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OBSERVATIONS FROM RECENT SEISMIC PRAs IN USA

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ABSTRACT / BACKGROUND

Following the Fukushima Incident of 2011, the USNRC Near Term Task Force (NTTF) made several recommendations. Recommendation 2.1 dealt with seismic assessment of operating plants. Initially USNRC had binned the US operating plants (67 sites) into three categories [1]. The categorization was based primarily on: (a) the significance of the new seismic hazard (principally, the increase of new Ground Motion Response Spectra [GMRS] over the design basis Safe Shutdown Earthquake [SSE]); (b) a generalized estimate of each plant's Core Damage Frequency (CDF) for the new seismic hazard; and (c) corresponding estimation of each plant's conditional containment failure probability. Based on initial screening assessment using the new GMRS submitted by all plant licensees, USNRC required 21 plants to perform a SPRA. Subsequently, based on further evaluation, and due to a few plant closures (planned), only 15 of the initial 21 plants ultimately submitted SPRAs by (or before) the required submittal date of December 31, 2019. The primary author of this paper had been involved in ten of the 15 SPRAs, either as a consultant to the utility or as an independent peer reviewer. Based on this experience and assessment of the generally available information submitted to the USNRC, the authors have prepared several lessons learned to be shared with the international nuclear power plant (NPP) community. Upon initial preparation of this paper (November 2019), nine (9) plants had submitted their SPRAs and were made available through the USNRC Library [1]. Lessons learned from the initial 9 plants were presented at an international conference, ICAPP 2021 [2]. All 15 SPRAs have since been submitted to USNRC, and for the present paper the authors have updated their earlier lessons-learned study to address four more (of the additional six) SPRAs; whereas only two additional NPPs remain to be included, the lessons presented here are considered as substantially valid and useful.

The SPRAs address BWRs and PWRs designed and built in the 1960's through 1980's. They cover a diverse range of site conditions (soil and rock) and different geographic locations of the US (East, Southeast, Midwest and West). The SPRAs were performed following the requirements specified in the ASME/ANS Standard [3] or the Code Case 1 [4] to this standard, and the guidelines provided in the EPRI SPID document [5]. The plants in the central and eastern parts of the US follow the seismic source characterization laid out in NUREG-2115 [6]. The two SPRA plants in the Western US performed site-specific seismic hazard analyses following the SSHAC Level-3 process [7, 8, 9].

Observation from the thirteen (13) SPRAs considered in the present study are organized according to the major SPRA technical elements: (i) Seismic Hazard Analysis, (ii) Seismic Fragility Analysis, and (iii) Plant Systems Response Analysis.

SEISMIC HAZARD ANALYSIS

PSHA studies for the SPRAs of plants in the Western United States applied site-specific source characterization and ground-motion models (GMMs) developed using a SSHAC Level-3 process [7, 8, 9]. Some of the GMMs included the treatments of complicated fault geometries and multiple ruptures. The site-response analyses for all western US sites included detailed consideration of site-specific geophysical and geotechnical data, and relied on Appendix B of the SPID document [5] in cases where better data were unavailable.

PSHA studies for the SPRAs in the Central and Eastern United States (CEUS) relied heavily on existing regional seismic source characterization (SSCh) models and GMMs that had been developed over several years by large groups of experts, frequently using the SSHAC Level-3 process conducted under the sponsorship of owners and regulators. In particular, the CEUS-SSCh (DOE/EPRI/NRC [2013]) [6] regional SSCh model was used as a starting point for the source characterization, and the EPRI (2013) [10] GMM was used to calculate the associated shaking. Due to their cost, complexity, and need for extensive peer review, these studies are not updated on a frequent basis. On the other hand, the ASME/ANS Standards [3, 4] call for the use of up-to-date and site-specific information in the PSHA. These requirements in the Standard create the need for updates and refinements to these existing models in order to make them current and to add site-specific detail. The Standard does not contain detailed guidance regarding the extent of, and the process for, these updates and refinements, although some guidance exists elsewhere (e.g., NUREG-2117 [7], NUREG-2213 [8]). These updates and refinements were not always performed in a consistent manner, which generated questions during the peer reviews.

One of the issues requiring update was the hazard from human-induced earthquakes, which may be caused by hydrocarbon production, deep injection of fluids, or water reservoirs. These earthquakes were not included in the CEUS-SSCh study, and the techniques for modelling seismic hazard from future human-induced earthquakes are not as well developed as those for natural earthquakes. The various plants addressed these issues using different approaches. Several peer reviewer's questions related to the treatment of man-made earthquakes.

It is generally accepted that most of the epistemic uncertainty in PSHA results is due to the lack of knowledge of the GMMs. This is especially the case in the CEUS, where the rate of earthquakes is low and there are very few records from earthquakes of engineering interest. A pointed example of this high epistemic uncertainty is the difference between the EPRI (2013) [10] GMM, which was used for all the CEUS SPRAs discussed in this paper, and the more recent NGA-East SSHAC Level-3 GMM [11]. Preliminary results using these GMMs (e.g., Toro et al., 2019) [12] indicate significant differences in the calculated hazard and risk. These differences depend on geographic location, site conditions, and on the structural frequencies that dominate the plant response. Industry and the USNRC have been further investigating these differences (e.g., Talaat et al., 2021) [13].

Another issue requiring update was the earthquake catalog, as the CEUS-SSCh catalog includes only events through December 31, 2008. This issue was addressed by compiling updated catalog data using the same approach followed in the CEUS studies. Various approaches were then used to confirm that re-calculation of the CEUS-SSCh recurrence parameters was not necessary. In addition, literature reviews were conducted, and experts were contacted to identify new information about potential new seismic sources in the site region or changes to the parameters of existing sources.

Site response was treated using the approach in Appendix B of the SPID [5], which contains procedures for the incorporation of uncertainty when the geophysical and geotechnical data are limited.

Table 1 summarizes the seismic design and hazard results for the 13 plants considered in this paper. Not shown in the table is the large variation in the spectral shapes associated with the SSE and GMRS (details are found in the individual plant reports [1]). The variation in SSE spectral shapes is largely a result of the plant vintage. The variation in the GMRS shapes arises from the site conditions and the proximity to seismic sources capable of producing large earthquakes. Figure 1 compares the normalized GMRS shapes, as calculated by the Licensees, for three plants with different site conditions in the Southeastern US, as documented by references in [1]. These differences can be large and must be considered in the SPRA.

TABLE 1 – PLANT SEISMIC HAZARD PARAMETERS

Plant	1	2	3	4	5	6	7	8	9	10	11	12	13
SSE PGA (g)	0.18	0.20	0.16	0.15	0.18	0.12	0.125	0.30	0.40	0.20	0.25	0.20	0.20
GMRS PGA (g)	0.37	0.44	0.57	0.37	0.38	0.40	0.19	0.50	0.86	0.263	0.248	0.184	0.248
Soil/Rock	Shallow soil	Soil	Mostly Rock	Mostly Rock	Shallow soil	Shallow soil	Soil	Shallow Soil	Shallow soil	Soil	Soil	Mostly Rock	Shallow Soil

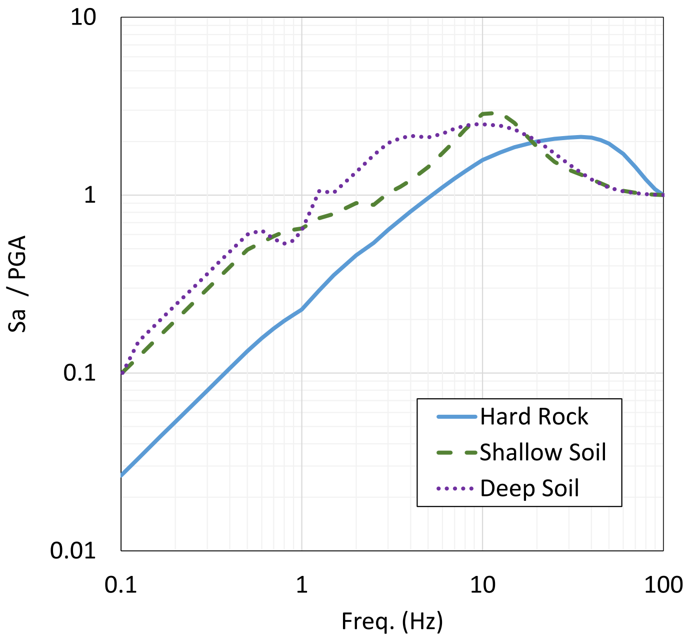

Figure 1. Normalized GMRS Spectra for Three US (South-East) Plants

Concerning seismic hazards, one area that generated many peer reviewer comments is the treatment of collateral or secondary hazards. This category of seismic hazards encompasses all geotechnical or hydrodynamic hazards caused by earthquakes other than direct shaking of safety-related structures or mechanical/electrical components in the plant. This category includes a number of physical processes, including liquefaction beneath safety-related structures (with the associated cyclic-mobility related settlements, lateral spreading, or loss of bearing capacity), dynamic bearing-capacity failure, failure of dams (either upstream dams that may cause flooding of the plant or downstream dams that may affect the integrity of cooling reservoirs), slope instability or cyclical-induced movement, fault displacement, tsunami, and seismic seiche. Generally, an attempt was made to screen out these hazards by performing a conservative (or bounding) analysis of the possible effects and/or estimating the annual probability of these events to be sufficiently low. A representative conservative (median-based) screening probability is 10^{-7} per year, although higher values may be justified on a plant-specific basis. If a hazard could not be screened out, its effects had to be included explicitly in the plant model. One general problem in performing these screening calculations (and in developing the associated fragility functions, if the secondary hazard could not be screened out) was that the probabilistic techniques to quantify each type of collateral hazard are typically not as well developed as for direct ground shaking. Also, many of the conventional geotechnical techniques for these hazards were developed for purposes of design and may include unclear levels of conservatism. In other cases, these techniques may not be properly calibrated or validated for the strong levels of shaking required for screening at 10^{-7} per year.

BUILDING AND SSI RESPONSE AND FRAGILITY ANALYSIS

Building Response Analysis

To obtain realistic demands for use in the fragility analysis, the mathematical models of a building should accurately represent the dynamic characteristics of the foundation soil profile, the building structure itself, and the components housed in them. Ideally from a structural modelling perspective, a 3D Finite Element Model (FEM) would be developed. Recognizing the time and cost involved, however, simpler Lumped-Mass Spring Models (LMSM) have also been used. For the 13 SPRAs reviewed for this paper, it was observed that all of them generally used existing LMSM (with justification) or newly developed 3D FEMs for vital structures, such as: Reactor and Containment Buildings, Auxiliary/Control Building, and Diesel Generator Buildings. For other structures, LMSMs were most often used. In a few cases, existing LMSMs were enhanced to consider potential torsional modes and flexibility of floor slabs (both in-plane and out-of-plane).

Tables 2 and 3 summarize the structural response and fragility parameters for all thirteen SPRAs. It is seen that all SPRAs accounted for SSI effects in the structural response analyses. This was typically done to consider the effects of ground motion incoherence, even for those structures founded on rock, which tends to lower the structural responses at higher frequencies (> 10 Hz).

TABLE 2 - PLANT STRUCTURAL RESPONSE AND FRAGILITY ANALYSIS (Plants 1 through 7)

Plant No. →	1	2	3	4	5	6	7
Ref. Earthquake (PGA or Freq)	GMRS (PGA)	1E-04 (PGA)	GMRS (PGA)	1E-05 (PGA)	GMRS (PGA)	GMRS (PGA)	GMRS (PGA)
Structural Models	FEM LMSM	FEM LMSM	FEM LMSM	LMSM FEM	FEM LMSM	FEM LMSM	FEM
SSI	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Concrete Cracking	Cracked	Partially Cracked	Partially Cracked	Partially Cracked	Partially Cracked	Cracked Uncracked	Cracked
Deterministic/ Probabilistic (No. of soil cases)	Determ (1)	Determ (3)	Prob (30)	Determ (3)	Prob (30)	Determ (3)	Determ (1)
No. of EQ Time Histories	1	5	30	1	30	5	1
No. of SEL Items	N/A*	N/A	800	N/A	1350	1300	900
CDFM or SoV	CDFM/ SoV	CDFM/ SoV	CDFM/ SoV	CDFM/ SoV	CDFM/ SoV	CDFM/ SoV	CDFM/ SoV

* N/A denotes Not Available

The Reference Earthquake (REs) used in the structural response analyses for the thirteen SPRAs is listed in Tables 2 and 3. These REs are conveyed as Uniform Hazard Response Spectra (UHRS) corresponding to a specific hazard level (typically in the range of 10-4 to 10-5 per yr). In some cases, it is seen that the RE is a scale factor times the 10-4/yr to 10-5/yr UHRS. Selection of an appropriate RE is critical to ensure that the fragilities of the risk-significant components are as realistic as possible, considering the spectral shape, and representative structural response and nonlinearity in soil response. The issue of suitable RE selection was identified and discussed in several SPRA peer reviews.

TABLE 3 - PLANT STRUCTURAL RESPONSE AND FRAGILITY ANALYSIS (Plants 8 through 13)

Plant No. →	8	9	10	11	12	13
Ref. Earthquake (PGA or Freq)	GMRS (PGA)	GMRS (5 Hz)	10-5 x 0.8* (PGA)	10-5 (2.5 Hz)	3xGMRS (PGA)	10-5 (PGA)
Structural Models	FEM LMSM	FEM	FEM LMSM	FEM LMSM	FEM	FEM LMSM
SSI	Yes	Yes	Yes	Yes	Yes	Yes
Concrete Cracking	Partially Cracked	Partially Cracked	Partially Cracked	Partially Cracked	Cracked	Partially Cracked
Deterministic/ Probabilistic (No. of soil cases)	Determ (3)	Prob (30)	Determ (3)	Prob (30)	Determ (5)	Determ (5)
No. of Eq. Time Histories	1	30	1	30	5	5
No. of SEL Items	750	1200	2000	1200	900	1670
CDFM or SoV	CDFM/ SoV	CDFM/ SoV	CDFM/ SoV	CDFM/ SoV	CDFM/ SoV	CDFM/ SoV

* 1×10^{-5} uniform hazard spectrum multiplied by 0.8

One source of structural nonlinear behavior addressed in all thirteen SPRAs was concrete cracking. The extent of cracking depends on the earthquake level. Three of the SPRAs assumed a fully cracked condition and ten of them considered a partially cracked condition. Partially cracked implies that some of the structural elements would be cracked while others would be uncracked. Appropriate stiffness and structural damping values were assumed for the cracked and uncracked elements. One plant performed the structural response analysis for both uncracked and cracked conditions. As the time and effort to perform structural response analyses is very significant, it is critical for resource efficiency that an appropriate level of cracking be estimated prior to performing the detailed analysis.

Fragility Analysis

Given that there are several hundreds to over a thousand components requiring fragility estimates, it would be impractical to do a detailed fragility analysis for all components in an SEL. Plants have thus taken a graded approach to perform the fragility analysis. Initially they used representative fragility values based on design-basis calculations, walkdown observations, experience data, and prior seismic margin evaluations (such as from IPEEE [Individual Plant Examination of External Events] studies). For those risk-significant components assessed from preliminary risk quantification analyses, the fragility values were determined using the CDFM approach. Based on the subsequent set of quantification analyses, a list of top risk contributors was developed. For those top contributors, more refined calculations using the SoV method were performed. The number of refined SoV calculations performed varied across the thirteen SPRA studies, from a few (3 to 4) to over 20.

The ASME PRA Standard [3], or the Code Case [4], does not specify how the fragility values are to be determined. It simply states that the fragility parameters for risk-significant components are to be calculated based on plant-specific data and that they are to be realistic. The Standard also states adequate justification should be provided if generic or conservatively based fragility values are used. Plants have generally substantiated the use of representative fragilities for risk-significant components using sensitivity analyses.

In addition to the fragility values, the controlling failure modes were also reported in the SPRA reports. The controlling failure modes typically include: anchorage, functional, structural, interaction (especially from non-safety masonry block walls), soil failures, etc. Of these failure modes, the most common ones are the anchorage and functional failure modes. For a few NPPs, especially regarding the evaluation of the seismic large early release frequency (LERF), soil failure modes (liquefaction, bearing capacity, etc.) and structural failure modes (failure of structural elements, building-to-building impact, etc.) were found to be important.

PLANT LOGIC ANALYSIS AND RISK QUANTIFICATION

The results of the plant response analysis for the thirteen NPPs are summarized in Tables 4 and 5. Table 4 includes the results for Plants 1 through 7 and Table 5 includes Plants 8 through 13. The plant response analysis and the quantification of seismic CDF and LERF generally involved Event Tree (ET) and Fault Tree (FT) analysis and in some cases the convolution of the seismic hazard and fragility values.

Plant Risk Results

An SPRA involves an iterative refinement process of feedback between the plant-logic model, quantification, and any indicated adjustments or refinements to the fragility analyses. Initial quantification of a plant systems model typically is based on the use of generic or representative fragility values for the components. In subsequent iterations, more realistic fragility values are calculated based on either the Hybrid (CDFM) or the SoV method, as mentioned above (§"Fragility Analysis"). It is generally the case that new top risk-significant components may be identified with the development of more realistic fragility values. The process is repeated until a stable risk profile is achieved. It was observed during the peer review of some of the SPRAs that the level of refinement in the fragility analysis approach varied, and some residual conservatisms were noted. Thus, the CDF and LERF values reported in Tables 3 may represent somewhat conservative estimates for some NPPs.

TABLE 4 - PLANT RESPONE ANALYSIS
(Plants 1 through 7)

Plant No. →	**1**	**2**	**3**	**4**	**5**	**6**	**7**
OL Issued	1996	1987	1978	1982	1980	1973	1976
Reactor Type	PWR	PWR	PWR	PWR	PWR	BWR	PWR
CDF	2.6E-06	2.8E-06	6.0E-05	4.0E-05	4.1E-06	2.1E-05	1.3E-05
LERF	1.7E-06	3.3E-07	1.6E-05	3.7E-06	2.6E-06	4.0E-06	6.1E-07
LERF/CDF	0.65	0.12	0.27	0.09	0.63	0.19	0.05
CDF – Top Contributors	-LOSP -SI Flood -FLEX DG -SS Blackout -125 V Batt Charger	-LOSP -125V DC Panel -CRDM -RCP -LOCA	-LOSP -VS-LOCA -480V Bus Relay Chatter -SWP Relay Chatter -SLOCA	-LOSP -Relay Group -VS-LOCA -Relay Group -SI Fire (Trans.)	-LOSP -Relay Chatter -RPV -VS-LOCA -120V AC Inverter	-LOSP -DC Batt -Hydro Plant -Elec Panels -Relay Gr	-LOSP -VS-LOCA -Demin. Water Tank -RWST -Transformer
LERF – Top Contributors	-LOSP -HRA Instr -SI Flood -Breaker Chatter - DG Block Walls	-LOCA -MLOCA -SLOCA -125V DC SGR -AC Inveter	-LOSP -S-LOCA -RS Pump Relay Chatter -RB Cont Bldg -Relay Chatter	-RB Isol. Valve -DC Battery -Relay Gr -120 V AC Dist Panel -Aux Bld Surrogate	-LOSP -Relay Chatter -480V Board -RPV -HRA MCR Instr	-LOSP -RPV Internals -DC Batt -Elec panels -Hydro Plant	-MS Isol Valve -LOSP -Turbine Bld -VS-LOCA - Demin. Water STank

TABLE 5 - PLANT RESPONE ANALYSIS
(Plants 8 through 13)

Plant No. →	8	9	10	11	12	13
OL Issued	1984	1984	1974	1984	1971	??
Reactor Type	PWR	PWR	PWR	BWR	BWR	BWR
CDF	5.6E-05	2.8E-05	2,4E-05	2.0E-05	5.8E-06	6.3E-06
LERF	3.41E-06	5.4E-06	5.7E-06	8.8E-06	2.9E-06	3.0E-06
LERF/CDF	0.07	0.19	0.23	0.44	0.05	0.48
CDF – Top Contributors	-LOSP - Service Water - Yard Transformer - Non-safety Components - 4.16kV Switchgear	-CST/RWST -Main Cntrl Bldg -Fire Water Tank -Aux Bldg -Process Control System	- LOSP -VS-LOCA - Relay Chatter - CR Ceiling - Battery Racks	- LOSP - Relay Chatter - RB/TB/Rad Bldg - MCC - RWCU	- LOSP - Control Panels - 125V Battery - RPV Internals - Instr. Racks	- LOSP - RHRSW Pumps - EECW Pumps - Unit Battery - 480V BD
LERF – Top Contributors	-Soil Failure -SG Supt - RB Penet. - LOSP - SI Flood	-Cont. Bldg. - SG - Fan Cooler - 125 V AC Panel - SSPS	- Aux. Bld - LOSP - Relay Chatter - Suppl. Diesel - SI Fire	- RB/TB/Rad Bldg - LOSP - Relay Chatter - MCC - RWCU	- LOSP - RPV Internals - 125V Battery - Instr. Racks - Control Panels	- LOSP - Init. Relays - Unit Battery - RPS Panels - EECW Pumps

It is seen in Tables 4 and 5 that the seismic CDF values for many of the plants are at about the 10^{-5}/r-yr level, whereas a few are at about the 10^{-6}/r-yr level. Given the evolution of NPP seismic design and plant configuration, one might expect the later vintage plants to have higher design margins as compared to the earlier vintage plants. However, the new seismic hazard of the plant site also plays a key part in the risk quantification. Thus, no direct correlation could be initially made between the seismic CDF or LERF and the vintage of the plant, whereas the authors intend to study and report various correlation insights in future work.

The ratio of (CDF/LERF) for the NPPs were computed and are also shown in Tables 4 and 5. It is seen that this ratio is the lowest (0.05) for Plants 7 and 12, and highest (0.65) for Plant 1. There are several reasons for such a broad range, including: plant specific differences (e.g., emergency response procedures, type of containment; type and number of containment penetrations and their possible sensitivity to relay chatter, among others); credit for any secondary containment; credit for operator actions; and so forth. Table 6 provides statistics for this ratio, derived from the preceding 13 values. Owing to the fact that this ratio, in general, represents a compounding of the various plant-specific safety barriers (to prevent large early release given core damage), its expected probability distribution is lognormal; the adequacy of the lognormal form was verified in this study using both method of moments and least squares fitting approaches.

TABLE 6 – (LERF/CDF) STATISTICS AND DISTRIBUTION

Mean (μ)	Median	σ	CoV=σ/μ	σ_{LN}	Distribution
0.27	0.19	0.21	0.81	0.91	Lognormal

As noted earlier, the SPRAs were prepared in response to the USNRC's NTTF 2.1 Recommendations and related request for information. For feasibility in resource allocation, SPRAs commonly employed some conservative and simplifying assumptions provided the risk values were still acceptable Some studies subsequently refined the assumptions, particularly in cases where the plant elected to appeal to available risk-informed guidelines for plant safety decisions. For the purpose of risk-informed design-basis development, USNRC R.G. 1.174 [14] provide some guidelines and threshold values for CDF (10^{-4}/r-yr) and LERF (10^{-5}/r-yr). Although this R.G. does not prescribe a fixed value for the LERF/CDF ratio, it is implied that the ratio would generally be about 0.1 ($10^{-5}/10^{-4}$). Such LERF/CDF ratio of 0.1 has historically been used based on internal events and other early SPRA studies. The present study has confirmed that (LERF/CDF) of approximately 0.1 roughly corresponds with the *mode* of a sensibly fitted lognormal distribution, as opposed to the mean or median. Based on the nature of the newly developed seismic hazard curves and the SPRAs performed, Table 6 suggests that the historic rule of thumb of (LERF/CDF)=0.10 may no longer be most valid as a guideline/approximating risk ratio in the context of SPRA. The authors suggest that the use of statistics in Table 6 provides an improved basis for such generic considerations and analysis as to risk-based containment performance; however, it is the authors' intent to develop (in a future paper) a further improved probabilistic characterization for this ratio in consideration also of the earlier SPRA results for US NPPs as well as (possibly) like results for NPPs in other countries.

Top Risk Contributors

Tables 4 and 5 list the top five (5) risk contributors to seismic CDF and LERF for each of the thirteen NPPs. The dominant risk contributors tend to be highly plant specific (and, considering the issue of potential residual conservatisms noted previously, perhaps study specific as well), as they are dependent on the site-specific seismic hazard and plant component fragility (capacity) data. However, some common actors are seen in the data and a brief related discussion is provided here.

As typical also of earlier SPRAs, LOSP (Loss of Offsite Power) appears as one of the top contributors to both seismic CDF and LERF in nearly all of the SPRAs. Though it is one of the top risk contributors, the fragility value (Am = 0.3g) used is based on seismic experience data from primarily non-nuclear facilities. The low fragility value is governed by the ceramic insulators used in the electrical distribution system; this observation has been well known throughout the history of NPP SPRAs.

The next common component that appears in several SPRAs as one of the top risk contributors is VSLOCA (Very Small Break LOCA). It is known that piping systems typically have high seismic capacity based on past seismic test data and earthquake experience data from non-nuclear facilities. Thus, one would not expect VSLOCA to be a top contributor. One of the reasons for this anomaly is the practical use of very conservative seismic capacity values in place of an expensive detailed fragility analysis that must include an extensive walkdown process to preclude failure modes associated with seismic interactions and other detailed plant-specific potential seismic vulnerabilities [15].

For similar reasons (i.e., the unavoidably low seismic capacity of ceramic insulators governing LOSP vulnerability and the typical cost-ineffectiveness of ruling out [or proving a high plant-specific capacity for] seismically induced small LOCAs), in the seismic margin assessment (SMA) methodology, selection of success paths for deriving plant-specific reliability insights typically assumes occurrence of LOSP and SLOCA. The simplified treatments of LOSP and SLOCA in both SPRA and SMA somewhat narrows the gap in practical risk/reliability resolution capability between the two approaches. As SMA develops plant-specific reliability models and insights beyond the assumption of LOSP and SLOCA, SPRA modeling and insights concerning key plant-specific risk contributors likewise extends beyond the reporting of LOSP and SLOCA as dominant risk contributors.

Relay groups also showed up as a top risk contributor for a few NPPs. Though relays may have been in some cases qualified based on functional plant-specific testing, the testing is usually at a level only somewhat above the required design basis earthquake levels and not at the higher earthquake levels causing failure. Most frequently, the broader database of experience data (a set of inferences from past earthquakes, and data from multiple testing programs) has been used and applied, in combination with application of spectral clipping and other intermediate capacity and demand factors in fragility analysis -- with one common approach being the use of GERS (generic equipment ruggedness spectra). In some cases a comparatively low relay capacity is owing to conservatism in the fragility approach for comparing the capacity and in-structure demand spectra (e.g., assuming the worst-case equipment frequency); in other cases of low relay capacity, the relay can be a known bad actor relay, yet conservatively taken (without further evaluation/verification) to be applied in a vulnerable configuration.

In a few plants, nuclear steam supply system (NSSS) components (reactor pressure vessel [RPV], steam generator [SG], etc.) showed up as the top risk contributors for both CDF and LERF. It was observed that, in the typical case, the structural modelling of the NSSS in the overall building dynamic models was generally based on simplified LMSMs dating back to the design-basis calculations. In addition, detailed fragility calculations of the NSSS components were not generated, as the details of the equipment were proprietary to the NSSS manufacturer. A few plants performed sensitivity analyses with assumed increases in the seismic capacity values of NSSS component, which then (as expected) showed reduction in CDF and LERF values.

There were several other classes of SEL items that showed up as top risk contributors: for example, battery and racks, instrument panel, switchgear, soil related failures, structural failures, etc. However, these tended to be very plant specific, with no clear trends indicated across the NPP fleet. Such observations are consistent with results of past SPRAs.

CONCLUSION

1. This paper presents results and general observations for thirteen of the recently completed SPRAs in the USA.

2. The level of refinement of the SPRAs has varied considerably and several sources of conservatism have been discussed.

3. Although the state of art of Seismic Hazard Analysis has steadily improved over some decades, there still exists large uncertainty in the ground motion models.

4. An important refinement in Structural Analysis included use of new 3D Finite Element models and implementation of SSI.

5. A graded approach (representative, hybrid based on CDFM and SoV) has continued to be applied in the SPRA Fragility Analyses.

6. Specific additional lessons and insights can be readily seen in the tables presented herein.

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