INSIGHTS FROM REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENTS IN THE CONTEXT OF 10 CFR 50.69

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Nuclear power reactor licensees that are regulated by the U.S. Nuclear Regulatory Commission (NRC) have the option of voluntarily adopting the regulation in Part 50.69 to Title 10 of the Code of Federal Regulations, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors" (hereafter referred to as 10 CFR 50.69). 10 CFR 50.69 allows categorization of structures, systems, and components (SSCs) as either high safety significance (HSS) or low safety significance (LSS). Certain regulatory requirements can be reduced for LSS components that were previously considered to be safety-related.

The number of applications from licensees seeking to adopt 10 CFR 50.69 has increased recently. The NRC staff has performed detailed reviews of the technical acceptability of seismic probabilistic risk assessments (SPRAs) and the use of those SPRAs by licensees in the categorization of SSCs per the requirements in 10 CFR 50.69. This paper will present insights gained from the NRC staff's review of SPRAs in the context of 10 CFR 50.69. Discussion will include insights on fragility evaluation, determination of importance measures across seismic 'bins', mapping of components between SPRAs and PRAs for other hazards, and the relation between reduced regulatory requirements allowed by 10 CFR 50.69 and SPRA maintenance.

I. BACKGROUND

The U.S. Nuclear Regulatory Commission (NRC) promulgated Part 50.69 of Title 10 to the Code of Federal Regulations (hereafter referred to as 10 CFR 50.69) in November 2004 (Ref. 1). The rule provides a voluntary alternative to compliance with several regulations that require "special treatment" for certain structures, systems, and components (SSCs). "Special treatments" are regulatory requirements which go beyond industrial controls, including specific inspection, testing, qualification, and reporting requirements. 10 CFR 50.69 requires a process for categorization of SSCs as being either High Safety Significance (HSS) or Low Safety Significance (LSS). The categorization process can result in increased requirements for HSS SSCs where such requirements did not exist previously (i.e., non-safety related SSCs that perform safety significant functions) and reduced requirements for LSS SSCs where such requirements existed previously (i.e., safety-related SSCs that perform low safety significant functions).

I.A. Categorization Based on Importance Measures

The NRC has endorsed the categorization process in Nuclear Energy Institute (NEI) 00-04, Revision 0 (Ref. 2) in Regulatory Guide (RG) 1.201, Revision 0 (Ref. 3). The categorization process includes plant-specific risk analyses which are used in combination with an Integrated Decisionmaking Panel (IDP) to determine whether the SSC has a low or high safety significance. The IDP, in its determination of the appropriate categorization of a particular SSC, considers the risk analyses in conjunction with non-risk related attributes such as the impact of the proposed categorization on the defense-in-depth and safety margins at the plant.

Probabilistic Risk Assessments (PRAs) for different hazards (such as internal events, internal fires, seismic events etc.) provide the quantitative input to the categorization. Crucial to the categorization process outlined in NEI 00-04 are risk importance measures, which are indicators of the risk significance of an SSC in a particular PRA. Different importance measures provide information about the risk significance from different perspectives. The Fussel-Vesely (F-V) importance measure represents the fractional change in the risk upon elimination of a particular failure mode of a SSC (i.e., setting the failure probability for a particular failure mode of a SSC to 0.0). It can also be expressed to represent the fractional contribution of a particular failure mode of a SSC to the total risk. The Risk Achievement Worth (RAW) importance measure represents the ratio of the risk under the assumption that a particular failure mode of a SSC is guaranteed to always occur (i.e., setting the failure probability for a particular failure mode of a SSC to logical 'TRUE') to the base risk. The fractional contribution in case of F-V and RAW is with respect to the total base risk, in terms of core damage frequency (CDF) or large early release frequency (LERF) for a particular hazard as quantified by the corresponding PRA. As a result, F-V and RAW are relative importance measures

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II. RISK FROM SEISMIC EVENTS AT NUCLEAR POWER PLANTS

The guidance in General Design Criterion (GDC) 2 necessitates nuclear power plants (NPPs) in the United States to be designed to withstand credible natural and manmade hazards, including seismic events (i.e., earthquakes). In addition, certain structures, systems, and components (SSCs) have to be designed such that they can operate and remain operational at the Safe Shutdown Earthquake (SSE), which represents the design basis earthquake for each nuclear power plant.

Since the original design of many NPPs, various efforts to assess the impact of seismic events on these plants, including the development of quantitative risk and risk insights, have been undertaken, including:

- Unresolved Safety Issue A-46, "Seismic Qualification of Equipment in Operating Plants," 1980 (Ref. 4),
- Individual Plant Examination of External Events (IPEEE), 1991 (Ref. 5),
- Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States," 2005 (Ref. 6), and
- Fukushima Dai-ichi Response, Near Term Task Force Recommendation 2.1, 2012 (Ref. 7).

Although these efforts have resulted in an improved understanding of seismic risk at U.S. NPPs and a number of safety improvement, the seismic risk assessment methodologies used in response to these programs have varied in detail as well as methodology and the assessments were not necessarily maintained beyond initial completion of the effort or program.

SPRA is one of the risk analyses options to account for the seismic risk contribution. SPRAs have evolved over the last few decades and are garnering increased safety and regulatory use as the technology matures. Some U.S. NPPs have developed high quality SPRAs and have chosen to use them to support the implementation of 10 CFR 50.69. The use of SPRAs by the licensees provides more realistic insights in the categorization process and consequently, avoids conservative categorization of SSCs as HSS if the other approach in the endorsed NEI 00-04 guidance is used.

III. KEY INSIGHTS FROM TECHNICAL ACCEPTABILITY REVIEWS OF SPRAs FOR USE IN 10 CFR 50.69 CATEGORIZATION

The NRC staff has been reviewing the use of SPRAs in the context of 10 CFR 50.69. The NRC staff's review includes determination of the technical acceptability of the SPRA for risk-informed decisionmaking and the application of SPRA in the categorization process. Because the use of SPRAs as part of the categorization for 10 CFR 50.69 rule has currently not been implemented widely, several insights have been identified with broad applicability to the use of SPRAs in those applications. NRC Standardized Plant Analysis Risk (SPAR) models have been used to support some of the insights.

III.A. Fragility Evaluations

The fragility of an SSC is the conditional probability of failure of that SSC at a particular reference seismic acceleration such as the peak ground acceleration (PGA). A SSC can have multiple seismically-induced failure modes each of which can have a separate fragility and a corresponding basic event in the SPRA.

SPRAs usually use the entire fragility curve for quantification. Truncation of the fragility curves at a level that does not significantly change the SCDF as well as the categorization is used to refine the fragility evaluations used in SPRA development. Structural mechanics, earthquake experience, and NPP SSC design (which, for certain SSCs, includes seismic design considerations) provide basis for the use of such truncation. In addition, SPRAs have demonstrated that the conditional risk (i.e., conditional core damage probability or conditional large early release probability) from low seismic acceleration ranges is minimal.

As stated earlier, importance measures are used in the categorization process. Because F-V captures the fractional contribution of a BE to the total risk, it is not expected to be substantially changed if the contribution from lower seismic accelerations up to a justified truncation level is excluded. On the other hand, RAW captures the impact of the guaranteed failure of a BE to the total risk, and consequently, assigning a guaranteed failure to the lower seismic accelerations up to a justified truncation level can have unrealistic impact on the importance measure. Ref. 8 explored the use of a truncated fragility curve for SSCs in SPRAs in the context of the categorization for 10 CFR 50.69. Ref. 8 used seven SPAR models to determine the appropriateness and the resulting impact on the SCDF of truncating the fragility curve. The results therefrom provide quantitative basis

for the selection of the appropriate truncation of the fragility curves in an SPRA.

III.B. Importance Measure Calculations

SPRAs discretize the seismic hazard curve into 'bins', usually 10 or more, for quantification purposes. In addition, SPRAs include seismic failure modes for components in addition to random failures, which are modeled as separate basic events. As a result, the determination of importance measures is not as straightforward as that for internal events and fire PRAs. Further, different modeling and therefore, quantification, schemes for developing a SPRA exist. As a result, a single cutset file is produced in one scheme which includes the initiating 'bin' for each cutset. In an alternate scheme each 'bin' is solved separately resulting in different cutset files for each 'bin' without explicit information of the initiating 'bin'.

Since F-V represents the fractional contribution of a particular BE to the total risk, it is reasonable to expect that the importance measure is additive for mutually independent failure modes which are represented as BEs. Therefore, the total F-V for a SSC from the SPRA is the addition of the F-V for the BEs representing each failure mode of that SSC.

Approach 1: In case of the 'one-top' model that produces one cutset file the addition does not add any complications and can provide the F-V for the SSC from the SPRA.

Approach 2: In case each 'bin' is solved separately, the F-V for the SSC due to seismically-induced failures can be performed by determining the F-V for such failures (i.e., BEs) for each 'bin' and then weighing the seismic F-V from each 'bin' with the ratio of the 'bin' SCDF to the total SCDF will provide the F-V for seismically induced failures for the SSC from the SPRA. Subsequently summing the result with the F-V for random failures will result in the F-V for the SSC from the SPRA. Mathematically, the operation can be expressed as,

$$FV_i = \left(\frac{\sum_k FV_{iSEIS,k} \times SCDF_k}{\sum_k SCDF_k}\right) + \left(\sum_j FV_{iSRAND,j}\right) \quad (1)$$

where,

FV_i = Fussell-Vesely for the ith component
k = number of seismic 'bins'
FV_{iSEIS} = Fussell-Vesely for the seismic failure of the ith component
FV_{iSRAND} = Fussell-Vesely for the random failure of the ith component
j = number of random failures
SCDF = seismic core damage frequency (from seismic and non-seismic failures)

Expanding the term in the first parenthesis in Equation (1) using the definition of FV results in,

$$\frac{\sum_k FV_{iSEIS,k} \times SCDF_k}{\sum_k SCDF_k} = \frac{1}{\sum_k SCDF_k} \times \sum_k \left(\frac{F(i)_{SEIS,k}}{SCDF_k} \times SCDF_k\right) = \frac{\sum_k F(i)_{SEIS,k}}{\sum_k SCDF_k} \quad (2)$$

Equation (2) translates to the definition of F-V and demonstrates that F-V is calculated correctly.

Since RAW represents the impact of guaranteed failure of a particular BE, which implies a particular failure mode of a SSC, to the total risk, the importance measure is not additive. Adding the RAW for individual BEs is a simple approach which provides an upper bound but will definitely lead to overestimation of the importance measure as well as potentially conservative categorization and diminishes the realism captured in the SPRA. Different approaches provide valid calculations for the RAW for a SSC from SPRAs.

Approach 1: In case each 'bin' is solved separately, the RAW can be determined by 'collecting' the cutsets from each file, use post-processing rules to set each BE for a particular SSC to logical 'TRUE', and determine the result SCDF. Such an implementation will work only if 'collecting' the cutsets for each 'bin' does not lose the initiator (i.e., 'bin') information. Using the maximum of the RAW from the individual BEs for a SSC as the RAW for that SSC from the SPRA. This approach is consistent with the logic model too where each failure mode is mutually exclusive and therefore, cannot occur together. The maximum RAW represents the maximum impact that the SSC can have on the plant risk due to any failure mode of the SSC. Alternately, all BEs for a particular SSC in the SPRA can be assumed to suffer guaranteed failure simultaneously (i.e., set all BEs to logical 'TRUE' at the same time), the resulting logic model is resolved, and the SCDF determined (or seismic LERF, as the case may be). The RAW can subsequently be calculated by taking the ratio of the seismic CDF calculated with all BEs for a SSC set to logical 'TRUE' to the base seismic CDF.

Caution needs to be exercised when implementing the second approach to ensure that the BEs are set to logical 'TRUE' and not 1.0. Use of 1.0 will result in incorrect determination of the SCDF and therefore, the RAW because the idempotent law will not be applicable. As an illustration consider two cutsets, A*B-FR*C*D + A*B-FS*C*D, in the final quantification. Setting B-FR and B-FS, either individually or simultaneously, to 'TRUE' will

result in a single cutset, A*C*D, due to the idempotent law. However, setting B-FR and/or B-FS to 1.0 will cause both cutsets to be retained leading to a conservative RAW calculation.

Approach 2: In case each 'bin' is solved separately, the calculation of RAW for the SSC due to seismically-induced failures can be performed by determining the RAW for such failures (i.e., BEs) for each 'bin' and then weighing those RAWs with the ratio of the 'bin' SCDF to the total SCDF. Using the maximum of the resulting RAW and the RAW for BEs representing random failures can provide the representative value for RAW for the SSC from the SPRA. Mathematically, the operation can be expressed as,

$$RAW_i = Max\left[\left(1+\frac{\sum_k (RAW_{iSEIS,k}-1)\times SCDF_k}{\sum_k SCDF_k}\right), RAW_{iSRAND,j}\right] \quad (3)$$

where,
RAW_i = RAW for the ith component
k = number of seismic 'bins'
RAW_{iSEIS} = RAW for the seismic failure of the ith component
RAW_{iSRAND} = RAW for the random failure of the ith component
j = number of random failures
SCDF = seismic core damage frequency (from seismic and non-seismic failures)

Expanding the term in the first parenthesis in Equation (3) using the definition of FV results in,

$$1+\frac{\sum_k (RAW_{iSEIS,k}-1)\times SCDF_k}{\sum_k SCDF_k} = 1+\frac{1}{\sum_k SCDF_k} \times \sum_k \left[\left(\frac{F(1)_{iSEIS,k}}{SCDF_k}-1\right)\times SCDF_k\right]$$

$$= 1+\frac{\sum_k F(1)_{iSEIS,k}-\sum_k SCDF_k}{\sum_k SCDF_k} = \frac{\sum_k F(1)_{iSEIS,k}}{\sum_k SCDF_k} \quad (4)$$

Equation (4) translates to the definition of RAW and demonstrates that RAW is calculated correctly.

The ability of the above approaches to calculate a RAW for a SSC from the SPRA was investigated using SPAR models. Table I provides sample results from the investigation and demonstrates the ability of the approaches to determine representative RAW values for an SSC from the SPRA for use in 10 CFR 50.69 categorization.

III.C. Mapping of Basic Events

Due to the nature of a seismic event, the SPRAs include SSCs as well as failure modes for SSCs that are not present in the PRA models for internal hazards as well as internal fire. Examples of such SSCs and failure modes include passive components like tanks, explicit modeling of electrical relay failures, and seismically-induced (e.g., structural) failure of pumps. Such failure modes and SSCs that are unique to SPRAs need special attention in the context of 10 CFR 50.69categorization. In other PRAs such SSCs and/or failure modes would either be considered as part of the 'super-component' boundary (e.g. a valve relay is considered part of the valve boundary) and inherently accounted for in the failure probability and associated importance measures for the 'super-component'. If the SPRA modeled the relay separately, the importance measures for the relay should be considered as a failure mode for the 'super-component' and included in determining the integrated importance measures assessment for determining the category of the 'super-component'. Therefore, a detailed PRA basic event-to-component mapping is crucial to the categorization process.

Table I. Illustration of the use of two approaches to determine RAW for an SSC from SPRAs

Failure Mode	RAW	Failure Mode RAW (Eq. 4)
Seismically-Induced in Bin 1 (Bin CDF = /year)	0.0	1.60
Seismically-Induced in Bin 2 (Bin CDF = /year)	1.95	
Seismically-Induced in Bin 3 (Bin CDF = /year)	1.23	
Seismically-Induced in Bin 4 (Bin CDF = /year)	1.0	
Seismically-Induced in Bin 5 (Bin CDF = /year)	1.0	
Fails to Start	1.02	1.02
Fails to Run	1.02	1.02
Test and Maintenance Unavailability	1.02	1.02
Common Cause Failure to Start	1.79	1.79
Common Cause Failure to Run	1.89	1.89
Approach 1 (Eq. 3)		1.89
Approach 2 (All Failure Modes Set to 'TRUE')		1.89

However, instances will exist where such mapping cannot be performed due to the unique functions of the

SSCs specific to seismic events. If such SSCs are determined to be HSS from the SPRA they can either (1) be subjected to the integrated importance measure determination (the denominator will be to sum of all hazard CDF values), or (2) conservatively assumed to be HSS and presented as such to the IDP for categorization.

Based on the above discussion, it is clear that SPRAs provide unique insights in the form of seismic failure modes that are not captured by other PRAs as well as unique SSCs that are not present in other PRAs and cannot be mapped to SSCs in other PRAs. Case studies performed using eight SPAR models that included internal events, internal fire, and seismic PRAs, provide additional evidence of such insights (Ref. 8).

III.D. Dominant Risk Contributors

SPRAs provide important insights on the risk profile of NPPs for seismic events. While each NPP's seismic risk profile will be dependent on plant-specific information such as the site's hazard, the design margin against the re-evaluated hazard, and assumptions in the NPP's SPRA, comparison of dominant contributors provide information on the similarity and differences between the dominant contributors for different NPPs.

Loss-Of-Offsite-Power (LOSP) sequences tend to dominate the majority of current SPRAs contributors. This insight is unchanged from past SPRAs and is driven, as previously, by the low generic fragility value for offsite power (ceramic insulators) within and outside the NPP switchyard as well as the consequent assumption of non-recovery of offsite power. Station Blackout (SBO) sequences, due to their being a subset of the LOSP sequences, also appear as important contributors. Seismically induced failures of emergency diesel generators (EDGs) leading to SBO sequences occur for the higher acceleration bins due to the high fragility of those components. However, current SPRA results appear to show that plant improvements, driven by regulatory and voluntary reasons, have decreased the impact and relative contribution of the SBO sequences as compared to past SPRAs. Such improvements include the SBO rule (10 CFR 50.63), installation of alternate EDGs (e.g., SBO DG, Severe Accident Mitigation Alternative DG), and installation of low leakage reactor coolant pump (RCP) seals.

As a result, current SPRAs also include initiators such as anticipated transients without scram (ATWS), and loss-of-coolant accidents (LOCAs; primarily very small, small, and medium breaks) as dominant contributors. It is recognized that small break LOCAs were identified as one of the initiators for which a success path was to be established for plants performing seismic margins analysis (SMA) during the Individual Plant Examination for External Events (IPEEE). However, the SSCs that need to fail for ATWS, small, and medium LOCA initiators to become important usually have high fragilities (i.e., fails at high ground accelerations) and the mitigation equipment also has a high likelihood of failure at high ground accelerations.

III.E. Performance Monitoring After Categorization

10 CFR 50.69(e) requires performance monitoring and a feedback 'loop' to ensure that the change in "special treatment" due to the categorization of certain components does not impact the inputs used for the PRA models. Further, in case of seismic design basis, the rule allows demonstration of seismic qualification by means other than testing. The inputs used for SPRA models which are different from other PRAs are related to seismically-induced failures. Such failures are represented by the corresponding fragilities. These fragilities are developed using a variety of information, such as design documents, testing data, and the 'as-exists' configuration, which can be impacted by the change to the treatment, permitted by 10 CFR 50.69, for SSCs categorized as LSS. The guidance in NEI 00-04 recommends performing a 'risk sensitivity study' with the purpose of evaluating the impact of changes to the treatment of LSS SSCs. The 'risk sensitivity study' is performed by increasing the probability of random failures (e.g., failure to start, failure to run after starting etc.) by a factor of 3 to 5 with the multiplicative factor acting as a surrogate for increased unreliability due to changes to the treatment of LSS SSCs.

NEI 00-04 guidance does not explicitly the seismically-induced failures, which are distinct from the random failures, in the recommended 'risk sensitivity study'. In addition, reduction in seismic capacity by 3 to 5 times simply due to change in the "special treatment" is not expected based on existing information. Therefore, the impact of the change in the "special treatment" as well as the effectiveness of the feedback 'loop' for SSCs that are LSS due to seismically-induced failures based on SPRA results are not explicitly considered using quantitative approaches.

Programs and procedures at NPPs, in conjunction with the PRA configuration control process as well as the ability of PRAs to determine the impact of changes via quantitative sensitivity studies, provide mechanisms for the feedback 'loop' in such cases. Such programs include periodic degradation monitoring, aging management, and plant configuration management (also known as design change control). These programs remain unaffected by the 10 CFR 50.69 categorization of SSCs as well as the resulting change to "special treatment." Degradation monitoring and aging management can identify issues that

can impact the inputs and assumptions in the SPRA such as the occurrence of concrete cracking and missing anchorage. If such issues were not included in the original SPRA development, their impact can be determined via sensitivity studies, and a determination of whether or not the original categorization remains valid can be made and incorporated in the program.

NPP's design change process includes impact assessments and explicit inclusion of the seismic design as well as SPRA impact during such assessments provide another means of ensuring that the SPRA inputs and assumptions, and consequently, the categorization therefrom remains valid. For changes that directly impact the seismic design, the SPRA impact assessment can include the need for determination of the seismic capacity of the new SSC (which prompted the design change) and a sensitivity study using the SPRA. NPPs can seek to strengthen their existing processes to handle the use of SPRAs in 10 CFR 50.69 categorization and ensure that the requirements of 10 CFR 50.69(e) are met.

IV. INSIGHTS IN THE CONTEXT OF 10 CFR 50.69 FOR PLANTS WITH LOW SEISMIC HAZARD

NPPs that are sited in areas with low seismic hazard are perceived to have low seismic risk. However, none of the programs listed in the previous section to assess the impact of seismic events at NPPs have resulted in a quantitative determination of the perceived low seismic risk in the categorization process. Ref. 8 proposed a general framework, consistent with the guidance in NEI 00-04, for determining whether the impact of the seismic risk for plants with low seismic hazard need to be considered in the 10 CFR 50.69 categorization process.

The approach in Ref. 8 sought to demonstrate that the SSCs necessary to achieve safe shutdown during a seismic event had capacity equal to or exceeding a 'plant' fragility that resulted in the seismic CDF (SCDF) being less than 1% of the NPP's internal events CDF. In order to illustrate the approach, a NPP with a low seismic hazard was selected. A 'reduced scope' walkdown was considered sufficient for the NPP as part of the IPEEE effort which was indicative of its low seismic hazard. Following the approach in Ref. 8, a median 'plant' fragility was iteratively determined such that, when convolved with the mean re-evaluated hazard curve at PGA for the NPP, the resulting SCDF was 1% of the NPP's internal events CDF. The internal events CDF used for the example presented here was obtained from the plant-specific NRC Standardized Plant Analysis Risk (SPAR) model. A composite lognormal standard deviation of 0.4 was used from the GI-199 study for the NPP. The resulting 'plant' fragility was determined to be 0.9g.

An example of a success path for safe shutdown during a seismic event is provided in Figure 3-1 of EPRI report 1009648, "Methodology and Case Study for Use of Seismic Margin Assessments in Quantitative Risk-Informed Decision Making," (Ref. 9) which is reproduced in Figure I below. Information from NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," (Ref. 10) as well as ERPI report 1002989 "Seismic Probabilistic Risk Assessment Implementation Guide," (Ref. 11), can be used to show that the median fragility of the components in the success path shown in Figure 1 is below the 'plant' fragility determined (i.e., the failure probability will be lower than that determined by the 'plant' fragility). As a result, there can be reasonable confidence that the SCDF will be less than 1% of the internal events CDF. The reasonable confidence stems from the fact that only a small subset of the equipment available for mitigation is considered and the entire mean seismic hazard curve at PGA has been used. The use of the simple average of the seismic CDF at different frequencies instead of the PGA used in the illustration as well as limited plant-specific fragility evaluations instead of reliance on generic values will provide stronger basis for the argument presented here.

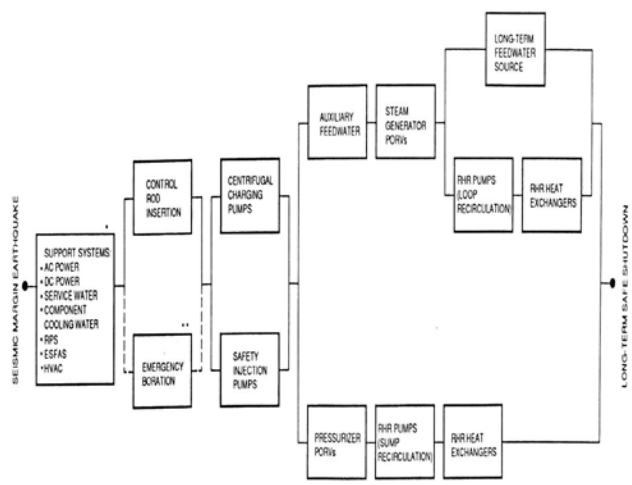

Figure I. Example success path for safe shutdown at a NPP during a seismic event (reproduced from Ref. 9)

V. CONCLUSIONS

The NRC staff performed detailed reviews of the technical acceptability of SPRAs for risk-informed decisionmaking and the implementation of those SPRAs by licensees in the categorization of SSCs per the requirements in 10 CFR 50.69. Insights gained from the NRC staff's review of 10 CFR 50.69 applications that include SPRAs were presented. Insights discussed include (1) fragility evaluation, (2) determination of importance

measures across seismic 'bins', (3) mapping of components between SPRAs and PRAs for other hazards, and (4) the relation between reduced regulatory requirements allowed by 10 CFR 50.69 and maintaining the quality of SPRAs. Refinements in fragility evaluations as well as approaches to demonstrate low seismic risk in the context of 10 CFR 50.69 and to calculate importance measures were developed and presented.

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