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SEISMIC PROBABILISTIC RISK ASSESSMENT OF NUCLEAR POWER PLANTS: 10 CFR 50.69 ASSUMPTIONS AND SOURCES OF UNCERTAINTY

Sara Lyons, Shilp Vasavada
U.S. Nuclear Regulatory Commission[1]
Rockville, MD, USA

ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) promulgated Part 50.69 to Title 10 of the Code of Federal Regulations (CFR), "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," in November 2004 (hereafter referred to as 10 CFR 50.69). The rule provides a voluntary alternative to compliance with many regulations which require "special treatment," or regulatory requirements which go beyond industrial controls, including: specific inspection, testing, qualification, and reporting requirements. The voluntary alternative includes a process for categorization of structures, systems, and components (SSCs) as having either low safety significance (LSS) or high safety significance (HSS). The categorization process can result in increased requirements for HSS SSCs which were previously treated as non-safety-related, and reduced requirements for LSS SSCs which were previously treated as safety-related.

The categorization process includes plant-specific risk analyses which are used in combination with an integrated decision-making panel (IDP) to determine whether the SSC has a low or high safety significance. Seismic probabilistic risk assessment (SPRA) is one of the risk analyses options to account for the seismic risk contribution. Because the 10 CFR 50.69 rule has currently not been implemented widely, the significance of various SPRA assumptions and sources of uncertainty to the categorization process has had limited evaluation for a broad spectrum of U.S. nuclear power plants.

This paper will assess the importance of certain aspects of the seismic risk contribution to the categorization process. NRC Standardized Plant Analysis Risk (SPAR) models will be used to perform sensitivity studies to quantify the impact of various assumptions and sources of uncertainty on the outcome of the categorization process.

BACKGROUND

Nuclear power plants (NPPs) in the United States have been designed to withstand credible natural and manmade hazards, including earthquakes. However, since the original design of many NPPs, the technical community's understanding of the seismic hazard in areas across the U.S. has continued to evolve such that seismic events in some regions of the country are now thought to be more likely. As a result, various efforts to assess the seismic risk of these plants have been undertaken, including:

- Unresolved Safety Issue A-46, "Seismic Qualification of Equipment in Operating Plants," 1980 [1],
- Individual Plant Examination of External Events (IPEEE), 1991 [2],
- Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States," 2005 [3], and
- Fukushima Dai-ichi Response, Near Term Task Force Recommendation 2.1, 2012 [4]

These efforts have resulted in an improved understanding of seismic risk at U.S. NPPs and a number of safety improvements. However, the seismic risk assessment methodologies used in response to these programs have varied and the assessments were not necessarily maintained beyond initial completion.

Some U.S. NPPs have developed high quality SPRAs and have chosen to use them to support the implementation of 10 CFR 50.69. This rule provides a voluntary alternative to compliance with many regulations which require "special treatment," or regulatory requirements which go beyond industrial controls, including specific inspection, testing, qualification, and reporting requirements. The voluntary alternative under 10 CFR 50.69 includes a process for categorization of SSCs as having either LSS or HSS. The categorization process can result in increased requirements for HSS SSCs where such requirements did not exist previously

[1] The views expressed herein are those of the authors and do not represent an official position of the U.S. NRC.

(termed non-safety related SSCs) and reduced requirements for LSS SSCs where such requirements existed previously (termed as safety-related SSCs).

The categorization process includes plant-specific risk analyses which are used in combination with an 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. 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. Because the 10 CFR 50.69 rule has currently not been implemented widely, the significance of various SPRA assumptions and sources of uncertainty to the categorization process has had limited evaluation for a broad spectrum of U.S. nuclear power plants.

This paper will assess the importance of certain aspects of the seismic risk contribution to the categorization process. NRC Standardized Plant Analysis Risk (SPAR) models will be used to perform sensitivity studies to quantify the impact of various assumptions and sources of uncertainty on the categorization process.

IMPORTANCE OF SEISMIC RISK TO 10 CFR 50.69

The importance of the seismic risk to the 10 CFR 50.69 categorization process can vary depending on several factors, including the geographic location, seismic hazard, design, construction codes, and vintage of the plant as well as the seismic risk relative to the risk from other hazards. In order to assess the impact of the seismic risk on the categorization process, 10 CFR 50.69 requires a systematic evaluation process and reasonable confidence that any potential increases in core damage frequency (CDF) or large early release frequency (LERF) are small. "Small" increases are discussed in Regulatory Guide (RG) 1.174 as resulting in a change in CDF (termed ΔCDF and read as delta CDF) between approximately 1E-6/year and 1E-5/year and change in LERF (ΔLERF) of between approximately 1E-7/year and 1E-6/year for NPPs that have a total CDF and LERF of less than approximately 1E-4/year and 1E-5/year, respectively. Based on these requirements, it may be inferred that seismic risk will be important to the categorization process for plants with a relatively high seismic CDF or seismic LERF. Such plants will likely need a high quality SPRA to support implementation of 10 CFR 50.69. Conversely, the seismic risk contribution may be negligible for those plants that are robust relative to the seismic hazard at their site, provided there is reasonable confidence that aspects that were used to determine that the as-built, as-operated plant is seismically robust are maintained.

An SPRA consists of three major parts: hazard analysis, fragilities evaluation, and plant response analysis. It is important to provide an overview of seismic fragility to support the discussion in the remainder of this paper. The seismic fragility of a SSC is the probability of failure of that SSC conditional on the seismic acceleration experienced by the SSC. The fragility of any SSC is usually expressed in the form of a family of fragility curves. The fragility curves are often described by a lognormal-lognormal distribution with parameters A_m (median capacity), β_R (randomness), and β_U (uncertainty). The lognormal standard deviation associated with randomness, β_R, is included to account for the aleatory uncertainty which is present due to the random nature of the seismic hazard and is described by the peak and valley variation of an actual earthquake. The lognormal standard deviation associated with uncertainty, β_U, is included to account for the epistemic uncertainty associated with the actual spectral shape of the earthquake as compared to the reference earthquake. Oftentimes, β_R and β_U are combined into a single composite lognormal standard deviation, β_C. In an SPRA, the plant response logic model is used to identify combinations of failures that may result in core damage and subsequent large early release of radionuclides. The risk of core damage and large early release is calculated by combining the appropriate fragility curves with the site hazard curves and plant response model. Both the fragility curves and site hazard curves must use the same reference earthquake intensity parameter, most commonly the peak ground acceleration (PGA) [5]. The physics of failure modeling approach is based on a stress-strength model where the SSC fails if the stress exceeds its capacity [6]. In SPRA fragility analysis, several variables are considered in estimating the seismic demand (i.e., stress) and capacity (i.e., strength) for various earthquakes at the PGA.

In this paper, nine NRC Standardized Plant Analysis Risk (SPAR) models will be used to explore the importance of seismic risk to the 10 CFR 50.69 categorization process. These models were developed to represent the as-built, as-operated plant, but have limitations with respect to plant representation and level of detail primarily due to the model update frequency. Furthermore, the SPAR models use generic seismic fragility values for SSCs and are not expected to be representative of refined plant-specific fragility calculations. Thus, the results of the SPAR models are used in this paper as examples to explore the importance of certain aspects of the seismic risk in the context of the categorization process. The results presented in this paper are not used and should not be used to draw definitive conclusions related to particular plants. For the purposes of this paper the risk contribution is considered from only internal events, internal fire, and seismic events as these hazards are generally expected to have the more significant effect on the categorization results. Figure 1 shows the relative risk contribution from each of these hazards for the nine example plants. These SPAR models were selected to represent a variety of NPPs, including pressurized and boiling water reactors, various containment designs, and some NPPs which are located in geographic regions which have had a significant increase in expected seismic hazard relative to that used for the plant's original design.

One method for performing categorization in accordance with 10 CFR 50.69 is described in Nuclear Energy Institute (NEI) Report 00-04, "10 CFR 50.69 SSC Categorization

Guideline" [8] which has been endorsed by the NRC in RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" [9]. NEI 00-04 allows for not considering the seismic risk in the categorization if that risk (expressed in seismic CDF) is less than 1% of the internal events plant risk (internal events CDF). Note that the CDF for Example Plant 9 is less than 1% of the total CDF. Some plants, such as Example Plant 9, which are expected to have a low seismic risk may not have a seismic margins analysis (SMA) or SPRA to support the categorization process. However, 10 CFR 50.69 does not restrict the evaluation approaches to SMA or SPRA. There can exist alternative systematic evaluations which may be sufficient to support 10 CFR 50.69 implementation. The systematic evaluation must recognize that the low hazard of the site does not directly translate into low seismic risk because the seismic design of the SSCs is based on the hazard. As a result, the convolution of the low hazard with SSC-specific fragilities may not result in low risk.

A possible semi-quantitative approach can be postulated by seeking the 'plant' level fragility (i.e., a fragility curve hypothesized to represent the failure of all basic events necessary to prevent core damage) that would result in the seismic CDF being less than 1% of the internal events CDF as follows:

- The target seismic CDF, 1% of the internal events CDF, is calculated and the plant-specific mean seismic hazard curve is known *a priori*.
- A composite lognormal standard deviation for the 'plant' level fragility curve can then be selected with supporting technical justification.
- A 'plant' level fragility can be found using the above information that, when convoluted with the plant-specific hazard curve, will result in the target seismic CDF.
- Limited fragility calculations, possibly for SSCs that were identified in historical evaluations as necessary for safe shutdown of the plant during seismic events, can be used to support the justification that the resulting 'plant' level fragility is equal to or above the calculated value.
- Sensitivity studies on the impact of the composite lognormal standard deviation can provide information about the variability in that parameter as well as the calculated 'plant' level fragility.

Regardless of whether or not an SPRA is used, information should be provided to the IDP with respect to the seismic hazard as well as the components necessary for safe shutdown under seismic events to inform their decision-making. The defense-in-depth aspect of categorization by the IDP is done relative to the design basis of the plant. The lack of an SPRA can result in a potential 'decoupling' between the current (and future) seismic hazard at the plant and the design basis seismic hazard. Therefore, it is important for the 10 CFR 50.69 implementation that the IDP is informed of the difference between the design basis hazard and the most recent seismic hazard so that the assumption of the robustness of the plant SSCs against the seismic hazard is re-evaluated, if necessary.

Due to the low seismic contribution, the results for Example Plant 9 are not expected to be meaningful for the evaluations in this paper and will not be considered further.

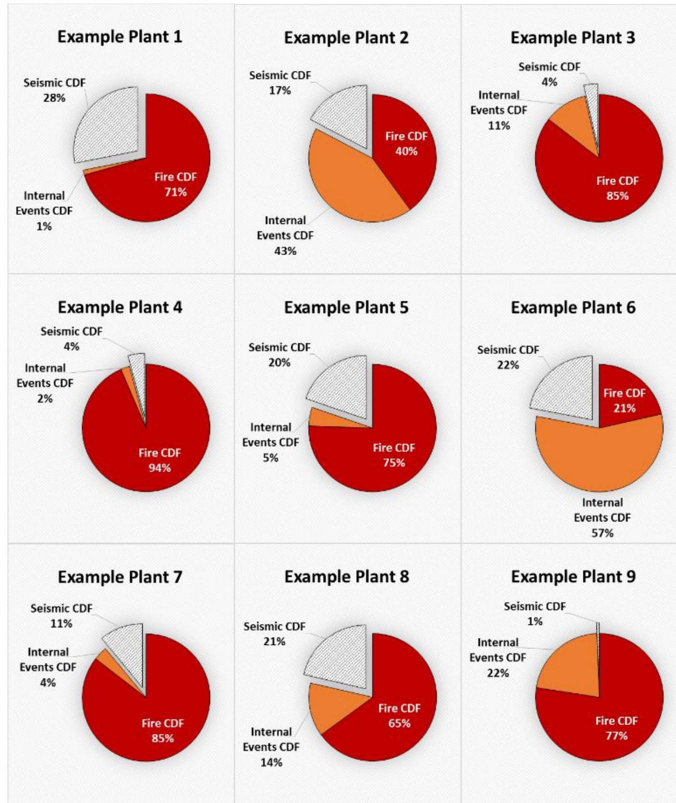

Figure 1. Example Plant Internal Event, Seismic, and Fire Risk Core Damage Frequency (CDF) Contribution

USE OF IMPORTANCE MEASURES FOR 10 CFR 50.69

The 50.69 categorization process described in NEI 00-04, as endorsed in RG 1.201, includes plant-specific risk analyses which are used in combination with the IDP. SSCs which meet certain criteria in the plant-specific risk analyses may not be re-categorized by the IDP, including SSCs which are identified as HSS by the integrated risk characterization portion of the process, the internal events PRA assessment, a non-PRA method to address external events or shutdown risk, or the defense-in-depth assessment.

The integrated risk characterization portion of the process identifies SSCs as HSS if the integrated importance measures meet the criteria to be determined HSS. That is, if the integrated Fussell-Vesely (F-V) importance measure is greater than 0.005, the integrated Risk Achievement Worth (RAW) is greater than 2, or the integrated RAW associated with common cause failure is greater than 20, the SSC will be categorized as HSS. NEI 00-04 states that the integrated importance measures for contribution to core damage frequency (CDF) are calculated per Equations (1) and (2), below, which are reproduced from NEI 00-04.

$$IFV_i = \frac{\sum_j (FV_{i,j} \times CDF_j)}{\sum_j CDF_j} \qquad (1)$$

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*

$$IRAW_i = 1 + \frac{\sum_j (RAW_{i,j} - 1) \times CDF_j}{\sum_j CDF_j} \qquad (2)$$

where,

$IRAW_i$ = Integrated RAW of component *i* over all CDF contributors

$RAW_{i,j}$ = RAW of component *i* for CDF contributor *j*

CDF_j = CDF of contributor *j*

The expressions in Equations (1) and (2) are also applicable for LERF related importance measures.

Under the process described in NEI 00-04, SSCs would be categorized as HSS based, in part, on the SPRA if (1) the integrated importance measure met the criteria identified above or (2) based on seismic PRA specific sensitivity analysis the IDP determined that the SSC would be categorized as HSS. There are other aspects of the NEI 00-04 process that would result in an SSC being identified as HSS, but those aspects would not necessarily be driven by the results of the seismic PRA.

EVALUATION APPROACH

There are several traditional engineering methods which provide the foundation and basis for the seismic PRA results. For example, the fragility curves for specific SSCs are based on the expected stress (seismic loading) and strength (capacity) of the component during a given seismic event. There are several inputs which are needed to support this analysis, including the probabilistic seismic hazard analysis, the spectral shape of the seismic event, the soil-structure interaction analysis, the propagation of the seismic load through any relevant structures, the analysis used to model the response of the SSC itself, and any correlation of the SSC with similar SSCs which are expected to be subject to similar loading during a given seismic event. As a result, many assumptions and sources of uncertainty can affect the importance measures associated with a specific basic event or SSC which may potentially impact the 10 CFR 50.69 categorization results. Similarly, the associated human reliability analyses, which considers the impact of the seismic event on NPP operator actions and responses, contain several assumptions and sources of uncertainty which may impact the 10 CFR 50.69 categorization results.

Eight SPAR models from the eight example plants which have a seismic risk contribution greater than 1% (Example Plants 1 – 8 in Figure 1) were selected to further understand the significance of potential assumptions and sources of uncertainty within the seismic PRA which may impact the categorization process. The following assessments were performed for all or a subset of these eight SPAR models:

- Case 1: Baseline Assessment
- Case 2: Use of truncated lognormal fragility curves
- Case 3: The probability of a loss of offsite power due to a seismic event was reduced
- Case 4: Select HEPs were decreased to guarantee success

The baseline assessment (Case 1) was performed to gauge the significance of the SPRA results in the context of the overall categorization results and to provide a control case for comparison purposes. The use of truncated lognormal fragility curves (Case 2) was evaluated as a potentially more realistic representation of the conditional failure of SSCs. The reduction in the probability of a loss of offsite power due to a seismic event (Case 3) was evaluated because it is generally a dominant contributor to seismic CDF and seismic LERF and its significance could be reduced at some NPPs in future years. The decrease in select HEPs (Case 4) was evaluated because the human reliability analysis can be significant to the SPRA results, but some currently used methods are simplified based on a bounding assessment. Cases 2, 3, and 4 represent areas of potential future refinement in the development and application of SPRAs.

The results of these sensitivity assessments are discussed below in the context of the impact of the seismic CDF on the categorization results. However, further study of the impact with respect to seismic LERF would be needed to reach a definitive conclusion. It has been observed from recent, high quality SPRA results of U.S. NPPs that seismic LERF can be a significant percentage of the seismic CDF due to dependencies associated with the seismic hazard.

CASE 1: BASELINE ASSESSMENT

The eight SPAR models were run to identify the SSCs that would likely be categorized as HSS based, in part, on the SPRA results using a methodology which is similar to the endorsed process described in NEI 00-04. This evaluation included solving the respective SPAR models, calculating the integrated importance measures for each basic event which was consistently defined across the hazards of interest, and evaluating the results. Basic events which did not meet the candidate HSS criteria based on SPRA results and those that could be determined not to meet the integrated importance measure results were excluded from further evaluation. The remaining results were divided into the following three groups:

- Result 1: Basic events that met the HSS criteria based on the integrated importance measures, and met the HSS criteria based on the internal events PRA model.
- Result 2: Basic events that met the HSS criteria based on the integrated importance measures, but did not meet the HSS criteria based on the internal events PRA model.
- Result 3: Basic events that met the candidate HSS criteria based on the SPRA results, but did not directly map to the fire and internal events PRA models for calculation of the integrated importance measures.

Of the SPAR models that were evaluated, many of the SPRA basic events which met the HSS criteria based on the integrated importance measure calculations also met the HSS criteria based on the internal events (IE) PRA model (Result 1). This result is not surprising because many of the same SSCs are relied upon to protect the plant, regardless of the hazard of concern. In the cases where the SSC would be found to be HSS based on the IE PRA results alone, the additional consideration of the SPRA results would offer minimal benefit if the process described in NEI 00-04 was used as the basis for 10 CFR 50.69 categorization, provided the seismic failure modes were also addressed by the internal events PRA categorization.

Four of the eight SPAR models had SPRA basic events which met the HSS criteria based on the integrated importance measure calculations but did not meet the HSS criteria based on the IE PRA model (Result 2). In many cases, the basic events that fell into this category appeared to align to SSCs which would have been categorized as HSS based on the IE PRA model had a full functional assessment and SSC mapping effort been completed. However, it is likely that the seismic failure modes and associated components (e.g., supports) would not have been appropriately considered in the absence of the SPRA results as would be necessary for accurate categorization results. The SPRA results appeared to influence the HSS determination for several component types, including: batteries, swing diesel generators, valves, tanks, and turbine driven pumps. This determination is expected to vary from plant to plant.

The standardized nature of the SPAR models allowed for simplified comparisons across hazards and between plants. Nonetheless, in some cases, components could not be directly mapped across hazards. All eight SPAR models had SPRA basic events that met the candidate HSS criteria, but did not directly map to the fire and internal events PRA models for calculation of the integrated importance measures (Result 3). The basic events which did not directly map to the fire and internal events PRA models were highly dependent on how the specific model was developed, and included:

- seismically-induced failure of buildings, containment, the polar crane, or cooling towers
- seismically-induced failure of RCS, vessel internals or secondary side SSCs
- seismically-induced loss of electrical equipment not modeled to a similar level of detail as in the associated internal events or fire PRA models
- failure of SSCs or failure modes not directly modeled in the associated internal events or fire PRA models: seismically-induced failure of heat exchangers, tanks, battery racks, walls, compressors, instrumentation, valves
- seismically-induced external events (e.g., dam failure)

Many of these basic events would have met the integrated importance measure HSS criteria, even if the contribution from other hazards was negligible.

CASE 2: TRUNCATED FRAGILITY CURVE

The fragility evaluation is a crucial aspect of SPRAs and can significantly influence the insights and metrics derived from an SPRA. As a result, the categorization under 10 CFR 50.69 using SPRAs will be influenced by such evaluations. Refinements in the fragility evaluation are sought in the development of SPRAs through the use of more detailed approaches and such refinements usually result in a decrease in seismic CDF.

In the context of the refinement in the fragility evaluations to support SPRA development and the corresponding impact on 10 CFR 50.69 categorization, it is conceivable that a truncated fragility curve for determining the plant-specific seismic CDF is used. The possibility of a fragility cutoff has been recognized in past studies, including NUREG/CR-4334, An Approach to the Quantification of Seismic Margins in Nuclear Power Plants [7]. In this study, the panel recognized that the conservative capacities are close to the lower-bound cutoff values below which there is no significant likelihood of failure. That study further stated that "although lower-bound capacity values have not been rigorously established, it is the belief of many engineers that lower bound capacity values do exist in the absence of major design and construction errors," and that "earthquakes below or near the [safe shutdown earthquake] SSE are found not to contribute significantly, which is not surprising in light of the generally conservative design practices used" [7]. Such an approach is customary in SMAs which do not require further consideration if the High Confidence of Low Probability of Failure (HCLPF[3]) capacity exceeds the seismic margin earthquake (SME; also known as the review level earthquake [RLE]) [7]. A similar screening approach was also used during the recent Expedited Seismic Evaluation Process (ESEP) which was initiated following the accident at Fukushima Dai-ichi NPP [11]. Therefore, the plant SSE, plant HCLPF value, or a fraction thereof can be considered to be a reasonable candidate for the

[3] HCLPF values represent the peak ground acceleration that corresponds to a 95% confidence level of a 5% or less probability of failure, which can be shown to correspond to a failure probability of 1% or less on the mean fragility curve. HCLPF values were originally developed using an expert elicitation approach and are currently more objectively defined in accordance with existing calculation procedures [7, 10].

truncation of the fragility curve for use in SPRAs if appropriately justified.

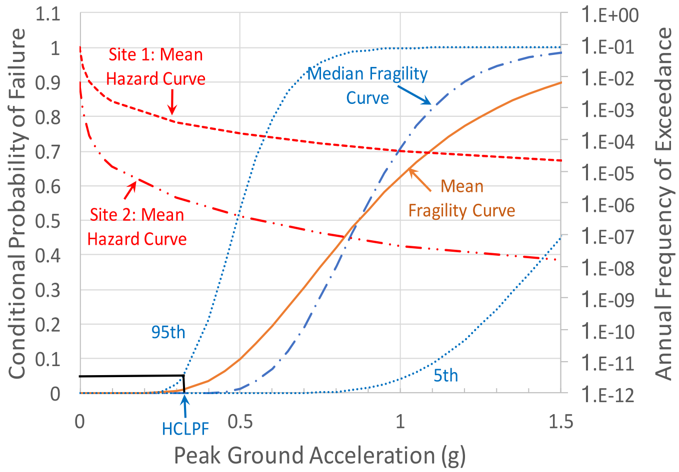

Figure 2. Notional Hazard and Fragility Curve

In Figure 2, example mean hazard curves for two sites and an example fragility curve is plotted. The hazard curve is plotted on a logarithmic scale because the annual exceedance frequency decreases significantly as the intensity of the earthquake increases. As a result of this relationship, the left tail of the fragility curve can contribute a significant portion of the resulting risk. According to a study by Shiu, up to 50% of the core damage frequency can be attributed to peak ground accelerations below the design level – or safe shutdown – earthquake [12]. This analytical result is contrary to the expected outcome because the SSCs are not expected to fail when exposed to earthquake events that are within their design capacity. The HCLPF is also indicated in Figure 2.

In order to gauge the potential of truncating the left tail of the fragility curve for use in an SPRA, an assessment of the potential risk contribution from the truncated portion has been performed. The results from the SPAR models discussed previously were used for the assessment. The intent of the assessment performed here is to illustrate the concept. Table 1 provides information on the SSE and the contribution of the seismic CDF at the SSE to the total calculated seismic CDF using the plant-specific SPAR model results. As noted from Table 1, certain plants, such as Plants 3, 5, and 7, have a relatively small contribution from accelerations at or below the SSE. However, the rest of the plants, notably Plant 4, have an appreciable contribution from acceleration at or below SSE to the total seismic CDF. Such an evaluation can be performed before developing an SPRA by using the available ‘plant’ level fragility curve, the plant-specific hazard curve, and the plant-specific SSE. The ‘plant’ level fragility curve and the plant-specific hazard curve can be combined using the convolution method to develop the seismic CDF as indicated in Equation 3 [13].

$$P_{f=} \int_0^\infty P_{(f|a)} \left(-\frac{dHa}{da}\right) da \qquad (3)$$

where,

$P_f =$ mean probability of failure

$P_{(f|a)} =$ conditional probability of failure from fragility curve with respect to PGA

$\frac{dHa}{da} =$ derivative of hazard curve with respect to PGA

The ASME/ANS PRA Standard [14] that is currently endorsed by NRC allows screening of sequences that contribute less than 1% to the hazard-specific CDF. Therefore, the SSE can represent a possible truncation level for the SPRA for a plant where an evaluation similar to that presented here shows a contribution that the PRA community considers as small. In such a case, the evaluation can also be extended to the plant HCLPF (or an intermediate value between the SSE and HCLPF) and a similar determination can be made. As noted previously, the numerical values presented in Table 1 and the above discussion are for illustration purposes only.

During a relatively low intensity seismic event, the dominant failure mode may be different than it was for the failures which were used to develop and validate the fragility curves. The introduction of new failure modes can be problematic for defining representative failure models. Since there is very limited test and experience data for failures which occur at PGA below the HCLPF value, the current fragility curve may not be representative of the true conditional probability of failure in this region [15].

Table 1. Illustration of Fragility Truncation

Example Plant #	SSE PGA (g)	Contribution of acceleration levels up to SSE to total seismic CDF (%)
1	0.25	7
2	0.12	5
3	0.15	1
4	0.2	38
5	0.12	<<0.1
6	0.15	2
7	0.2	<<0.1

It is possible that the left tail of the fragility curve may be dominated by low cycle fatigue or even random failures. It has been hypothesized that SSCs do not fail at earthquake intensities that are low relative to the component capacities [7, 10]. If this is the case, removing this failure contribution may result in a more realistic seismic risk profile.

Certain caveats on the above evaluation are in order:

- It needs to be stressed that any truncation, including the assessment described above, will represent an important assumption in the development of the SPRA model. It is normal and expected practice to confirm the validity of such assumptions prior to using the SPRA results.

- Because the technical community's understanding of the seismic hazard at a particular NPP may increase in the future, it may be worthwhile for plants with even moderate hazards to continue to refine and develop more realistic modeling approaches.

CASE 3: DECREASE FRAGILITY DRIVING LOSS OF OFFSITE POWER GIVEN SEISMIC EVENT

Three SPAR models (representing Example Plants 1, 2, and 7) were run with an improved LOOP fragility to identify whether the categorization results would likely change based in part on the SPRA results using a process which is similar to the endorsed process described in NEI 00-04. The improved LOOP fragility was developed by shifting the fragility curve such that the conditional failure probability associated with the Bin 1 hazard was assigned to Bin 2, the conditional failure probability associated with the Bin 2 hazard was assigned to Bin 3, etc. The new Bin 1 fragility was reduced by an order of magnitude from the original value. The results were evaluated using a methodology similar to that discussed in the baseline evaluation and compared to the baseline results.

Because importance measures are relative, shifting the LOOP fragility resulted in variations in the importance measures of many basic events. However, this sensitivity did not result in changes which appeared to significantly alter the categorization results. While there were some changes, as compared to the baseline evaluation with respect to the basic events which met the criteria of Result 1, 2, or 3, it did not appear the changes would result in significantly different categorization results if a full functional assessment and SSC mapping effort been completed. However, it was noted that a change in the categorization result would be possible for SSCs that were near the threshold values. Interestingly, the resulting seismic CDF was at least 95% of the baseline seismic CDF for all three models.

CASE 4: DECREASE OPERATOR RECOVERY ACTIONS GIVEN SEISMIC EVENT

The same three SPAR models that were used to evaluate Case 3 (representing Example Plants 1, 2, and 7) were run and all HEPs which represented failure to manually align and actuate were set to zero implying guaranteed success of those actions. The goal of this evaluation was to identify whether a more refined seismic human reliability analysis would likely change the categorization results based in part on the SPRA results using a process which is similar to the endorsed process described in NEI 00-04. This simplified assessment was intended to gauge the potential significance of the HEPs to the categorization results. It does not address the significant changes to the modeling logic which may occur as the HRA methodology is refined and additional SSCs are credited. The results were evaluated using a methodology similar to that discussed in the baseline evaluation and compared to the baseline results.

Similar to the results of Case 3, reducing the probability of a select class of HEPs resulted in variations in the importance measures of many other basic events. However, this sensitivity did not result in changes which appeared to significantly alter the categorization results. Again, it did not appear that the changes would result in significantly different categorization results if a full functional assessment and SSC mapping effort been completed. However, it was noted that a change in the categorization result would be possible for SSCs that were near the threshold values and may be likely in cases where the refined HRA results in different failure sequences.

CONCLUSIONS

10 CFR 50.69 requires a systematic evaluation process for categorization of SSCs based on their risk significance from different hazards including the seismic hazard. The importance of seismic risk to the 10 CFR 50.69 categorization process can vary based on various plant-specific attributes. A possible semi-quantitative approach was postulated and presented for plants which do not have a SMA or SPRA due to historical reasons. The postulated approach was formulated based on the endorsed guidance that allows screening of the seismic risk if the seismic CDF is less than 1% of the internal events CDF.

NRC SPAR models were used to explore the importance of seismic risk to the 10 CFR 50.69 categorization process. In order to further understand the significance of potential assumptions and sources of uncertainty within the SPRA which may impact the outcome of the categorization process, the following assessments were performed:

- Case 1: Baseline Assessment
- Case 2: Use of truncated lognormal fragility curves
- Case 3: The probability of a loss of offsite power due to a seismic event was reduced
- Case 4: Select HEPs were decreased to guarantee success

The baseline assessment (Case 1) was performed to gauge the significance of the SPRA results in the context of the overall categorization results and to provide a control case for comparison purposes. The use of truncated lognormal fragility curves (Case 2) was evaluated as a potentially more realistic representation of the conditional failure of SSCs. The reduction in the probability of a loss of offsite power due to a seismic event (Case 3) was evaluated because it is generally a dominant contributor to seismic CDF and seismic LERF and its significance could be reduced at some NPPs in future years. The decrease in select HEPs (Case 4) was evaluated because the human reliability analysis can be significant to the SPRA results, but some currently used methods are simplified based on a bounding assessment. Cases 2, 3, and 4 represent areas of potential future refinement in the development and application of SPRAs.

The baseline assessment showed that, for the SPAR models that were evaluated, many of the SPRA basic events which met the HSS criteria based on the integrated importance measure calculations also met the HSS criteria based on the IE PRA model. This result is not surprising because many of the same SSCs are relied upon to protect the plant, regardless of the hazard of concern. However, four of the eight SPAR models had SPRA basic events which met the HSS criteria based on the integrated importance measure calculations but did not meet the HSS criteria based on the IE PRA model. In many cases, the basic events that fell into this category appeared to align with SSCs which would have been categorized as HSS based on the IE PRA model if a full functional assessment and SSC mapping effort been completed. However, in such circumstances, it is important to ensure that seismic failure modes and associated components (e.g., supports) are appropriately considered in the absence of the seismic PRA results.

The truncated fragility curve assessment showed that truncating the fragility curve (i.e., not considering the structural failures below a certain threshold), may be technically defensible and may result in a reduction in calculated risk. Further, the comparison of the likelihood of structural failure as compared to random failures can also be used to support such an argument. Such an assumption would need to be adequately validated prior to use.

The assessments that varied the LOOP fragility and select HEPs demonstrated that, because importance measures are relative, variations in the importance measures of many other basic events were noted. However, the sensitivities did not result in changes which appeared to significantly alter the categorization results. It was noted that a change in the categorization result would be possible for SSCs that were near the threshold values.

The assessments in this paper were focused on the impacts on seismic CDF and the categorization therefrom. However, further study of the impact with respect to seismic LERF is needed. It has been observed from recent, high quality SPRA results of U.S. NPPs that seismic LERF can be a significant due to dependencies associated with the seismic hazard.

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