage switchgear spread over two Seismic Fragility Groups (SEIS_0-24 and SEIS_0-25), which make up the majority of items not explicitly identified as HSS in the FPIE and/or the Fire PRA. There are also two exhaust fans that have seismically correlated failures and four traveling screens that have seismically correlated failures.

These insights contributed to the creation of the approach described in Section 2.3.1 to account for the possibility of seismically correlated failures or seismic interaction related failures.

Finally, the Plant D results identify four seismic fragility groups associated with FLEX that are HSS in the SPRA but not HSS in the FPIE or Fire PRAs. Section 3.6.4 describes FLEX considerations within the 50.69 categorization process.

Table 3-11
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
AC Power	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR CNTL & AUX Vent BD 1A1-A	1-BKR-212-A001/10B-A	Chatter	✓	✓					
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	480 V Shutdown BD 1A1A Nor Feed	1-BKR-212-A001/1B-A	Chatter	✓	✓					
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR FOR RX MOV BD 1A1-A (1-MCC-213-A1)	1-BKR-212-A001/8B-A	Chatter	✓	✓					
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm Supply from 6.9KV SD BD 1A-A	1-BKR-212-A002/1B-A	Chatter	✓	✓					
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for RX MOV BD 1A1-A (1-MCC-213-A1)	1-BKR-212-A002/8B-A	Chatter	✓	✓					
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for C&A Vent BD 1B1 (1-MCC-215-B1)	1-BKR-212-B001/10B-B	Chatter	✓	✓					
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	1-BKR-212-B001/1B-B, Norm Supply from 6.9KV SD BD 1B-B	1-BKR-212-B001/1B-B	Chatter	✓					✓	Chatter correlated failure

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
AC Power	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for RX MOV BD 1B1-B (1-MCC-213-B1)	1-BKR-212-B001/8B-B	Chatter	✓					✓	Chatter correlated failure
AC Power	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	1-BKR-212-B002/1B-B, Norm Supply from 6.9KV SD BD 1B-B	1-BKR-212-B002/1B-B	Chatter	✓					✓	Chatter correlated failure
AC Power	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for RX MOV BD 1B2-B (1-MCC-213-B2)	1-BKR-212-B002/8B-B	Chatter	✓					✓	Chatter correlated failure
AC Power	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm Supply from 6.9KV SD BD 2B-B	2-BKR-212-B002/1B-B	Chatter	✓	✓					
AC Power	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR CNTL & AUX Vent BD 2A1-A	2-BKR-212-A001/10B-A	Chatter	✓					✓	Chatter correlated failure
AC Power	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	480 V Shutdown BD 2-A1A NOR FEED;	2-BKR-212-A001/1B-A	Chatter	✓					✓	Chatter correlated failure
AC Power	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for RX MOV BD 2A1-A (2-MCC-213-A1)	2-BKR-212-A001/8B-A	Chatter	✓					✓	Chatter correlated failure

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
AC Power	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm Supply from 6.9KV SD BD 2A-A	2-BKR-212-A002/1B-A	Chatter	✓					✓	Chatter correlated failure
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for RX MOV BD 2A2-A (2-MCC-213-A2)	2-BKR-212-A002/8B-A	Chatter	✓					✓	Chatter correlated failure
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for C&A Vent BD 2B1-B (2-MCC-214-B1)	2-BKR-212-B001/10B-B	Chatter	✓					✓	Chatter correlated failure
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Nor Supply from 6.9KV SD BD 2B-B	2-BKR-212-B001/1B-B	Chatter	✓					✓	Chatter correlated failure
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for RX MOV BD 2B1-B (2-MCC-213-B1)	2-BKR-212-B001/8B-B	Chatter	✓					✓	Chatter correlated failure
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for RX MOV BD 2B2-B (2-MCC-213-B2)	2-BKR-212-B002/8B-B	Chatter	✓					✓	Chatter correlated failure
	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for VIT BATT CHGR III (0-CHGR-236-3)	0-BKR-236-0003-A	Chatter	✓	✓					

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
AC Power	SEIS_0-24	Breaker Chatter, Low Voltage Switchgear	Norm FDR for VITAL BATTCHGR IV (0-CHGR-236-4)	0-BKR-236-0004A-B	Chatter	✓	✓					
AC Power	SEIS_0-25	Breaker Chatter, medium voltage switchgear	Norm Supply from 6.9KV COMMON SWG C	1-BKR-211-1716/16-A	Chatter	✓	✓					
AC Power	SEIS_0-25	Breaker Chatter, medium voltage switchgear	Norm Supply from 6.9KV COMMON SWG D	1-BKR-211-1728/16-B	Chatter	✓	✓					
AC Power	SEIS_0-25	Breaker Chatter, medium voltage switchgear	1-BKR-212-B001-B, 480V Shutdown XFMR 1B1 (1-OXF-212-B1)	1-BKR-212-B001-B	Chatter	✓					✓	Chatter correlated failure
AC Power	SEIS_0-25	Breaker Chatter, medium voltage switchgear	1-BKR-212-B002-B, 480V Shutdown XFMR 1B2 (1-OXF-212-B2)	1-BKR-212-B002-B	Chatter	✓					✓	Chatter correlated failure
AC Power	SEIS_0-25	Breaker Chatter, medium voltage switchgear	6.9kV SDBD Breaker 1816, 1828	1-BKR-211-1816/16-A, 1-BKR-211-1828/16-B	Chatter	✓					✓	Chatter correlated failure

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
AC Power	SEIS_0-25	Breaker Chatter, medium voltage switchgear	ALT Supply from 6.9KV Common SWG C, D	1-BKR-211-1934/1-B, 1-BKR-211-1932/1-A	Chatter	✓					✓	Chatter correlated failure
	SEIS_0-25	Breaker Chatter, medium voltage switchgear	1-BKR-212-A001-A, 480V Shutdown XFMR 1A1 (1-OXF-212-A1), 1B2 (1-OXF-212-A2)	1-BKR-212-A001-A, 1-BKR-212-A002-A	Chatter	✓					✓	Chatter correlated failure
	SEIS_0-25	Breaker Chatter, medium voltage switchgear	ALT Supply from 6.9KV Common SWG C, D	2-BKR-211-1938/1-B, 2-BKR-211-1936/1-A	Chatter	✓					✓	Chatter correlated failure
	SEIS_0-25	Breaker Chatter, medium voltage switchgear	6.9kV Supply to Transformer 2A1A (2-BKR-212-A1-A)	2-BKR-212-A001/A	Chatter	✓					✓	Chatter correlated failure
	SEIS_0-25	Breaker Chatter, medium voltage switchgear	480V SHDN Trans 2A2-A (2-OXF-212-A2-A), 2B1-B (2-OXF-212-B1-B), 2B2-B (2-OXF-212-B2-B)	2-BKR-212-A002-A, 2-BKR-212-B001-B, 2-BKR-212-B002-B	Chatter	✓					✓	Chatter correlated failure

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
AC Power	SEIS_3-1	AUX 480V Inverter	120V AC Vital Inverter 1-I, 1-II, 1-III, 1-IV	1-INV-235-0001-D, 1-INV-235-0002-E, 1-INV-235-0003-F, 1-INV-235-0004-G	Anchorage	✓		✓				
	SEIS_3-1	AUX 480V Inverter	120V AC Vital Inverter 2-I, 2-II, 2-III, 2-IV	2-INV-235-0001-D, 2-INV-235-0002-E, 2-INV-235-0003-F, 2-INV-235-0004-G	Anchorage	✓		✓				
	N/A[7]	N/A[7]	6.9KV Normal Supply Breaker for Shutdown Board	1-BKR-211-1716/16-A, 1-BKR-211-1728/16-B	Random	✓	✓					

[7] This entry is not a Seismic Fragility Group. It is random failure of the SSC to function (i.e. start, run, or other PRA functional failure)

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
DC Power	SEIS_2-1	125VDC Vital Battery	125VDC Vital Battery I, II, III, IV	0-BAT-236-0001-D, 0-BAT-236-0002-E, 0-BAT-236-0003-F, 0-BAT-236-0004-G	Functional	✓		✓				
DC Power	SEIS_3-3	125V Vital Battery Charger	125V Vital Battery Charger I, II, III, IV	0-CHGR-236-0001-D, 0-CHGR-236-0002-E, 0-CHGR-236-0003-F, 0-CHGR-236-0004-G	Functional	✓	✓	✓				

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
Emergency Diesel Generator	N/A[8]	N/A[8]	EDG random failure to start and/or run	1-GEN-082-0001A-A, 1-GEN-082-0001B-B	Random	✓						EDGs would be treated as HSS in the 50.69 defense-in-depth review. See Section 3.6.5 for additional discussion of defense-in-depth.
Emergency Diesel Generator	N/A[8]	N/A[8]	Diesel Generator Exhaust Fan	1-FAN-030-0447-A, 1-FAN-030-0449-B, 1-FAN-030-0451-A, 1-FAN-030-0453-B, 1-FAN-030-0459-A	Random	✓	✓					
Component Cooling Water	SEIS_19-10	CCS Surge Tank A	CCS Surge Tank A	1-TANK-070-0001	Anchorage	✓	✓					

[8] This entry is not a Seismic Fragility Group. It is random failure of the SSC to function (i.e. start, run, or other PRA functional failure)

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
Onsite Water Sources	SEIS_19-14	Refueling Water Storage Tank	Refueling Water Storage Tank	1-TANK-063-0046	Anchorage	✓	✓					
	SEIS_19-9	SIS Boron Injection Tank	SIS Boron Injection Tank	1-TANK-063-0036	Anchorage	✓				✓		
Component Cooling Water	SEIS_20-1	CCS Heat Exchanger	CCS Heat Exchanger A, B	1-HTX-070-0185, 1-HTX-070-0186	Anchorage	✓	✓					
AC Relay Panel	SEIS_5-1	6.9 Logic Relay Panel	6900V STDN LOG REL PNL 1A-A, 1B-B	1-PNL-211-A-A, 1-PNL-211-B-B	Functional	✓	✓					

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
Auxiliary Feedwater	SEIS_11-6	Aux Feedwater Pump	Aux Feedwater Pump 1A-A, 1B-B	1-PMP-003-0118-A, 1-PMP-003-0128-B	Anchorage	✓	✓					
	SEIS_17-4	AFW Exhaust Fan	TD AFW Pump Room 125V DC EMERG EXH FAN	1-FAN-030-0214	Functional	✓					✓	Seismic fails both AC and DC Fans together (correlated seismic failure)
	SEIS_17-4	AFW Exhaust Fan	TD AFW Pump Room 120V AC EMERG EXH FAN	1-FAN-030-0217	Functional	✓					✓	Seismic fails both AC and DC Fans together (correlated seismic failure)
	SEIS_5-17	TDAFWP Control Panels	AUX FW TURBINE FLOW (BECKMAN DWG 797492)	1-PNL-276-L381	Anchorage	✓	✓					
	SEIS_5-18	Aux Feedwater Controls	Aux Feedwater Control	1-PNL-276-L381A	Functional	✓	✓					

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
Emergency Raw Cooling Water	SEIS_24-1	ERCW Traveling Screen	ERCW Traveling Screen 1A-A, 1B-B, 2A-A, 2B-B	1-TWS-067-0434-A, 1-TWS-067-0445-B, 2-TWS-067-0439-A, 2-TWS-067-0451-B	Anchorage	✓	✓					Internal Events Ranking was based on individual component, but Internal Events RAW for Common cause of TWS is 23
MCR Panels	SEIS_5-10	Main Control Room Panel	Generator & Aux Power	1-PNL-278-M001	Functional	✓	✓					
	SEIS_5-10	Main Control Room Panel	120VAC PREFERRED POWER RACK UNIT 1	1-PNL-278-M007	Functional	✓	✓					
	SEIS_5-12	Main Control Room Panel	0-PNL-278-M026A-A, DSL Gen 1A-A Main Cont RM	0-PNL-278-M026A	Functional	✓	✓					
	SEIS_5-12	Main Control Room Panel	0-PNL-278-M026B-B, DSL GEN 1B-B MAIN CONT RM	0-PNL-278-M026B	Functional	✓	✓					
	SEIS_5-12	Main Control Room Panel	0-PNL-278-M026C-A, DSL GEN 2A-A MAIN CONT RM	0-PNL-278-M026C	Functional	✓	✓					

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
MCR Panels	SEIS_5-12	Main Control Room Panel	0-PNL-278-M026D-B, DSL GEN 2B-B MAIN CONT RM	0-PNL-278-M026D	Functional	✓	✓					
	SEIS_5-12	Main Control Room Panel	ERCW MAIN CNTL RN PNL	0-PNL-278-M027A	Functional	✓	✓					
	SEIS_5-12	Main Control Room Panel	COMP COOL WATER PNL	0-PNL-278-M027B	Functional	✓	✓					
FLEX	SEIS_3MW FLEXDG	3MW FLEX DGs	6900V 3MW FLEX Diesel Generator 3A, 3B	0-DG-360-0003A, 0-DG-360-0003B	Anchorage	✓						FLEX is modeled in Internal events for LOOP but does not show up as important. See Section 3.6.4 for additional discussion of FLEX components.

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
FLEX	SEIS_480 VFLEXDG	480V FLEX DGs	480V FLEX/ESBO 225 KVA DIESEL GENERATO R	0-DG-360-000A, 0-DG-360-000B	Anchorage	✓						FLEX is modeled in Internal events for LOOP but does not show up as important. See Section 3.6.4 for additional discussion of FLEX components.
	SEIS_FLE XBUS	480 V FLEX DG Buses	0-PNL-360-FP/A, 480V FLEX Fuse Panel A, B	0-PNL-360-FP/A, 0-PNL-360-FP/B	Functional	✓						FLEX is modeled in Internal events for LOOP but does not show up as important. See Section 3.6.4 for additional discussion of FLEX components.

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
FLEX	SEIS_FLEXTANK	FLEX Fuel Tanks	0-TANK-360-0113, 360-0213, 6900V 3MW FLEX DG Fuel oil Storage Tank	0-TANK-360-0113, 0-TANK-360-0213	Anchorage	✓						FLEX is modeled in Internal events for . See Section 3.6.4 for additional discussion of FLEX components.
TDAFW Pump	N/A[9]	N/A[9]	Turbine Driven Auxiliary Feedwater Pump	1-PMP-003-0001A-S	Random	✓	✓					

[9] This entry is not a Seismic Fragility Group. It is random failure of the SSC to function (i.e. start, run, or other PRA functional failure)

Table 3-11 (continued)
Sensitivity Study Results for Plant D

System	Seismic Fragility Group	Description of Fragility Group	Component from Fragility Group that Governs the Fragility			HSS in Risk Evaluations					Correlation Review	Comments
			Component Description	Component ID	Failure Mode of SSC	Seismic PRA	FPIE PRA	Fire Risk	Implicit Modeling	Passive Cat.		
Contain-ment Penetrations	SEIS_ CONPEN	Seismically-induced Failure of Containment Penetrations	Containment penetrations	Various	Structural	✓						Containment penetrations would be treated as HSS in the 50.69 defense-in-depth review. See Section 3.6.5 for additional discussion of defense-in-depth.
					Totals	64	31	4	0	1	24	
							35 Seismic Fragility Groups classified as HSS via overlapping 50.69 criteria, including 2 items addressed by defense-in-depth criteria					

3.6 Summary of Sensitivity Study Insights

Sections 3.2 through 3.5 describe trial 50.69 categorization evaluations performed at four plants to determine how the seismic related categorization insights compare with categorization insights at the same plants using their FPIE PRAs and fire PRAs, if available. Overall conclusions are summarized in this section.

3.6.1 Limited Unique Seismic High Safety Significant Structures, Systems, and Components

In all four trial studies, there were either no components or very few components identified as HSS in the SPRA that were not also HSS for another reason. Therefore, the seismic risk insights provided only limited unique insights into the 50.69 categorization process. And those unique insights were generally associated with SSCs that would be treated as seismically correlated failures in an SPRA. This suggests that the SSCs most important in responding to a seismic event are included within the set of SSCs necessary to respond to other events.

This result should not be interpreted to suggest that there are no SSCs that would be HSS from a seismic hazard. In each study, there were a significant number of HSS SSCs identified using the SPRAs. However, those same SSCs were also HSS for other reasons.

The trial studies indicate that the overall benefits, in terms of seismic risk insights in the 50.69 categorization process, do not warrant the cost of performing an SPRA.

3.6.2 Seismic Correlated Failures

Some of the trial studies identified a limited number of seismic-related HSS SSCs due to the way seismically correlated failures are typically treated in SPRAs. For example, if two pumps performing the same function are located side by side in the plant, they are both assumed to fail with the same seismic fragility. These correlated failures can contribute unique seismic insights into the 50.69 categorization process.

In the case of passive items such as tanks, two similar tanks located side by side would generally also be assumed to fail with the same seismic fragility. This correlated failure is not identified by the 50.69 passive categorization process, which relies on the FPIE PRA, which does not model common cause failure of tanks.

The trial studies indicate that special considerations may be necessary for evaluating the potential of seismically correlated failures to influence the categorization process at sites where the correlated failures may be likely.

3.6.3 Relays

Relays are important components in NPP seismic evaluations. Many FPIE PRAs do not explicitly include relays in their models and they are usually added to the model for an SPRA. However, important relays such as those in the emergency power systems are critical to the success of the backup AC power function and therefore would be implicitly addressed by the FPIE PRA insights in the 50.69 categorization process. For example, all four trial evaluations identified parts of the emergency power system as HSS in the SPRA, the FPIE PRA and/or the Fire PRA. If the relays within those systems, or the electrical enclosures housing the relays, are not explicitly modeled in the FPIE PRA, then their 50.69 categorization would be derived by identifying the importance of the system function and correlating those functions with the specific components. Section 5 of NEI 00-04 provides the following guidance.

> Some systems and structures are implicitly modeled in the PRA. It is important that PRA personnel that are knowledgeable in the scope, level of detail, and assumptions of the plant-specific PRA make these determinations. As outlined in Section 1, by focusing on the significance of system functions and then correlating those functions to specific components that support the function, it is possible to address even implicitly modeled components.

Therefore, in the case of relays in the emergency power system, the relays would be implicitly modeled in the FPIE PRA and their function within the system would need to be evaluated to perform 50.69 categorization down to the component level.

3.6.4 FLEX Components

As noted in Section 3.5, one of the sensitivity studies identified that some FLEX equipment exceeded the quantitative categorization thresholds within the SPRA but the FLEX equipment was not identified as HSS in the FPIE PRA model. With respect to 50.69 programs, inclusion of FLEX equipment in the PRA model can impact a 50.69 program in two ways; categorization results and application of alternate treatment.

With respect categorization results, inclusion of FLEX equipment in the PRA model (for example, FPIE PRA, seismic PRA) would act to, at worse, make some modeled non-FLEX equipment appear to be less safety significant as compared to the PRA results with the FLEX equipment not modeled. This is because the 50.69 categorization process uses relative risk metrics (that is RAW, F-V) and if the FLEX equipment is providing relative value (for example, reducing CDF), then the other modeled equipment (non-FLEX) would become less important. That is, some previously categorized RISC-1 components could become RISC-3 components when the FLEX equipment is included in the PRA model. Thus, not including FLEX equipment in the PRA model for 50.69 categorization is at worst conservative from a RISC-3 assignment perspective.

From an alternative treatment perspective, for plants that chose to categorize FLEX equipment, these components will be categorized as either RISC-2 (non-safety related / safety significant) or RISC-4 (non-safety related / non- safety significant). The rule [1] requires that for RISC-2 components:

> The licensee … shall ensure that SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance.

10 CFR 50.69 was published in 2004. As such, while the rule requirement to "ensure that the treatment of RISC-2 SSCs is consistent with the assumed performance in the PRA" is a valid position, it does not reflect the maturing of PRAs over the ensuing years such as the need to reflect Regulatory Guide 1.200, Rev 2 and later revisions of the ASME / ANS PRA Standard [24]. Additionally, each 50.69 license amendment request contains a section on PRA Technical Adequacy which assures that the plant-specific PRA is adequate to support the 50.69 categorization effort, including a complete treatment of causes of system failures, reliability and unavailability of modeled SSCs. Thus, by meeting RG1.200 [17] and ASME/ANS Standard [24] guidance, it is assured that the performance assumed in the PRA for FLEX equipment is consistent with plant practices.

Further, in response to post-Fukushima actions, licensees are required to demonstrate that FLEX equipment is stored, tested, maintained and procedures are in place so that the FLEX equipment can fulfill their stated missions.

3.6.5 Defense-in-Depth Assessment

NEI 00-04 [2] Sections 6.1 (Core Damage Defense-in-Depth) and 6.2 (Containment Defense-in-Depth) provide guidance for incorporating considerations to assure that defense in depth is preserved when categorizing an SSC as low safety significant.

With respect to core damage, the assessment considers both the level of defense-in-depth in preventing core damage and the frequency of the events being mitigated. This ensures that adequate defense-in-depth is available to mitigate design basis events given their likelihood of occurrence, including consideration of diverse and redundant trains and systems in the overall categorization process.

With respect to containment, the assessment considers SSCs that play a role in preventing large, early releases, such as interfacing systems LOCA (BWR and PWR), steam generator tube leak (PWR), containment isolation failures (BWR and PWR), and early hydrogen burns (ice condenser and Mark III containments). Containment defense-in-depth is also assessed for SSCs that play a role in preventing large containment failures (for example, due to loss of containment heat removal).

3.6.6 Civil Structures

NEI 00-04 [2] requires that both F-V and RAW importance measures be considered in 50.69 categorization. The RAW importance measure is calculated assuming the SSC (or basic event) is always failed. Although this is a useful importance measure for bounding discussions and for FPIE PRAs, in SPRAs RAW implies that the SSC has no seismic capacity and the RAW insights should be considered with some care when used in an SPRA.

When applied literally for Category 1 civil structures such as Reactor Buildings or Auxiliary Buildings that house critical systems and components, high RAW values can be expected because it implies that the structure is failed. The RAW metric can also be sensitive to cutset truncation depending upon the base probability of the basic event in question and the cutsets in which the basic event participates.

It is recognized that civil structures containing PRA credited equipment (for example, Reactor Building) are likely important to safety because their failure can fail the credited equipment functions. Therefore, if a licensee chooses to categorize structures under 50.69 using the guidance in this report, the recommended practice is to consider civil structures housing HSS SSCs to be HSS themselves, unless otherwise justified. Note that this does not imply that everything inside an HSS structure should then be considered HSS.

4 SUMMARY AND CONCLUSIONS

The NRC's 10 CFR 50.69 process [1] allows a plant to categorize the safety significance of its SSC using a robust categorization process defined in NEI 00-04 [2], as endorsed by NRC in Regulatory Guide 1.201 [3]. The risk-informed categorization process helps focus attention on SSCs that are the most important to plant safety while allowing increased operational flexibility for SSCs that are less important to plant safety.

One of the screening criteria evaluated in the categorization process specified in NEI 00-04 is seismic risks, which can be evaluated using an SPRA, or an SMA if an SPRA is not available, or screened out if the SCDF and SLERF are very small compared to the FPIE PRA CDF and LERF. There are a number of plants that do not have an acceptable SPRA or SMA and cannot screen out of seismic considerations, therefore a need exists to consider alternatives for considering the insights of seismic risks in the 50.69 categorization process.

This report develops alternate approaches for plants to provide the necessary seismic risk insights within the 50.69 categorization process. Trial 50.69 categorization evaluations are performed at four plants with SPRAs and high GMRS compared to their SSEs to determine the seismic related categorization insights. Those insights are compared with categorization insights at the same plants using their FPIE PRAs and fire PRAs if available to determine the degree to which the seismic insights produce unique categorization insights.

The trial case results show that there were either no components or very few components identified as HSS in the SPRA that were not also HSS for another reason. Therefore, the seismic risk insights provided only limited unique insights into the 50.69 categorization process. And those unique insights were generally associated with SSCs that would be treated as seismically correlated failures in an SPRA.

These insights are used to develop a three-tiered graded approach for considering seismic risks in the 50.69 categorization process. The tiers are defined based on the degree to which the plant GMRS exceeds the plant SSE, which influences the likelihood that unique seismic-related HSS SSCs will be identified. The tiers and recommended seismic risk evaluation processes are described in Table 4-1.

For Tier 2 seismic hazard plants, a new seismic risk evaluation process is developed to use the FPIE PRA to determine the categorization insights appropriate for seismically correlated failures. SSCs that would be treated as seismically correlated in an SPRA are identified through a series of reviews and seismic walkdowns, those SSCs are modeled with new common cause failure basic events in the FPIE PRA, and sensitivity studies are performed to determine if specific SSCs should be HSS.

Table 4-1
Alternate Approach Seismic Tiers and Seismic Rick Evaluation Process

Tier	Tier Criteria	Seismic Risk Evaluation Processes
1	Plants where the GMRS peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0 Hz and 10 Hz. At these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are expected to be minimal.	At Tier 1 sites, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low. Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the FPIE PRA and other risk evaluations along with the required Defense-in-Depth and Integrated Decision-making Panel (IDP) qualitative considerations are expected to adequately identify the safety-significant functions and SSCs required for those functions and no additional seismic reviews are necessary for 50.69 categorization.
2	Plants where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3. At these sites, the unique seismic categorization insights are expected to be limited.	At Tier 2 sites, there may be a limited number of unique seismic insights, most likely attributed to the possibility of seismically correlated failures, appropriate for consideration in determining HSS SSCs. The special seismic risk evaluation process recommended using a Common Cause impact approach in the FPIE PRA can identify the appropriate seismic insights to be considered with the other categorization insights by the Integrated Decision-making Panel (IDP) for the final HSS determinations.
3	Plants where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is high enough that the NRC required the plant to perform an SPRA to respond to the Fukushima 50.54(f) letter [6].	At Tier 3 sites, the available methods in NEI 00-04 [2] can be used to provide seismic inputs to the categorization process. These methods include the use of an SPRA or an SMA as described in NEI 00-04 Section 5.3.

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A

IDENTIFYING SEISMIC CORRELATED OR SEISMIC INTERACTION SCENARIOS FOR CONSIDERATION IN 50.69 CATEGORIZATION

Seismic Probabilistic Risk Assessments (SPRAs) have been conducted for a large number of nuclear power plants worldwide in the last 35 years. The methodology has progressed during that period of time and is currently well established with several technical references documenting the methodology. Seismic PRA is different from a full power internal events (FPIE) PRA in two important ways: (a) all possible levels of earthquakes along with their frequencies of occurrence and consequential damage to plant systems and components should be considered, and (b) earthquakes can simultaneously damage multiple redundant components due to the common cause effect of the earthquake. This common cause effect has traditionally been referred to as seismic correlation in the SPRA technical literature. A separate but related potential common cause effect from earthquakes consists of a phenomenon referred to as seismic systems interactions. Seismic systems interaction consists of the failure, displacement or action (for example, spraying from failed piping, impacts from failing block walls) of a structure, system or component (SSC) that negatively affects the credited function of other SSCs in the SPRA.

One of the key seismic insights from the trial studies documented in Section 3 of this report is the importance of considering seismic correlation effects on the 50.69 HSS categorization of plant SSCs. The correlation of SSCs that exists in a seismic event is not typically captured in the FPIE PRA or the fire PRA. As a result, for 50.69 categorization purposes, the Tier 2 plants meeting the criteria in Section 2.3 should include these seismic correlation and interaction insights when performing system categorization. Figure 2.4 provides an overview of the process for the seismic correlated failure assessment and Section 2.3.1 provides the details of that seismic correlation assessment. The purpose of this appendix is to provide an approach to identify the SSCs considered to be seismically correlated (Item 5a in Figure 2.4) and to identify seismic interaction scenarios (Item 5b in Figure 2.4) which could affect the 50.69 categorization.

A.1 Background on Seismic Correlation Considerations in SPRAs

The first attempt to programmatically treat the dependencies between seismic responses and between seismic capacities of components was in the seismic risk methodology developed for the Seismic Safety Margins Research Program (SSMRP) at the Lawrence Livermore National Laboratory [27]. The local responses of different components located at different elevations in various buildings were represented by a joint lognormal distribution; similarly, the seismic capacities of these components were also represented by a joint lognormal distribution. This SSMRP methodology consisted of a detailed set of calculations to develop partial correlation coefficients for each of the structures, systems and components (SSCs) in the plant logic model. Because the application of this methodology was both computationally intensive and data intensive, it was not used in subsequent SPRAs.

Using the results of the SSMRP methodology to perform two SPRAs as trial applications (for Zion and LaSalle), a Sandia Laboratory study [28] developed simplified rules for assigning the response correlation coefficient, thus bypassing the case-by-case SSC partial correlation computations. These simplified rules are provided in Table A-1.

Table A-1
Correlation Guidance from Sandia National Laboratory Study

Rule #	Correlation Guidance
1	Components on the same floor slab and sensitive to the same spectral frequency range (*that is*, ZPA, 5-10 Hz. or 10-15 Hz) will be assigned response correlation = 1.0.
2	Components on the same floor slab sensitive to different ranges of spectral acceleration will be assigned response correlation = 0.5.
3	Components on different floor slabs (but in the same building) and sensitive to the same spectral frequency range (ZPA, 5-10 Hz or 10-15 Hz) will be assigned response correlation = 0.75.
4	Components on the ground surface (outside tanks, etc.) shall be treated as if they were on the grade floor of an adjacent building
5	"Ganged" valve configurations (either parallel or series) will have response correlation = 1.0.
6	All other configurations will have response correlation equal to zero.

Recent reports by the NRC [29] and EPRI [30] propose methods to address seismic correlation. Neither correlation methodology has been piloted to date and will need to be evaluated in order to understand the costs, benefits and limitations of the recommended approaches.

While the studies summarized above proposed a partial correlation characterization, the state of the practice in SPRAs consists of a binary (zero or one) correlation factor. Debate among the SPRA practitioners as to the accuracy, the cost/benefit and the lack of pilot applications of the partial correlation approaches has led to the use of this more simplified binary approach for most SPRAs conducted to date. However, the research associated with these partial correlation studies have laid the foundation for the decisions made in the simplified binary approach and serve to guide the decisions in SPRAs practiced today. For the 50.69 categorization program, the current state of practice (binary approach) for the treatment of correlation is applied.

A.2 Approach

Figure 2-3 outlines the approach for identifying unique HSS components for Tier 2 plants. Step 5 of the process consists of a seismic walkdown focused on identifying those SSCs in the system being categorized that would either be (1) evaluated to be seismically correlated in the event of an earthquake or (2) evaluated to be subject to common cause seismic interactions. Confirmatory seismic walkdowns integrated with plant documentation reviews (that is general arrangement drawings and previous seismic walkdown documentation) are the basis for identifying correlations and seismic interactions within Step 5 of the recommended 50.69 categorization process.

Seismic walkdown methods have been documented in past SPRA and SMA methodology reports, including [19], [18] and [31]. In addition, EPRI offers a training course [32] that focuses on the seismic walkdown and also provides additional information on the identification and assessment of seismic interactions. The methods and qualifications for these seismic walkdowns and seismic interaction reviews are well documented in these references and will not be repeated within this report.

The seismic fragility of an SSC can be broken down into two fundamental elements: the seismic capacity is a measure of the strength of the SSC and the seismic demand is a measure of the accelerations/displacements induced by the earthquake at the SSC location. The binary method for the identification of correlated/uncorrelated SSCs in an SPRA is to assign either 100% correlation or 0% correlation for the fragilities associated with each set of SSCs being addressed. Since many SSCs could be judged to have some degree of correlation in either the seismic capacity or the seismic response, the ultimate binary correlation decision is typically a judgment of engineers experienced in both seismic capacity and seismic response fields.

The following guidance from references [18] and [31] summarize the seismic correlation process and judgments made in most SPRAs and are the recommended guidance for identifying seismically correlated conditions for moderate seismic hazard plants as described in Section 2.3.1.

1. Review available documentation (general arrangement drawings, previous seismic walkdown documentation, etc.) in advance of the walkdown to support the correlation assessments.
2. Perform a confirmatory walk down of the system being categorized to confirm the characteristics described below.
3. Similar SSCs subject to similar seismic response are assumed to be perfectly correlated (factor = 1.0) and should be included in the Section 2.3.1 evaluation. This includes the following conditions.
    a. Similar equipment on the same floor of a structure are typically judged to be fully correlated.
    b. Similar equipment on adjacent floors of a structure (resulting in similar demand) are also typically considered to be correlated [29, 30] if the equipment have similar failure modes and fundamental frequencies. As summarized in [18], the Diablo Canyon Long Term Seismic Program performed a more detailed review of seismic correlations and concluded that a high degree of correlation existed between items of similar natural frequencies located on different floors in the same structure.
    c. Similar equipment in different but similarly constructed buildings on the same basemat are also judged to be correlated based on the assumption of similar seismic demand.
4. SSCs with different types of failure modes are treated as uncorrelated (correlation factor = 0) and do not need to be included in the Section 2.3.1 evaluation.

5. Similar SSCs but with significantly different seismic demands are treated as uncorrelated (correlation factor = 0) and do not need to be included in the Section 2.3.1 evaluation. Examples include:
   a. Similar SSCs with similar failure modes but located in different structures, and
   b. Similar SSCs with similar governing failure modes located in the same structure, but with significantly different seismic responses.

These correlation guidelines are provided to assist in the identification of SSCs judged to be seismically correlated. Additional guidance is provided in [18] and [31] to support the decisions made on the walkdown. Following completion of the walkdown, the list of correlated SSCs identified should be placed into Step 6 from Figure 2-3.

The second part of the 50.69 categorization walkdown includes the evaluation for seismic interactions which could cause correlated failures within the system being categorized. Potential seismic interactions should be evaluated during the system walkdown to assess whether any credible interactions could result in correlated failures of equipment within the system being categorized. As mentioned above, the approaches for evaluating seismic interactions are well documented in technical references and will not be repeated in this appendix. For purposes of describing the process recommended in this appendix, it is informative to define terminology associated with seismic interaction assessments:

- Interaction Source – the source is a structure, system or component (SSC) that causes the seismic interaction. The sources of seismic interactions can be based on falling items, deflecting items or flood initiators. So an example of a typical seismic interaction source would be an unreinforced block wall or a failed water storage tank that floods an area.
- Interaction Target – the target is the SSC that is being evaluated and is required to maintain its safety function or pressure boundary as part of the seismic risk assessment being conducted. For purposes of this 50.69 correlation evaluation, the equipment in the system being categorized will generally be considered as the targets.

The process for assessing the potential for correlated seismic interactions during the walkdown should consist of the following steps:

1. Review available documentation (general arrangement drawings, previous seismic walkdown documentation, etc.) in advance of the walkdown to support the seismic interaction assessments
2. Perform a confirmatory walk down of the system being categorized to confirm the characteristics described below.
3. Determine if any credible seismic interactions exist in the vicinity of the SSCs being categorized. The walkdown team should screen out those seismic interaction sources not deemed credible based on their experience and training. The walkdown team should also screen out credible sources that would not be expected to damage/fail the target equipment in the system being categorized.
4. Those seismic interaction sources that are not screened out during the walkdown) should be assessed using the methods documented in Appendix B to determine if they may be screened out as high capacity seismic interaction sources.

5. The remaining seismic interactions that could represent a common cause event (affecting more than a single SSC in the system being categorized) should be added to the list of correlated SSCs identified and evaluated per the Step 6 diamond from Figure 2-3.

Past SPRAs have identified several SSC categories [29] that have been frequently classified as being correlated and, at the same time, their correlation or dependency made a difference to baseline seismic CDF or to the safety insights.

Typical Interaction Sources

- Masonry walls
- Non-safety related structures housing safety related equipment
- Large tanks: condensate storage tanks or other similar tanks (flooding source)

Typical Interaction Targets

- Batteries and racks
- Electrical cabinets: motor control centers and switchgear
- Small tanks: diesel generator fuel oil day tanks
- Heat exchangers: such as component cooling water heat exchangers
- Mechanical equipment: long shafted service-water pumps, horizontal auxiliary feed water pumps

This list is not intended to be a limiting set for this assessment, instead it serves as operating experience from past SPRAs to be used in the walkdown and correlation assessment to ensure these items are given the appropriate focus.

B
CRITERIA FOR CAPACITY-BASED SCREENING FOR HIGH CAPACITY SSCS

Seismic risk insights from past SPRA and SMA studies have shown that high seismic capacity SSCs from the SPRA Seismic Equipment List (SEL) do not typically contribute to the seismic risk. Similarly, those seismic interaction scenarios (for example, block walls, falling objects, and displacements which cause impact with nearby elements) which can be demonstrated to have high seismic capacities, have also not resulted in significant risk contribution in past seismic studies. Therefore, these high seismic capacity SSCs and interactions are unlikely to be categorized as HSS and can be screened out from the 50.69 seismic categorization process. This high seismic capacity screening fits into Step 5c of the flow chart in Figure 2-3. The process for screening individual SSCs documented in EPRI 1025287 [11] (the SPID) will form the backbone for this screening approach. Following this approach, SSCs with a HCLPF capacity greater than the calculated screening level HCLPF could be categorized as low safety significant (LSS).

B.1 Approach

As part of the effort to develop the SPID [11], seismic capacity-based criteria were developed to determine which SSCs should have component specific calculated fragility values to ensure that proper focus was given to those SSCs with the potential to be risk-significant. These criteria were developed using a parametric/sensitivity study [33] which provided the basis for the SPID recommendations. SSCs with capacities above the calculated screening level are not expected to have significant impact on the result of the SPRA analyses, the ranking of accident sequences, or the calculated sequence- or plant-level seismic CDF or LERF values. As such, SSCs with capacities above that screening level would also not be expected to be high safety significant (HSS) components within the 50.69 categorization process.

Section 6.4.3 of the SPID [11] identifies the approach to develop a screening HCLPF value for these higher capacity fragilities. Following the SPID approach, a screening HCLPF value is calculated by convolving the fragility of a single element with the site-specific hazard curve such that the SCDF is at most about 5E-7 per year. This can be done with trial and error runs using a quantification code or with a spreadsheet with an assumed composite variability (for example, β_c= 0.4) as described in [11]. This 5E-7 screening criteria was developed for the higher seismic hazard plants where seismic typically has a corresponding higher resulting risk. For a medium to low seismic hazard site this screening level of 5E-7 could potentially be unconservative, therefore an SCDF value of approximately ½ of the SPID value, or 2.5E-7 is judged to be more appropriate for purposes of this 50.69 categorization screening assessment. Other appropriately justified site-specific screening values may be used.

To apply his approach, a seismic fragility must be developed for each SSC that is being assessed as part of the categorization process and compared to screening level developed as described above. The fragility methodology is well established and there are numerous references in the literature describing the methods. Four EPRI reports that collectively capture the fragility process are listed in Table B-1.

Table B-1
Seismic Fragility References

Topic	Title	Reference
Seismic Fragility Guidance	Seismic Fragility Applications Guide Update	EPRI Report 1019200 (2009) [34]
	Seismic Fragility Application Guide	EPRI 1002988 (2002) [35]
	Methodology for Developing Seismic Fragilities	EPRI TR-103959 (1994) [36]
	A Methodology for Assessment of Nuclear Plant Seismic Margin	EPRI NP 6041-SL (1991) [19]

For nuclear plants without existing SPRAs, one challenge will be to produce in-structure seismic responses for use in these fragilities. Development of finite element models and generation of new seismic response analyses using the current seismic hazard shape at the plant site is one option, however, more simplified and conservative approaches could be used when justified by experienced engineers within the structural dynamics field. These approaches include:

- Scaling of existing plant seismic response analyses where the shapes of the uniform hazard response spectra (UHRS) are similar [35, 19]
- Estimation of high frequency seismic response using an approach in EPRI 3002004396 [37] which describes a process to estimate seismic responses for hard rock sites that have ground response spectral peaks in the high frequency part of the response spectrum

In addition, it may also be possible for fragility analysts to conservatively estimate seismic demands using simplified approaches documented in ASCE 7 [38] for justifying additional SSCs that would have HCLPF capacities above the screening threshold. Assessments made would have to be necessarily conservative (biased towards higher in-structure response spectra (ISRS)) and account for potential variability of ISRS results based on the use of these approximate methods.

B.2 Justification

While the SPID capacity-based screening approach is intended as a tool to be used for seismic risk assessments to focus fragility resources on risk-significant SSCs, the concept can be extended to 50.69 categorization. The capacity-based screening approach from the SPID is purposely conservative and is based on a single element leading directly to core damage. In addition, the recommended approach in this Appendix conservatively reduces the SPID target SCDF of 5E-7 by 50%, resulting in a more conservative SCDF value of 2.5E-7. If it is possible to demonstrate a component has a HCLPF above the calculated screening threshold, that component is not expected to be risk-significant in an SPRA. So even in the absence of a formal risk assessment, it is possible to identify certain SSCs with high seismic capacity that would not be expected to be risk-significant.

B.3 Conclusion

Use of the capacity-based screening approach based on a similar approach documented in the SPID is an acceptable method to screen SSCs into the LSS category for 50.69 categorization. When SSCs are determined to have HCLPFs greater than this screening level HCLPF, it can be concluded that they would not be risk significant in an SPRA; therefore, those SSCs can be classified as LSS rather than HSS.

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Electric Power Research Institute
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com