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10 CFR 50.54(f)

October 30, 2019

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

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Seismic Probabilistic Risk Assessment Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident

References:

1. Letter from E. J. Leeds (NRC) to all Licensees, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 12, 2012 (ML12053A340)

2. EPRI Report 1025287, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," dated November 27, 2012 (ML12333A170)

3. Letter from G. T. Kaegi (Exelon Generation Company, LLC (EGC)) to NRC, "Exelon Generation Company, LLC, Seismic Hazard and Screening Report (Central and Eastern United States (CEUS) Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 31, 2014 (ML14091A012)

4. Letter from T. Govan (NRC) to B. Hanson (EGC), "Dresden Nuclear Power Station, Units 2 and 3 - Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Reevaluations Relating to Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident (TAC Nos. MF3877 and MF3878)," dated April 27, 2015 (ML15097A519)

5. Letter from W. M. Dean (NRC) to Power Reactor Licensees, "Final Determination of Licensee Seismic Probabilistic Risk Assessments Under the Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendation 2.1 'Seismic' of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated October 27, 2015 (ML15194A015)

6. Letter from P. R. Simpson (EGC) to NRC, "Request for Extension of Due Date for Seismic Probabilistic Risk Assessment Submittal," dated May 14, 2018 (ML18134A224)

7. Letter from L. Lund (NRC) to B. C. Hanson (EGC), "Dresden Nuclear Power Station, Units 2 and 3 – Response to Request for Extension of Seismic Probabilistic Risk Assessment Submittal," dated September 17, 2018 (ML18236A262)

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued a request for information pursuant to 10CFR 50.54(f) associated with the recommendations of the Fukushima Near-Term Task Force (NTTF) (Reference 1). Enclosure 1 of Reference 1 requested each licensee to reevaluate the seismic hazards at their sites using present-day NRC requirements and guidance, and to identify actions taken or planned to address plant-specific vulnerabilities associated with the updated seismic hazards.

Reference 2 contains industry guidance developed by Electric Power Research Institute (EPRI) that provides the screening, prioritization and implementation details (SPID) for the resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic. The SPID (Reference 2) was used to compare the reevaluated seismic hazard to the design basis hazard. The Dresden Nuclear Power Station (DNPS), Units 2 and 3 reevaluated seismic hazard (Reference 3) concluded that the ground motion response spectrum (GMRS) exceeded the design basis seismic response spectrum in the 1 to 10 Hz range, and therefore a seismic probabilistic risk assessment was required.

Reference 4 contains the NRC Assessment of the DNPS, Units 2 and 3 seismic hazard submittal which concluded that the reevaluated seismic hazard prepared for DNPS, Units 2 and 3 is suitable for other activities associated with the NTTF Recommendation 2.1: Seismic.

Reference 5 provided the NRC final seismic hazard evaluation screening determination results and the associated schedules for submittal of the remaining seismic hazard evaluation activities for DNPS, Units 2 and 3. Reference 5 indicated that the DNPS, Units 2 and 3 Seismic Probabilistic Risk Assessment (SPRA) was expected to be submitted by June 30, 2019. In Reference 6, Exelon Generation Company, LLC requested an extension of the DNPS Units 2 and 3 SPRA submittal date to December 31, 2019. This extension request was approved by the NRC in Reference 7.

The enclosure to this letter contains the DNPS, Units 2 and 3 SPRA Summary Report which provides the information requested in Enclosure 1, Item (8) B. of the 10 CFR 50.54(f) letter. This letter completes the remaining actions associated with Regulatory Commitment No. 1 of Reference 3.

This letter contains no new regulatory commitments or revisions to existing regulatory commitments.

If you have any questions regarding this report, please contact Mitchel Mathews at (630) 657-2819.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 30th day of October 2019.

Respectfully,

Patrick R Simpson

Patrick R. Simpson
Sr. Manager Licensing
Exelon Generation Company, LLC

Enclosure: Dresden Nuclear Power Station, Units 2 and 3 Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic, dated October 2019

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station
NRC Project Manager, NRR – Dresden Nuclear Power Station
Mr. Milton Valentin-Olmeda, NRR/DLP/PBMB, NRC
Illinois Emergency Management Agency- Division of Nuclear Safety

ENCLOSURE

Dresden Nuclear Power Station, Units 2 and 3
Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic

October 2019

(206 Pages)

DRESDEN NUCLEAR POWER STATION (DRE) UNITS 2 AND 3 SEISMIC PROBABILISTIC RISK ASSESSMENT IN RESPONSE TO 50.54(F) LETTER WITH REGARD TO NTTF 2.1 SEISMIC

October 2019

DRE UNITS 2 AND 3 SEISMIC PROBABILISTIC RISK ASSESSMENT SUMMARY REPORT

Table of Contents

1. Purpose and Objective

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter on March 12, 2012 [1], requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance.

A comparison between the reevaluated seismic hazard and the design basis for Dresden Nuclear Power Station (DRE) has been performed, in accordance with the guidance in Electric Power Research Institute (EPRI) 1025287, "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" [2], and previously submitted to NRC [3]. That comparison concluded that the ground motion response spectrum (GMRS), which was developed based on the reevaluated seismic hazard, exceeds the design basis seismic response spectrum in the 1 to 10 Hz range, and a seismic risk assessment is required. A seismic PRA (SPRA) has been developed to perform the seismic risk assessment for DRE in response to the 50.54(f) letter, specifically Item (8) in Enclosure 1 of the 50.54(f) letter.

This report describes the seismic PRA developed for DRE and provides the information requested in item (8)(B) of Enclosure 1 of the 50.54(f) letter and in Section 6.8 of the SPID [2]. The SPRA model has been peer reviewed (as described in Appendix A) and found to be of appropriate scope and technical capability for use in assessing the seismic risk for DRE, identifying which structures, systems, and components (SSCs) are important to seismic risk, and describing plant-specific seismic issues and associated actions planned or taken in response to the 50.54(f) letter.

This report provides summary information regarding the SPRA as outlined in Section 2.

The level of detail provided in the report is intended to enable NRC to understand the inputs and methods used, the evaluations performed, and the decisions made as a result of the insights gained from the DRE seismic PRA.

2. Information Provided in This Report

The following information is requested in the 50.54(f) letter [1], Enclosure 1, "Requested Information" Section, paragraph (8)B, for plants performing an SPRA.

(1) The list of the significant contributors to SCDF for each seismic hazard interval, including importance measures (e.g., Fussell-Vesely)
(2) A summary of the methodologies used to estimate the SCDF and LERF, including the following:
    i. Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions
    ii. SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information
    iii. Seismic fragility parameters
    iv. Important findings from plant walkdowns and any corrective actions taken
    v. Process used in the seismic plant response analysis and quantification, including the specific adaptations made in the internal events PRA model to produce the seismic PRA model and their motivation
    vi. Assumptions about containment performance
(3) Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews
(4) Identified plant-specific vulnerabilities and actions that are planned or taken

Note that 50.54(f) letter Enclosure 1 paragraphs 1 through 6, regarding the seismic hazard evaluation reporting, also apply, but have been satisfied through the previously submitted DRE Seismic Hazard Submittal [3]. Further, 50.54(f) letter Enclosure 1 paragraph 9 requests information on the Spent Fuel Pool. This information was submitted separately [45].

Table 2-1 provides a cross-reference between the 50.54(f) reporting items noted above and the location in this report where the corresponding information is discussed.

The SPID [2] defines the principal parts of an SPRA, and the DRE SPRA has been developed and documented in accordance with the SPID. The main elements of the SPRA performed for DRE in response to the 50.54(f) Seismic letter correspond to those described in Section 6.1.1 of the SPID [2], i.e.:

- Seismic hazard analysis
- Seismic structure response and SSC fragility analysis
- Systems/accident sequence (seismic plant response) analysis
- Risk quantification

Table 2-2 provides a cross-reference between the reporting items noted in Section 6.8 of the SPID [2], other than those already listed in Table 2-1, and provides the location in this report where the corresponding information is discussed.

The DRE SPRA and associated documentation has been peer reviewed against the PRA Standard [4] in accordance with the process defined in NEI 12-13 [5], as documented in the DRE SPRA Peer Review Report [23]. The DRE SPRA, complete SPRA documentation, and details of the peer review are available for NRC review.

This submittal provides a summary of the SPRA development, results and insights, and the peer review process and results, sufficient to meet the 50.54(f) information request in a manner intended to enable NRC to understand and determine the validity of key input data and calculation models used, and to assess the sensitivity of the results to key aspects of the analysis.

The content of this report is organized as follows:

Section 3 provides information related to the DRE seismic hazard analysis.

Section 4 provides information related to the determination of seismic fragilities for DRE SSCs included in the seismic plant response.

Section 5 provides information regarding the plant seismic response model (seismic accident sequence model) and the quantification of results.

Section 6 summarizes the results and conclusions of the SPRA, including any identified plant seismic issues and actions taken or planned.

Section 7 provides references.

Section 8 provides a list of acronyms used.

Appendix A provides an assessment of SPRA Technical Adequacy for Response to NTTF 2.1 Seismic 50.54(f) Letter, including a summary of DRE SPRA peer review.

Table 2-1 Cross-Reference for 50.54(f) Enclosure 1 SPRA Reporting

50.54(f) Letter Reporting Item	Description	Location in this Report
1	List of the significant contributors to SCDF for each seismic acceleration hazard interval, including importance measures	Section 5
2	Summary of the methodologies used to estimate the SCDF and LERF	Sections 3, 4, 5
2i	Methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions	Section 4
2ii	SSC fragility values with reference to the method of seismic qualification, the dominant failure mode(s), and the source of information	Tables 5.4-2, 5.4-3, 5.5-2, and 5.5-3 provide fragilities (Am and beta), failure mode information, and method of determining fragilities for the top risk significant SSCs based on standard importance measures such as Fussell-Vesely (FV). Seismic qualification reference is not provided as it is not relevant to development of SPRA.
2iii	Seismic fragility parameters	Tables 5.4-2, 5.4-3, 5.5-2, and 5.5-3 provide fragility (Am and beta) information for the top risk significant SSCs based on standard importance measures such as FV.
2iv	Important findings from plant walkdowns and any corrective actions taken	Section 4.2 addresses walkdowns and walkdown insights.
2v	Process used in the seismic plant response analysis and quantification, including specific adaptations made in the internal events PRA model to produce the seismic PRA model and their motivation	Sections 5.1 and 5.2
2vi	Assumptions about containment performance	Sections 5.3, 5.4 and 5.5. Section 4.3 provides information related to dynamic analysis of the Reactor Building.

Table 2-1 Cross-Reference for 50.54(f) Enclosure 1 SPRA Reporting

50.54(f) Letter Reporting Item	Description	Location in this Report
3	Description of the process used to ensure that the SPRA is technically adequate, including the dates and findings of any peer reviews	App. A describes the assessment of SPRA technical adequacy for the 50.54(f) submittal and results of the SPRA peer review.
4	Identified plant-specific vulnerabilities and actions that are planned or taken	Section 6.

Table 2-2 Cross-Reference for Additional SPID Section 6.8 SPRA Reporting

SPID Section 6.8 Item [1] Description	Location in this Report
A report should be submitted to the NRC summarizing the SPRA inputs, methods, and results.	Entirety of the report addresses this.
The level of detail needed in the submittal should be sufficient to enable NRC to understand and determine the validity of all input data and calculation models used	Entirety of the report addresses this. The key methods of analysis and referenced codes and standards are identified in this report.
The level of detail needed in the submittal should be sufficient to assess the sensitivity of the results to all key aspects of the analysis	Entirety of the submittal addresses this. Sensitivities are discussed in the following sections: • 4.4 (SSC Fragility Analysis) • 5.7 (SPRA model sensitivities)
The level of detail needed in the submittal should be sufficient to make necessary regulatory decisions as a part of NTTF Phase 2 activities.	Entirety of the report addresses this.
It is not necessary to submit all of the SPRA documentation for such an NRC review. Relevant documentation should be cited in the submittal and be available for NRC review in easily retrievable form.	Entirety of report addresses this. This report summarizes important information from the SPRA, with detailed information in lower tier documentation.
Documentation criteria for a SPRA are identified throughout the ASME/ANS Standard [4]. Utilities are expected to retain that documentation consistent with the Standard.	This is an expectation relative to documentation of the SPRA that the utility retains to support application of the SPRA to risk-informed plant decision-making. This information has been retained and is available for NRC review.

Note (1): The items listed here do not include those designated in SPID Section 6.8 as “guidance”.

3. DRE Seismic Hazard and Plant Response

This section provides summary site information and pertinent features including location and site characterization. The subsections provide brief summaries of the site hazard and plant response characterization.

DRE is located 15 miles southwest of Joliet, Illinois, in the northeast quarter of the Morris 15-minute quadrangle, Goose Lake Township, Grundy County, adjacent to where the Des Plaines and Kankakee Rivers converge to form the Illinois River. The site is within the Central Stable Region of the North American Continent [6; 60]. The regional and local site geology is described in additional detail in DRE NTTF 2.1 Seismic Hazard submittal [3].

The site is located just west of the area where the Des Plaines and Kankakee Rivers flow together to form the Illinois River. The terrain is slightly hilly with a maximum relief at the site of about 25 feet. Regional relief is on the order of 200 feet. The site area is within the Central Lowland Physiographic Province [60].

A thin (less than 10-foot) mantle of soil, mostly glacial drift, overlies bedrock at the site. The upper unit of bedrock is the Spoon formation of the Pennsylvanian age (300 million years before present [MYBP]). The Spoon is sandstone that varies in thickness beneath the site from 0 to 45 feet. A thin soil horizon is present below the Spoon overlying rocks of the Upper Ordovician (450 to 430 MYBP) Marquoketa formation. The Marquoketa consists of a 20- to 45-foot thick upper limestone member, the Fort Atkinson limestone, and a 70-foot thick lower shale member, the Scales shale. Below the Marquoketa formation are approximately 1000 feet of limestone, dolomites, and sandstones ranging in age from Middle Ordovician (450 MYBP) to Cambrian (570 MYBP). These rocks lie on the Precambrian crystalline basement [60].

The Dresden site lies within the Central Stable Region of the North American Continent. This region extends from the Rocky Mountains to the Appalachian Plateaus and is relatively undeformed tectonically. It is characterized by a pattern of large basins, domes, and arches which formed throughout the Paleozoic Era (570 to 225 MYBP). The site is located on the northeast flank of one of these structures, the Illinois Basin. The north-northwest striking LaSalle anticlinal belt, a major structural element within the Illinois Basin, lies a few miles west of the site. The LaSalle anticline is a band of echelon folds which formed during the Mississippian and Pennsylvanian periods (345 to 280 MYBP). The northwest trending Kankakee Arch forms the northeastern boundary of the Illinois Basin and intersects the Wisconsin Arch to the North [60].

The Ground Motion Response Spectrum (GMRS) at DRE is defined at the foundation control point corresponding to EL 472.5 feet reference (re:) Mean Sea Level (MSL) [6]. The following two Foundation Input Response Spectra (FIRS) are developed for the structures as summarized [6]:

- *FIRS1* – corresponds to the soil column outcrop response at EL 472.5 feet re: MSL. This corresponds to foundation elevations for individual buildings included in the combined Reactor Building-Turbine Building (RB-TB) model varying in range from EL 463 feet re: MSL for the lower portion of the Turbine Buildings, up to EL 514.5

feet re: MSL for the control room structure. In addition, the foundation elevation for the majority of the Unit 2 and Unit 3 Reactor Buildings is at EL 472.5 re: MSL. FIRS1 is also used for Unit 2/3 Crib House. FIRS1 is considered as the GMRS and is referred to as GMRS/FIRS1.

- *FIRS2* – corresponds to the soil column surface response at EL 515 feet re: MSL. FIRS2 is equivalent to the previous GMRS developed by EPRI in the DRE NTTF 2.1 Seismic Hazard submittal [3]. This FIRS is used for any near surface founded structures (Isolation Condenser Pumphouse and Station Blackout Building). This FIRS is also considered appropriate for components credited in the SPRA that are not housed in plant structures, but rather located at grade in the yard.

The PSHA methodology used in this study allows for the explicit inclusion of epistemic uncertainty and aleatory variability in components of the model, including seismic source characterization and ground motion estimation. Uncertainties in models and parameters are incorporated into the PSHA through the use of logic trees. Because the sites are located on firm rock, a site-specific site response analysis was also performed to assess the effects on the probabilistic hard rock hazard results.

Additional site description and profile development are described in the DRE NTTF 2.1 Seismic Hazard submittal [3].

3.1 Seismic Hazard Analysis

This section discusses the seismic hazard methodology, presents the final seismic hazard results used in the SPRA, and discusses important assumptions and important sources of uncertainty.

The seismic hazard analysis determines the annual frequency of exceedance for selected ground motion parameters. The analysis involves use of earthquake source models, ground motion attenuation models, characterization of the site response (e.g. soil column), and accounts for the uncertainties and randomness of these parameters to arrive at the site seismic hazard. Detailed information regarding the DRE site hazard was provided to NRC in the seismic hazard information submitted to NRC [3] in response to the NTTF 2.1 Seismic information request [1]. As further discussed below, a supplemental seismic hazard analysis has been performed for DRE [6].

3.1.1 Seismic Hazard Analysis Methodology

A probabilistic seismic hazard analysis was performed [6] to support the DRE Seismic PRA in lieu of the NTTF 2.1 submittal [3] since the site analysis develops the additional elements required for the Seismic PRA such as FIRS, hazard-consistent strain-compatible properties, and vertical ground motions.

To perform the site response analyses for DRE, a random vibration theory approach was employed. This process is consistent with existing NRC guidance and the SPID [2]. The guidance contained in Appendix B of the SPID [2] on incorporating epistemic uncertainty in shear-wave velocities, non-linear dynamic properties and source spectra was followed for DRE in addition to development of High Frequency (HF) and Low Frequency (LF) controlling earthquakes (control motions) per recommendations in Regulatory Guide 1.208 [61] for mean annual frequency of exceedance (MAFE) corresponding to 1E-02, 1E-03, 1E-04, 1E-05, and 1E-06.

The GMRS at DRE is defined as the soil column outcrop response at EL 472.5 feet re: MSL [6]. FIRS were developed for additional structures at the elevations described in Section 3.0.

As discussed in the NTTF 2.1 Seismic Hazard submittal [3], to accommodate uncertainty in depth to hard rock (Precambrian basement) two depths were considered: 1,000 feet (305 meters) randomized ± 300 feet (92 meters) (Profile [P]1, P2, and P3) and 5,000 feet (1,525 meters) randomized ± 1,500 feet (460 meters) (P4, P5, and P6). The depth randomization reflects ± 30% of the depth and was included to provide a realistic broadening of the fundamental resonance at deep sites rather than reflect actual random variations to basement shear-wave velocities across a footprint. The best estimate shear wave velocity profiles were identical to the NTTF 2.1 Seismic Hazard submittal [3] shear wave velocity profiles with the epistemic uncertainty represented by the six shear wave velocity profiles and are presented in Figures 3.1.1-1(a), 3.1.1-1(b), 3.1.1-2(a), and 3.1.1-2(b) [6].

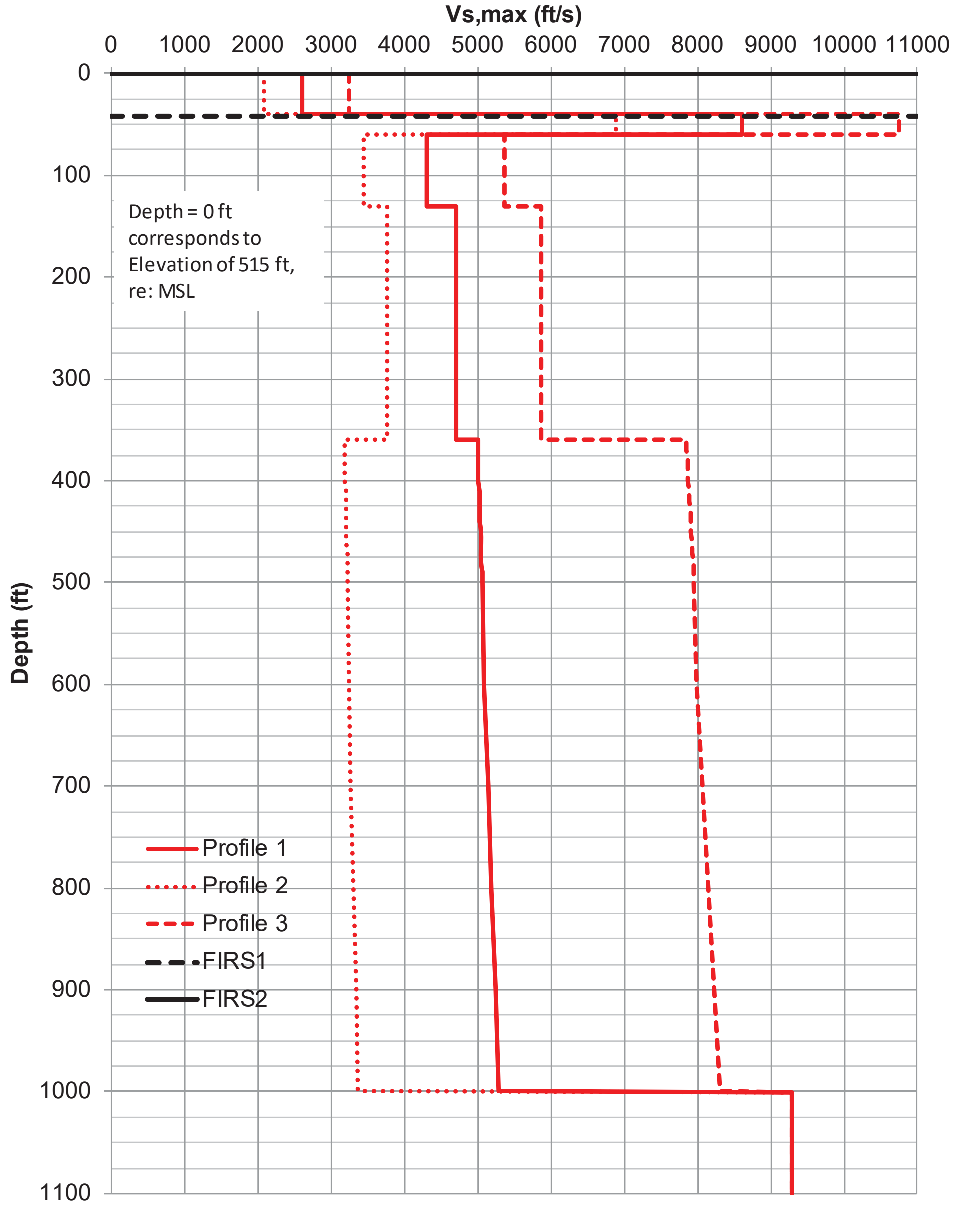

Figure 3.1.1-1(a) Idealized Shear Wave Velocity (Vs) Profiles for Profiles 1, 2, and 3 Representing Epistemic Uncertainty

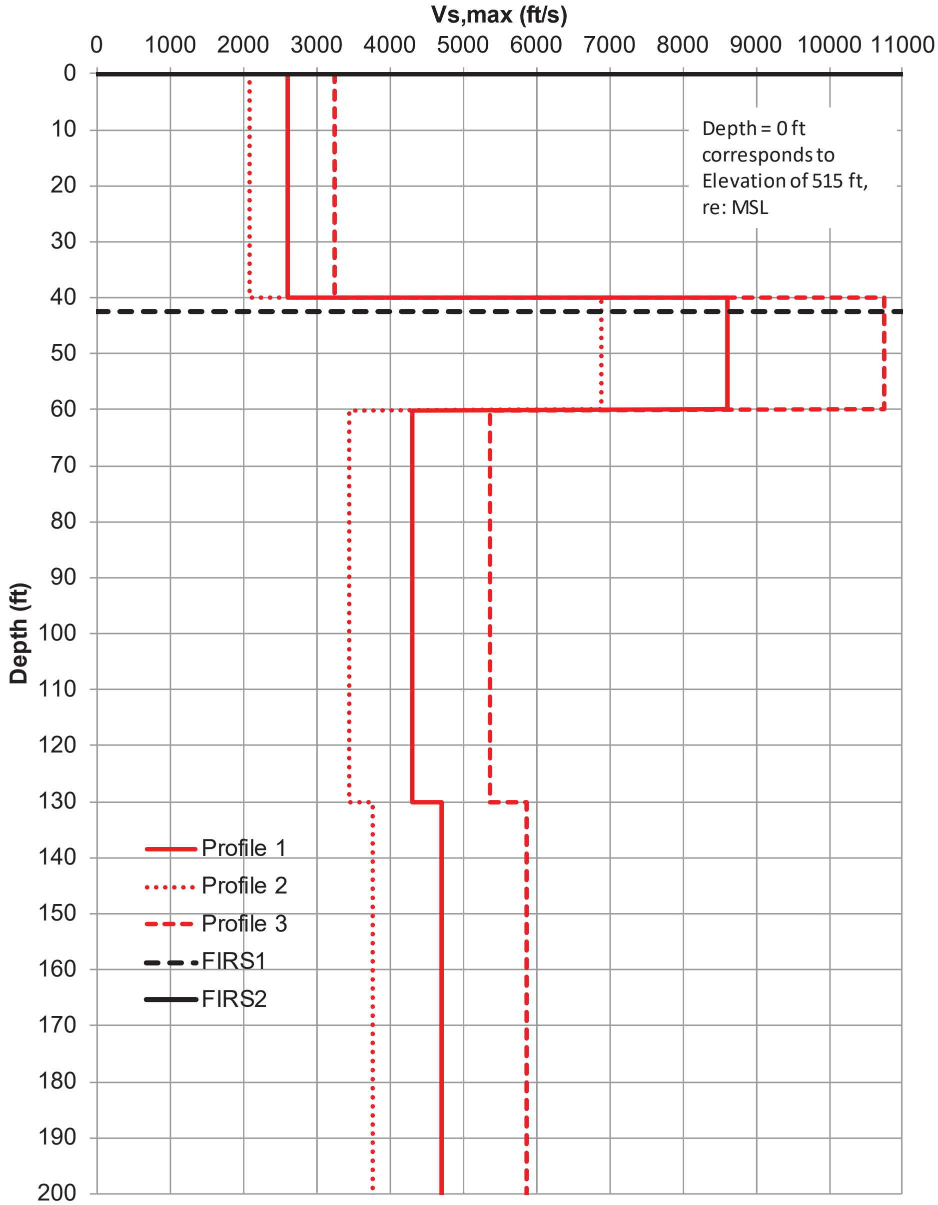

Figure 3.1.1-1(b) Idealized Shear Wave Velocity (Vs) Profiles for Profiles 1, 2, and 3 Representing Epistemic Uncertainty (Top 200 ft)

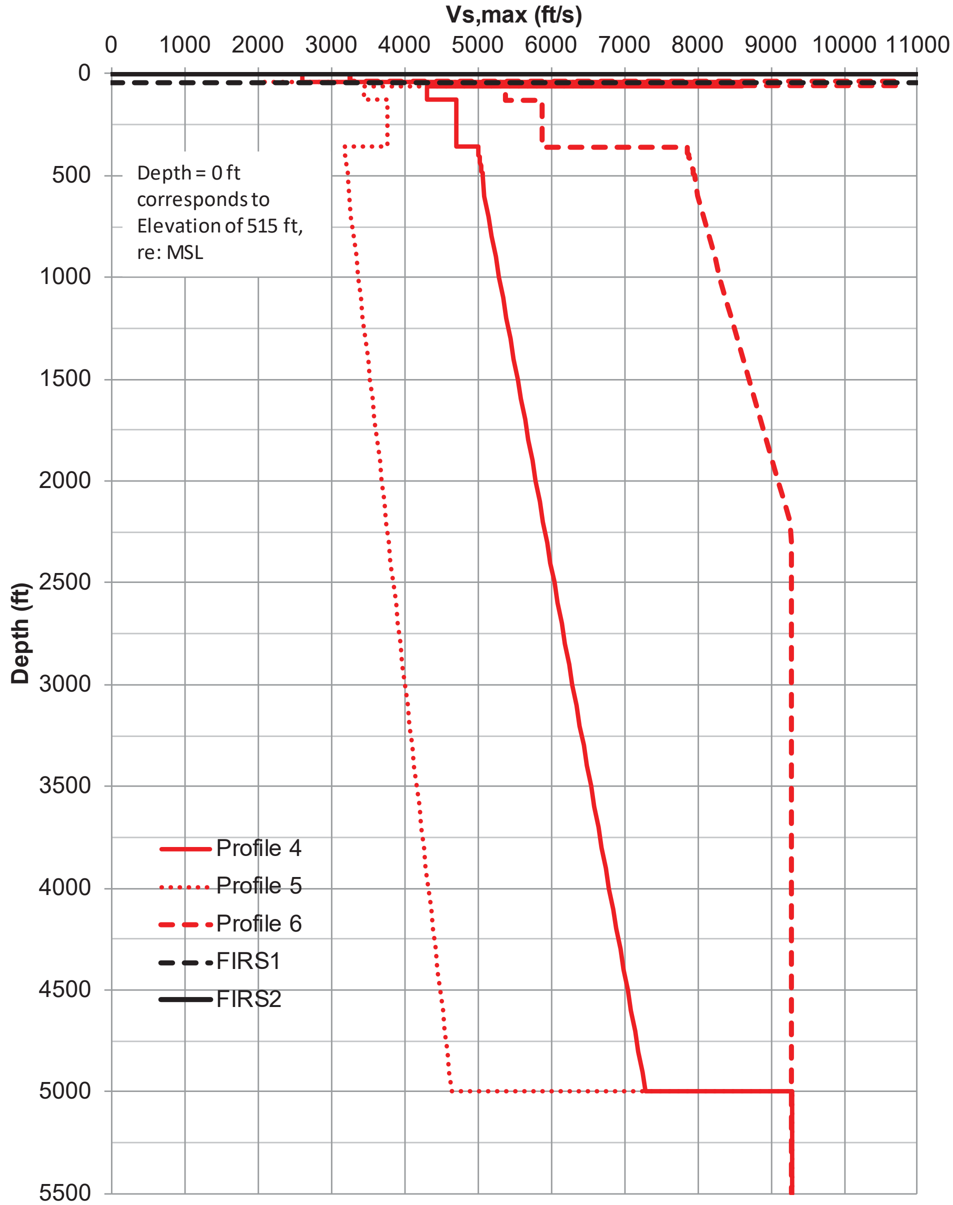

Figure 3.1.1-2(a) Idealized Shear Wave Velocity (Vs) Profiles for Profiles 4, 5, and 6 Representing Epistemic Uncertainty

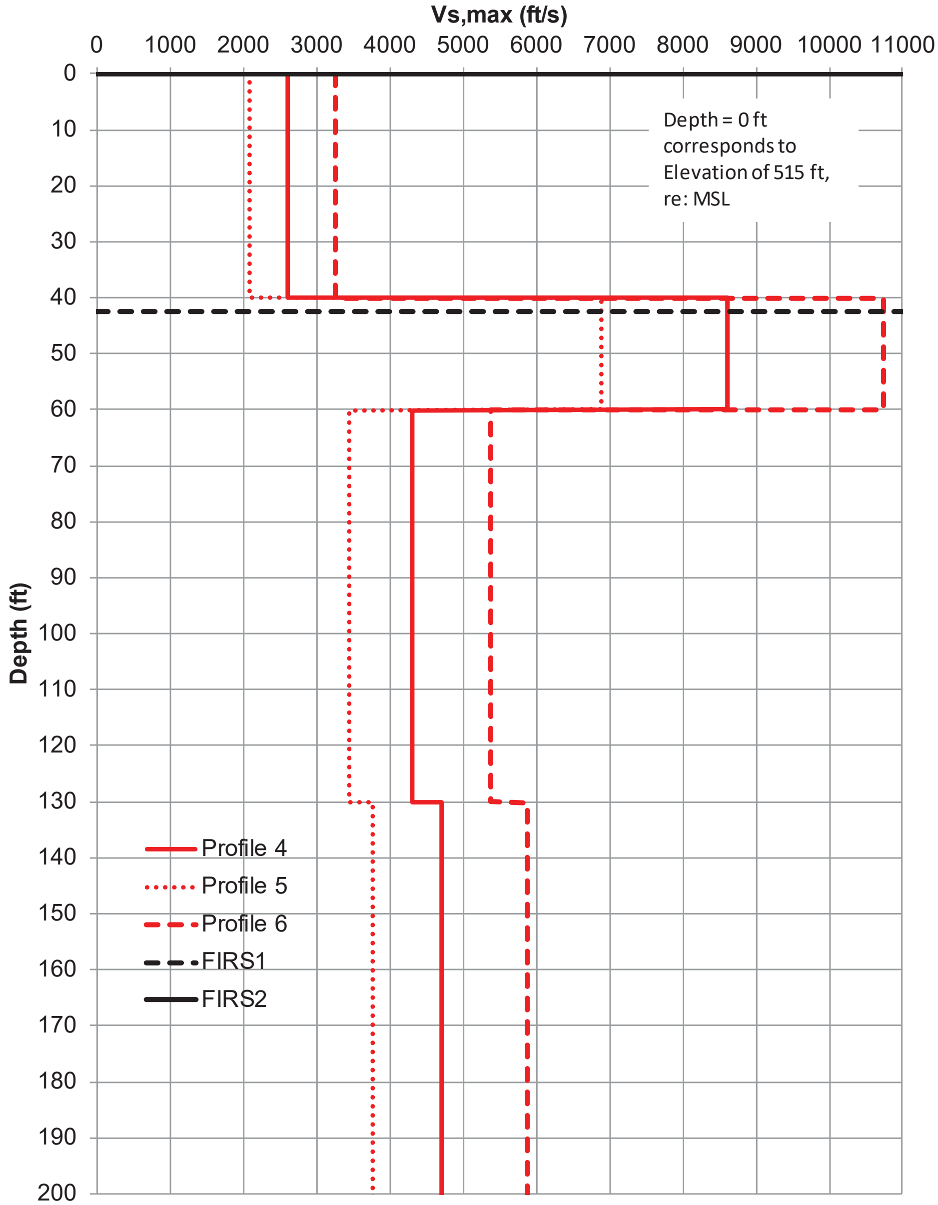

Figure 3.1.1-2(b) Idealized Shear Wave Velocity (Vs) Profiles for Profiles 4, 5, and 6 Representing Epistemic Uncertainty (Top 200 ft)

Consistent with NTTF 2.1 Seismic Hazard submittal [3], the standard deviation of the natural logarithm of Vs is 0.25 over the upper 50 feet (15 meters) and 0.15 below that depth with correlation coefficients between the natural logarithm of Vs of adjacent layers implemented consistent with the subsurface stratigraphy [6].

To accommodate the full range in expected dynamic material behavior for the firm rock profiles, linear analyses, as well as nonlinear analyses, were included in the site response analyses, with equal weights given to each approach. This approach is consistent with the approach of the NTTF 2.1 Seismic Hazard submittal [3].

The results of the site response analyses consist of amplification factors which describe the amplification (or de-amplification) of hard reference rock motion as a function of frequency and input reference rock amplitude. The amplification factors are presented in terms of a median amplification value and an associated standard deviation (sigma) for each oscillator frequency and input rock amplitude. Consistent with the SPID [2], a minimum median amplification value of 0.5 was employed in the present analysis. Table 3.1.1-1 and Figure 3.1.1-3 present the mean and fractile exceedance frequencies for hard reference rock (shear wave velocity equal to or greater than 9,200 feet per second) at 100 Hz [6]. Sample amplification factors are presented in Figure 3.1.1-4 [6].

Table 3.1.1-1 DRE Mean and Fractile Exceedance Frequencies – Hard Reference Rock PGA (100 Hz)

Amplitude (g)	Mean	Exceedance Frequency				
		0.05	0.16	0.50	0.84	0.95
0.0001	1.687E-01	6.995E-02	1.087E-01	1.748E-01	2.215E-01	2.624E-01
0.00025	1.166E-01	4.310E-02	7.414E-02	1.171E-01	1.568E-01	1.896E-01
0.0005	7.951E-02	2.690E-02	4.791E-02	7.716E-02	1.126E-01	1.361E-01
0.00075	6.056E-02	1.959E-02	3.591E-02	5.828E-02	8.778E-02	1.092E-01
0.001	4.878E-02	1.539E-02	2.782E-02	4.606E-02	7.148E-02	9.099E-02
0.0015	3.489E-02	1.068E-02	1.899E-02	3.247E-02	5.125E-02	7.065E-02
0.002	2.702E-02	8.203E-03	1.421E-02	2.468E-02	4.013E-02	5.706E-02
0.003	1.850E-02	5.541E-03	9.149E-03	1.605E-02	2.727E-02	4.188E-02
0.005	1.126E-02	3.164E-03	5.136E-03	9.361E-03	1.700E-02	2.714E-02
0.0075	7.425E-03	1.995E-03	3.081E-03	5.964E-03	1.103E-02	1.920E-02
0.01	5.406E-03	1.444E-03	2.077E-03	4.146E-03	7.928E-03	1.517E-02
0.015	3.302E-03	8.286E-04	1.213E-03	2.360E-03	4.909E-03	9.913E-03
0.02	2.253E-03	5.431E-04	8.025E-04	1.498E-03	3.378E-03	7.064E-03
0.03	1.276E-03	2.816E-04	4.070E-04	8.032E-04	1.901E-03	4.015E-03
0.05	6.082E-04	1.208E-04	1.823E-04	3.832E-04	8.522E-04	1.846E-03
0.075	3.317E-04	6.436E-05	9.216E-05	2.106E-04	4.769E-04	9.750E-04
0.1	2.138E-04	4.066E-05	5.924E-05	1.353E-04	3.144E-04	6.208E-04
0.15	1.129E-04	2.057E-05	3.142E-05	7.136E-05	1.701E-04	3.376E-04
0.2	7.010E-05	1.245E-05	1.901E-05	4.401E-05	1.057E-04	2.145E-04
0.3	3.420E-05	5.234E-06	9.057E-06	2.128E-05	5.218E-05	1.072E-04
0.5	1.255E-05	1.607E-06	3.086E-06	7.856E-06	2.091E-05	4.041E-05
0.75	5.148E-06	5.398E-07	1.141E-06	3.095E-06	8.849E-06	1.704E-05
1	2.578E-06	2.146E-07	5.124E-07	1.471E-06	4.343E-06	8.658E-06
1.5	8.846E-07	4.943E-08	1.367E-07	5.070E-07	1.461E-06	3.048E-06
2	3.843E-07	1.406E-08	4.632E-08	2.079E-07	6.481E-07	1.360E-06
3	1.055E-07	7.517E-10	7.385E-09	4.675E-08	1.691E-07	4.164E-07
5	1.664E-08	6.706E-27	3.199E-10	5.529E-09	2.456E-08	7.879E-08
7.5	3.195E-09	2.598E-29	4.635E-27	8.410E-10	4.308E-09	1.721E-08
10	8.996E-10	1.706E-29	2.798E-29	1.860E-10	1.153E-09	4.959E-09

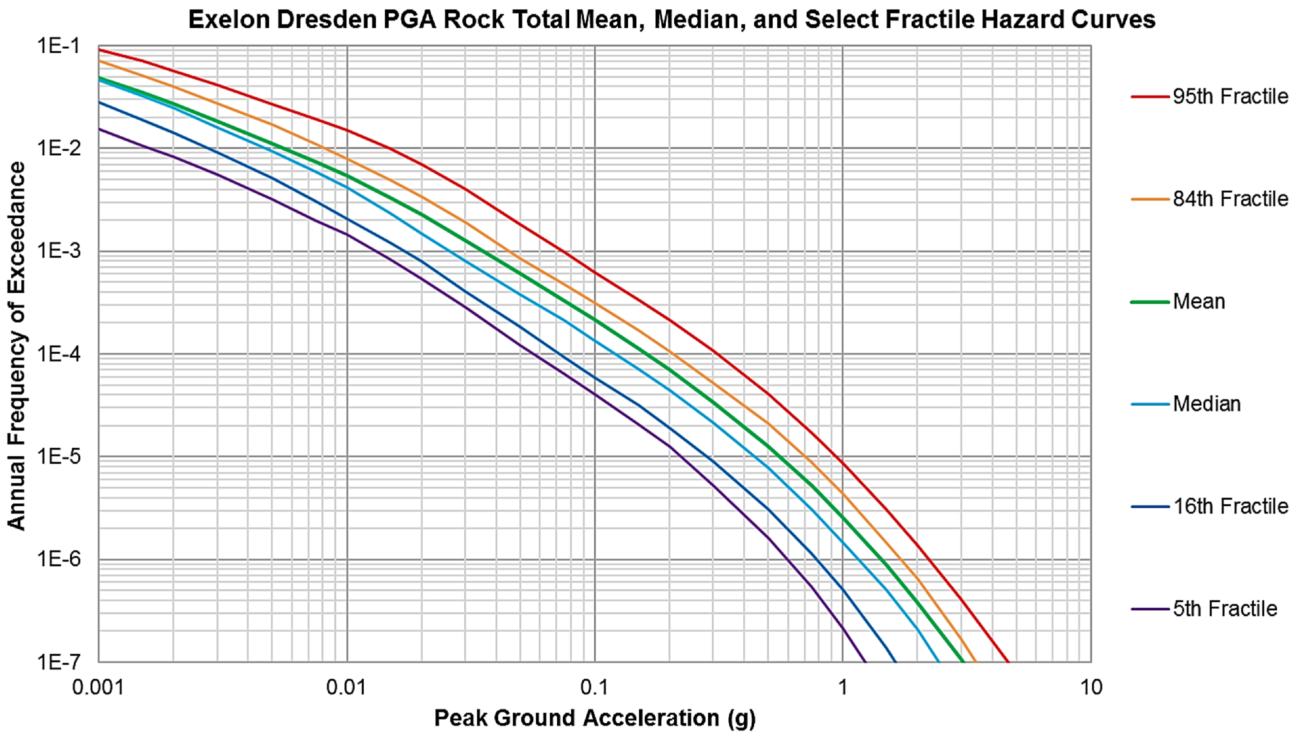

Figure 3.1.1-3 PGA (100 Hz) Fractile Hazard Curves for DRE (Hard Reference Rock)

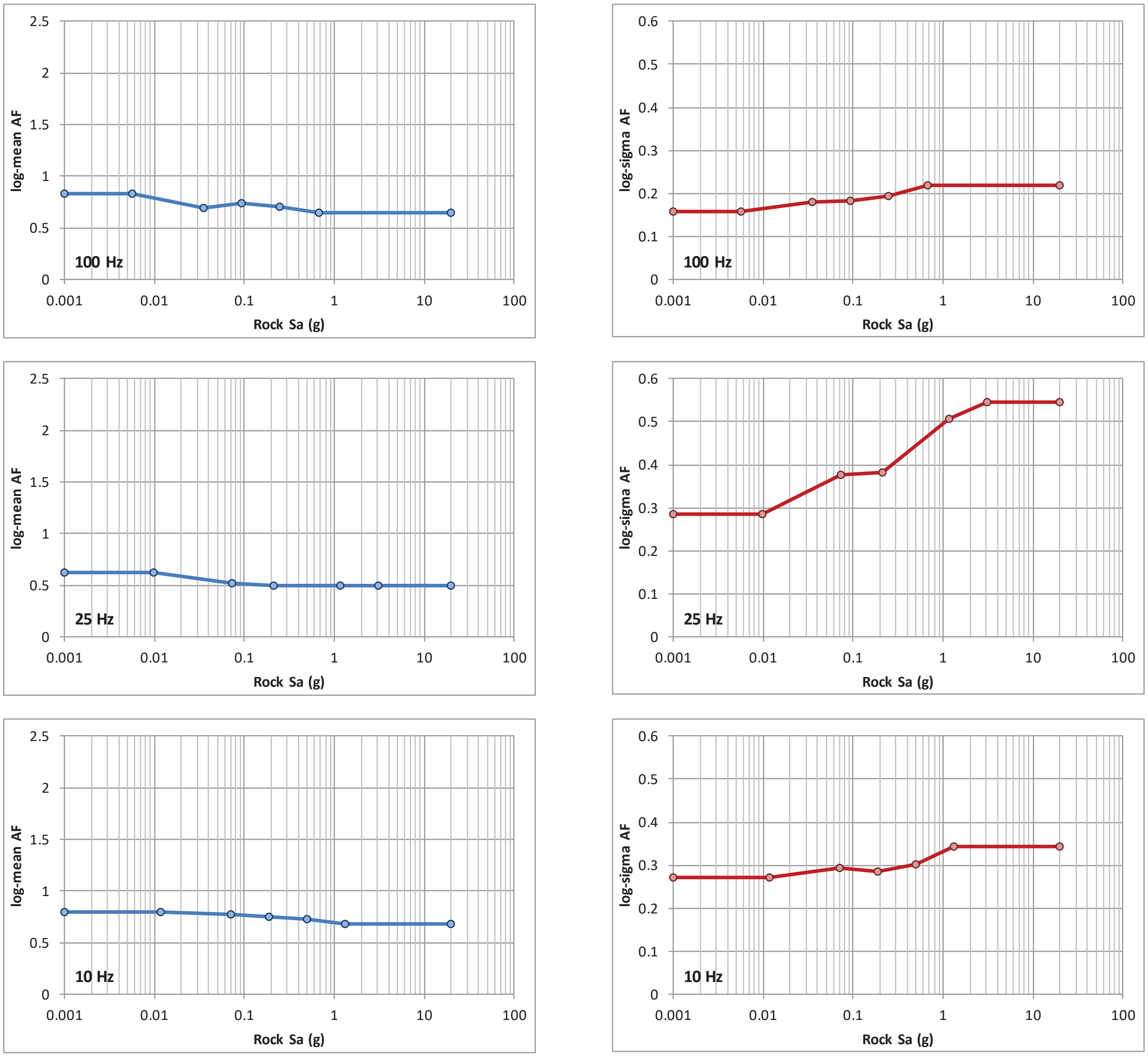

Figure 3.1.1-4 GMRS/FIRS1 Soil Profile Site Amplification Factor and Logarithmic Sigmas (100 Hz, 25 Hz, and 10 Hz)

GMRS/FIRS1 and FIRS2 were developed in accordance with Regulatory Guide 1.208 [61]. Sixty randomizations were performed for the site response for each epistemic branch in the soil logic tree, compared to a minimum of thirty recommended in the SPID [2]. The site response analyses were completed using the HF and LF control motions. Site-specific horizontal hazard curves for each of the FIRS site conditions were used and were developed using Approach 3 of NUREG/CR-6728 [7].

The reference earthquake ground motion to which the majority of fragilities are referenced is represented by the horizontal three times GMRS (3xGMRS) seismic hazard level. Risk significant components in the combined Reactor Building – Turbine Building (RB-TB) are referenced to 3xGMRS. A small number of SSCs are referenced to 1E-05 or GMRS hazard levels, consistent with their seismic failure levels [21]. See Section 4.3 and Appendix A for further discussion.

Peak Ground Acceleration (PGA) is the ground motion parameter used for the Seismic PRA.

Vertical ground motions were developed by applying Vertical/Horizontal (V/H) ratios to the horizontal GMRS/FIRS1 and FIRS2. For GMRS/FIRS1 profile, the V/H ratios were developed at both the GMRS and 3xGMRS levels. A logic tree was adopted to incorporate epistemic uncertainty by weighting the equivalent Central and Eastern United States (CEUS) Rock V/H ratios from NUREG/CR-6728 [7] and Western United States (WUS) V/H ratios (Campbell and Bozorgnia (2003) [77] Bozorgnia and Campbell (2004) [78] and Gülerce and Abrahamson (2011) [79]) shifted in the frequency domain by a factor of three (3) to match the peak in the CEUS V/H ratios. For GMRS/FIRS1, an 80% weight was given to the CEUS Rock V/H ratios [7] and 20% to the WUS frequency shifted V/H ratios. For FIRS2, equal weightage was given to both relations.

Tables 3.1.1-2 and 3.1.1-3 and Figures 3.1.1-5 and 3.1.1-6 provide the horizontal and vertical GMRS/FIRS1 and FIRS2, respectively [6].

Table 3.1.1-2 Smoothed Horizontal and Vertical GMRS/FIRS1

Frequency (Hz)	Horizontal GMRS/FIRS1 (g)	Vertical GMRS/FIRS1 (g)
0.1	1.07E-02	7.00E-03
0.125	1.38E-02	9.02E-03
0.15	1.72E-02	1.12E-02
0.2	2.41E-02	1.57E-02
0.3	3.67E-02	2.39E-02
0.4	4.92E-02	3.21E-02
0.5	6.19E-02	4.04E-02
0.6	7.27E-02	4.74E-02
0.7	8.16E-02	5.33E-02
0.8	8.97E-02	5.85E-02
0.9	9.70E-02	6.32E-02
1	1.05E-01	6.82E-02
1.25	1.22E-01	7.93E-02
1.5	1.37E-01	8.96E-02
2	1.61E-01	1.05E-01
2.5	1.80E-01	1.17E-01
3	2.05E-01	1.34E-01
4	2.57E-01	1.68E-01
5	2.88E-01	1.88E-01
6	3.08E-01	2.01E-01
7	3.19E-01	2.08E-01
8	3.25E-01	2.12E-01
9	3.27E-01	2.13E-01
10	3.26E-01	2.13E-01
12.5	3.21E-01	2.11E-01
15	3.18E-01	2.11E-01
20	3.20E-01	2.20E-01
25	3.11E-01	2.25E-01
30	2.86E-01	2.13E-01
35	2.57E-01	1.99E-01
40	2.36E-01	1.90E-01
45	2.21E-01	1.82E-01
50	2.11E-01	1.77E-01
60	1.99E-01	1.71E-01
70	1.91E-01	1.65E-01
80	1.87E-01	1.57E-01
90	1.84E-01	1.49E-01
100	1.83E-01	1.41E-01

Table 3.1.1-3 Smoothed Horizontal and Vertical FIRS2

Frequency (Hz)	Horizontal FIRS2 (g)	Vertical FIRS2 (g)
0.1	1.07E-02	7.12E-03
0.125	1.38E-02	9.19E-03
0.15	1.72E-02	1.14E-02
0.2	2.41E-02	1.60E-02
0.3	3.67E-02	2.44E-02
0.4	4.92E-02	3.27E-02
0.5	6.20E-02	4.12E-02
0.6	7.28E-02	4.84E-02
0.7	8.19E-02	5.44E-02
0.8	9.00E-02	5.98E-02
0.9	9.74E-02	6.47E-02
1	1.05E-01	6.99E-02
1.25	1.23E-01	8.15E-02
1.5	1.39E-01	9.24E-02
2	1.64E-01	1.09E-01
2.5	1.86E-01	1.24E-01
3	2.17E-01	1.44E-01
4	2.85E-01	1.90E-01
5	3.37E-01	2.24E-01
6	3.83E-01	2.55E-01
7	4.29E-01	2.85E-01
8	4.75E-01	3.16E-01
9	5.17E-01	3.44E-01
10	5.50E-01	3.66E-01
12.5	5.88E-01	3.91E-01
15	5.81E-01	3.91E-01
20	5.13E-01	3.63E-01
25	4.41E-01	3.33E-01
30	3.79E-01	3.03E-01
35	3.35E-01	2.80E-01
40	3.08E-01	2.70E-01
45	2.91E-01	2.66E-01
50	2.80E-01	2.62E-01
60	2.66E-01	2.50E-01
70	2.58E-01	2.40E-01
80	2.53E-01	2.31E-01
90	2.50E-01	2.21E-01
100	2.48E-01	2.15E-01

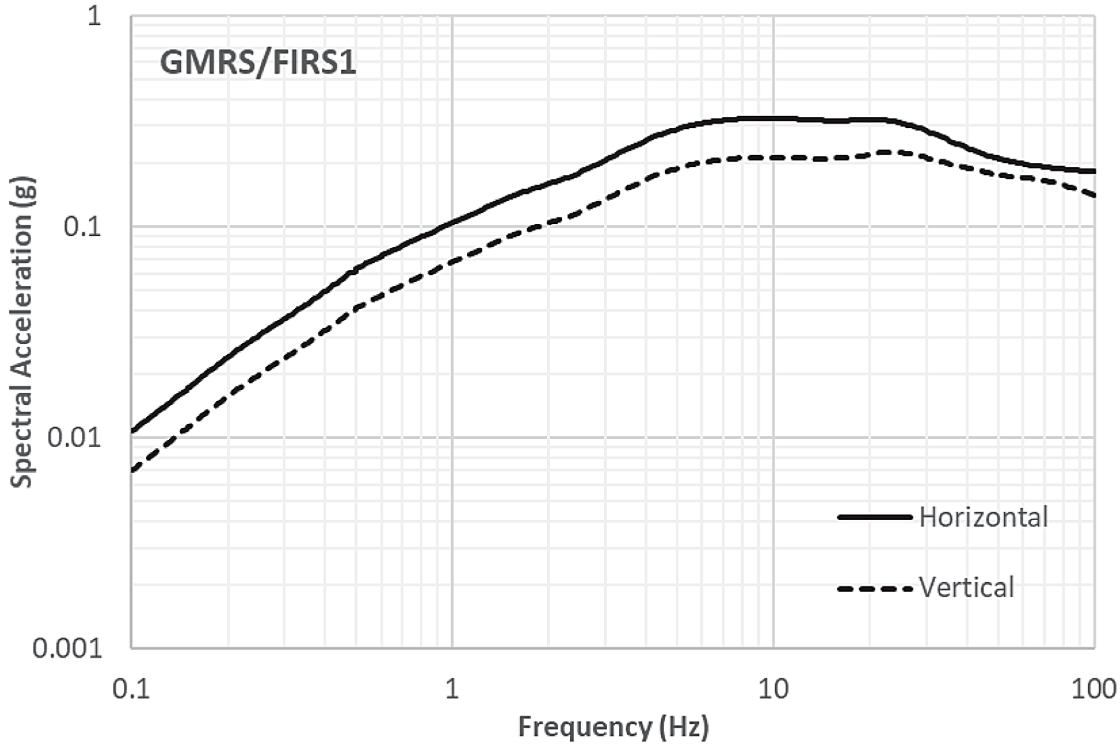

Figure 3.1.1-5 Horizontal and Vertical GMRS/FIRS1

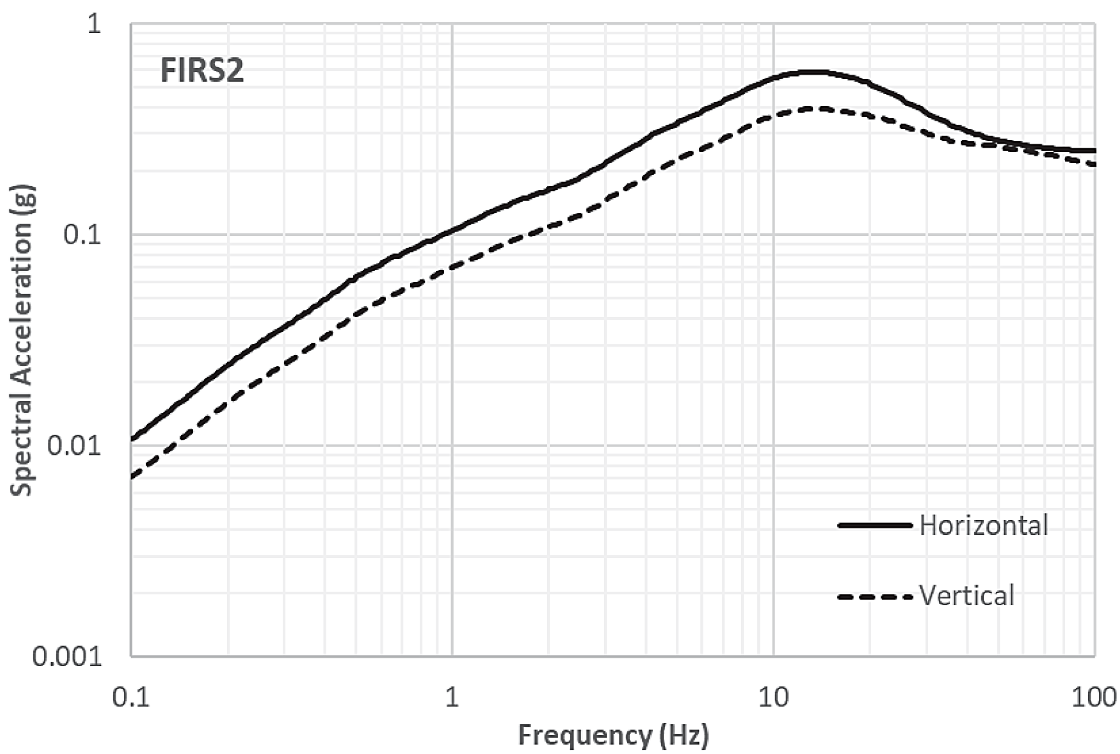

Figure 3.1.1-6 Horizontal and Vertical FIRS2

3.1.2 Seismic Hazard Analysis Technical Adequacy

The DRE hazard analysis was subjected to an independent peer review against the pertinent requirements in the PRA Standard [4]. The Seismic PRA was peer reviewed relative to Capability Category II for the full set of requirements in the Standard. After completion of the peer review and the disposition of the peer review findings, the full set of supporting requirements was met [23]. The seismic hazard analysis was determined to be acceptable for use in the Seismic PRA.

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A.

3.1.3 Seismic Hazard Analysis Results and Insights

Table 3.1.1-1 and Figure 3.1.1-3 provide the final seismic hazard results used as input to the DRE Seismic PRA, in terms of exceedance frequencies as a function of PGA level for the mean and several fractiles at hard reference rock [6].

For the low frequencies (1+2.5 Hz), the background source zones are the main contributors to the seismic hazard at DRE. The New Madrid Fault System RLME has considerable high peaks at the 1E-02, 1E-03, and 1E-04 MAFE levels, noticeable peaks at the 1E-05 MAFE level, and small peaks at the 3xGMRS and 1E-06 MAFE levels. Wabash Valley has high peaks at the 1E-02 MAFE level and small but noticeable peaks at the 1E-03, 1E-04, 1E-05, 3xGMRS, and 1E-06 levels [6].

For high frequencies (5+10 Hz), the background source zones are the main contributors to seismic hazard at DRE. The New Madrid Fault System has considerable high peaks at the 1E-02 MAFE level, moderate noticeable peaks at the 1E-03 MAFE level, and small but noticeable peaks at the 1E-04 MAFE level. Wabash Valley have high peaks at the 1E-02 MAFE level and small but noticeable peaks at the 1E-03 and 1E-04 MAFE levels [6].

Sensitivities of the hard rock hazard to the ground motion models [34] and most significant portions of the seismic source model were performed [6]. The sensitivity analyses indicate a large uncertainty in the rock hazard due to the suite of ground motion models. Also, the sensitivity analyses indicate that the ground motion models for the background seismic source zones and the seismicity rates for the dominant background zone contribute the most to the uncertainty for spectral frequencies corresponding to the PGA (100 Hz) and 1 Hz.

Sensitivity analyses were also performed on significant portions related to the site response analyses including alternate randomization techniques and removal of interbedded stiffer layers [6]. The sensitivity analyses confirmed that the results were not sensitive to the assumptions adopted in the model.

The Central and Eastern United States Seismic Source Characterization (CEUS-SSC) [8; 9] concluded its data gathering efforts in 2008. As a result, a literature search of published and unpublished data was completed to identify any data that may have an impact on the SSC or any other site-specific modifications based on new information. The CEUS-SSC [8] developed comprehensive up-to-date databases including a comprehensive earthquake catalog through December 31, 2008 and a compilation of paleo-seismic data. For the CEUS-SSC Project, comprehensive Data Evaluation Tables were prepared [8]. Literature that post-dates the CEUS-SSC was evaluated to confirm the lack of local seismic sources [6]. An updated earthquake catalog post-dating the CEUS-SSC through July 31, 2016 was developed along with induced seismicity [6]. After the review and studies of new information, it was concluded that the CEUS-SSC model did not require an update [6].

The PSHA [6] performed incorporated the entire CEUS-SSC logic tree published in NUREG-2115 [8] with its revisions published in 2015 [9]. The only 'simplification' performed to the entire CEUS-SSC was related to using point sources for the background sources. No seismic sources were screened out of the analyses. The use of point sources for modeling the background sources is supported by the sensitivities presented in NUREG-2115 [8].

3.1.4 Horizontal and Vertical GMRS

This section provides the control point horizontal and vertical GMRS.

The horizontal and vertical GMRS at the control point is tabulated in Table 3.1.1-2 and presented in Figure 3.1.1-5. The development of the control point response spectra is summarized in Section 1.1.1 and further described in detail in the DRE PSHA report [6].

4. Determination of Seismic Fragilities for the SPRA

This section provides a summary of the process for identifying and developing fragilities for SSCs that participate in the plant response to a seismic event for the DRE SPRA. The subsections provide brief summaries of these elements.

4.1 Seismic Equipment List

For the DRE SPRA, a seismic equipment list (SEL) [51] was developed that includes those SSCs that are important to achieving safe shutdown following a seismic event, and to mitigating radioactivity release if core damage occurs, and that are included in the SPRA model. The methodology used to develop the SEL is generally consistent with the guidance provided in EPRI 3002000709 [10].

4.1.1 SEL Development

The DRE SPRA SEL [51] is developed consistent with the requirements and guidance identified in the following industry references:

- Part 5 (Addenda B) of the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) PRA Standard (RA-Sb-2013) [4]
- Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, EPRI Report 1025287 [2]
- Seismic Probabilistic Risk Assessment Implementation Guide, EPRI Report 3002000709, December 2013 [10]

The EPRI 2013 Seismic PRA Implementation Guide (SPRAIG) [10] provides the following general guidance as one method to develop an initial SEL:

1. Identify SSCs important to safe shutdown from Full-Power PRA Models
2. Identify SSCs from Review of Seismic evaluation performed for the IPEEE
3. Identify structures and passive components important to seismic response (including identification of SSCs from secondary hazard considerations)
4. Identify Additional SSCs from Plant Walkdown
5. Disposition SSCs on SEL
6. Review and document SEL

The above EPRI approach is followed for the DRE SPRA SEL development.

The DRE SPRA SEL is developed by using the DRE existing full-power PRA models as the starting point. Use of the PRA models as a starting point for SSCs to consider for fragility analysis is a rational starting point as the PRA models have already identified and modeled SSCs that cover all the critical safety functions and are appropriate for modeling in PRA core damage frequency (CDF) and release frequency models. Basic events in the PRA models are used as the vehicle to identify the starting list of SSCs and operator action pathways to be walked down for the purpose of developing fragilities for input to the SPRA model.

The PRA model files used as input for the SEL development are the DRE full-power internal events PRA (which also includes internal flooding models) and the internal fires PRA. These models include both Level 1 (CDF) and Level 2 (LERF) full power PRA related equipment.

In addition to internal flooding and internal fires, these PRA models used as input for the initial phases of the SEL development cover the following types of initiating events:

- Transients
- Loss of support systems (e.g., loss of DC bus, loss of AC bus, loss of instrument air, etc.)
- Loss of offsite power (LOOP)
- Loss of coolant accidents (LOCA) inside primary containment (including excessive LOCA)
- Interfacing Systems LOCAs (ISLOCA)
- Loss of coolant accidents outside primary containment (Break Outside Containment (BOC))

All these initiated states are included in the DRE SPRA with seismic-induced SSC failures. Given the low capacity of offsite power, seismic-induced transients (i.e., offsite power remains intact) were not explicitly modeled in the peer reviewed SPRA as the plant likely would remain at power (not trip) or if a trip did occur the likelihood of seismic-induced failure of significant mitigation equipment would be very low; this is a common SPRA approach [10]. In response to an F&O from the DRE SPRA peer review, the final DRE SPRA incorporated accident sequence modeling of seismic-induced transients. As such, equipment on the initial SEL that is powered only from non-emergency AC power is included in the final SEL [51].

The Very Small LOCA initiator is added to the SEL and included in the DRE SPRA model. The excessive LOCA is addressed by a fragility for the RPV supports. The RPV recirculation pumps were added to support fragility evaluation for seismic-induced Large LOCA. Failure to scram (Anticipated Transient Without Scram (ATWS)) is addressed by fragility calculations of the RPV internals [21].

Initiating events for plant shutdown configurations (e.g., loss of SFP Cooling) are not covered by these models and this is consistent with the scope of this full-power DRE SPRA (and consistent with Reference [2]).

These PRA models also cover all the requisite Level 1 and Level 2 critical safety functions:

- Reactivity control
- Reactor pressure control
- Reactor coolant inventory control (including RPV depressurization)
- Containment pressure control (including vapor suppression)
- Primary and secondary containment isolation

The frontline systems modeled in the DRE SPRA as a function of critical safety function are summarized in Table 4.1.1-1. The support systems used in the DRE SPRA are not listed in Table 4.1.1-1. The support systems modeled in the DRE SPRA are [50]:

- Auxiliary AC power
- Emergency AC power (including EDGs and SBO DGs)
- 125V and 250V safety DC
- Containment Cooling Service Water (CCSW)
- Service Water (SW)
- Diesel Generator Cooling Water (DGCW)
- Reactor Building Closed Cooling Water (RBCCW)
- Turbine Building Closed Cooling Water (TBCCW)
- Pneumatic supplies (Instrument Air, Service Air, N2 systems)
- Condensate Transfer (e.g., Condensate Storage Tank (CST))
- Room Cooling (e.g., HPCI rooms, CCSW B&C rooms, Emergency Diesel Generator (EDG) and SBO Diesel Generator (DG) rooms)

In addition to the initial development stages described above, the SEL development is supplemented by the following efforts:

- Review of system drawings to identify items not explicitly included in the PRA models
- Review of the internal flooding PRA to identify internal flooding sources of potential significance
- Review of plant drawings and Human Reliability Analysis to identify operator action pathways
- Identification of block walls in buildings containing SPRA equipment
- Identification of flammable sources (e.g., hydrogen, fuel oil, lube oil)
- Identification of potential seismic-induced electrical fire sources (including non-safety electrical, with the assumption that arcing may occur prior to loss of offsite power)
- Component chatter assessment (separate topic discussed below)
- Identification of buildings of interest to SPRA
- Identification of above ground tanks
- Identification of buried items
- Plant walkdowns

Table 4.1.1-1 DRE SPRA Frontline Systems per Safety Function [48]

Critical Safety Function	Systems (1),(2)
Reactivity Control	RPS ARI RPT SLC
RPV Pressure Control	TBVs Isolation Condenser HPCI Steam lines Electromatic Relief Valves (ERVs) / SRVs / SVs RPT
RPV Coolant Inventory Control (High Pressure)	Feedwater Isolation Condenser HPCI Control Rod Drive (CRD)
RPV Coolant Inventory Control (Low Pressure)	Condensate / SBCS LPCI Core Spray (CS) Fire Protection Water
RPV Depressurization	TBVs Isolation Condenser HPCI Steam lines ERVs / SRVs
Containment Pressure and Temperature Control	Main Condenser Shutdown Cooling Torus Cooling (LPCI/CCSW) Containment Sprays Primary Containment Venting
Vapor Suppression	WW-DW Vacuum Breakers Containment Sprays ERVs / SRVs TBVs

Table 4.1.1-1 DRE SPRA Frontline Systems per Safety Function [48]

Critical Safety Function	Systems (1),(2)
Containment Isolation	Primary Containment Isolation System and associated valves Primary containment structure Reactor building structure

Notes to Table 4.1.1-1:

1. Support systems (e.g., electric power) are not listed in this table.
2. Some of the critical safety functions also are modeled with FLEX equipment. FLEX can supply emergency AC power to various functions and FLEX is used as an alternative injection system in the SPRA (refer to Accident Sequence discussions in Section 5.3.2 and the use of FLEX in accident sequence mitigation).

Structures that house or spatially interact with identified SSCs, as well as those that involve ex-Control Room actions credited in the SPRA, are included in the SEL for fragility consideration. A disposition of all structures on the site is performed and documented in the SEL report. The following buildings and structures were identified for inclusion on the SEL (no earthen structures were identified for inclusion on the SEL):

- Drywell, Vents, Torus, and Penetrations (Primary Containment): Houses NSSS and key equipment in the SPRA. NSSS line items included separately on SEL for RCS piping (LOCAs) and RPV supports.
- Reactor Buildings, including Unit 2/3 HPCI Building: House key equipment in the SPRA (e.g., isolation condenser, HPCI, LPCI and CS pumps, 2/3 DG, 4 kV AC Buses).

  The U1 Reactor Building was also added for fragility investigation in the event electrical cabling existed below or through the building and was determined to be used in the SPRA. Subsequently determined not to be the case and fragility not important to SPRA.
- Reactor Vessel Support Pedestal: Houses RPV and control rods.
- Turbine Buildings: House balance-of-plant equipment (which are low risk contributors to the SPRA) but also houses 125 and 250 VDC electrical equipment, 4 kV AC Buses, EDG2, EDG3 and CCSW pumps (as well as other SSCs) which are key to the SPRA. The Turbine Buildings are structurally connected to each other and to the Reactor Buildings. The Main Control Room Complex (including the Control Room and Aux. Electric Equipment Room) is also contained within the Turbine Building. The Main Control Room ceiling was also added as a separate line item.

The U1 Turbine Building was also added for fragility investigation in the event electrical cabling below or through the building was determined to be used in the SPRA. Subsequently determined not to be the case and fragility not important to SPRA.

- Isolation Condenser Pumphouse: Houses key equipment in the SPRA (e.g., diesel driven IC makeup pumps).
- 2/3 Crib House: Houses key equipment in the SPRA (i.e., SW and DGCW pumps, Unit 2/3 diesel driven fire pump).
- SBO Building (formerly Unit 1 HPCI Building): Houses key plant equipment (Unit 2 and Unit 3 Station Blackout, SBO, DGs).
- Radwaste Building (Unit 2/3): This structure does not contain any SPRA equipment and no operator actions are performed in this area for the SPRA model. However, identified initially in the event electrical cabling below or inside this building that is used in the SPRA. Subsequently determined not to be the case and the only potential influence on the SPRA is the NSW overhead piping that is attached to the exterior of the Radwaste building.
- Station Chimneys: The U2/3 station chimney is a PRA credited vent path for a large torus or drywell vent for prevention and mitigation of postulated core damage accidents; it also is considered for collapse impacts on surrounding structures. The U1 chimney is not used in the SPRA but was identified for potential collapse impacts on U2/3 structures.
- Dresden Lock and Dam: The downstream dam supports maintaining adequate river level as the ultimate heat sink to support normal suction to the 2/3 Crib House.
- FLEX Buildings A and B: The new FLEX A building is east of the SBO building and the new FLEX B building is southwest of the U3 reactor building. They house FLEX equipment.
- Miscellaneous Switchyard Areas and related Switchgear Buildings: The switchyard and the miscellaneous outdoor switchgear structures are addressed by the “Offsite Power” line item on the SEL.

The following buried items were identified for inclusion on the SEL:

- Buried SW piping
- Buried DGCW piping
- Buried Fire Protection water piping
- Buried EDG Fuel Oil Transfer piping
- EDG Fuel Oil Storage tanks
- SBODG Fuel Oil Storage tank
- U2/3 Station Chimney buried exhaust
- Buried CST piping

Based on discussions with the fragility personnel on the DRE SPRA team, the level of relative displacement of soil near the surface (where buried cable would exist) at seismic levels of interest is very low. These small displacements are not considered to have any potential ability to cause failure of cables. As such, individual buried cable items are not included as specific line items on the SEL.

Every cable tray, pipe and HVAC duct in the plant was not specifically itemized; the DRE SPRA used fragility walkdowns to search for outliers, to assess the ruggedness of these distributed systems, and to calculate fragilities in certain cases.

In addition to the above, SSCs from the previous seismic related assessments were added to the DRE SEL for consideration:

- DRE IPEEE Success Path Equipment List [53]
- DRE NTTF 2.3 Seismic Walkdown Equipment List (SWEL) [71,72]
- DRE Expedited Seismic Equipment List [29]
- DRE Initial Seismic PRA Model – An earlier “Phase I” seismic PRA performed for DRE in 2011 [87]. Initial “Phase I” seismic PRA models were developed for Exelon sites following the events at Fukushima in anticipation of potentially developing more detailed seismic PRA models in the future.
- Other plant SEL [73]

The total number of line items on the SEL is approximately 6000 and covers initiating events, operator actions, various basic event types, and specific pieces of equipment and structures. A disposition process of each line item is used to identify those line items that can be screened and those that are to be carried forward for SSC fragility evaluation. The following disposition codes are used to disposition the DRE SEL line items:

- S0a: Non-applicable initiating event to SPRA
- S0b: Type A and B HEPs
- S0c: Type C dependent HEP (Type C independent HEPs already provide the necessary information on action pathways)
- S0d: Function Recovery and Repair basic events
- S0e: Test and maintenance basic events
- S0f: Common Cause Failure (CCF) basic events
- S0g: Flag basic events (i.e., PRA basic events set to TRUE or FALSE to model specific plant conditions)
- S0h: Other basic events that need not be carried forward in the SEL development process for the identification of SSCs (e.g., plant configuration probabilities, phenomena events).
- S0x: Additional Failure Mode basic events that can exist in the PRA models for a given SSC that need not be carried forward in the SEL development process.
- S1: SSC not included in SPRA model
- S1-BOP: SSC not included in SPRA model (Balance of Plant equipment)
- S2: Post-initiator operator actions performed in Main Control Room (Main Control Room structure and control panels already included on SEL)
- S3a: Inherently rugged SSC

- S3b: Rugged SSC based on observation
- S4a: Subsumed into fragility component boundary – circuit breakers
- S4b: Subsumed into fragility component boundary – relays
- S4c: Subsumed into fragility component boundary – misc. instrument and control items
- S4d: Subsumed into fragility component boundary – rule of the box
- F1: SPRA post-operator actions performed outside Main Control Room (these define the operator action pathways that need to be investigated)
- F2: SSC requiring fragility evaluation
- F2-S3b: SSCs that were originally dispositioned as S3b (e.g., valves identified as rugged based on observations). The DRE SPRA Fragility Team calculated fragilities for SSCs that need to change state (e.g., Motor Operated Valves (MOVs) and Air Operated Valves (AOVs)).

The disposition codes beginning with the letter "S" indicate SEL line items that need not be carried forward for fragility calculations for the variety of reasons indicated (e.g., other line items already capture that SSC; or that line item is within the fragility component boundary of another line item on the SEL, etc.). The line items with the "S3b" disposition code were walked down to determine if they can be properly classified as rugged and not require a fragility evaluation. However, fragility evaluations were performed for approximately 400 SEL items with the "F2-S3b" disposition code covering both units where the SSC needs to actively change state (e.g., MOV or AOV needs to open or close to support the system mitigation suction in the PRA model). The SEL line items with the "F2" disposition code identify the SSCs requiring fragility evaluation. There are over 600 "F2" SSC line items (i.e., not including the "F2-S3b" line items) on the DRE SEL covering both units. Of the approximately 6000 line items on the SEL covering both units, fragility information was provided that covered over 1000 SSCs (not counting relay chatter fragilities) associated with disposition code "F2" and disposition code "F2-S3b" for valves needing to change state.

4.1.2 Relay Evaluation

During a seismic event, vibratory ground motion can cause relays to chatter. The chattering of relays potentially can result in spurious signals to equipment. Most relay chatter is either acceptable (does not impact the associated equipment), is self-correcting, or can be recovered by operator action. An extensive relay chatter evaluation was performed for the DRE SPRA, in accordance with SPID [2], Section 6.4.2 and ASME/ANS PRA Standard [4], Section 5-2.2 and is documented in reference [32]. Note that "relay" is used in sections of this report to mean relays as well as other contacts and contact devices that have the potential to chatter, including circuit breakers and motor starters. The term relay should be taken to mean any or all of these different electrical devices that are potentially sensitive to chatter. The evaluation resulted in most relay chatter scenarios being screened from further evaluation based on no impact to component function. The relays, circuit breakers and other contact devices that were not screened are listed in Table 4.1.2-1, along with their function and disposition in the SPRA with appropriate seismic fragility or operator action.

The unscreened contact chatter scenarios provided in the contact chatter evaluation [32] (approximately 600) are considered and evaluated for inclusion in the SPRA model based on the identified system impact (e.g., divisional diesel fails to start or load). Given this high number of unscreened contact chatter scenarios, not all contact chatter scenarios are explicitly included in the SPRA model. Initial SPRA model quantifications helped identify the risk impact of individual or correlated contact chatter scenarios based on associated system impact and fragility value. Of the contact chatter scenarios which were not screened via the chatter evaluation [32], Table 4.1.2-1 lists the subset of contact chatter scenarios that are explicitly modeled in the SPRA based on initial SPRA model quantifications and risk insights.

Table 4.1.2-1 Summary of Disposition of Unscreened Relays [50]

Relay	Function	Disposition
2391 Auto Isolation Relays	HPCI auto isolation - This contact would have to chatter long enough for the TDE coil to pick up (3-9 sec). Assuming the TDE is met, chatter will cause the valve to fully close and remain locked out until reset.	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of HPCI with CDFM fragility. Credit for potential operator recovery of the seismic induced relay chatter event is based on insights from plant specific operator interviews and detailed Human Reliability Analysis (HRA).
86 Lock Out Relays	Chatter of the 86 relays at 4KV VAC switchgear will trip individual Emergency Core Cooling System (ECCS) pumps and lock out the restart.	**Modeled in SPRA** due to potential risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of individual ECCS pumps with CDFM fragility. Conservatively assume no credit for proceduralized operator recovery of the seismic induced relay chatter event based on relatively low risk significance from subsequent interim quantification results.
151N Ground Overcurrent Relays	Chatter of the 151N ground overcurrent will trip the individual ECCS pump and lock out the pump.	**Modeled in SPRA** due to potential risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of individual or multiple ECCS pumps with CDFM fragility. Credit for potential operator recovery of the seismic induced relay chatter event is based on insights from plant specific operator interviews and detailed HRA.
186 Lock Out Relays	Chatter of the 186 lockout relay trips the diesel and the output breaker.	**Modeled in SPRA** due to potential risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of individual or multiple EDGs with CDFM fragility. Credit for potential operator recovery of the seismic induced relay chatter event is based on insights from plant specific operator interviews and detailed HRA.

Table 4.1.2-1 Summary of Disposition of Unscreened Relays [50]

Relay	Function	Disposition
VSR EDG excitation start relay	The generator excitation starts relay seal-in closes to maintain the VSR signal to the excitation circuit. VSR seal-in could lead to spurious generator excitation before the motor reaches 800 rpm. Overexcitation has the potential to damage the EDG.	**Modeled in SPRA** due to potential risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of individual or multiple EDGs with CDFM fragility. No credit for potential operator recovery of the seismic induced relay chatter event is based on circuit analyses.
7641 Fire Detection Relays	Chatter of Fire Detection Relays will trip DG HVAC fans. Room temperature limits may be exceeded and this has the potential to damage the EDG.	**Modeled in SPRA** due to potential risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of multiple EDGs with CDFM fragility. No credit for potential operator recovery of the seismic induced relay chatter event is based on circuit analyses.
1530 Relays	Chatter of the relay will trip the breaker to individual EDGs or ECCS pumps.	**Modeled in SPRA** due to potential risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of individual or multiple EDGs or ECCS pumps with CDFM fragility. Credit for potential operator recovery of the seismic induced relay chatter event impacting the ECCS pumps is based on insights from plant specific operator interviews and detailed HRA. No credit for potential operator recovery of the seismic induced relay chatter event impacting the EDGs is based on circuit analyses.
51N Overcurrent Trip Relays	Chatter will cause the breaker to trip. The RBCCW pumps will trip. An overcurrent trip signal would occur and would need to be reset locally.	**Modeled in SPRA** due to potential risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of all RBCCW pumps with CDFM fragility. Conservatively assume no credit for proceduralized operator recovery of the seismic induced relay chatter event based on low risk significance from subsequent interim quantification results.

Table 4.1.2-1 Summary of Disposition of Unscreened Relays [50]

Relay	Function	Disposition
151A, 151B, 151C Overcurrent Relays	Chatter of the overcurrent relays will trip the breaker to EDGs 3 and 2/3.	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability EDGs 3 and 2/3 with CDFM fragility. Credit for potential operator recovery of the seismic induced relay chatter event is based on insights from plant specific operator interviews and detailed HRA.
CR2871, CR29B, CR3871 and CR39B Breaker Trip Relay	Chatter will trip the breaker for Bus 29 (39) feeding MCC 29-7 (39-7).	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of power from Bus 29 (39) feeding MCC 29-7 (39-7) with CDFM fragility. Conservatively assume no credit for proceduralized operator recovery of the seismic induced relay chatter event based on low risk significance from subsequent interim quantification results.
27XTD EDG Time Delay Relay	The contact has a two-second time delay pickup on diesel start. The time delay avoids the potential of the breaker closing prior to the bus going dead. Chatter could cause the breaker to close prior to the two-second pickup.	**Modeled in SPRA** due to potential risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of multiple EDGs with CDFM fragility. No credit for potential operator recovery of the seismic induced relay chatter event is based on circuit analyses.
151N-3426 Trip Relay	Chatter will trip the breaker for 4KV Bus 34-1 feeding 480 VAC Bus 39.	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of power from 4KV Bus 34-1 feeding 480 VAC Bus 39 with CDFM fragility. Conservatively assume no credit for proceduralized operator recovery of the seismic induced relay chatter event based on low risk significance from subsequent interim quantification results.

Table 4.1.2-1 Summary of Disposition of Unscreened Relays [50]

Relay	Function	Disposition
MC, 42-C, 42-MC	Chatter of the motor starter relay may cause spurious closure of the IC inlet valve, spurious closure of pump flow path valve, or tripping of the HPCI gland seal condenser exhaust fan.	**Modeled in SPRA** due to potential risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of individual or multiple ECCS pumps or valves with CDFM fragility. Credit for potential operator recovery of the seismic induced chatter of the 42-C relays impacting the IC inlet valves is based on insights from plant specific operator interviews and detailed HRA. No credit for potential operator recovery of the seismic induced chatter of the MC and 42-MC relays impacting the other system pumps and valves (e.g., CS/LPCI) is based on circuit analyses.
Medium-voltage circuit breakers	Spurious tripping of 4KV AC power circuit breakers due to contact chatter.	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of power to all 4KV busses with CDFM fragility. Conservatively assume no credit for proceduralized operator recovery of the seismic induced relay chatter event based on low risk significance from subsequent interim quantification results.
Low voltage circuit breakers	Spurious tripping of 480V AC power circuit breakers due to contact chatter.	**Modeled in SPRA** due to calculated risk impact based on initial quantification results. Modeled as seismic induced relay chatter unavailability of power to all 480 VAC busses with CDFM fragility. Conservatively assume no credit for proceduralized operator recovery of the seismic induced relay chatter event based on low risk significance from subsequent interim quantification results.

4.2 Walkdown Approach

This section provides a summary of the methodology and scope of the seismic walkdowns performed for the SPRA. Walkdowns were performed by personnel with appropriate qualifications as defined in the SPID [2]. Walkdowns of those SSCs included on the seismic equipment list were performed to assess the as-installed condition of these SSCs for use in determining their seismic capacity and performing initial screening.

Walkdowns were performed in accordance with guidance in SPID Section 6.5 and the associated requirements in the PRA Standard [4]. These walkdowns were documented in [31].

Several previous seismic walkdowns for DRE have been documented. The information gathered during these previous walkdowns and the results and conclusions contained in the walkdown information was used where applicable to supplement plant drawings and calculations for the SPRA as discussed in this report. These previous walkdowns include:

- SQUG/IPEEE – Performed in 1995-97 time frame in support of the Individual Plant Examination of External Events (IPEEE) and in response to Unresolved Safety Issue (USI) A46 [53, 54, and 64], using the methodology developed by the Seismic Qualification Utility Group (SQUG) and contained in the SQUG Generic Implementation Procedure (GIP) [26] and the guidelines contained in EPRI NP 6041-SL [12].
- NTTF 2.3, Seismic – Performed in response to Near-Term Task Force (NTTF) Recommendation 2.3, Seismic. This walkdown was completed in late 2012 [55, 56, 57, 71, & 72].
- ESEP – Performed during 2014 in support of the Expedited Seismic Evaluation Process (ESEP) [29, 30].

  In addition to ESEP walkdowns, DRE has completed a Mitigating Strategies Assessment (MSA) [85; 88] for the impacts of the reevaluated seismic hazard to determine if the mitigating (FLEX) strategies developed, implemented and maintained in accordance with NRC Order EA-12-049 [35, 36, 37, 38] remain acceptable at the reevaluated seismic hazard levels. Consistent with Section H.5 of NEI 12-06 Revision 4 [39], the evaluations and walkdowns included FLEX equipment storage buildings and Non-Seismic Category I Structures that could impact FLEX implementation, operator pathways walkdown, tie down of FLEX portable equipment, seismic interaction not included in ESEP that could affect FLEX strategies and haul paths walkdown.
- Seismic PRA – Performed to develop input to an earlier “Phase I” seismic PRA performed for DRE in 2011-12 [40]. Initial “Phase I” seismic PRA models were developed for Exelon sites following the events at Fukushima

in anticipation of potentially developing more detailed seismic PRA models in the future.

Information from these walkdowns was gathered and reviewed to obtain inputs and insights for the development of component fragilities. To ensure that the information remained valid and to include components that had not been walked down previously, all components on the SEL, including those walked down previously were included in the scope of the current SPRA walkdowns. However, for components which had been walked down previously and for which sufficient information was available to permit development of a fragility, the walkdown was limited to a walk-by of the individual components.

Detailed walkdowns were performed for components which had not been walked down previously. During a detailed walkdown, the caveats from the SQUG GIP [26] were verified and sufficient information was gathered to allow a fragility to be developed. This included information on anchorage, configuration, weight, dimensions, load path and other structural information. In addition, the walkdown team focused on potential adverse seismic interaction issues including the potential for seismically induced fire and flood and seismic II/I concerns such as masonry block walls in the vicinity of the components.

More simplified walk-bys were performed for components which had been walked down previously. During walk-bys, the walkdown team inspected the component to ensure that there were no obvious changes to the component since the previous walkdown that would adversely impact the seismic capacity of the component. In particular, the walkdown team focused on potential seismic interaction concerns and conditions that might adversely impact the component. In general, walk-bys were less detailed and less intrusive than walkdowns.

The walkdowns were performed in accordance with Table 6.5 of the SPID [2]. Information contained in the SQUG GIP [26] and EPRI NP 6041-SL [12] was used to supplement the guidance provided in the SPID. The SPRA walkdown meets or exceeds the requirements for a Capability Category 2 SPRA established in the current ASME/ANS risk assessment standard updated through ASME/ANS RA-Sb-2013 [4]. This standard is endorsed by the United States Nuclear Regulatory Commission (USNRC) Regulatory Guide (RG) 1.200, Revision 2 [13], for seismic risk analysis. Insights from other industry programs on seismic testing and earthquake experience, such as those catalogued in EPRI NP 6041-SL [12], EPRI NP-7149-D [41], NUREG/CR-4659 [42], EPRI Technical Report 1025286 [43], NUREG/CR-7040 [44], and other industry documents as applicable are applied to compliment individual seismic walkdown and fragility analysis experience.

During the course of the seismic PRA, three phases of walkdowns were performed. The first phase included the outage walkdowns to fulfil the SPRA fragility walkdown requirements for the items that are not accessible during normal plant operation, primarily the SEL components located inside Drywell. Two separate outage walkdowns were performed for each unit during their respective refueling outage. The second phase included non-outage walkdowns to fulfill the SPRA fragility walkdown requirements for the SEL items that are located throughout the plant and in areas outside the containment that are accessible during normal plant operation. This phase was used for grouping similar or commonly mounted components and assigning SEL components to general fragility categories. The third phase of walkdowns was performed for specific components that require more detailed examination as part of the component fragility analysis. As part of Phase 3, separate walkdowns were performed to assess operator pathways used to perform operator actions, to obtain detailed information related to in-cabinet amplification factors for relays and to provide specific inputs to the fragility team such as nozzle loads. In addition, even though the walkdown team focused on potential for seismically induced fire and flood concerns during the Phase 2 walkdowns, a separate walkdown [31] was conducted to specifically evaluate the potential for seismically induced fires due to electrical faults and to cover the unscreened relays for chatter and operator paths required for Human Reliability Analysis (HRA) [49].

During the walkdowns, the walkdown team focused on seismic issues that could potentially affect the assignment of a seismic capacity to individual components. This included anchorage details, compliance with the caveats contained in the SQUG GIP [26] associated with each equipment class, seismic interaction due to falling or displacement, existence of block walls in proximity to the components and potential for seismically induced flood and fire. Walkdown documentation for equipment and structures consisted of noting the existing conditions, taking photographs and recording findings, if any.

4.2.1 Significant Walkdown Results and Insights

Consistent with the guidance from EPRI NP 6041-SL [12], no significant findings were noted during the DRE seismic walkdowns. Observations made during the walkdowns are documented in the walkdown report [31].

Components on the SEL were evaluated for seismic anchorage and interaction effects (including block walls and other items that might cause a reduction in seismic capacity), effects of component degradation, such as corrosion and concrete cracking, for consideration in the development of SEL fragilities. In addition, walkdowns were performed to assess operator pathways. The potential for seismic-induced fire and flooding scenarios was assessed independently of the walkdowns for individual components on the SEL. Potential seismic induced internal flood scenarios were incorporated into the DRE SPRA model and fragilities were assigned to events that would cause these events to occur. No seismic induced internal fire scenarios were required to be incorporated into the SPRA model due to low calculated risk contribution. The walkdown observations were adequate for use in developing all the SSC fragilities for the SPRA.

4.2.2 Seismic Equipment List and Seismic Walkdowns Technical Adequacy

The DRE SPRA SEL development [51] and walkdowns [31] were subjected to an independent peer review against the pertinent requirements (i.e., the relevant SFR and SPR requirements) in the PRA Standard [4]. The peer review was performed relative to Capability Category II for the full set of requirements in the PRA Standard.

The peer review assessment [23], and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the DRE SPRA SEL and seismic walkdowns are suitable for this SPRA application.

4.3 Dynamic Analysis of Structures

This section summarizes the dynamic analyses of structures that contain systems and components important to achieving a safe shutdown, using fixed-base and/or Soil Structure Interaction analysis (as applicable).

4.3.1 Fixed-base Analyses

Fixed-base analyses were not performed for the Reactor Building – Turbine Building (RB-TB) and the Station Blackout (SBO) Building, i.e., SSI analysis was performed for the most significant structures analyzed for the SPRA. Note that fixed-base analyses were performed as a verification step in development of the RB-TB and SBO SSI models [14], [15], [58] as well as for limited seismic demand determination for structural fragility evaluation of low-significance structures such as the Unit 1 and Unit 2/3 chimneys [21]. Fixed-base analyses were also performed for the U2/3 Crib House which is founded on hard rock.

Unit 2/3 Crib House

A fixed base analysis was performed for the Unit 2/3 Crib House. The hazard range of interest (HROI) was selected to be the 1E-05 hazard level based on insights from incremental risk quantifications, especially regarding the relative risk-significance of different acceleration intervals and individual components.

Time histories (matched to GMRS hazard level FIRS1) were scaled to HROI of U2/3 Crib House (1E-05) by using the ratio of 1E-05 FIRS1 PGA to GMRS Level (in between 1E-04 and 1E-05) FIRS1 PGA. These scaled time histories were used as input to the analysis of the Unit 2/3 Crib House. The spectral shapes of the 1E-05 and the GMRS spectra are similar and the use of scaling to develop a hazard compatible time history is considered appropriate.

Crib House analysis documentation is provided in Reference [16].

A list of structures and description of relevant parameters is provided in Table 4.3-1.

4.3.2 Soil Structure Interaction (SSI) Analyses

SSI analyses were performed for the RB-TB and the SBO Building.

Reactor Building – Turbine Building (RB-TB) Complex

SSI analyses considering ground motion incoherence were performed for the Reactor Building – Turbine Building (RB-TB) Complex (which includes the Reactor Buildings, Turbine Buildings, Main Control Room, and HPCI building). Structural and soil properties were defined consistent with their response at two distinct representative acceleration HROI selected via coordination with fragility and PRA analysts. The HROI were initially selected to be (1) the GMRS level, and (2) the 3x GMRS level, based on insights from incremental risk quantifications, especially regarding the relative risk-significance of different acceleration intervals and individual components.

The RB-TB is founded at variable shallow depths within sandstone over the footprint on the combined structure, with approximately half of the TB essentially surface founded. The RB foundation slab, and deepest portions of the TB, are founded on a layer of firm rock (shear wave velocity ~8000 fps) which is above softer rock, as defined in the PSHA [6].

A baseline set of SSI analyses was first performed for the GMRS HROI with uncracked concrete elements, which is appropriate considering the expected stress-state at the GMRS-level. Subsequently, a second, supplemental set of SSI analyses was performed for the 3x GMRS HROI, with consideration for concrete cracking and initial post-yield degradation of secondary TB walls (i.e., simulated with reduced stiffness and increased damping per ASCE/SEI 43-05 [18]) commensurate with the expected stress-state and load re-distribution at the 3x GMRS-level.

Initially, fragilities were developed based on a weighting approach using results of both the analysis performed at the GMRS level and at the 3x GMRS level. However, as the development of the SPRA proceeded, it became apparent that many important contributors to risk had fragilities more closely related to the 3x GMRS level. Therefore, structural response results primarily from the supplemental (3x GMRS) analyses were considered during component fragility analysis based on the consideration that seismic demands of the components should be reasonably consistent with those experienced at their failure levels.

The RB-TB SSI analyses consider soil and structural property variation via use of Best Estimate (BE), Lower Bound (LB), and Upper Bound (UB) structure models and BE, LB, and UB soil models such that five analysis cases are developed: BE_{soil}-$BE_{structure}$, LB_{soil}-$BE_{structure}$, UB_{soil}-$BE_{structure}$, BE_{soil}-$LB_{structure}$, BE_{soil}-$UB_{structure}$. Ground motion variability is considered via use of five independent sets of time histories for each analysis case. The five sets of time histories were provided as part of the PSHA [6]. Soil properties for each layer of each variable soil case (BE, LB, and UB) were defined consistent with the results of the probabilistic site response analysis performed with the PSHA [6].

The SC-SASSI analysis code [27] was used to perform the SSI analyses. Cutoff frequency for the SSI analyses was chosen to be 50 Hz, and the SSI models were sufficiently refined to transmit frequencies up to at least 50 Hz through the soil/rock-foundation interface. All SSI analyses utilized the SASSI Direct Method and the analyses in the three spatial directions were performed simultaneously.

RB-TB SSI analysis documentation is provided in references [14] and [58].

<u>Station Blackout (SBO) Building</u>

SSI analyses without considering ground motion incoherence were performed for the SBO Building. The HROI was selected to be 1E-05 hazard level based on insights from incremental risk quantifications, especially regarding the relative risk-significance of different acceleration intervals and individual components.

Time histories (matched to GMRS hazard level FIRS2) were scaled to HROI of SBO Building (1E-05) by using the ratio of 1E-05 FIRS2 PGA to GMRS Level (in between 1E-04 and 1E-05) FIRS2 PGA before being used in the SSI analysis.

The spectral shapes of the 1E-05 and the GMRS spectra are similar and the use of scaling to develop a hazard compatible time history is considered appropriate. Based on the cracking assessment, a portion of the SBO Building is modeled with cracked section properties. Hazard consistent strain-compatible soil properties at 1E-05 are used.

The SBO SSI analyses consider soil and structural property variation via use of BE, LB, and UB structure models and BE, LB, and UB soil models such that five analysis cases are developed: BE_{soil}-$BE_{structure}$, LB_{soil}-$BE_{structure}$, UB_{soil}-$BE_{structure}$, BE_{soil}-$LB_{structure}$, BE_{soil}-$UB_{structure}$. Ground motion variability is considered via use of five independent sets of time histories for each analysis case. The five sets of time histories, which were matched to GMRS hazard level FIRS2 spectra were provided as part of the PSHA [6]. Soil properties for each layer of each variable soil case (BE, LB, and UB) were defined consistent with the results of the probabilistic site response analysis performed with the PSHA [6].

The MTR/SASSI analysis code [59] was used to perform the SSI analyses. Cutoff frequency for the SSI analyses was chosen to be 50 Hz, and the SSI models were sufficiently refined to transmit frequencies up to at least 50 Hz through the soil/rock-foundation interface. All SSI analyses utilized the SASSI Direct Method and the analyses in the three spatial directions were performed simultaneously.

SBO SSI analysis documentation can be found in reference [15].

A list of structures and description of relevant parameters are provided in Table 4.3-1.

4.3.3 Structure Response Models

Reactor Building – Turbine Building (RB-TB) Complex

A detailed 3D finite-element seismic model of the RB-TB was developed based on industry codes and standards (ASCE/SEI 4-16 [22] and ASCE/SEI 43-05 [18]) to obtain median-centered response analyses including SSI effects at the GMRS and 3x GMRS hazard levels. The model was sufficiently refined to capture building torsion effects, out-of-plane floor response, and in-plane floor diaphragm stiffness. Mass sources included self-weight, equipment, distributive systems, and seismic live load. Concrete and steel material properties were building-specific and based on plant data.

For the RB-TB, the individual structures share a common foundation, and the separate buildings are constructed monolithically (continuous and/or dowelled walls and slabs). Therefore, a new, combined RB-TB detailed 3D finite element model (FEM) was generated to consider the coupled construction and structural behavior and the common foundation beneath the entire RB-TB. In the baseline

(GMRS) analyses which considered uncracked concrete, the structural damping value used (as a percentage of critical damping) was 4% for concrete. For the supplemental analyses considering some cracked concrete where appropriate, concrete damping was based on stress-state / degradation-level: 4% of critical damping for uncracked concrete elements, 7% for cracked concrete elements, and 10% for further degraded concrete elements.

Structural model verification was performed by comparing the total mass and fixed-base fundamental frequencies to the existing LMSMs, as well as performing static analyses considering 1.0g acceleration forces in the vertical and two horizontal directions to confirm reasonable structural behavior. SSI model verification was performed by mass comparison and careful review of transfer functions in all directions and all structure/soil cases. Transfer function review included, for example, confirmation that low frequency response approached 1.0 for on-axis directions and 0.0 for off-axis directions, reasonableness of amplification with increased building elevation, and comparison of resonant peaks to fixed-base frequency analyses of the structure and site response analyses of the soil column.

Following the frequency-domain SSI analyses for the RB-TB, the in-structure response spectra (ISRS) were developed using spectrally matched time-histories. Both horizontal and vertical ISRS were computed from the time-history motions at various floor levels and other important locations. Selection of the locations at which response was calculated was based on equipment location within the buildings. For both the baseline and supplemental ISRS, small plant areas / rooms were defined to capture each component location, and the responses at representative nodes within each area were included in the response at that area.

The ISRS were calculated in the frequency range of 0.1 Hz to 100 Hz and are the algebraic sum of the response obtained for each of the three directions of input ground motion.

For the DRE RB-TB dynamic analyses, both median (~50^{th}%) and conservative (~84^{th}%) estimates of ISRS were developed from a series of structural response analyses which separately considered variability in structural properties, soil properties, and ground motion characteristics. The separate analysis cases were combined to capture the collective effect of such independent variabilities on the median and conservative response. For the RB-TB, a multi-case deterministic approach was used where the structural frequency and soil properties were varied. Each of the five analysis cases were analyzed using five time-histories. The median ISRS were developed by averaging the response from the time-histories for each individual analysis case, and then separately averaging the response from the varied soil cases and structure cases and enveloping those two averages. Conservative ISRS were developed by averaging the response from the time-histories for each individual analysis case, and then separately enveloping the response from the varied soil cases and structure cases and enveloping those two envelopes.

Station Blackout (SBO) Building

A detailed 3D finite-element seismic model of the SBO Building was developed based on industry codes and standards (ASCE/SEI 4-16 [22] and ASCE/SEI 43-05 [18]) to obtain median-centered response analyses including SSI effects at the 1E-05 hazard level. The model was sufficiently refined to capture building torsion effects, out-of-plane floor response, and in-plane floor diaphragm stiffness. Mass sources included self-weight, equipment, distributive systems, and seismic live load. Concrete and steel material properties were building-specific and based on plant data.

In the analyses which considered uncracked concrete, the structural damping value used (as a percentage of critical damping) was 4% for concrete. For the analyses considering cracked concrete, the structural damping value used was 7% for concrete.

Structural model verification was performed by comparing the static analyses considering 1.0g acceleration forces in the vertical and two horizontal directions to confirm reasonable structural behavior. SSI model verification was performed by mass comparison and careful review of transfer functions in all directions and all structure/soil cases. Transfer function review was performed in the same way as described for the RB-TB.

Following the frequency-domain SSI analyses for the SBO Building, the in-structure response spectra (ISRS) were developed using spectrally matched time-histories. Both horizontal and vertical ISRS were computed from the time-history motions at various floor levels and other important locations. Selection of the locations at which response was calculated was based on equipment location within the buildings.

The ISRS were calculated in the frequency range of 0.1 Hz to 100 Hz and are the algebraic sum of the response obtained for each of the three directions of input ground motion.

For the SBO Building dynamic analyses, both median (~50th%) and conservative (~84th%) estimates of ISRS were developed from a series of structural response analyses which separately considered variability in structural properties, soil properties, and ground motion characteristics. The separate analysis cases were combined to capture the collective effect of such independent variabilities on the median and conservative response. For the SBO Building, a multi-case deterministic approach was used where the structural frequency and soil properties were varied. Each of the five analysis cases were analyzed using five time-histories. The median ISRS were developed by averaging the response from the time-histories for each individual analysis case, and then separately averaging the response from the varied soil cases and structure cases and enveloping those two averages. Conservative ISRS were developed by averaging the response from the time-histories for each individual analysis case, and then separately enveloping the response from the varied soil cases and structure cases and enveloping those two envelopes.

Crib House

A detailed 3-D finite element seismic model was developed for the Crib House. A deterministic response analysis is performed for the Crib House to obtain to median centered seismic response of the structure at 1E-05 Hazard Level. Randomly, one time-history set is selected from the five sets available in the PSHA report [6]. The approach used in here is judged to be reasonable and beneficial given the low volume, high capacities and risk-significance of components housed inside the Crib House.

Table 4.3-1 summarizes the type of analysis and model used for each of the major structures modeled in the SPRA.

Table 4.3-1 Description of Structures and Dynamic Analysis Methods for DRE SPRA [6, 14, 15, 16, 58]

Structure	Foundation Condition	Type of Model	Analysis Method	Comments/Other Information
RB-TB Complex (includes Reactor Buildings, Turbine Buildings, Main Control Room, and HPCI Building)	Variable foundation and embedment depths. Partially embedded in soft rock. Deepest portion founded on hard rock overlying softer rock	Detailed 3D coupled FEM	Multi-case Deterministic SSI	Shear Wave velocity ≈2,600 ft/sec for top ~40 ft. below grade and ~8,000 ft/sec at foundation level; SSI analysis performed with incoherence, 5 soil-structure cases used (BE, UB, LB cases for structure and soil), 5 time-histories (T-H) for each case
SBO Building	Top of ground floor is 6 inch above grade. Near surface founded.	Detailed 3D coupled FEM	Multi-case Deterministic SSI	5 soil-structure cases used (BE, UB, LB cases for structure and soil), 5 time-histories (T-H) for each case
Crib House	Variable foundation. Rock founded.	Detailed 3D coupled FEM	Fixed base deterministic analysis	One soil-structure case, one time-history.

4.3.4 Seismic Structure Response Analysis Technical Adequacy

The DRE SPRA Seismic Structure Response and Soil Structure Interaction Analyses were subjected to an independent peer review against the pertinent requirements in the PRA Standard [4]. The peer review was performed relative to Capability Category II for the full set of requirements in the PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the DRE SPRA Seismic Structure Response and Soil Structure Interaction Analyses are suitable for this SPRA application.

4.4 SSC Fragility Analysis

The SSC seismic fragility analysis considers the impact of seismic events on the probability of SSC failures at a given value of a seismic motion parameter defined as PGA. The fragilities of the SSCs that participate in the Seismic PRA accident sequences, i.e., those included on the SEL are addressed in the model. Seismic fragilities for the significant risk contributors (i.e., those which have an important contribution to plant risk, are realistic and plant-specific based on actual current conditions of the SSCs in the plant) are confirmed through the detailed walkdown of the plant.

This section summarizes the fragility analysis methodology and presents a tabulation of the fragilities with appropriate parameters for those SSCs determined to be sufficiently risk important, based on the final Seismic PRA quantification (as summarized in Section 5). Important assumptions and important sources of uncertainty, and any particular fragility-related insights identified, are also discussed.

4.4.1 SSC Screening Approach

The DRE SEL, consisting of approximately 6000 components, was reviewed, analyzed, and then reduced to about 900 components after various screens and walkdowns. The process of reducing the SEL is an iterative and multi-step process as summarized below.

First, the SEL provided to the Seismic Review Team (SRT) was reduced by removing components judged to be non-contributors to the overall response of the SPRA. This includes components deemed to be unnecessary for safe plant operation or shutdown.

Components that are judged inherently rugged were also screened out from requiring development of fragility information. These items included manual valves, check valves, cables and reset pushbuttons. These components are driven by the system they are mounted on as they are typically more rugged. Passive valves are small, lightweight, robust, and are for the most part mounted in line with piping. They do not need to change state during or after an event and have no external vulnerabilities. While the failure of one of these valves can contribute to the results of the SPRA, they will be bounded by the fragility of the distribution system they are attached to. No fragility value is specifically developed for passive valves, but fragility for piping is developed. Piping is walked by as part of the distribution system walkdown.

The typical SPRA practice for certain SSCs is to confirm by visual observation that seismic-induced failures of such items are low likelihood and non-significant risk contributors. These SSCs include MOVs, AOVs, dampers, piping items (pipes, orifices, flanges, reducers, etc.), filters and strainers. These components at DRE site were identified based on walkdowns and walk-bys and they were assigned 2g HCLPF capacity. Inherently rugged items were also reviewed on an area basis during the walkdown to ensure they did not have any adverse seismic interaction concerns such as being mounted to a block wall.

The components that reside inside other components are screened by the rule-of-the-box. Examples include lube oil pump or heat exchanger on a steam-driven turbine pump skid. Like active valves, these components are still addressed in the fragility analysis, but a walkdown of the host component is all that is necessary. These devices are modeled in the SPRA with the fragility value of their box to which they are assigned. It was assured that host components containing devices will be within the SEL themselves.

4.4.2 SSC Fragility Analysis Methodology

For the DRE SPRA, the following methods were used to determine seismic fragilities for SSCs included in the SPRA:

Consistent with the requirements in ASME/ANS PRA Standard [4], the fragility analysis for the selected SSCs is based on the methodology in EPRI guidelines. The strategy for developing the fragilities for the complete set of SSCs on the Seismic PRA SEL follows the recommendations of EPRI NP-6041-SL [12], EPRI 1019200 [20], EPRI 103959 [19] and EPRI 3002000709 [10] and proceeds progressively from using experienced-based capacities to component-specific-evaluations. Regardless of the method, the development of fragility estimates uses plant-specific information based on SSC conditions, as confirmed through detailed walkdowns. Generic fragilities are not assigned to any individual component or structure.

Components are first binned into equipment classes, e.g. EPRI classes presented in Appendix F of EPRI NP-6041-SL [12] and then grouped according to similarity and location. Representative samples in each equipment group are then evaluated to obtain fragility estimates for all the items in the group.

The SPRA approach used at DRE initially utilized three quantifications. In addition to these formal quantifications, various sensitivity studies were performed during the course of the effort to help identify important risk contributors. After each quantification and completion of the sensitivity studies, components identified as risk significant were selected and evaluated further in an attempt to improve their calculated fragilities in order to reduce their risk significance. This approach has been successfully implemented at several plants and is in compliance with the ASME Standard [4] and the SPID [2]. All three quantifications and numerous sensitivity studies were performed prior to the peer review. Subsequent to the peer review and in an effort to address peer review findings, additional quantifications were performed. After each quantification, the results were reviewed to determine if additional insights were obtained and to determine if further refinement of fragilities associated with top risk contributors would improve the results and yield a more realistic model.

For the first quantification, site specific representative fragilities (referred to as 'representative' throughout) were typically developed by scaling existing design basis calculations to account for available margins in the design. This is the margin between allowable values associated with design requirements and values associated with HCLPF evaluations. These margins were used to develop a Safety Factor which is anchored to the PGA of the GMRS to estimate a HCLPF fragility value. Additionally, the results from the USI A-46 [64] and IPEEE evaluations [53, 54] were used to estimate the fragility parameters. The generic values of aleatory variability and epistemic uncertainty from the SPID [2] were applied to the HCLPF to obtain the median fragility value.

For the second quantification, "enhanced" fragilities were provided for top risk contributors to both SCDF and SLERF. The top risk contributors were determined based on the Fussell-Vesely Importance measure (FV) numbers from the initial quantification and subsequent sensitivity studies. The cutoff FV value for selecting components from the first quantification was 5E-03 for both SCDF and SLERF, consistent with the threshold from the ASME Standard. The fragilities were calculated using the Conservative Deterministic Failure Margin (CDFM) method to determine the HCLPF. The generic uncertainty values, as recommended in Table 6.2 of the SPID for various SSCs, were used to estimate the median fragility value, with the generic uncertainty values adjusted if needed to account for specific conditions. Site specific information obtained from walkdowns and plant documentation, including actual anchorage and configuration details, were used along with ISRS at the location of the individual components to develop the HCLPF.

For the third quantification, refined fragilities were developed for the dominant risk contributors. The dominant risk contributors were selected by reviewing the FV numbers from the second quantification followed by a series of sensitivity studies performed after the second quantification. The ASME/ANS RA-Sb-2013 [4] PRA Standard does not prescribe any specific methodology for calculating fragility, only that they need to be realistic. The guidance provided in EPRI SPID [2] was used to select a handful of high-risk contributors for developing the detailed fragilities. Detailed fragilities were developed using detailed CDFM Method or Hybrid Separation of Variables (Hybrid SoV) Method. In the Hybrid SoV approach, the HCLPF capacity and the median capacity are calculated separately first. Then, the logarithmic standard deviations are back-calculated from the ratio of the median capacity and the HCLPF capacity.

Subsequent to the peer review, additional quantifications were performed to further refine the SPRA model and to respond to peer review findings. For the post-peer review quantification, refined fragilities were developed for the dominant risk contributors. The dominant risk contributors were selected by reviewing the FV numbers from the third quantification followed by a series of sensitivity studies performed after the post-peer review quantification. Detailed fragilities were developed using Separation of Variables (SoV) Method for a handful of equipment with significant risk contribution. Using the SoV approach, the median capacity was directly calculated. The variabilities were not based on generic values. A unique set of randomness and uncertainty variabilities were computed for each equipment for which SoV calculations were performed. The ASME Standard requires the fragilities for the dominant risk contributors to be realistic. This approach is more realistic and will provide more realistic fragility values. The fragilities for the remaining components determined to be significant contributors to risk following the various quantifications were developed using the Hybrid SoV Method and/or CDFM Method. Table 4.4.2-1 provides a summary of the number of components for which fragilities were developed for each quantification.

Critical failure modes were identified, (structure/anchorage or functionality or block wall), and fragility calculations were performed for the median capacity Am for each of the failure modes. The lowest, governing Am was selected and input to the model. When two or more failure modes were close (i.e. their median capacities within 20% of each other and the failure modes were determined to not be correlated), the assigned fragility was based on a combined probability of failure, considering the closely-spaced modes together.

Table 4.4.2-1 Approximate Numbers of Refined SSC and Relay Fragilities for Each Risk Quantification [21]

Quantification	Count of SSC (1)
Q1	~1500
Q2	~800
Q3	~150
Post Peer Review	~29

(Note 1: The number for the first quantification includes ~900 SSC fragilities and ~600 relay fragilities. Other numbers for the later quantifications are a combination of SSC and relay fragilities, as well.)

Representative seismic fragilities for DRE structures were developed as input to the first risk quantification (Q1). The representative seismic fragility evaluations were less detailed and designed to be slightly conservatively biased. From the results of Q1 and Q2, potentially risk-significant structures were identified and a more refined seismic fragility evaluation for these structures was performed in order to ensure realistic fragility input parameters for risk-significant SSCs.

The enhanced structural seismic fragilities were calculated using the CDFM method. The shear walls were generally evaluated for three failure modes: diagonal shear cracking, in-plane shear flexure, and shear friction.

Structure fragilities for Reactor Building – Turbine Building (RB-TB) structure complex, Station Blackout Building (SBO), Crib House, Isolation Condenser Pump House (ICPH), Ventilation Chimney for Unit 1 and Ventilation Chimney for Units 2 and 3, and The Dresden Island Lock and Dam were calculated. FLEX storage buildings fragility was calculated based on scaling from the design calculation.

4.4.3 SSC Fragility Analysis Results and Insights

The final set of fragilities for the risk important contributors to SCDF and SLERF are summarized in Section 5. Refer to Tables 5.4-2 and 5.4-3 for SSCs which are important contributors to SCDF and Tables 5.5-2 and 5.5-3 for SSCs which are important contributors to SLERF. Detailed (SoV, Hybrid SoV, or detailed CDFM) calculations have generally been performed for the highest risk significant SSCs, as well as for selected other components.

4.4.4 SSC Fragility Analysis Technical Adequacy

The DRE Seismic PRA SSC Fragility Analysis [21] was subjected to an independent peer review against the pertinent requirements in the PRA Standard [4]. The SSC fragility analysis was peer reviewed relative to Capability Category II for the full set of supporting requirements in the standard. The peer review assessment and subsequent disposition of peer review findings are described in Appendix A and establishes that the DRE SPRA SSC Fragility Analysis is suitable for this SPRA application.

5. Plant Seismic Logic Model

This section summarizes the adaptation of the DRE internal events at power PRA model to create the seismic PRA plant response (logic) model.

The seismic plant response analysis models the various combinations of structural, equipment, and human failures given the occurrence of a seismic event that could initiate and propagate a seismic core damage or large early release sequence. This model is quantified to determine the overall SCDF and SLERF and to identify the important contributors, e.g., important accident sequences, SSC failures, and human actions. The quantification process also includes an evaluation of sources of uncertainty and provides a perspective on how such sources of uncertainty affect SPRA insights.

5.1 Development of the SPRA Plant Seismic Logic Model

The DRE seismic response model was developed by starting with the 2017 DRE internal events at power PRA model of record as of May 4, 2018 [75], and adapting the model in accordance with guidance in the SPID [2] and PRA Standard [4], including adding seismic fragility-related basic events to the appropriate portions of the internal events PRA, eliminating some parts of the internal events model that do not apply or that were screened-out, and adjusting the internal events PRA model human reliability analysis to account for response during and following a seismic event.

For the DRE SPRA, the following sections discuss the methods used to develop the seismic plant response model. The elements of the analysis are as follows:

- The seismic initiators are derived from the site specific seismic hazard analysis.
- The seismic accident sequences are developed by using a Seismic Initiating Event Tree (SIET) and, a set of Level 1 (core damage) and Level 2 (post-core damage) accident sequence event trees based on the DRE specific FPIE PRA model.
- The seismic system fault trees that support the event tree quantification are based on the DRE specific FPIE PRA model.
- The fragility analysis that is performed to characterize the seismic induced failure modes of SSCs is used to model seismic induced system failure modes in the event tree and fault tree models.
- The interface of the operators with accident mitigation systems is incorporated into the seismic system fault trees as modified by the fragility analysis.
- The software is used to process the above information into a cohesive framework and quantify the models. (See Section 5.3.1)

Initiating Events

The frequency of earthquakes at the DRE site is based on site-specific probabilistic seismic hazard analysis developed by Fugro [6]. The mean hazard curve is divided into eight ground motion ranges (seismic hazard intervals) for use in developing and quantifying the SPRA. Each seismic hazard interval initiator in the DRE seismic evaluation is assigned an initiator ID (e.g., %G4, "Seismic Initiating Event 0.4g to <0.5g PGA") and an initiator frequency. The frequency for the seismic hazard interval initiator is calculated as the exceedance frequency of the beginning point of the ground motion range minus the exceedance frequency of the end point of the ground motion range. The frequency of the last (highest) ground motion interval is the exceedance frequency at the beginning point of that interval. The seismic initiating events developed for the DRE SPRA are documented in the DRE Seismic PRA Initiating Event Notebook [47].

Accident Sequences

Event trees and fault trees are used to model the SPRA accident sequences. The accident sequence model accounts for the unique failure modes caused by seismic induced ground motion in addition to combinations of non-seismic failure modes. The sequence models address all the mitigation responses necessary to bring the plant to a safe shutdown. Event trees are a useful logic tool for displaying the seismic accident sequences.

The SPRA model process uses a seismic pre-tree, i.e., the Seismic Initiating Event Tree (SIET), to sort out the more pervasive effects of a seismic event that can lead directly to core damage or to a degraded plant condition (e.g., induced LOOP, induced large LOCA). The second tier of the event trees are systemic event trees (identical to those in the Level 1 internal events PRA) that evaluate the plant response and mitigation capability given the preconditions established in the SIET. Sequence logic transfers directly from the SIET into the systemic event trees to ensure that no information is lost in these transfers. The event trees are used to define the accident sequence progression and the assigned end state of the Level 1 events.

The methodology to group and transfer core damage sequences from the Level 1 event trees to the Level 2 Containment Event Trees (CETs) is identical to the FPIE PRA methodology. In addition, the seismic PRA is judged to create no unique Level 2 accident scenarios such that the SPRA Level 2 CETs are also identical to the FPIE CETs. The SPRA Level 2 CETs employ the identical definition for LERF timing and radionuclide release categories as the FPIE CETs. The Level 1 and Level 2 seismic accident sequence evaluation is documented in the DRE Seismic PRA Event Tree Notebook [48]. A sensitivity study has been performed to evaluate the potential risk impact on the SPRA results if the LERF definition is revised to consider seismic induced impacts of sheltering and evacuation offsite.

System Fault Trees

The SPRA system models reflect the as-built and as-operated plant. The internal events system fault tree models derived directly from the internal events model are used as a starting point for development of the SPRA system fault tree models.

The internal events PRA system fault trees are modified to reflect the unique aspects of the seismic hazard challenge. Therefore, both seismic and random SSC failures are accounted for in the SPRA model. These seismic response modifications include the following specific seismic attributes:

- Seismic hazard interval initiating events are inserted as the initiating event logic of the SIET sequences, as well as into system fault tree structures.
- SSC fragilities that would lead to a system or train failure are added to the system models.
- Effects on operator error probabilities due to the seismic induced changes to performance shaping factors are incorporated in the HEP calculations.
- Each of the above effects varies with seismic hazard intensity, i.e., varies by seismic hazard interval initiating event.

Specific aspects of the SSC fragility modeling and impacts include, but are not limited to, the following:

- For seismic induced LOOP events, recovery of offsite power is not credited for any hazard interval (e.g., failure of ceramic insulators).
- A fragility for seismic induced Very Small LOCA is explicitly modeled (e.g., potential to model the equivalent impact of a recirculation pump seal LOCA).
- The unscreened contact chatter scenarios provided in Table 4.1.2-1 are the ones explicitly included in the SPRA model based on the identified system impact (e.g., divisional diesel fails to start or load). Given the high number of total unscreened contact chatter scenarios (i.e., approximately 600), not all contact chatter scenarios are explicitly included in the SPRA model. Initial SPRA model quantifications helped identify the risk impact of individual or correlated contact chatter scenarios based on associated system impact and fragility value. In addition, a Human Reliability Analysis (HRA) is performed to evaluate the potential credit for operator recovery of the contact chatter scenario (e.g., locally reset diesel) in the SPRA model. The DRE Seismic PRA Fragility Modeling Notebook [50] and the DRE Seismic PRA Methodology Notebook [46] provide further details on the methodology for including contact chatter events in the SPRA model.
- One of the aspects of the seismic hazard is that it could induce either a fire or a flood event. Because of this possibility, an assessment of these induced hazards is needed. The DRE SPRA approach to identification and

assessment of postulated seismic-fire and seismic-flood interactions follows the SPRA Implementation Guide 3002000709 [10] and ASME/ANS RA-Sb–2013 [4] Supporting Requirements SFR-E4, SFR-E-5 and SPR-B9. This includes use of DRE fire PRA and internal flooding PRA information as well as plant walkdowns and drawing reviews to identify sources for consideration. The postulated seismic-induced sources for assessment includes non-safety electrical cabinets (although these are powered by offsite AC, it may be postulated that such cabinets may experience seismic-induced arcing prior to seismic-induced loss of offsite power). Walkdowns were performed to identify additional sources as well as to assess the sources of seismic induced fire or flood events and to characterize their potential risk for inclusion in the seismic PRA model [31]. Hazards identified in the internal flood study and the internal fire analysis [86] were considered by the walkdown team.

- Seismic-induced flooding from tanks and piping systems was also assessed. Those flooding scenarios of potential significance to the SPRA include piping systems with a significant suction source volume and which can cause flow without auxiliary power and flood areas with equipment used in the SPRA (e.g., fire protection piping containing Victaulic couplings). Other potential scenarios were investigated and determined to be non-significant risk contributors either due to limited consequences or due to the piping having sufficiently high seismic-capacity.
- SSCs with a potential impact on containment integrity (e.g., containment bypass scenarios) were also evaluated and modeled accordingly for the Level 2 LERF model.

The PRA fault tree models contain the basic Boolean logic regarding SSC failure modes and their associated probabilities. For the DRE Seismic PRA, three types of fault tree models are developed:

- System Fault Trees
- Event Tree Nodal Fault Trees
- Integrated Fault Tree to model CDF and LERF accident sequences

<u>Fragilities</u>

Seismic fragility of a structure or equipment item is defined as the conditional probability of its failure at a given value of the seismic input or response parameter (e.g., PGA, stress, moment, or spectral acceleration). Seismic fragilities are needed in an SPRA to estimate the conditional seismic-induced failure probabilities of structures and mitigating systems (including their support systems) given a seismic initiating event. The fragilities are calculated using the methodologies discussed in Section 4.4.

SSC's of the same type that also possess the same location, elevation, and orientation are assigned to a single, correlated group. Due to the widespread nature of a seismic event, if a single SSC in a correlated group were to fail, it can be assumed that all SSC's in the group would fail. This is consistent with the current state of practice.

Over 200 fragility groups are modeled in the DRE SPRA. Of the over 200 fragility groups, approximately 120 involve correlated fragility groups. Fully correlated response of the same or very similar equipment in the same structure and elevation is assumed. The SPRA does not model any partial correlation of fragility groups. Some of the risk significant correlated fragility groups include, but are not limited to, the following:

- Various Control Room Instrument Panels (6 SSCs)
- 125 VDC Buses (4 SSCs)
- SBO DG #2 and #3 Batteries 6A and 7A (2 SSCs)
- 4160V Buses (2 SSCs)

The development of the fragility groups, fragility correlation groups, and how they are incorporated into the SPRA model is documented in the DRE Seismic PRA Fragility Modeling Notebook [50].

Human Reliability Analysis

The scope of the Seismic PRA HRA is focused on the post-initiator operator actions. The pre-initiator Human Interactions (HI) are performed prior to a seismic event and are therefore not affected by the seismic event. Therefore, the assessment of the pre-initiator HIs remain the same as in the Internal Events PRA HRA (the pre-initiator HEPs existing in the FPIE system fault tree models propagate through the SPRA accident sequence logic and quantification).

The DRE Internal Events PRA uses a systematic approach for the identification and evaluation of operator actions in response to postulated accidents. The methods used are well established and are applied appropriately to the internal events models through use of the EPRI HRA Calculator® [65]. The seismic HRA uses these operator actions and these base calculations of Human Error Probabilities (HEPs) as input to the seismic HRA. DRE uses the EPRI HRA Calculator® for the internal events PRA, the fire PRA and the seismic PRA.

The human actions that are modeled in the Level 1 and Level 2 internal events PRA are included as basic events in the fault trees. The Human Error Probabilities (HEPs) generated from the Human Reliability Analysis (HRA) have been assigned unique basic event names. Additional actions specific to seismic conditions (relay chatter recovery actions) are incorporated into the SPRA.

The approach used for the SPRA HRA is to develop an integrated performance shaping factor (IPSF) for each HEP that is representative of the seismic accident sequence and apply the additional performance shape factor (i.e., IPSF) to the detailed internal events PRA HEPs based on EPRI guidance documents [66]. For HEPs identified to potentially have a high-risk contribution based on SPRA model quantifications, more detailed HEP evaluations (incorporating seismic impact adjustments) are developed using the EPRI HRA Calculator®. The dependent HEP probabilities are then also re-calculated using the seismic-adjusted HEPs. The details are documented in the DRE Seismic PRA HRA Notebook [49].

5.2 SPRA Plant Seismic Logic Model Technical Adequacy

The DRE SPRA seismic plant response methodology and analysis were subjected to an independent peer review against the pertinent requirements in the PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the DRE SPRA seismic plant response analysis is suitable for this SPRA application.

5.3 Seismic Risk Quantification

In the SPRA risk quantification the seismic hazard is integrated with the seismic response analysis model to calculate the frequencies of core damage and large early release of radioactivity to the environment. This section describes the SPRA quantification methodology and important modeling assumptions.

5.3.1 SPRA Quantification Methodology

For the DRE SPRA, the following approach was used to quantify the seismic plant response model and determine seismic CDF and LERF:

The analytic tools for the development of a quantified model are the EPRI CAFTA code suite augmented by the ACUBE Binary Decision Diagram (BDD) software. The EPRI CAFTA code suite [67] is well tested and widely used in various industries in numerous countries. The ACUBE code [68] is still expanding its capability, and this will increase the number of cutsets that can be precisely calculated using the BDD algorithm and less reliance on the Minimum Cut Upper Bound (MCUB) approximation.

The DRE SPRA model has been developed so that it is modular. Event trees (convertible to fault trees), event tree top logic (nodal fault trees), and system-level fault trees all have been developed as distinct files. In addition, the FRANX tool from the EPRI CAFTA suite has been used to develop a relational database for linking individual fragility events to existing modeled basic events. This modular structure allows individual files to remain manageable and reviewable. A single-top model used for quantification is developed by merging the previously described files. Merging the files into a single-top model is a standard CAFTA modeling technique and is performed by the user.

The model is quantified using PRAQuant, which is a code within the total CAFTA software suite. Also, due to the special circumstances within seismic modeling (i.e., over-counting caused by numerous high failure probability events), the ACUBE code, which uses the BDD algorithm, is used in model quantification to obtain a realistic assessment of the total CDF/LERF risk metric.

5.3.2 SPRA Model and Quantification Assumptions

This section discusses modeling assumptions made as part of the seismic PRA quantification. In addition, potential conservatisms that remain in the SPRA risk profile calculation include the following:

Seismic Human Reliability Analysis (HRA)

- As expected of a SPRA, the post-initiator FPIE-based human error probabilities (HEPs) in the SPRA are reconsidered and adjusted upward in failure probability to consider various seismic performance shaping factors. The approach used sets most post-initiator HEPs (except for FLEX actions) directly to 1.0 failure probability for the two (2) highest hazard intervals (i.e., %G7 for 0.8g to 1.0g and %G8 for >1.0g). Based on anecdotal information from seismic events, this approach is likely conservatively biased.

Seismic Correlation

- 100% fragility correlation is assumed for like equipment installed similarly and located on the same elevation of the same building. This applies to non-significant and significant risk contributors (e.g., various instrument control panels in the main control room). This approach defeats design redundancies if the redundancies are the same equipment and in the same general location. This modeling is a common SPRA practice; in fact, most SPRAs in the world use this method of applying a binary approach to fragility correlation modeling. Assigning partial correlation factors to SSC fragility groups throughout the SPRA would likely create a model that cannot be quantified at a reasonable truncation limit and will introduce another significant element of modeling uncertainty (i.e., bases for the various partial correlation factors).

Accident Sequence Modeling

Generally, there are limited success states for a seismic induced SBO because repair/recovery of seismic-induced failures (including seismic-induced loss of offsite power) is typically not credited in the SPRA. This is a typical SPRA approach. If recovery of offsite power were credited using an extreme weather related OSP non-recovery curve (which would be reflective of downed lines and poles) then the calculated SCDF and SLERF may potentially reduce by a few percentage points (extreme weather related OSP non-recovery curves have very high failure probabilities in the first 24 hours). However, the DRE SPRA model explicitly credits FLEX mitigation strategies [35-38; 85]. If the FLEX equipment can be aligned in a timely manner and successfully operates as designed, then a success state (i.e., no core damage) for a seismic induced SBO can be achieved.

The following FLEX strategies are incorporated into the SPRA (with system logic, seismic fragilities and human actions for the alignments):

- IC makeup using FLEX pumps
- RPV injection using FLEX pumps
- FLEX 480V AC electrical power restoration using FLEX generators

The Dresden at-power SPRA does not incorporate the FLEX strategy to align the FLEX pump for spent fuel pool (SFP) makeup. This strategy has no direct relationship to at-power SCDF and SLERF accident sequences.

- No credit is modeled for isolation of seismic-induced breaks outside containment. The seismic-induced BOC sequence is modeled as leading directly to core damage and LERF. This is a small conservatism. If isolation of such a break could be credited with a proper basis in the modeling, the impact on SCDF and SLERF would be negligible because of the high calculated Am for the associated piping.
- Assignment of LERF to certain Level 2 PRA accident sequences may be conservative. Certain phenomena exist that recent studies show, such as the NRC SOARCA studies [69], may require reconsideration in the DRE PRA models. Examples include the timings and magnitudes of severe accidents involving RPV melt-through and subsequent drywell shell melt-through. The degree of potential conservatisms in these types of Level 2 sequences is discussed in sensitivity cases in Section 5.7 of this report.

<u>Quantification Process</u>

- The SPRA quantification process makes use of the EPRI ACUBE software module (which employs a binary decision diagram, BDD, algorithm) to minimize “overcounting” in the Boolean summation of result cutsets. A very minor level of over-counting in the SCDF and SLERF metrics (essentially the true values are achieved) exists in the base quantification. This level of precision can be challenged by individual risk applications that may set equipment to “failed” or high failure rates, but such challenges will be addressed as they arise in application of the model to risk informed decision making.

5.4 SCDF Results

The seismic PRA performed for DRE shows that the point estimate mean seismic CDF is 5.8E-06/yr for Unit 2 and also 5.8E-06/yr for Unit 3 [52]. The Unit 2 Seismic CDF of 5.8E-06/yr is calculated with a single top CAFTA model at a truncation that ranges from 1E-07/yr to 1E-12/yr depending on the seismic hazard interval quantified. The single top PRA model could not be quantified at a consistent truncation limit for all seismic hazard intervals due to quantification limitations. Refer to Section 5.7 for a summary of the quantification truncation limits that support convergence of the DRE SPRA model for both SCDF and SLERF quantifications.

Given the similarities in the Unit 2 and Unit 3 SCDF values, the remainder of this section focuses on the Unit 2 results, except as noted. In general, DRE Unit 2 and Unit 3 are symmetrical.

The single top model accounts for both the accident sequence failure logic as well as the success logic. This calculation is then refined by the use of the ACUBE computer code operating on the cutsets from the single top to reduce any over counting of failures in the cutsets due to high failure probabilities in the cutsets.

Important Seismic Initiating Event Contributors

Table 5.4-1 summarizes the Unit 2 SCDF contributors by seismic initiating event. Figure 5.4-1 displays the results of Table 5.4-1 in graphical pie chart form, i.e., the CDF contributors by initiating event. Figure 5.4-2 shows the initiating event contribution in the form of a bar graph.

As can be seen from the graphical display, the seismic initiators %G4, %G5, %G6, and %G8 are the dominant seismic risk contributors. Seismic hazard interval initiator %G7 contributes less to SCDF than does %G8.

The seismic initiating event interval with the highest contribution relative to the CDF risk metric is %G6 (0.6g to 0.8g) with a contribution of 30%.

Conditional Core Damage Probability (CCDP) values were also calculated for the initiators. These CCDP values are displayed in Figure 5.4-3. Figure 5.4-3 shows the CCDP for the %G6-%G8 initiators (0.6g->1.0g) as nearly 1.0. These ground motion values are close to or greater than the median capacity values (A_m) for some of the safety related SSCs at Dresden (e.g., core shroud tie rods, 125V DC battery racks). Thus, it is deemed reasonable that the CCDP for these initiators is very high.

The Unit 3 SCDF contributors by seismic initiating event are similar to those shown for Unit 2.

Important Contributors to CDF

Table 5.4-2 provides the Unit 2 SCDF Fussell-Vesely (FV) importance measures for SSC fragilities. The risk importances are calculated using cutset results (as typical in an R&R workstation environment) and using the EPRI ACUBE software to determine the individual basic event risk importance values. The SCDF FV values for SSC fragilities are based on a weighted sum of the individual SSC FV values calculated for the individual hazard intervals, excluding the G8 interval which is assumed to lead directly to core damage and large early release. Sensitivity case 7c in section 5.7 evaluates the impact of this assumption on risk importance measures. In other words, the total FV of an SSC fragility is the weighted sum of the associated seven (7) SSC fragility basic events (one per hazard interval, except G8). The SSC FV values for each hazard interval are calculated based on the cutset importance measures for each hazard interval as calculated by ACUBE. The weighted sum is a summation of the individual SSC FV values for an individual hazard interval multiplied by the ratio of the associated ACUBE SCDF for an individual hazard interval and the total ACUBE SCDF for hazard intervals G1-G7.

Note: The term FV is used here but the ACUBE software actually produces the Criticality Importance (CI) risk measure in place of FV. The CI and FV measures are very close numerically such that any minor difference in their values is non-significant for typical decision-making purposes. A discussion of the relationship of CI and FV is contained in the DRE SPRA Quantification Notebook. [52]

Consistent with past SPRA models, the top SCDF FV contributors are associated with AC and DC power supplies.

The top 8 contributors to the Unit 2 SCDF FV are as follows [52]:

Normal offsite power (FV = 5.86E-01)

Normal offsite power is expected to have a high FV because there is a high probability for the seismic event to fail offsite power (Am = 0.3g). The fragility is based on a "generic" value for loss of offsite AC power consistent with industry guidance [10].

Various Control Room Panels (FV = 7.76E-02)

Correlated control panel group C20-4 (consisting of Unit 2 Panels 902-15, -16, -17, -18, -19, and -20) has a high risk impact due to the relatively low median capacity of 0.78g and the important components the panels are modeled to fail including various safety related components including diesel generator circuit breakers and AC Buses.

Unit 2 125 VDC Battery Racks (FV = 7.28E-02)

The Unit 2 125 VDC batteries have a high-risk impact because their failure results in loss of the Unit 2 station EDG (i.e., EDG2) and the Unit 2 IC. The risk impact for Loss of the Unit 2 125 VDC is exacerbated when failed in combination with other SSCs (e.g., Unit 3 125 VDC SSCs) that provide redundant defense-in-depth capabilities for mitigation systems such as HPCI and other EDGs (e.g., EDG3 or EDG2/3).

SCRAM (RPV Internals) (FV = 6.80E-02)

Seismic failure of the RPV internals is modeled to prohibit successful insertion of the control rods into the reactor (SCRAM), resulting in an ATWS scenario. The governing failure mode in the fragility calculation is identified as the upper and lower clamps on the core shroud tie rods (Am = 0.75g).

Instrument Rack 2202-7 (FV = 4.68E-02)

The governing seismic failure mode of instrument rack 2202-7 is block wall failure resulting in a relatively low median capacity of 0.39g. This low median capacity combined with seismic failure of this instrument rack modeled to fail LPCI valves and pumps results in high risk significance.

Unit 2 125 VDC Train B Buses (FV = 1.59E-02)

Correlated seismic failure of Unit 2 Buses 2B, 2B-1, and 2B-2 results in similar consequences to seismic failure of the Unit 2 125 VDC Battery Racks described above.

SBO DG 2 Battery 6A and SBO DG 3 Battery 7A (FV = 1.57E-02)

Seismic failure of the Unit 2 and Unit 3 SBO DG batteries results in failure of the Unit 2 and Unit 3 SBO DGs to start and provide power to Bus 23/24 and Bus 33/34. The risk impact for loss of the SBO DGs is exacerbated when failed in combination with other SSCs supporting onsite AC power (e.g., other station EDGs).

Unit 3 125 VDC Battery Racks (FV = 1.52E-02)

Seismic failure of the Unit 3 125 VDC battery racks results in loss of HPCI support systems and loss of EDG3 and EDG2/3 control power supplies. The risk impact for Loss of the Unit 3 125 VDC is exacerbated when failed in combination with other SSCs (e.g., Unit 2 125 VDC SSCs) that provide redundant defense-in-depth capabilities for these mitigation systems.

The quantitative results showed that the only SSC with risk significant non-seismic failure contribution to SCDF (i.e., failures to start, run, etc. with FV > 5E-03) was EDG2/3 failing to run with a Unit 2 SCDF FV = 9.3E-03 and a Unit 3 SCDF FV = 1.2E-02. Non-seismic failure of EDG2 and EDG3 have the same probability as EDG2/3, but EDG2/3 supports power to 4KV Busses 23-1 and 33-1 on Unit 2 and Unit 3, respectively, resulting in greater importance of random failure when combined with the general low seismic capacity of the SBODG support systems (e.g., SBODG 125 VDC batteries with Am ~0.3g). Furthermore, fragility groups modeled to fail EDG2/3 have relatively high median capacities (fragility group S-DGDG3 models seismic failure of EDG2/3 and has Am = 1.61g) while the non-seismic failure probability to run is 4.37E-02, making random failure to run the dominating failure mode for hazard intervals G1-G6.

Table 5.4-3 provides the Unit 3 SCDF Fussell-Vesely (FV) importance measures for SSC fragilities. The Unit 3 SCDF FV contributors are similar to the Unit 2 contributors with the exception of the addition of fragility groups S-INCP04-1-(correlated failure of various Unit 3 control panels), S-ACBS15 (correlated failure of 4160V Buses 33 and 34), and S-DCBC3 (Unit 3 125 VDC battery charger #3) to the Top 10 Unit 3 contributors. The Unit 3 4KV buses 33/34 were calculated to have a lower Am of 0.69g compared to the Unit 2 4KV buses 23/24 which have an Am of 0.76g, to explain why Unit 3 4KV buses 33/34 have a higher FV in the U3 model (FV = 2.03E-02) compared to the Unit 2 4KV buses 23/24 in the U2 model (FV = 1.19E-02).

Table 5.4-4 provides the Unit 2 SCDF FV importance measures for the operator actions. Similar to the total FV for the SSC fragilities, the total FV for the operator actions is the sum of the individual FV values for the G1-G7 hazard intervals.

Two of the risk significant operator actions for Unit 2 SCDF are relay chatter recovery actions. This indicates that seismic induced relay chatter is a dominant failure mode for risk significant components like HPCI and the EDGs.

The top four risk significant operator actions for Unit 2 SCDF are described below:

- Failure to control containment venting (FV = 5.18E-02). The top operator action contributor to the Unit 2 SCDF FV is an operator failure to control containment venting leading to a class 2V accident where decay heat is not removed post containment challenge, leading to core damage. A similar operator action for failure to vent containment using the hard pipe vent is also risk significant with FV = 8.58E-03.

- Failure to inject through 'A' LPCI loop given 'B' LPCI loop failure (FV = 1.85E-02). The second highest operator action contributor to Unit 2 SCDF is operator failure to switch LPCI injection loops when the LPCI loop chosen by the LPCI loop selection logic fails. This action is risk significant because several of the risk significant control panel and instrument rack fragility groups are modeled to fail LPCI injection valves and pumps. The high probability of failure for LPCI valves and pumps increase the importance of operator action to switch to the functioning LPCI loop injection paths.

- Operator fails to recover from relay chatter impacting EDG 2, 3, and/or EDG2/3 (FV = 1.67E-02). Limited credit is provided for operator recovery from any relay chatter scenarios impacting the EDGs due to the time available and time required to perform the necessary actions. A single operator action addresses relay chatter scenario impacting individual EDGs or correlated failure of multiple EDGs.

- Crew fails to align RWCU for letdown (FV = 1.56E-02). This operator action supports reducing RPV water inventory through RWCU to prevent overfill of the RPV and assumed failure of HPCI due to water intrusion into the HPCI steam supply line. At Dresden, the HPCI steam supply line is not from the Main Steam Line like at many US Boiling Water Reactors (BWRs). The HPCI steam supply is a separate penetration approximately 1 foot above the Level 8 trip. Therefore, Dresden is more susceptible to water intrusion events into the HPCI steam line, as observed from plant specific operating experience.

Table 5.4-5 provides the Unit 3 SCDF Fussell-Vesely (FV) importance measures for the operator actions. The Unit 3 SCDF FV contributors are very similar to the Unit 2 contributors.

Top 10 SCDF Cutset Evaluation

Table 5.4-6 provides the Top 10 Unit 2 SCDF cutsets because the Unit 2 results are slightly more limiting for the DRE SPRA model. The Top 10 Unit 3 SCDF cutsets are similar to the Unit 2 Top 10 SCDF cutsets so the Unit 3 cutsets are not explicitly provided. The cutset result file combines the cutsets from all seismic hazard intervals (i.e., %G1 through %G8). The SCDF values identified for each of the cutsets is based on the independent calculated cutset frequency. The integrated SCDF when combining the cutsets using the EPRI ACUBE software results in a much lower total SCDF.

Cutset #1 (SCDF = 9.08E-07/yr): This cutset contains the %G8 initiator (with the availability factor included in the initiating event frequency) along with an accident class and sequence tag. The %G8 interval (>1.0g) was unable to quantify with currently available processing power and is assumed to lead directly to core damage. Assuming that the highest seismic interval leads directly to core damage is consistent with typical industry SPRA models.

As shown in Table 5.4-6, the Accident Class and sequence are shown to be Class V and sequence SIET-020 (seismic induced failure of the RPV supports leading to core damage due to assumed inability to maintain core cooling and subsequent bypass of containment). Having the %G8 initiator lead directly to core damage (i.e., by setting all seismic induced fragilities to TRUE) masks the ability to readily identify the contribution to individual core damage accident classes and sequences. Different accident class contributors would be identified (e.g., Class 1A for loss of makeup with the RPV at high pressure or Class 1B for Station Blackout) if the %G8 initiator and cutsets were quantified in a more detailed manner. However, Class V is identified because it is consistent with assuming that the %G8 initiator leads directly to both a CDFM and LERF end state.

Cutset #2 (SCDF = 7.25E-07/yr): This cutset involves a %G7 seismic initiating event (seismic magnitude 0.8 to 1.0g) leading to a Dual Unit Loss of Offsite Power (DLOOP) accident sequence with HPCI, the IC, and IC makeup initially available. The IC is assumed not available long term to satisfy the 24 hour PRA mission time during a DLOOP event. (This is a modeling assumption carried over from the FPIE PRA model and may be potentially conservative.) Operator failure to initiate SPC early results in unavailability of HPCI long term due to procedural direction to depressurize the RPV prior to reaching the Heat Capacity Temperature Limit (HCTL). CRD is unavailable for long term RPV makeup (e.g., insufficient CST volume during dual unit LOOP), but LPCI or CS are initially available following RPV depressurization. Consistent with typical industry SPRA models, offsite AC power recovery is not credited. Operator failure to initiate SPC late results in the need to vent the primary containment. Primary containment venting is successful, but the operators subsequently fail to control the containment venting evolution (i.e., maintain primary containment pressure in a high pressure band). Failure to control the containment venting results in unavailability of LPCI or CS due to phenomenological issues (e.g., steam binding of ECCS suction flow). Unavailability of RPV makeup following successful primary containment venting results in core damage (Accident Class 2V).

A sensitivity study (i.e., Case 3a in Section 5.7) has been performed to evaluate the potential conservatism associated with the FPIE PRA assumption that the IC is not credited to be available long term to satisfy the 24 hour PRA mission time during a DLOOP event. The sensitivity study supports that the FPIE PRA modeling assumption has a negligible impact on the calculated SCDF.

Cutset #3 (SCDF = 7.25E-07/yr): This %G7 cutset describes a DLOOP scenario with early SPC and IC initially available but no HPCI or IC makeup. A dependent HEP group contains operator actions for the following:

- Failure to align RWCU for letdown and Failure to Close the HPCI steam line isolation valve results in failure of HPCI due to water intrusion into the HPCI steam line
- Failure to initiate IC makeup results in unavailability of the IC long term
- Failure to depressurize the RPV precludes credit for low pressure RPV makeup.

Loss of all high and low pressure RPV makeup results in core damage (Accident Class 1A).

Cutset #4 (SCDF = 7.25E-07/yr): This %G7 cutset describes a DLOOP scenario with failure of the operator to link 4KV Busses 23 and 24 to EDG-powered 4KV Busses 23-1 and 24-1, respectively. Unavailability of power to 4KV Busses 23 and 24 precludes operation of the CCSW pumps to support SPC. Similar to Cutset #2, the IC is not credited to be available long term to satisfy the 24 hour PRA mission time during a DLOOP event and HPCI is not available long term due to RPV depressurization prior to reaching HCTL. Failure of the operator to maintain the RPV in a depressurized state precludes credit for low pressure RPV makeup. Loss of all high and low pressure RPV makeup results in core damage (Accident Class 1BL – late core damage during DLOOP or Station Blackout).

Cutset #5 (SCDF = 7.25E-07/yr): Cutset #5 is similar to Cutset #4 except SPC is unavailable due to operator failure to initiate SPC early. This cutset leads to core damage (Accident Class 1BL – late core damage during DLOOP or Station Blackout).

Cutset #6 (SCDF = 7.25E-07/yr): This cutset is similar to cutset #3 except IC makeup is failed due to operator failure to locally open valves to align makeup from IC makeup pumps and operator failure to align SW makeup to the IC along with seismic failure of the U2/3 diesel fire pump.

Cutset #7 (SCDF = 7.25E-07/yr): This cutset is similar to cutset #6 except IC makeup from FPS is failed due to seismic failure of the fire pump day tank.

Cutset #8 (SCDF = 7.25E-07/yr): Cutset #8 is similar to Cutset #6 except IC makeup from SW is failed due to operator failure to restore SW following the DLOOP caused by seismic failure of offsite AC power.

Cutset #9 (SCDF = 7.25E-07/yr): Cutset #9 is similar to Cutset #7 except IC makeup from SW is failed due to operator failure to restore SW following the DLOOP caused by seismic failure of offsite AC power.

Cutset #10 (SCDF = 6.98E-07/yr): This %G6 cutset represents a seismic induced ATWS scenario due to failure of the core shroud tie-rods. RPV overpressure protection and early SLC injection are successful. However, HPCI is unavailable due to RPV overfill. The operator successfully depressurizes the RPV in a controlled manner, but the operator subsequently fails to adequately control RPV water level with low pressure systems following RPV depressurization. This cutset leads to core damage (Accident Class 1C).

Although the cutsets may appear conservative because of the many HEPs set to 1.0, the cutsets are consistent with the DRE SPRA modeling assumptions/approaches in this regard (i.e., operator error probabilities increase to 1.0 or close to 1.0 as the hazard magnitude increases; refer to the SPRA HRA Notebook, [49]).

A review of sample cutsets from each decade of quantification results did not identify any improper cutsets.

<u>SCDF Accident Class Contributors</u>

The dominant contributors to SCDF by Level 1 accident class include the following:

- Class 1BL (Late Station Blackout, core damage at greater than 4 hours) – 23%
- Class 1A (Failure of RPV makeup with the RPV at high pressure) - 22%
- Class 2V (Loss of containment heat removal except the vent operates as designed, suppression pool is saturated but intact) – 17%
- Class 1D (Failure of RPV makeup with the RPV at low pressure) - 13%
- Class 1BE (Early Station Blackout, core damage at less than 4 hours) – 13%

The Level 1 accident class definitions for the DRE SPRA are based on those defined for the DRE FPIE PRA model. The Level 1 accident classes are described in Table 3-1 of the DRE SPRA Methods Notebook [46].

The accident class contributions are calculated based on the FV values of accident class basic events that are included in each cutset. Each accident class basic event has a probability of 1.0. The FV values are estimates based on the ACUBE results. The FV values have significant uncertainty because the ACUBE software was not able to fully process 100% of the Level 1 CDF cutsets because of software and hardware limitations. Nevertheless, the accident class contributions appear to be generally reasonable.

As shown in Table 5.4-6, the Accident Class for the top CDF cutset is shown to be Class V (seismic induced failure of the RPV supports leading to core damage due to assumed inability to maintain core cooling and subsequent bypass of containment). This is due to assuming that the %G8 initiator leads directly to core damage (i.e., by setting all seismic induced fragilities to TRUE) and masks the ability to readily identify the contribution to individual core damage accident classes and sequences. In order to evaluate the accident class contributions, a sensitivity case was performed to explicitly quantify the %G8 cutsets. The SCDF contribution results discussed below are due to explicitly calculating the %G8 contribution to individual accident classes (e.g., Class 1A, Class 1BE, Class 1BL) instead of assuming that the %G8 initiator leads directly to core damage with an assumed Class V accident class. The Class V accident class is not shown to have an actual high contribution to Level 1 SCDF because the fragility for the RPV supports is relatively high (i.e., Am=4.6g).

Class 1BE and 1BL (Early or Late Station Blackout) accidents have the highest combined contribution to the DRE Level 1 SCDF. Seismic induced LOOP events are generally amongst the highest contributors for typical SPRA models because recovery of offsite AC power is generally not credited. In addition, some of the highest contributors to SCDF include seismic induced failure of SSCs supporting AC and DC power (e.g., 125 VDC system, AC distribution, relay chatter impacting EDGs).

Class 1A (Failure of RPV makeup with the RPV at high pressure) is due to failure of RPV depressurization following loss of all high pressure makeup. The dominant contributor to failure to depressurize the RPV is due to operator action to manually depressurize the RPV, either as an independent operator action or as part of a dependent operator action group.

Class 2V (Loss of Containment Heat Removal with successful Containment Venting) involves loss of all Containment Heat Removal (e.g., SPC), but Primary Containment Venting is available. Following successful Containment Venting, continued RPV makeup is not available leading to core damage. One of the primary reasons for loss of continued RPV makeup from CS or LPCI with suction aligned to the suppression pool is due to loss of adequate NPSH via operator failure to control the Containment Venting evolution within a high pressure band. RPV makeup is credited post Containment Venting, but unavailability of CRD or SBCS for external injection would result in core damage.

Class 1D (Failure of RPV makeup with the RPV at low pressure) is similar to Class 1B, but emergency AC power from at least one (1) EDG or SBODG remains available. All high pressure makeup is unavailable and following successful RPV depressurization, all low pressure makeup is also unavailable. Seismic induced failure of AC or DC power to support CS and LPCI contribute to Class 1D core damage scenarios.

Table 5.4-1 DRE Unit 2 CDF Contributors by Seismic Hazard Interval Initiating Event [52]

Seismic Hazard Interval	Description	Interval Frequency (/yr)	Interval CDF (/yr)	% of Total SCDF	Cumulative SCDF (/yr)
%G1	%G1 - Hazard Curve: DRE SPRA - PGA Range: 0.1g to 0.2g	9.38E-05	2.13E-09	0%	2.13E-09
%G2	%G2 - Hazard Curve: DRE SPRA - PGA Range: 0.2g to 0.3g	2.27E-05	4.33E-08	1%	4.55E-08
%G3	%G3 - Hazard Curve: DRE SPRA - PGA Range: 0.3g to 0.4g	8.23E-06	3.05E-07	5%	3.50E-07
%G4	%G4 - Hazard Curve: DRE SPRA - PGA Range: 0.4g to 0.5g	3.71E-06	8.77E-07	15%	1.23E-06
%G5	%G5 - Hazard Curve: DRE SPRA - PGA Range: 0.5g to 0.6g	2.00E-06	1.23E-06	21%	2.46E-06
%G6	%G6 - Hazard Curve: DRE SPRA - PGA Range: 0.6g to 0.8g	1.85E-06	1.73E-06	30%	4.19E-06
%G7	%G7 - Hazard Curve: DRE SPRA - PGA Range: 0.8g to 1g	7.68E-07	7.39E-07	13%	4.93E-06
%G8	%G8 - Hazard Curve: DRE SPRA - PGA Range: > 1g	9.41E-07	9.08E-07	15%	5.84E-06

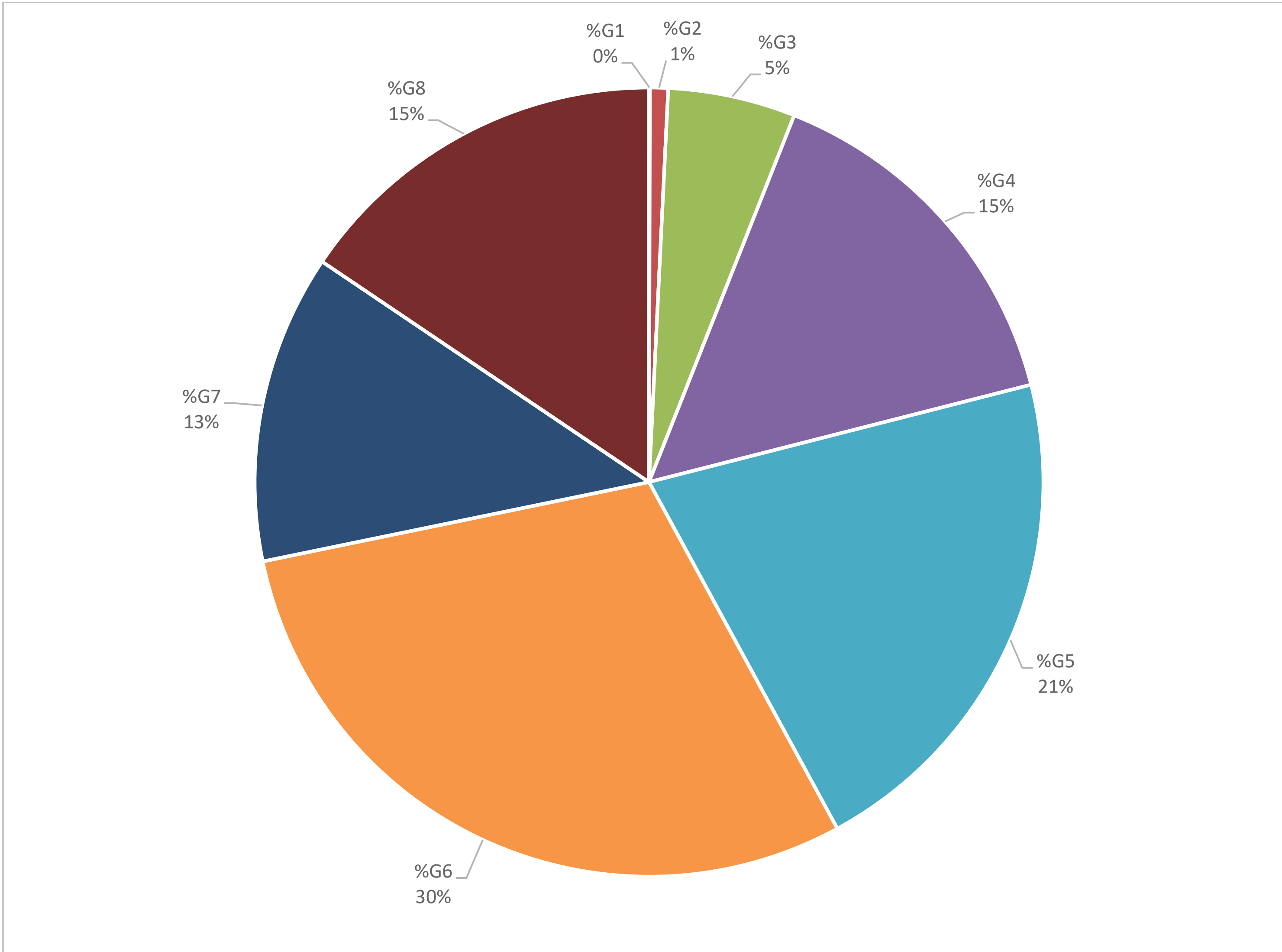

Figure 5.4-1 DRE SPRA Unit 2 SCDF by Initiating Event [52]

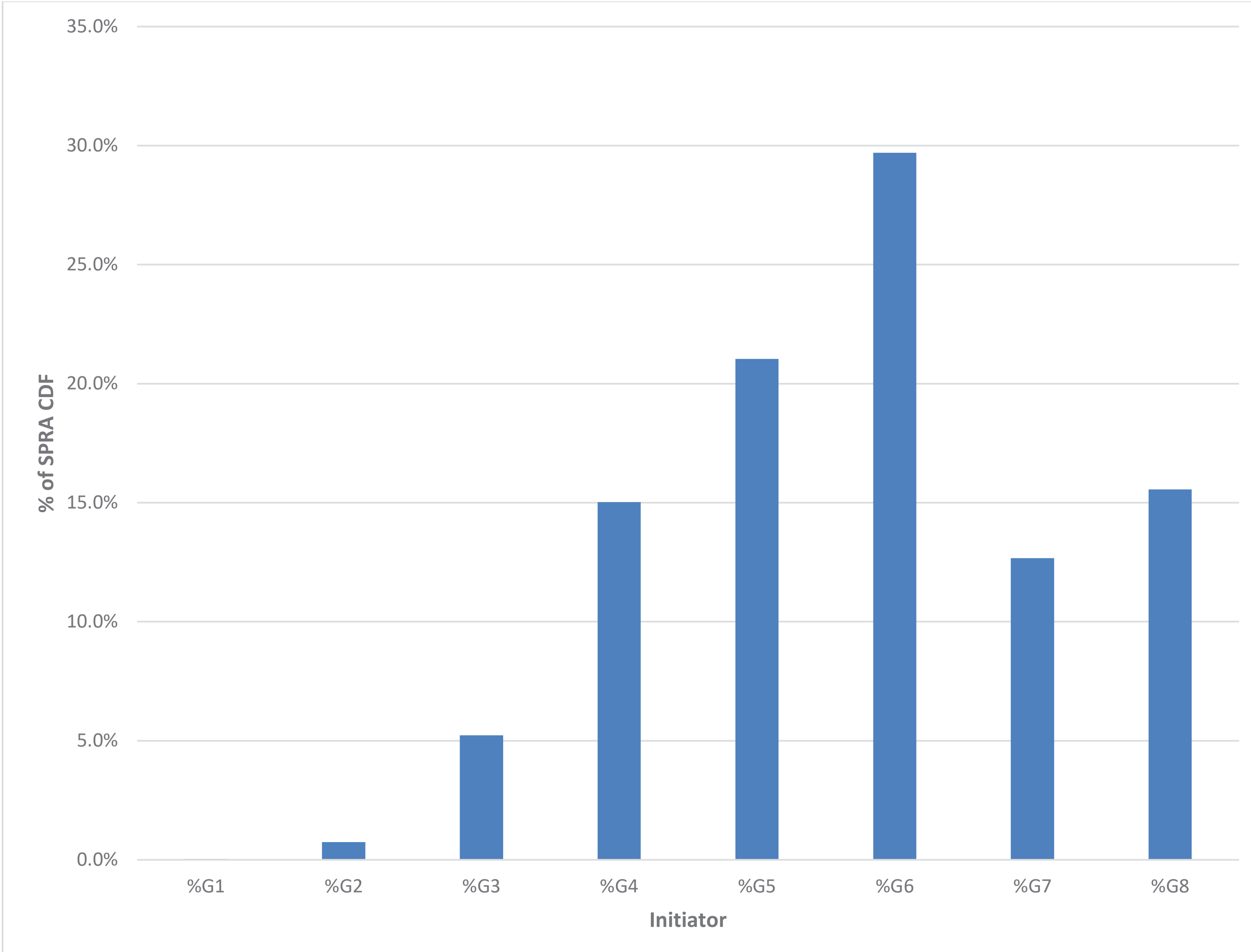

Figure 5.4-2 DRE SPRA Unit 2 SCDF by Initiating Event [52]

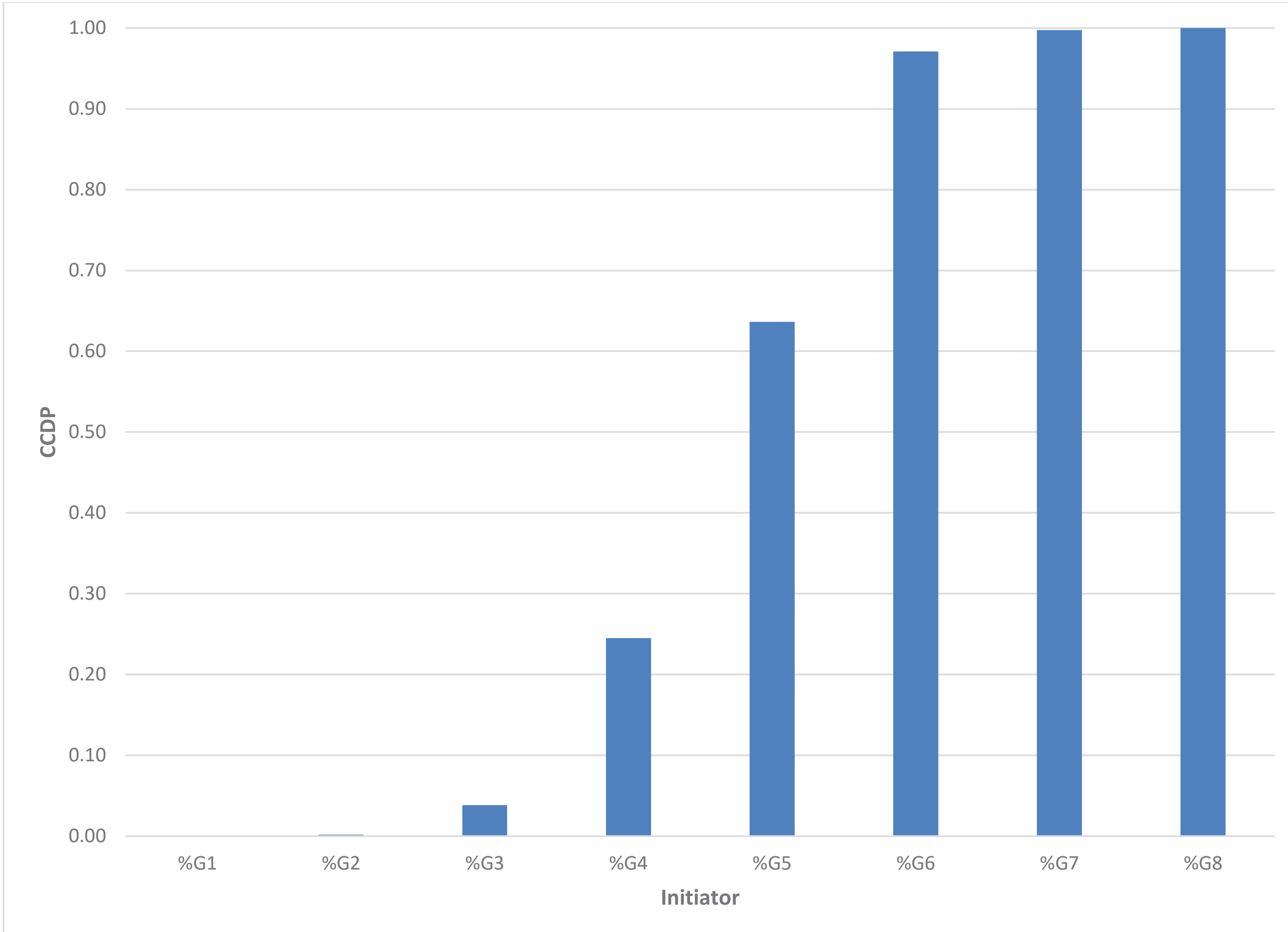

Figure 5.4-3 DRE SPRA Unit 2 CCDP by Initiating Event [52]

Table 5.4-2 DRE Unit 2 SCDF Fussell-Vesely Importance Measures for SSC Fragilities [52]

Fragility Group ID	Fragility Group Description	FV Total	Am (g)	βr	βu	Failure Mode	Fragility Method
OSP	Offsite Power	5.86E-01	0.3	0.3	0.45	Functional	Generic
S-INCP04-	Control Panel Group C20-4 (Panels 902-15, 902-17, 902-18, 902-19, 902-16, 902-20)	7.76E-02	0.78	0.24	0.5	Anchorage	Refined (SoV)
S-DCBY1	Unit 2 125 VDC Battery (549 TB)	7.28E-02	0.72	0.24	0.48	Anchorage	Refined (SoV)
SCRAM	RPV Internals (Scram)	6.80E-02	0.75	0.24	0.29	Anchorage	Refined (SoV)
S-INIR18-9-	Instrument Rack Group 18-9-1-1 (2202-7)	4.68E-02	0.39	0.24	0.32	Block Wall	Refined (CDFM)
S-DCBU5	Unit 2 125 VDC TRAIN B BUSSES (2-83125) - 549 TB	1.59E-02	0.79	0.24	0.32	Functional	Refined (CDFM)
S-DCBY3	SBODG2 Battery 6A and SBODG3 Battery 7A	1.57E-02	0.29	0.22	0.43	Functional	Refined (SoV)
S-DCBY2	Unit 3 125 VDC Battery (551 TB)	1.52E-02	0.72	0.22	0.29	Anchorage	Refined (CDFM)
S-INCP03-	Control Panel Group C20-3 (Panels 903-8, 902-8)	1.43E-02	1.09	0.24	0.32	Anchorage	Refined (CDFM)
S-ACBS01	4160V Buses 23, 24	1.19E-02	0.76	0.24	0.32	Functional	Refined (CDFM)
CRIB	Crib House	1.15E-02	0.99	0.24	0.26	Structure	Refined (CDFM)
S-DCBC3	Unit 3 125 VDC Battery Charger #3 - 538 TB	9.67E-03	0.58	0.24	0.32	Anchorage	Refined (CDFM)
S-CH101	Relay Chatter ID 101 (HPCI-Recov.)	8.11E-03	0.45	0.24	0.32	Functional	Refined (CDFM)
DAM	Dresden Lock and Dam	7.82E-03	0.82	0.24	0.32	Structure	Refined (CDFM)
S-CH613	Relay Chatter ID 613 (CS B-Unrecov.)	7.32E-03	0.65	0.24	0.32	Functional	Refined (CDFM)
S-ACBS10	480V MCC 35-2, 38-2, 38-3	6.90E-03	0.74	0.24	0.32	Functional	Refined (CDFM)
S-DCBU2	Unit 2 125 VDC TRAIN A BUSSES (2-83125) - 549 TB	6.84E-03	1.2	0.29	0.4	Functional	Refined (SoV)
S-INCP01-	Control Panel Group C20-1 (Panels 902-3, 903-3, 902-4, 903-4)	6.84E-03	1.27	0.24	0.32	Anchorage	Refined (CDFM)
S-INCP07-	Control Panel Group C20-7 (Panels 902-33, 903-33, 902-32)	6.78E-03	1.07	0.21	0.28	Functional	Refined (CDFM)

Table 5.4-2 DRE Unit 2 SCDF Fussell-Vesely Importance Measures for SSC Fragilities [52]

Fragility Group ID	Fragility Group Description	FV Total	Am (g)	βr	βu	Failure Mode	Fragility Method
S-CH361A	Relay Chatter ID 361A (EDG2, EDG3-Recov.)	5.95E-03	0.91	0.24	0.32	Functional	Refined (CDFM)
S-ACBS19	4160V AC/ Switchgear 40	5.02E-03	0.99	0.24	0.32	Functional	Refined (CDFM)

Table 5.4-3 DRE Unit 3 SCDF Fussell-Vesely Importance Measures for SSC Fragilities [52]

Fragility Group ID	Fragility Group Description	FV Total	Am (g)	βr	βu	Failure Mode	Fragility Method
OSP	Offsite Power	6.68E-01	0.3	0.3	0.45	Functional	Generic
SCRAM	RPV Internals (Scram)	7.03E-02	0.75	0.24	0.29	Anchorage	Refined (SoV)
S-DCBY2	Unit 3 125 VDC Battery (551 TB)	5.29E-02	0.72	0.22	0.29	Anchorage	Refined (CDFM)
S-INCP04-	Control Panel Group C20-4 (Panels 902-15, 902-17, 902-18, 902-19, 902-16, 902-20)	4.27E-02	0.78	0.24	0.5	Anchorage	Refined (SoV)
S-DCBY1	Unit 2 125 VDC Battery (549 TB)	3.75E-02	0.72	0.24	0.48	Anchorage	Refined (SoV)
S-INCP04-1-	Control Panel Group C20-4-1 (Panels 903-15, 903-17, 903-18, 903-19, 903-16, 903-20)	3.58E-02	0.78	0.24	0.5	Anchorage	Refined (SoV)
S-ACBS15	4160V Buses 33, 34	2.03E-02	0.69	0.24	0.32	Functional	Refined (CDFM)
S-DCBC3	Unit 3 125 VDC Battery Charger #3 - 538 TB	1.84E-02	0.58	0.24	0.32	Anchorage	Refined (CDFM)
S-DCBY3	SBODG2 Battery 6A and SBODG3 Battery 7A	1.51E-02	0.29	0.22	0.43	Functional	Refined (SoV)
S-INCP03-	Control Panel Group C20-3 (Panels 903-8, 902-8)	1.36E-02	1.09	0.24	0.32	Anchorage	Refined (CDFM)
CRIB	Crib House	1.20E-02	0.99	0.24	0.26	Structure	Refined (CDFM)
S-CH521	Relay Chatter ID 521 (CS B-Recov.)	1.04E-02	0.75	0.24	0.32	Functional	Refined (CDFM)
S-CH451	Relay Chatter ID 451 (Bus 29 feed to 480 VAC MCC 29-7) (1)	9.91E-03	0.47	0.24	0.32	Functional	Refined (CDFM)
S-CH462	Relay Chatter ID 462 (Bus 29 feed to 480 VAC MCC 29-7)(2)	9.91E-03	0.47	0.24	0.32	Functional	Refined (CDFM)
S-INCP08-	Control Panel Group C20-8 (Panels 903-32)	8.15E-03	1.04	0.21	0.28	Anchorage	Refined (CDFM)
S-ACBS19	4160V AC/ Switchgear 40	7.62E-03	0.99	0.24	0.32	Functional	Refined (CDFM)
DAM	Dresden Lock and Dam	7.37E-03	0.82	0.24	0.32	Structure	Refined (CDFM)
S-INCP01-	Control Panel Group C20-1 (Panels 902-3, 903-3, 902-4, 903-4)	7.29E-03	1.27	0.24	0.32	Anchorage	Refined (CDFM)
S-INIR18-9-	Instrument Rack Group C18-9-1 (2202-7)	6.83E-03	0.39	0.24	0.32	Block Wall	Refined (CDFM)

Table 5.4-3 DRE Unit 3 SCDF Fussell-Vesely Importance Measures for SSC Fragilities [52]

Fragility Group ID	Fragility Group Description	FV Total	Am (g)	βr	βu	Failure Mode	Fragility Method
S-INCP07-	Control Panel Group C20-7 (Panels 902-33, 903-33, 902-32)	6.48E-03	1.07	0.21	0.28	Functional	Refined (CDFM)
S-CH520A	Relay Chatter ID 520A (CS A and B-Recov.)	5.53E-03	1.18	0.24	0.32	Functional	Refined (CDFM)
S-DGPA3	EDG #3 Excitation Cabinet (3-2253-21)	5.50E-03	0.69	0.21	0.28	Anchorage	Refined (CDFM)
S-DCBU5	Unit 2 125 VDC TRAIN B BUSSES (2-83125) - 549 TB	5.39E-03	0.79	0.24	0.32	Functional	Refined (CDFM)
S-CH386	Relay Chatter ID 386 (EDG2-Recov.) (3)	5.12E-03	0.75	0.24	0.32	Functional	Refined (CDFM)

Notes to Table 5.4-3:

(1) SPRA fragility group S-CH451 used as a surrogate for relay IDs 465 and 475 (Bus 39 feed to 480 VAC MCC 39-7).

(2) SPRA fragility group S-CH462 used as a surrogate for relay ID 476 (Bus 39 feed to 480 VAC MCC 39-7).

(3) SPRA fragility group S-CH386 used as a surrogate for relay IDs 361 and 381 (EDG3-Recov.)

Table 5.4-4 DRE Unit 2 SCDF Fussell-Vesely Importance Measures for Operator Actions [52]

OPERATOR ACTION ID	OPERATOR ACTION DESCRIPTION	FV TOTAL
2CVOP-CNTROL-H--	FAILURE TO CONTROL CONTAINMENT VENTING	5.18E-02
2LIOPLIA-INJ-H--	FAILURE TO INJECT THROUGH A LOOP GIVEN B LOOP FAILURE	1.85E-02
BDGOPCHATREC1H--	Operator Fails to Recover From Relay Chatter Impacting EDG 2, 3, and/or 2/3 (SEISMIC)	1.67E-02
2RWOPBLOWDWN-H--	CREW FAILS TO ALIGN RWCU FOR LETDOWN	1.56E-02
2HIOP-HP-ISOLH--	FAILURE TO CLOSE HPCI STEAM LINE ISOLATION VALVE TO PREVENT WATER INTO HPCI TURBINE OR AUXILIARIES	1.56E-02
2ADOPINHIBIT-H--	FAILURE TO INHIBIT Automatic Depressurization System (ADS) (NO HP INJECTION) (ATWS)	9.02E-03
2CVOP-EM-VNT-H--	FAILURE TO EMERCENCY VENT CONTAINMENT USING HARD PIPE VENT	8.58E-03
2HIOPCHATREC1H--	Operator Fails to Recover From Relay Chatter Impacting HPCI (SEISMIC)	8.09E-03
2LIOP-LPLVL--H--	FAILURE TO CONTROL RPV LEVEL LOW (ATWS)	6.82E-03

<u>Notes to Table 5.4-4:</u>

(1) This table covers independent and dependent post-initiator HEPs and their risk contribution; however, if dependent HEPs do not show up in this table that is because their FV value is below 5E-03.

(2) The independent post-initiator HEP FV values presented in this table do not include the risk contribution from the independent HEPs appearing in dependent HEPs.

Table 5.4-5 DRE Unit 3 SCDF Fussell-Vesely Importance Measures for Operator Actions [52]

OPERATOR ACTION ID	OPERATOR ACTION DESCRIPTION	FV TOTAL
2CVOP-CNTROL-H--	FAILURE TO CONTROL CONTAINMENT VENTING	5.90E-02
2HIOP-HP-ISOLH--	FAILURE TO CLOSE HPCI STEAM LINE ISOLATION VALVE TO PREVENT WATER INTO HPCI TURBINE OR AUXILIARIES	3.39E-02
2RWOPBLOWDWN-H--	CREW FAILS TO ALIGN RWCU FOR LETDOWN	3.39E-02
BDCOPPORTCHRGH--	FAILURE TO ALIGN PORTABLE BATTERY CHARGERS	1.48E-02
BDGOPCHATREC1H--	Operator Fails to Recover From Relay Chatter Impacting EDG 2, 3, and/or 2/3 (SEISMIC)	1.43E-02
2CVOP-EM-VNT-H--	FAILURE TO EMERCENCY VENT CONTAINMENT USING HARD PIPE VENT	1.08E-02
2ADOPINHIBIT-H--	FAILURE TO INHIBIT ADS (NO HP INJECTION) (ATWS)	8.14E-03
BDCOPALT-BATDH--	FAILURE TO ALIGN ALTERNATE BATTERY GIVEN DUAL UNIT LOOP	6.23E-03
2LIOP-LPLVL--H--	FAILURE TO CONTROL RPV LEVEL LOW (ATWS)	6.15E-03

Notes to Table 5.4-5:

(1) This table covers independent and dependent post-initiator HEPs and their risk contribution; however, if dependent HEPs do not show up in this table that is because their FV value is below 5E-03.

(2) The independent post-initiator HEP FV values presented in this table do not include the risk contribution from the independent HEPs appearing in dependent HEPs.

Table 5.4-6 DRE Unit 2 Top 10 Seismic CDF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
1	9.08E-07	9.08E-07	%G8	Seismic Initiating Event (>1g)
		1.00E+00	RCVCL-5	ACCIDENT CLASS V
		1.00E+00	RCVSEQ-SIET-020	ACCIDENT SEQUENCE SIET-020
2	7.25E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		1.00E+00	2FPPH-FPS-HD-F--	FPS INADEQUATE FLOW TO PREVENT CORE DAMAGE
		1.00E+00	2LIPH-BINDNG-F--	FAILURE TO CONTROL VENT CAUSES STEAM BINDING IN ECCS SUCTION
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-2V	ACCIDENT CLASS IIV
		1.00E+00	RCVSEQ-DLP-026	ACCIDENT SEQUENCE DLP-026
		1.00E+00	SRX07_2CCOP-CNTC2--H--	S-HEP G7: FAILURE TO INITIATE CONTAINMENT COOLING (EARLY)
		1.00E+00	SRX07_2CCOP-CNTCCN-H--	S-HEP G7: FAILURE TO INITIATE CONTAINMENT COOLING LATE GIVEN EARLY FAILURE
		1.00E+00	SRX07_2CVOP-CONDCTRH--	S-HEP G7: OPERATOR FAILS TO CONTROL CONTAINMENT VENT (CONDITIONAL)
3	7.25E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		1.00E+00	2HIOP-CONTROL	CREW FAILS TO CONTROL LEVEL LOW IN THE LEVEL BAND
		1.00E+00	2HIPH-EXPAND	COLD WATER EXPANDS ABOVE HPCI/IC STEAM LINES
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1A	ACCIDENT CLASS IA
		1.00E+00	RCVSEQ-DLP-005	ACCIDENT SEQUENCE DLP-005
		1.00E+00	SDP07_COMBINATION_5_CDF	S-DHEP G7: Dependent HEP for 2ICOP-IC-MU1-H--,2ADOP-ACT-ADSH--,2RWOPBLOWDWN-H--,
		1.00E+00	SRX07_2ADOP-ACT-ADSH--	S-HEP G7: FAILURE TO DEPRESSURIZE THE RPV (ADS) (NON-ATWS)
		1.00E+00	SRX07_2HIOP-HP-ISOLH--	S-HEP G7: FAILURE TO CLOSE HPCI STEAM LINE ISOLATION VALVE TO PREVENT WATER INTO
		1.00E+00	SRX07_2ICOP-IC-MU1-H--	S-HEP G7: FAILURE TO INITIATE IC SHELL SIDE MAKEUP WITHIN 20 MIN.
		1.00E+00	SRX07_2RWOPBLOWDWN-H--	S-HEP G7: CREW FAILS TO ALIGN RWCU FOR LETDOWN
4	7.25E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1BL	ACCIDENT CLASS IBL
		1.00E+00	RCVSEQ-DLP-049	ACCIDENT SEQUENCE DLP-049

Table 5.4-6 DRE Unit 2 Top 10 Seismic CDF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	SRX07_2ACOP-LINK-BSH--	S-HEP G7: FAILURE TO LINK BUSES (23 & 23-1) (24 & 24-1) (33 & 33-1) (34 & 34-1)
		1.00E+00	SRX07_2ADOP-ACT-ADSH--	S-HEP G7: FAILURE TO DEPRESSURIZE THE RPV (ADS) (NON-ATWS)
5	7.25E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1BL	ACCIDENT CLASS IBL
		1.00E+00	RCVSEQ-DLP-049	ACCIDENT SEQUENCE DLP-049
		1.00E+00	SRX07_2ADOP-ACT-ADSH--	S-HEP G7: FAILURE TO DEPRESSURIZE THE RPV (ADS) (NON-ATWS)
		1.00E+00	SRX07_2CCOP-CNTC2--H--	S-HEP G7: FAILURE TO INITIATE CONTAINMENT COOLING (EARLY)
6	7.25E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		1.00E+00	2HIOP-CONTROL	CREW FAILS TO CONTROL LEVEL LOW IN THE LEVEL BAND
		1.00E+00	2HIPH-EXPAND	COLD WATER EXPANDS ABOVE HPCI/IC STEAM LINES
		1.00E+00	2ICPH-DGCWMU-F--	DGCW M/U CAPACITY TO IC INSUFFICIENT FOR GREATER THAN 8 HOURS
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1A	ACCIDENT CLASS IA
		1.00E+00	RCVSEQ-DLP-005	ACCIDENT SEQUENCE DLP-005
		1.00E+00	SDP07_COMBINATION_987_CDF	S-DHEP G7: Dependent HEP for 2FPOP-3906-SWH--,2ICOP-LOCAL--H--,2ADOP-ACT-ADSH--,
		9.99E-01	S-FPBU1-C-%G7	SEISMIC FRAGILITY FOR %G7: U2/3 Diesel Fire Pump
		1.00E+00	SRX07_2ADOP-ACT-ADSH--	S-HEP G7: FAILURE TO DEPRESSURIZE THE RPV (ADS) (NON-ATWS)
		1.00E+00	SRX07_2FPOP-3906-SWH--	S-HEP G7: FAILURE TO OPEN SW TO FPS CROSS TIE (MOV 2-3906) FROM CONTROL ROOM
		1.00E+00	SRX07_2HIOP-HP-ISOLH--	S-HEP G7: FAILURE TO CLOSE HPCI STEAM LINE ISOLATION VALVE TO PREVENT WATER INTO
		1.00E+00	SRX07_2ICOP-LOCAL--H--	S-HEP G7: FAILURE OF LOCAL MANIPULATION OF VALVES TO SUPPORT IC MAKEUP
		1.00E+00	SRX07_2RWOPBLOWDWN-H--	S-HEP G7: CREW FAILS TO ALIGN RWCU FOR LETDOWN
7	7.25E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		1.00E+00	2HIOP-CONTROL	CREW FAILS TO CONTROL LEVEL LOW IN THE LEVEL BAND
		1.00E+00	2HIPH-EXPAND	COLD WATER EXPANDS ABOVE HPCI/IC STEAM LINES
		1.00E+00	2ICPH-DGCWMU-F--	DGCW M/U CAPACITY TO IC INSUFFICIENT FOR GREATER THAN 8 HOURS
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power

Table 5.4-6 DRE Unit 2 Top 10 Seismic CDF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	RCVCL-1A	ACCIDENT CLASS IA
		1.00E+00	RCVSEQ-DLP-005	ACCIDENT SEQUENCE DLP-005
		1.00E+00	SDP07_COMBINATION_987_CDF	S-DHEP G7: Dependent HEP for 2FPOP-3906-SWH--,2ICOP-LOCAL--H--,2ADOP-ACT-ADSH--,
		9.99E-01	S-FPTK1-C-%G7	SEISMIC FRAGILITY FOR %G7: U2/3 Fire Pump Day Tank
		1.00E+00	SRX07_2ADOP-ACT-ADSH--	S-HEP G7: FAILURE TO DEPRESSURIZE THE RPV (ADS) (NON-ATWS)
		1.00E+00	SRX07_2FPOP-3906-SWH--	S-HEP G7: FAILURE TO OPEN SW TO FPS CROSS TIE (MOV 2-3906) FROM CONTROL ROOM
		1.00E+00	SRX07_2HIOP-HP-ISOLH--	S-HEP G7: FAILURE TO CLOSE HPCI STEAM LINE ISOLATION VALVE TO PREVENT WATER INTO
		1.00E+00	SRX07_2ICOP-LOCAL--H--	S-HEP G7: FAILURE OF LOCAL MANIPULATION OF VALVES TO SUPPORT IC MAKEUP
		1.00E+00	SRX07_2RWOPBLOWDWN-H--	S-HEP G7: CREW FAILS TO ALIGN RWCU FOR LETDOWN
8	7.25E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		1.00E+00	2HIOP-CONTROL	CREW FAILS TO CONTROL LEVEL LOW IN THE LEVEL BAND
		1.00E+00	2HIPH-EXPAND	COLD WATER EXPANDS ABOVE HPCI/IC STEAM LINES
		1.00E+00	2ICPH-DGCWMU-F--	DGCW M/U CAPACITY TO IC INSUFFICIENT FOR GREATER THAN 8 HOURS
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1A	ACCIDENT CLASS IA
		1.00E+00	RCVSEQ-DLP-005	ACCIDENT SEQUENCE DLP-005
		1.00E+00	SDP07_COMBINATION_991_CDF	S-DHEP G7: Dependent HEP for 2ICOP-LOCAL--H--,2ADOP-ACT-ADSH--,2RWOPBLOWDWN-H--,
		9.99E-01	S-FPBU1-C-%G7	SEISMIC FRAGILITY FOR %G7: U2/3 Diesel Fire Pump
		1.00E+00	SRX07_2ADOP-ACT-ADSH--	S-HEP G7: FAILURE TO DEPRESSURIZE THE RPV (ADS) (NON-ATWS)
		1.00E+00	SRX07_2HIOP-HP-ISOLH--	S-HEP G7: FAILURE TO CLOSE HPCI STEAM LINE ISOLATION VALVE TO PREVENT WATER INTO
		1.00E+00	SRX07_2ICOP-LOCAL--H--	S-HEP G7: FAILURE OF LOCAL MANIPULATION OF VALVES TO SUPPORT IC MAKEUP
		1.00E+00	SRX07_2RWOPBLOWDWN-H--	S-HEP G7: CREW FAILS TO ALIGN RWCU FOR LETDOWN
		1.00E+00	SRX07_2SWOP-DLOOP--H--	S-HEP G7: FAILURE TO RESTORE SW GIVEN DLOOP
9	7.25E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		1.00E+00	2HIOP-CONTROL	CREW FAILS TO CONTROL LEVEL LOW IN THE LEVEL BAND
		1.00E+00	2HIPH-EXPAND	COLD WATER EXPANDS ABOVE HPCI/IC STEAM LINES
		1.00E+00	2ICPH-DGCWMU-F--	DGCW M/U CAPACITY TO IC INSUFFICIENT FOR GREATER THAN 8 HOURS

Table 5.4-6 DRE Unit 2 Top 10 Seismic CDF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1A	ACCIDENT CLASS IA
		1.00E+00	RCVSEQ-DLP-005	ACCIDENT SEQUENCE DLP-005
		1.00E+00	SDP07_COMBINATION_991_CDF	S-DHEP G7: Dependent HEP for 2ICOP-LOCAL--H--,2ADOP-ACT-ADSH--,2RWOPBLOWDWN-H--,
		9.99E-01	S-FPTK1-C-%G7	SEISMIC FRAGILITY FOR %G7: U2/3 Fire Pump Day Tank
		1.00E+00	SRX07_2ADOP-ACT-ADSH--	S-HEP G7: FAILURE TO DEPRESSURIZE THE RPV (ADS) (NON-ATWS)
		1.00E+00	SRX07_2HIOP-HP-ISOLH--	S-HEP G7: FAILURE TO CLOSE HPCI STEAM LINE ISOLATION VALVE TO PREVENT WATER INTO
		1.00E+00	SRX07_2ICOP-LOCAL--H--	S-HEP G7: FAILURE OF LOCAL MANIPULATION OF VALVES TO SUPPORT IC MAKEUP
		1.00E+00	SRX07_2RWOPBLOWDWN-H--	S-HEP G7: CREW FAILS TO ALIGN RWCU FOR LETDOWN
		1.00E+00	SRX07_2SWOP-DLOOP--H--	S-HEP G7: FAILURE TO RESTORE SW GIVEN DLOOP
10	6.98E-07	1.79E-06	%G6	Seismic Initiating Event (0.6g to <0.8g)
		1.00E+00	2HIOP-CONTROL	CREW FAILS TO CONTROL LEVEL LOW IN THE LEVEL BAND
		1.00E+00	2HIPH-EXPAND	COLD WATER EXPANDS ABOVE HPCI/IC STEAM LINES
		9.39E-01	OSP-C-%G6	SEISMIC FRAGILITY FOR %G6: Offsite Power
		1.00E+00	RCVCL-1C	ACCIDENT CLASS IC
		1.00E+00	RCVSEQ-ATW6-31	ACCIDENT SEQUENCE ATW6-31
		4.17E-01	SCRAM-C-%G6	SEISMIC FRAGILITY FOR %G6: RPV Internals (Scram)
		1.00E+00	SRX06_2HIOP-HP-ISOLH--	S-HEP G6: FAILURE TO CLOSE HPCI STEAM LINE ISOLATION VALVE TO PREVENT WATER INTO
		1.00E+00	SRX06_2LIOP-LPLVL--H--	S-HEP G6: FAILURE TO CONTROL RPV LEVEL LOW (ATWS)
		1.00E+00	SRX06_2RWOPBLOWDWN-H--	S-HEP G6: CREW FAILS TO ALIGN RWCU FOR LETDOWN

5.5 SLERF Results

The seismic PRA performed for DRE shows that the point estimate mean seismic LERF is 2.9E-06/yr for Unit 2 and 2.8E-06/yr for Unit 3 [52]. The Unit 2 Seismic LERF of 2.9E-06/yr is calculated with a single top CAFTA model at a truncation that ranges from 1E-07/yr to 1E-13/yr. The seismic LERF of 2.9E-06/yr represents approximately 49% of the seismic CDF of 5.8E-06/yr.

The Unit 3 SLERF is 2.8E-06/yr, which is within 1% of the Unit 2 SLERF. Given the similarities in the Unit 2 and Unit 3 SCDF and SLERF values, the remainder of this section focuses on the Unit 2 results, except as noted. In general, DRE Unit 2 and Unit 3 are symmetrical. In addition, the fragility analysis supports that the dominant risk contributors to seismic LERF are similar for both units (e.g., seismic induced failure to SCRAM, seismic induced failure of 125 VDC batteries).

Important Seismic Initiating Event Contributors

Table 5.5-1 summarizes the LERF contributors by seismic initiating event. Figure 5.5-1 displays the results of Table 5.5-1 in graphical pie chart form, i.e., the LERF contributors by initiating event. Figure 5.5-2 shows the initiating event contribution in the form of a bar graph.

As can be seen from the graphical display, the seismic initiators %G6, %G7, and %G8 are the dominant seismic risk contributors. These initiators span the range from 0.60g to >1.0g. Their combined contribution is approximately 79% of the seismic LERF. Seismic hazard interval initiator %G7 contributes less to SLERF than %G8 because the initiator frequency of %G7 is lower than that of %G8 (i.e., %G7 is a bounded hazard interval and %G8 is the unbounded final hazard interval).

Conditional Large Early Release Probability (CLERP) values were also calculated for the initiators. These CLERP values are displayed in Figure 5.5-3. Figure 5.5-3 shows the CLERP for the %G7 initiator (0.8g to 1.0g) is approximately 0.75. The CLERP for the %G8 initiator (>1.0g) is 1.0 because %G8 seismic events are assumed to lead directly to CDF and LERF.

The Unit 3 SLERF contributors by seismic initiating event are similar to those shown for Unit 2.

Important Contributors to Large Early Release Frequency

Table 5.5-2 provides the Unit 2 SLERF Fussell-Vesely (FV) importance measures for SSC fragilities. The SLERF FV risk importance values are calculated in the same manner as that discussed in Section 5.4 for SCDF FV values, except that the SLERF cutset results are used.

Consistent with past SPRA models, many of the top SLERF FV contributors are associated with AC and DC power supply. Failure to scram (ATWS) scenarios are also a significant contributor to SLERF because of the relatively low Am value for seismic induced failure to scram and the modeling of ATWS scenarios in the Level

2 SPRA. The Level 2 SPRA is based on the Level 2 FPIE PRA model, which incorporates potentially conservative assumptions for ATWS mitigation.

The top 5 contributors to Unit 2 SLERF FV are as follows:

Normal offsite power (FV = 9.99E-01)

Normal offsite power is expected to have a high FV because there is a high probability for the seismic event to fail offsite power (Am = 0.3g).

SCRAM (RPV Internals) (FV = 6.50E-01)

The SCRAM (RPV Internals) has a high-risk impact because unmitigated failure to SCRAM events result in significant hydrodynamic loads on the containment. Failure to SCRAM events are modeled in the base DRE FPIE PRA model to have a high likelihood of leading to early containment failure and a Large Early Release.

Unit 2 125 VDC Battery Racks (FV = 1.20E-01)

The Unit 2 125 VDC batteries have a high-risk impact because failure results in loss of the Unit 2 station EDG (i.e., EDG2) and the Unit 2 IC. The risk impact for Loss of the Unit 2 125 VDC is exacerbated when failed in combination with other SSCs (e.g., Unit 3 125 VDC SSCs) that provide redundant defense-in-depth capabilities for mitigation systems such as HPCI and other EDGs (e.g., EDG3 or EDG2/3). Loss of the Unit 2 125 VDC batteries contributes to scenarios with loss of all RPV makeup in an early time frame, which leads to early RPV failure, Mark I shell liner failure, and a Large Early Release.

Instrument Rack 2202-7 (FV = 3.70E-02)

The governing seismic failure mode of instrument rack 2202-7 is block wall failure resulting in a relatively low median capacity of 0.39g. This low median capacity combined with seismic failure of this instrument rack modeled to fail LPCI valves and pumps results in high risk significance.

Various Instrument Control Panels (FV = 2.47E-02)

Correlated control panel group C20-4 (consisting of Unit 2 panels 902-15, -16, -17, -18, -19, and -20) has a high risk impact due to the relatively low median capacity of 0.78g and the important components the panels are modeled to fail including various safety related components including diesel generator circuit breakers and AC Buses.

Table 5.5-3 provides the Unit 3 FV importance measures for SSC fragilities. The Unit 3 SLERF FV contributors are similar to the Unit 2 contributors with the exception of the addition of fragility groups S-INCP08, S-INCP04-1-, S-CH521, S-ACBS14, S-CH451, S-CH462, and S-CH364 and the omission of groups S-INIR18-9-, S-ACBS10, S-DCBU2, S-DCBU5, S-CH483, S-CH101, S-INCP03-, CRIB, and S-DCBY3. The top two (2) new fragility groups for the Unit 3 fragility importance measures are discussed below:

- Fragility group S-INCP08 models only Unit 3 control panel 903-32. Therefore, its failure is more impactful on the Unit 3 model. Unit 3 control panel 903-32 impacts various AC and DC support system as well as various Div. 1 ECCS equipment (e.g., Div. 1 LPCI pumps).

- S-INCP04-1- consists of the Unit 3 counterparts to the control panels that make up group S-INCP04-.

Differences in risk significance for relay chatter groups between the U2 and U3 model arise because Am values are changed for some relay chatter groups that do not model correlated failure of components in both units. For example, S-CH101 models relay chatter failing HPCI with Am = 0.45g in the U2 model and is risk significant, however this relay chatter group affecting U3 HPCI has Am = 1.73g and is not risk significant for U3.

Table 5.5-4 provides the Unit 2 SLERF FV importance measures for the operator actions. Similar to the total FV for the SSC fragilities, the total FV for the operator actions is the sum of the individual FV values for the G1-G7 range of the hazard intervals.

The top five (5) operator action contributors to the Unit 2 SLERF FV are examined below:

Failure to align portable battery chargers (FV = 2.28E-02).

The portable battery chargers support the U2 and U3 125 VDC batteries, which in turn support the IC, HPCI, EDG2, EDG3, and EDG2/3. Given the high risk significance of the U2 and U3 125 VDC battery chargers, this operator action is an important backup strategy for supplying power to critical equipment.

Failure to inhibit ADS (no high pressure injection) (ATWS) (FV = 2.19E-02).

Failure of the operator to satisfactorily perform this action directly influences the assessment of subsequent, more difficult operator actions included in the event tree model such as the control of the low pressure injection systems. Consequently, ADS inhibit failure is assumed to lead directly to a Class IV because of the inability to successfully control the large volume of makeup from low pressure injection systems that would rapidly displace boron out of the core region and cause prompt recriticality or wash boron from the RPV.

Failure to inject through ‘A’ LPCI loop given ‘B’ LPCI loop failure (FV = 1.47E-02).

The third highest operator action contributor to Unit 2 SLERF is operator failure to switch LPCI injection loops when the LPCI loop chosen by the LPCI loop selection logic fails. This action is risk significant because several of the risk significant control panel and instrument rack fragility groups are modeled to fail LPCI injection valves and pumps. The high probability of failure for LPCI valves and pumps increase the importance of operator action to switch to the functioning LPCI loop injection paths.

Operator fails to recover from relay chatter impacting EDG2, EDG3, and/or EDG2/3 (FV = 1.23E-02).

The EDGs provide electric motive power as well as long term DC control power for systems essential to preventing core damage and radiological release, including HPCI, IC, LPCI and CS. These EDGs depend on successful operation of relays which are susceptible to seismic failure. This operator action models the ability of the operators to recover seismically affected relays and prevent EDG failure.

Operator fails to depressurize the RPV before vessel failure (FV = 8.45E-03).

Depressurization during the in-vessel core melt progression has the potential benefits of allowing the use of low pressure injection system to inject to the RPV to prevent or mitigate continued core melt progression as well as preventing high pressure blowdown induced failure modes of containment when the RPV is breached. Failure to depressurize increases the likelihood that a core damage event will result in a large early release.

Some of the above operator actions are different than the top operator actions contributing to Unit 2 SCDF FV due to differences in the types of dominant accident scenarios contributing to either SCDF or SLERF. The significant operator actions contributing to SLERF involve controlling RPV pressure and aligning RPV injection to maintain water level.

The quantitative results showed that there were no SSCs with significant non-seismic failure contribution to SLERF (i.e., no random failures to start, run, etc. with FV > 5E-03).

However, the Level 2 SPRA model includes a number of non-seismic failures related to phenomenological issues that are based on information from the FPIE DRE PRA model. Based on the SLERF FV importance measures, significant contributors from non-seismic failure events include the following:

- LOCA NOT INDUCED VIA HIGH TEMP, HIGH PRESSURE, OR SORV (2OPPH-NOLOCA-F--) (FV = 0.22). This basic event models phenomenological issues associated with the Level 2 accident progression resulting in a LERF end state. Successful depressurization of the RPV (e.g., due to inducing a LOCA) has a significant impact in precluding a LERF end state.

Table 5.5-5 provides the Unit 3 SLERF Fussell-Vesely (FV) importance measures for the operator actions. The Unit 3 SLERF FV contributors are similar to the Unit 2 contributors.

Top 10 SLERF Cutsets

Table 5.5-6 provides the Top 10 Unit 2 SLERF cutsets. The Top 10 Unit 2 SLERF cutsets are generally similar to the Unit 3 Top 10 SLERF cutsets so the Unit 3 cutsets are not explicitly provided. Similar to the top SCDF cutsets, the top SLERF cutsets involve a %G8 seismic initiating event (seismic magnitude >1.0g). A discussion of the top 10 cutsets is as follows:

Cutset #1 (9.08E-07/yr): This cutset contains the %G8 initiator (with the availability factor included in the initiating event frequency) along with accident class, sequence, and LERF release tags. The %G8 interval (>1.0g) was unquantifiable with currently available processing power and is assumed to lead directly to core damage and a Large Early Release. The Level 1 Accident Class is assumed to be Class V (i.e., containment bypass). Assuming that the highest seismic interval leads directly to core damage and a Large Early Release is consistent with typical industry SPRA models.

Having the %G8 initiator lead directly to SLERF (i.e., by setting all seismic induced fragilities to TRUE) masks the ability to readily identify the contribution to LERF from individual core damage accident classes and sequences. Different accident class contributors would be identified (e.g., Class 1BE for Early Station Blackout, Class 1C for ATWS with loss of RPV makeup, or Class 4A for ATWS with reactivity control) if the %G8 initiator and cutsets were quantified in a more detailed manner. However, Class V is identified because it is consistent with assuming that the %G8 initiator leads directly to a SLERF end state.

Cutset #2 (3.54E-07/yr): Cutset #2 is a %G7 cutset that represents a seismic induced ATWS scenario due to failure of the core shroud tie-rods. RPV overpressure protection and early SLC and HPCI injection are successful. The operators successfully inhibit ADS and control RPV level, but seismic-induced failure of the Unit 3 125 VDC battery racks combined with operator failure to link Unit 3 EDG-powered 4KV Bus 34-1 to Unit 2 4KV Bus 24-1 and failure to align alternate and portable battery chargers results in failure of the IC and failure to depressurize the RPV due to loss of power to the ADS valves, leading to a class 1C core damage event. After core damage, the RPV is able to depressurize with ERVs/SRVs. However, operators fail to recover injection before the RPV melts due to injection system hardware failures and the drywell shell fails, leading to a large early release.

Cutset #3 (3.54E-07/yr): Cutset #3 is similar to Cutset #2 except the alternate 125 VDC power supply is lost because operators fail to load shed to allow sufficient time to align the alternate batteries.

Cutset #4 (3.54E-07/yr): Cutset #4 is similar to Cutset #2 but in this accident sequence, 4KV Bus 24-1 is powered by EDG2 and operators fail to cross-tie power from Bus 24-1 to Bus 34-1, contributing to loss of the electrical support systems.

Cutset #5 (3.54E-07/yr): Cutset #5 is similar to Cutset #3 with load shed failure leading to loss of the alternate 125 VDC batteries. Cutset #5 differs from Cutset #3 and is similar to Cutset #4 in that 4KV Bus 24-1 is powered by EDG2 and operators fail to cross-tie power from Bus 24-1 to Bus 34-1.

Cutset #6 (3.54E-07/yr): Cutset #6 is similar to Cutset #2 but involves a different failure mode for loss of power to the DC battery chargers. In this accident sequence, operators fail to switch to reserve DC power sources instead of failing to align alternate AC power sources to supply power to the DC battery chargers.

Cutset #7 (3.54E-07/yr): Cutset #7 is similar to Cutset #6 except the alternate 125 VDC power supply is unavailable because operators fail to load shed to allow sufficient time to align the alternate batteries.

Cutset #8 (3.54E-07/yr): Cutset #8 is similar to Cutset #2 but involves operator failure to align alternate injection systems such as condensate and standby coolant supply to stop the core melt progression rather than hardware failure of those systems.

Cutset #9 (3.54E-07/yr): Cutset #9 is similar to Cutset #8 except the alternate 125 VDC power supply is lost because operators fail to load shed rather than failing to align the alternate batteries.

Cutset #10 (3.54E-07/yr): Cutset #10 is similar to Cutset #8 but in this accident sequence, 4KV Bus 24-1 is powered by EDG2 and operators fail to cross-tie Bus 24-1 with Bus 34-1, contributing to loss of the IC due to lack of DC power to operate IC valves.

Although the cutsets may appear conservative because of the many HEPs set to 1.0, the cutsets are consistent with the DRE SPRA modeling assumptions/approaches in this regard (i.e., operator error probabilities increase to 1.0 or close 1.0 as the hazard magnitude increases; refer to the SPRA HRA Notebook) [49].

A review of sample cutsets from each decade of quantification results did not identify any improper cutsets.

<u>SLERF Accident Class Contributors</u>

The dominant Level 2 accident class contributors to the DRE SLERF include the following:

- Class 1C (ATWS with failure of RPV makeup) – 55%
- Class 1BE (Station Blackout, core damage at less than 4 hours) - 28%
- Class 4 (ATWS with failure of reactivity control) - 17%

Similar to the discussion in Section 5.4 for the calculation of accident class contributions to SCDF, the accident class contributions to SLERF are calculated based on the FV values of accident class basic events that are included on each cutset. The accident class contributions appear to be generally reasonable.

As shown in Table 5.5-6, the Accident Class for the top LERF cutset is shown to be Class V (seismic induced failure of the RPV supports leading to core damage due to assumed inability to maintain core cooling and subsequent bypass of containment). This is due to assuming that the %G8 initiator leads directly to LERF (i.e., by setting all seismic induced fragilities to TRUE) and masks the ability to readily identify the contribution to individual core damage accident classes and sequences. In order to evaluate the accident class contributions, a sensitivity case was performed to explicitly quantify the %G8 cutsets. The SLERF contribution results discussed below are due to explicitly calculating the %G8 contribution to individual accident classes (e.g., Class 1C, Class 1BE, Class 4A) instead of assuming that the %G8 initiator leads directly to SLERF with an assumed Class V accident class. The Class V accident class is not shown to have an actual high contribution to Level 2 SLERF because the fragility for the RPV supports is relatively high (i.e., Am=4.6g).

Class 1C (ATWS with failure of RPV makeup) accidents have the highest contribution to SLERF due to the relatively low fragility for seismic induced failure to scram (Am = 0.75g) combined with failures of RPV level control (e.g., operator action) in the Level 1 PRA that lead to early core damage scenarios and potentially impact RPV level control in the Level 2 PRA.

Class 1BE (Station Blackout) is a significant contributor to both the DRE Level 1 SCDF and Level 2 SLERF. Some of the highest contributors to Class 1BE include seismic induced failure of SSCs supporting AC and DC power (e.g., 125 VDC system, AC distribution, relay chatter impacting EDGs) that result in failure of all high pressure and low pressure RPV makeup in a short time frame (i.e., core damage in 4 hours or less). No recovery of RPV makeup results in RPV failure and Mark I shell liner failure, resulting in a Large Early Release. The degree of potential conservatisms in these types of Level 2 sequences is discussed in sensitivity cases in Section 5.7 of this report.

Class 4A (ATWS with failure of reactivity control) accidents have a high contribution to SLERF due to the relatively low fragility for seismic induced failure to scram (Am = 0.75g) combined with failure of adequate reactivity control, (e.g., operator action or seismic induced failure of SSCs supporting SLC injection). An ATWS with failure of reactivity control is modeled to result in containment failure due to overpressure from the high core power generation and assumed loss of all RPV makeup following containment failure. Recovery of RPV makeup is not credited in Level 2 PRA for ATWS scenarios, resulting in a Large Early release.

Table 5.5-1 DRE Unit 2 SLERF Contributors by Seismic Hazard Interval Initiating Event [52]

Seismic Hazard Interval	Description	Interval Frequency (/yr)	Interval LERF (/yr)	% of Total SLERF	Cumulative SLERF (/yr)
%G1	%G1 - Hazard Curve: DRE SPRA - PGA Range: 0.1g to 0.2g	9.38E-05	2.10E-11	0%	2.10E-11
%G2	%G2 - Hazard Curve: DRE SPRA - PGA Range: 0.2g to 0.3g	2.27E-05	4.47E-09	0%	4.49E-09
%G3	%G3 - Hazard Curve: DRE SPRA - PGA Range: 0.3g to 0.4g	8.23E-06	4.54E-08	2%	4.98E-08
%G4	%G4 - Hazard Curve: DRE SPRA - PGA Range: 0.4g to 0.5g	3.71E-06	1.51E-07	5%	2.01E-07
%G5	%G5 - Hazard Curve: DRE SPRA - PGA Range: 0.5g to 0.6g	2.00E-06	4.00E-07	14%	6.01E-07
%G6	%G6 - Hazard Curve: DRE SPRA - PGA Range: 0.6g to 0.8g	1.85E-06	8.11E-07	28%	1.41E-06
%G7	%G7 - Hazard Curve: DRE SPRA - PGA Range: 0.8g to 1g	7.68E-07	5.47E-07	19%	1.96E-06
%G8	%G8 - Hazard Curve: DRE SPRA - PGA Range: > 1g	9.41E-07	9.08E-07	32%	2.87E-06

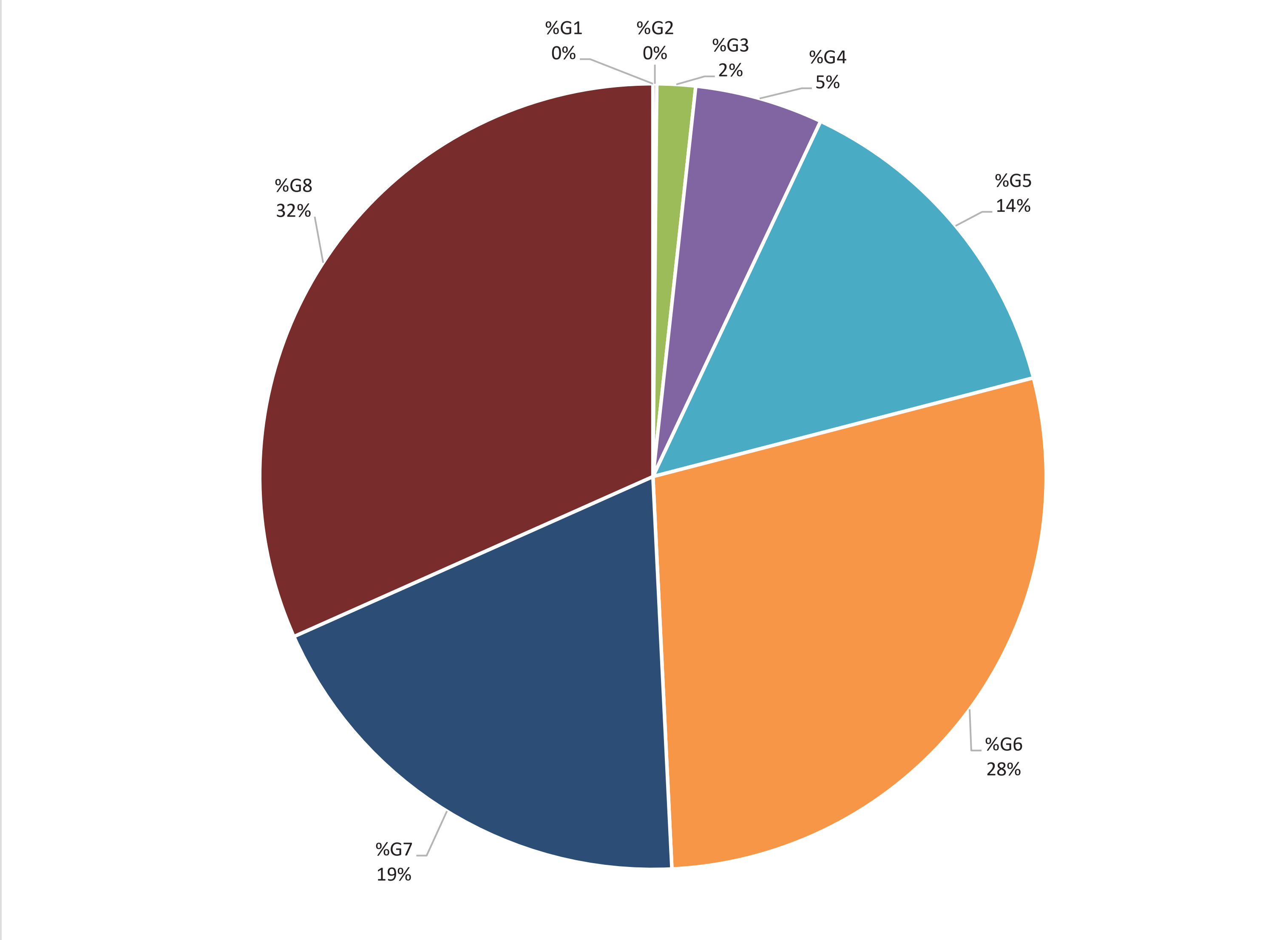

Figure 5.5-1 DRE SPRA Unit 2 SLERF by Hazard Interval Initiating Event [52]

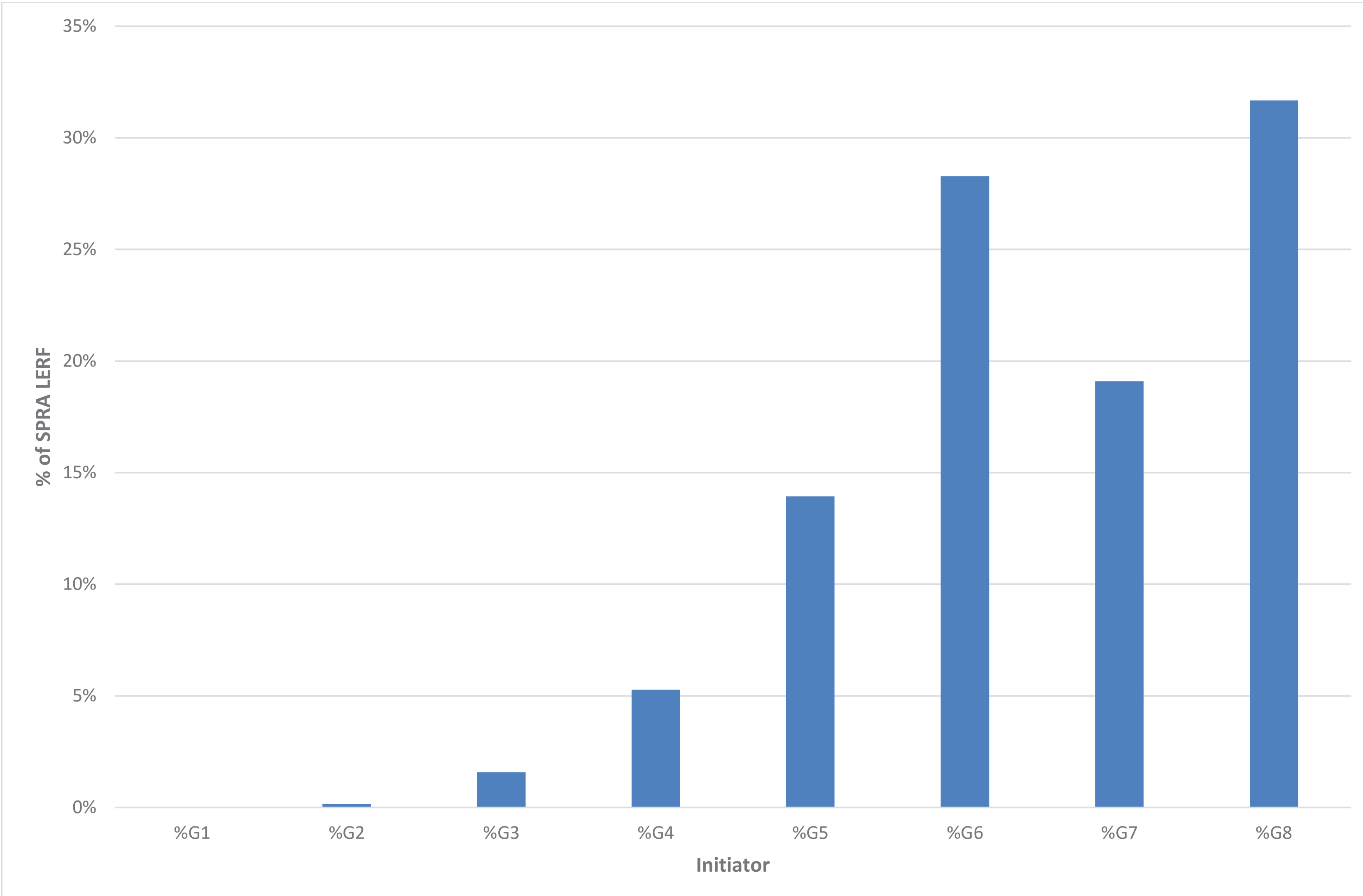

Figure 5.5-2 DRE SPRA Unit 2 SLERF by Hazard Interval Initiating Event [52]

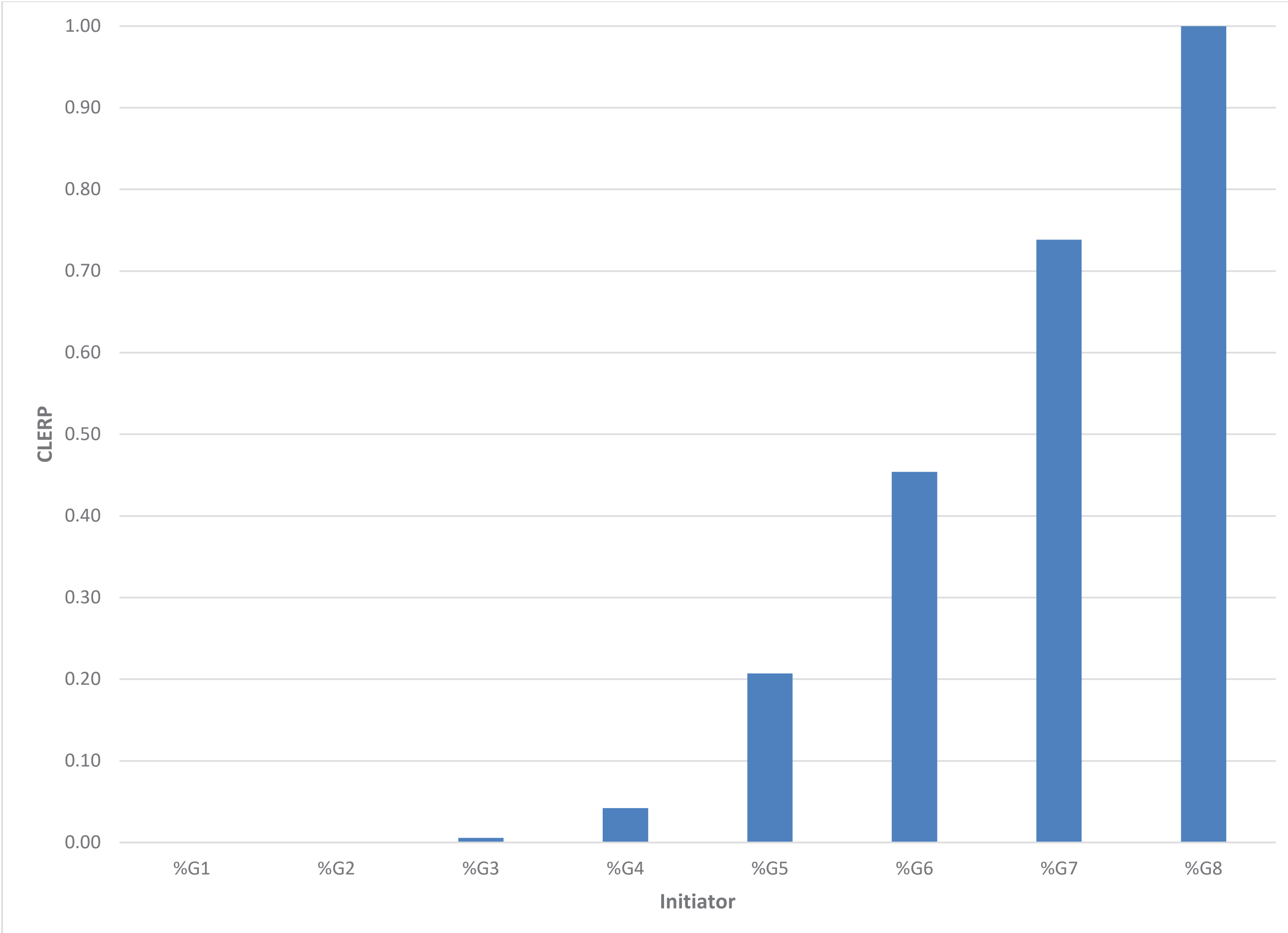

Figure 5.5-3 DRE SPRA Unit 2 CLERP by Hazard Interval Initiating Event [52]

Table 5.5-2 DRE Unit 2 SLERF Fussell-Vesely Importance Measures for SSC Fragilities [52]

Fragility Group ID	Fragility Group Description	FV Total	Am (g)	βr	βu	Failure Mode	Fragility Method
OSP	Offsite Power	9.99E-01	0.3	0.3	0.45	Functional	Generic
SCRAM	RPV Internals (Scram)	6.50E-01	0.75	0.24	0.29	Anchorage	Refined (SoV)
S-DCBY1	Unit 2 125 VDC Battery (549 TB)	1.20E-01	0.72	0.24	0.48	Anchorage	Refined (SoV)
S-INIR18-9-	Instrument Rack Group 18-9-1-1 (2202-7)	3.70E-02	0.39	0.24	0.32	Block Wall	Refined (CDFM)
S-INCP04-	Control Panel Group C20-4 (Panels 902-15, 902-17, 902-18, 902-19, 902-16, 902-20)	2.47E-02	0.78	0.24	0.5	Anchorage	Refined (SoV)
S-DCBY2	Unit 3 125 VDC Battery (551 TB)	1.82E-02	0.72	0.22	0.29	Anchorage	Refined (CDFM)
S-ACBS10	480V MCC 35-2, 38-2, 38-3	1.58E-02	0.74	0.24	0.32	Functional	Refined (CDFM)
S-DCBU2	Unit 2 125 VDC TRAIN A BUSSES (2-83125) - 549 TB	1.43E-02	1.2	0.29	0.4	Functional	Refined (SoV)
S-DCBU5	Unit 2 125 VDC TRAIN B BUSSES (2-83125) - 549 TB	1.34E-02	0.79	0.24	0.32	Functional	Refined (CDFM)
S-DCBC3	Unit 3 125 VDC Battery Charger #3 - 538 TB	1.21E-02	0.58	0.24	0.32	Anchorage	Refined (CDFM)
S-INCP07-	Control Panel Group C20-7 (Panels 902-33, 903-33, 902-32)	1.18E-02	1.07	0.21	0.28	Functional	Refined (CDFM)
S-INCP01-	Control Panel Group C20-1 (Panels 902-3, 903-3, 902-4, 903-4)	9.57E-03	1.27	0.24	0.32	Anchorage	Refined (CDFM)
S-CH483	Relay Chatter ID 483 (CS A-Recov.)	6.76E-03	0.91	0.24	0.32	Functional	Refined (CDFM)
S-CH101	Relay Chatter ID 101 (HPCI-Recov.)	6.34E-03	0.45	0.24	0.32	Functional	Refined (CDFM)
S-ACBS19	4160V AC/ Switchgear 40	5.65E-03	0.99	0.24	0.32	Functional	Refined (CDFM)
S-INCP03-	Control Panel Group C20-3 (Panels 903-8, 902-8)	5.43E-03	1.09	0.24	0.32	Anchorage	Refined (CDFM)
CRIB	Crib House	5.42E-03	0.99	0.24	0.26	Structure	Refined (CDFM)
S-DCBY3	SBODG2 Battery 6A and SBODG3 Battery 7A	5.28E-03	0.29	0.22	0.43	Functionality	Refined (SoV)
S-ACBS14	480V MCC 28-2 and 28-3	5.06E-03	0.71	0.24	0.32	Functionality	Refined (CDFM)

Table 5.5-3 DRE Unit 3 SLERF Fussell-Vesely Importance Measures for SSC Fragilities [52]

FRAGILITY GROUP ID	FRAGILITY GROUP DESCRIPTION	FV TOTAL	Am (g)	βr	βu	Failure Mode	Fragility Method
OSP	Offsite Power	9.99E-01	0.3	0.3	0.45	Functional	Generic
SCRAM	RPV Internals (Scram)	6.63E-01	0.75	0.24	0.29	Anchorage	Refined (SoV)
S-DCBY2	Unit 3 125 VDC Battery (551 TB)	1.04E-01	0.72	0.22	0.29	Anchorage	Refined (CDFM)
S-DCBY1	Unit 2 125 VDC Battery (549 TB)	3.13E-02	0.72	0.24	0.48	Anchorage	Refined (SoV)
S-DCBC3	Unit 3 125 VDC Battery Charger #3 - 538 TB	2.24E-02	0.58	0.24	0.32	Anchorage	Refined (CDFM)
S-INCP04-	Control Panel Group C20-4 (Panels 902-15, 902-17, 902-18, 902-19, 902-16, 902-20)	1.63E-02	0.78	0.24	0.5	Anchorage	Refined (SoV)
S-INCP08-	Control Panel Group C20-8 (Panels 903-32)	1.50E-02	1.04	0.21	0.28	Anchorage	Refined (CDFM)
S-INCP01-	Control Panel Group C20-1 (Panels 902-3, 903-3, 902-4, 903-4)	8.25E-03	1.27	0.24	0.32	Anchorage	Refined (CDFM)
S-INCP04-1-	Control Panel Group C20-4-1 (Panels 903-15, 903-17, 903-18, 903-19, 903-16, 903-20)	7.57E-03	0.78	0.24	0.5	Anchorage	Refined (SoV)
S-CH521	Relay Chatter ID 521 (CS B-Recov.)	7.28E-03	0.75	0.24	0.32	Functional	Refined (CDFM)
S-ACBS14	480V MCC 28-2 and 28-3	6.96E-03	0.71	0.24	0.32	Functional	Refined (CDFM)
S-ACBS19	4160V AC/ Switchgear 40	6.78E-03	0.99	0.24	0.32	Functional	Refined (CDFM)
S-CH451	Relay Chatter ID 451 (Bus 29 feed to 480 VAC MCC 29-7)(1)	5.21E-03	0.47	0.24	0.32	Functional	Refined (CDFM)
S-CH462	Relay Chatter ID 462 (Bus 29 feed to 480 VAC MCC 29-7)(2)	5.21E-03	0.47	0.24	0.32	Functional	Refined (CDFM)
S-INCP07-	Control Panel Group C20-7 (Panels 902-33, 903-33, 902-32)	5.10E-03	1.07	0.21	0.28	Functional	Refined (CDFM)
S-CH364	Relay Chatter ID 364 (EDG2/3-Recov.)	5.06E-03	1.05	0.24	0.32	Functional	Refined (CDFM)

Notes to Table 5.5-3:

(1) SPRA fragility group S-CH451 used as a surrogate for relay IDs 465 and 475 (Bus 39 feed to 480 VAC MCC 39-7).

(2) SPRA fragility group S-CH462 used as a surrogate for relay ID 476 (Bus 39 feed to 480 VAC MCC 39-7).

Table 5.5-4 DRE Unit 2 SLERF Fussell-Vesely Importance Measures for Operator Actions [52]

OPERATOR ACTION ID	OPERATOR ACTION DESCRIPTION	FV TOTAL
BDCOPPORTCHRGH--	FAILURE TO ALIGN PORTABLE BATTERY CHARGERS	2.28E-02
2ADOPINHIBIT-H--	FAILURE TO INHIBIT ADS (NO HP INJECTION) (ATWS)	2.19E-02
2LIOPLIA-INJ-H--	FAILURE TO INJECT THROUGH A LOOP GIVEN B LOOP FAILURE	1.47E-02
BDGOPCHATREC1H--	Operator Fails to Recover From Relay Chatter Impacting EDG 2, 3, and/or 2/3 (SEISMIC)	1.23E-02
2OPOP-DEPRESSH--	OPERATOR FAILS TO DEPRESSURIZE THE RPV BEFORE VESSEL FAILURE	8.45E-03
2SLOP-IN-ERLYH--	FAILURE TO INITIATE SLC EARLY	7.61E-03
BDCOPLOADSHEDH--	FAILURE TO SHED 125V DC LOAD (UNDER SBO CONDITIONS)	7.09E-03
2ADOP-ATWSADSH--	FAILURE TO DEPRESSURIZE THE RPV (ADS) (ATWS)	6.55E-03
2HIOPCHATREC1H--	Operator Fails to Recover From Relay Chatter Impacting HPCI (SEISMIC)	6.34E-03

Notes to Table 5.5-4:

(1) This table covers independent and dependent post-initiator HEPs and their risk contribution; however, if dependent HEPs do not show up in this table that is because their FV value is below 5E-03.

(2) The independent post-initiator HEP FV values presented in this table do not include the risk contribution from the independent HEPs appearing in dependent HEPs.

Table 5.5-5 DRE Unit 3 SLERF Fussell-Vesely Importance Measures for Operator Actions [52]

OPERATOR ACTION ID	OPERATOR ACTION DESCRIPTION	FV TOTAL
BDCOPPORTCHRGH--	FAILURE TO ALIGN PORTABLE BATTERY CHARGERS	4.22E-02
2ADOPINHIBIT-H--	FAILURE TO INHIBIT ADS (NO HP INJECTION) (ATWS)	2.01E-02
2OPOP-DEPRESSH--	OPERATOR FAILS TO DEPRESSURIZE THE RPV BEFORE VESSEL FAILURE	1.12E-02
BDGOPCHATREC1H--	Operator Fails to Recover From Relay Chatter Impacting EDG 2, 3, and/or 2/3 (SEISMIC)	9.71E-03
2ADOP-ATWSADSH--	FAILURE TO DEPRESSURIZE THE RPV (ADS) (ATWS)	7.48E-03
2SLOP-IN-ERLYH--	FAILURE TO INITIATE SLC EARLY	6.99E-03
BDCOPLOADSHEDH--	FAILURE TO SHED 125V DC LOAD (UNDER SBO CONDITIONS)	5.84E-03
BDCOPALT-BATDH--	FAILURE TO ALIGN ALTERNATE BATTERY GIVEN DUAL UNIT LOOP	5.46E-03

Notes to Table 5.5-5:

(1) This table covers independent and dependent post-initiator HEPs and their risk contribution; however, if dependent HEPs do not show up in this table that is because their FV value is below 5E-03.

(2) The independent post-initiator HEP FV values presented in this table do not include the risk contribution from the independent HEPs appearing in dependent HEPs.

Table 5.5-6 DRE Unit 2 Top 10 Seismic LERF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
1	9.08E-07	9.08E-07	%G8	Seismic Initiating Event (>1g)
		1.00E+00	LERF	ACCIDENT CLASS V MARKER
		1.00E+00	RCVCL-5	ACCIDENT CLASS V
		1.00E+00	RCVL2-V-002	ACCIDENT SEQUENCE V-002
2	3.54E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.90E-01	2GVPH-INERT--X--	CONTAINMENT INERTED; VENTING NOT REQUIRED
		1.00E+00	2RXPH-CRDSYS-F--	CRD SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FIRESYSF--	FIRE SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FWSYS--F--	FEEDWATER SYSTEM UNAVAILABLE
		1.00E+00	2SIHU-RCVR---H--	FAILURE TO RECOVER A WATER SYSTEM
		1.00E+00	2SIPH-BARRIS-F--	DW BARRIERS FAIL TO PREVENT DEBRIS FROM CONTACTING SHELL
		1.00E+00	2SIPH-SUMPOV-F--	MELT OVERFLOWS SUMP
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1C	ACCIDENT CLASS IC
		1.00E+00	RCVL2-IC-041	ACCIDENT SEQUENCE IC-041
		1.00E+00	RCVSEQ-ATW6-27	ACCIDENT SEQUENCE ATW6-27
		6.80E-01	SCRAM-C-%G7	SEISMIC FRAGILITY FOR %G7: RPV Internals (Scram)
		7.24E-01	S-DCBY2-C-%G7	SEISMIC FRAGILITY FOR %G7: Unit 3 125 VDC Battery (551 TB)
		1.00E+00	SRX07_2ACOP-28-29T-H--	S-HEP G7: FAILURE TO X-TIE BUSES 28 & 29 (TRANSIENT)
		1.00E+00	SRX07_2ACOP-U2U3EDGH--	S-HEP G7: CROSS TIE AC POWER FROM 34-1 TO 24-1 WHEN 34-1 IS POWERED FROM EDG 3
		1.00E+00	SRX07_2RXRX-FRECINJH--	S-HEP G7: OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		1.00E+00	SRX07_BDCOPALT-BATDH--	S-HEP G7: FAILURE TO ALIGN ALTERNATE BATTERY GIVEN DUAL UNIT LOOP
		1.00E+00	SRX07_BDCOP-ALTCHRGH--	S-HEP G7: FAILURE TO SWITCH TO ALTERNATE BATTERY CHARGER
		1.00E+00	SRX07_BDCOPPORTCHRGH--	S-HEP G7: FAILURE TO ALIGN PORTABLE BATTERY CHARGERS
3	3.54E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.90E-01	2GVPH-INERT--X--	CONTAINMENT INERTED; VENTING NOT REQUIRED
		1.00E+00	2RXPH-CRDSYS-F--	CRD SYSTEM UNAVAILABLE

Table 5.5-6 DRE Unit 2 Top 10 Seismic LERF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	2RXPH-FIRESYSF--	FIRE SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FWSYS--F--	FEEDWATER SYSTEM UNAVAILABLE
		1.00E+00	2SIHU-RCVR---H--	FAILURE TO RECOVER A WATER SYSTEM
		1.00E+00	2SIPH-BARRIS-F--	DW BARRIERS FAIL TO PREVENT DEBRIS FROM CONTACTING SHELL
		1.00E+00	2SIPH-SUMPOV-F--	MELT OVERFLOWS SUMP
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1C	ACCIDENT CLASS IC
		1.00E+00	RCVL2-IC-041	ACCIDENT SEQUENCE IC-041
		1.00E+00	RCVSEQ-ATW6-27	ACCIDENT SEQUENCE ATW6-27
		6.80E-01	SCRAM-C-%G7	SEISMIC FRAGILITY FOR %G7: RPV Internals (Scram)
		7.24E-01	S-DCBY2-C-%G7	SEISMIC FRAGILITY FOR %G7: Unit 3 125 VDC Battery (551 TB)
		1.00E+00	SRX07_2ACOP-28-29T-H--	S-HEP G7: FAILURE TO X-TIE BUSES 28 & 29 (TRANSIENT)
		1.00E+00	SRX07_2ACOP-U2U3EDGH--	S-HEP G7: CROSS TIE AC POWER FROM 34-1 TO 24-1 WHEN 34-1 IS POWERED FROM EDG 3
		1.00E+00	SRX07_2RXRX-FRECINJH--	S-HEP G7: OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		1.00E+00	SRX07_BDCOP-ALTCHRGH--	S-HEP G7: FAILURE TO SWITCH TO ALTERNATE BATTERY CHARGER
		1.00E+00	SRX07_BDCOPLOADSHEDH--	S-HEP G7: FAILURE TO SHED 125V DC LOAD (UNDER SBO CONDITIONS)
		1.00E+00	SRX07_BDCOPPORTCHRGH--	S-HEP G7: FAILURE TO ALIGN PORTABLE BATTERY CHARGERS
4	3.54E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.90E-01	2GVPH-INERT--X--	CONTAINMENT INERTED; VENTING NOT REQUIRED
		1.00E+00	2RXPH-CRDSYS-F--	CRD SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FIRESYSF--	FIRE SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FWSYS--F--	FEEDWATER SYSTEM UNAVAILABLE
		1.00E+00	2SIHU-RCVR---H--	FAILURE TO RECOVER A WATER SYSTEM
		1.00E+00	2SIPH-BARRIS-F--	DW BARRIERS FAIL TO PREVENT DEBRIS FROM CONTACTING SHELL
		1.00E+00	2SIPH-SUMPOV-F--	MELT OVERFLOWS SUMP
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1C	ACCIDENT CLASS IC

Table 5.5-6 DRE Unit 2 Top 10 Seismic LERF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	RCVL2-IC-041	ACCIDENT SEQUENCE IC-041
		1.00E+00	RCVSEQ-ATW6-27	ACCIDENT SEQUENCE ATW6-27
		6.80E-01	SCRAM-C-%G7	SEISMIC FRAGILITY FOR %G7: RPV Internals (Scram)
		7.24E-01	S-DCBY2-C-%G7	SEISMIC FRAGILITY FOR %G7: Unit 3 125 VDC Battery (551 TB)
		1.00E+00	SRX07_2ACOP-28-29T-H--	S-HEP G7: FAILURE TO X-TIE BUSES 28 & 29 (TRANSIENT)
		1.00E+00	SRX07_2ACOP-U2U3---H--	S-HEP G7: FAILURE TO X-TIE U2 TO U3 AC
		1.00E+00	SRX07_2RXRX-FRECINJH--	S-HEP G7: OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		1.00E+00	SRX07_BDCOPALT-BATDH--	S-HEP G7: FAILURE TO ALIGN ALTERNATE BATTERY GIVEN DUAL UNIT LOOP
		1.00E+00	SRX07_BDCOP-ALTCHRGH--	S-HEP G7: FAILURE TO SWITCH TO ALTERNATE BATTERY CHARGER
		1.00E+00	SRX07_BDCOPPORTCHRGH--	S-HEP G7: FAILURE TO ALIGN PORTABLE BATTERY CHARGERS
5	3.54E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.90E-01	2GVPH-INERT--X--	CONTAINMENT INERTED; VENTING NOT REQUIRED
		1.00E+00	2RXPH-CRDSYS-F--	CRD SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FIRESYSF--	FIRE SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FWSYS--F--	FEEDWATER SYSTEM UNAVAILABLE
		1.00E+00	2SIHU-RCVR---H--	FAILURE TO RECOVER A WATER SYSTEM
		1.00E+00	2SIPH-BARRIS-F--	DW BARRIERS FAIL TO PREVENT DEBRIS FROM CONTACTING SHELL
		1.00E+00	2SIPH-SUMPOV-F--	MELT OVERFLOWS SUMP
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1C	ACCIDENT CLASS IC
		1.00E+00	RCVL2-IC-041	ACCIDENT SEQUENCE IC-041
		1.00E+00	RCVSEQ-ATW6-27	ACCIDENT SEQUENCE ATW6-27
		6.80E-01	SCRAM-C-%G7	SEISMIC FRAGILITY FOR %G7: RPV Internals (Scram)
		7.24E-01	S-DCBY2-C-%G7	SEISMIC FRAGILITY FOR %G7: Unit 3 125 VDC Battery (551 TB)
		1.00E+00	SRX07_2ACOP-28-29T-H--	S-HEP G7: FAILURE TO X-TIE BUSES 28 & 29 (TRANSIENT)
		1.00E+00	SRX07_2ACOP-U2U3---H--	S-HEP G7: FAILURE TO X-TIE U2 TO U3 AC
		1.00E+00	SRX07_2RXRX-FRECINJH--	S-HEP G7: OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT

Table 5.5-6 DRE Unit 2 Top 10 Seismic LERF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	SRX07_BDCOP-ALTCHRGH--	S-HEP G7: FAILURE TO SWITCH TO ALTERNATE BATTERY CHARGER
		1.00E+00	SRX07_BDCOPLOADSHEDH--	S-HEP G7: FAILURE TO SHED 125V DC LOAD (UNDER SBO CONDITIONS)
		1.00E+00	SRX07_BDCOPPORTCHRGH--	S-HEP G7: FAILURE TO ALIGN PORTABLE BATTERY CHARGERS
6	3.54E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.90E-01	2GVPH-INERT--X--	CONTAINMENT INERTED; VENTING NOT REQUIRED
		1.00E+00	2RXPH-CRDSYS-F--	CRD SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FIRESYSF--	FIRE SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FWSYS--F--	FEEDWATER SYSTEM UNAVAILABLE
		1.00E+00	2SIHU-RCVR---H--	FAILURE TO RECOVER A WATER SYSTEM
		1.00E+00	2SIPH-BARRIS-F--	DW BARRIERS FAIL TO PREVENT DEBRIS FROM CONTACTING SHELL
		1.00E+00	2SIPH-SUMPOV-F--	MELT OVERFLOWS SUMP
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1C	ACCIDENT CLASS IC
		1.00E+00	RCVL2-IC-041	ACCIDENT SEQUENCE IC-041
		1.00E+00	RCVSEQ-ATW6-27	ACCIDENT SEQUENCE ATW6-27
		6.80E-01	SCRAM-C-%G7	SEISMIC FRAGILITY FOR %G7: RPV Internals (Scram)
		7.24E-01	S-DCBY2-C-%G7	SEISMIC FRAGILITY FOR %G7: Unit 3 125 VDC Battery (551 TB)
		1.00E+00	SRX07_2DCOP-ALTRES1H--	S-HEP G7: FAILURE TO SWITCH TO RESERVE DC POWER
		1.00E+00	SRX07_2RXRX-FRECINJH--	S-HEP G7: OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		1.00E+00	SRX07_BDCOPALT-BATDH--	S-HEP G7: FAILURE TO ALIGN ALTERNATE BATTERY GIVEN DUAL UNIT LOOP
		1.00E+00	SRX07_BDCOP-ALTCHRGH--	S-HEP G7: FAILURE TO SWITCH TO ALTERNATE BATTERY CHARGER
		1.00E+00	SRX07_BDCOPPORTCHRGH--	S-HEP G7: FAILURE TO ALIGN PORTABLE BATTERY CHARGERS
7	3.54E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.90E-01	2GVPH-INERT--X--	CONTAINMENT INERTED; VENTING NOT REQUIRED
		1.00E+00	2RXPH-CRDSYS-F--	CRD SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FIRESYSF--	FIRE SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FWSYS--F--	FEEDWATER SYSTEM UNAVAILABLE

Table 5.5-6 DRE Unit 2 Top 10 Seismic LERF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	2SIHU-RCVR---H--	FAILURE TO RECOVER A WATER SYSTEM
		1.00E+00	2SIPH-BARRIS-F--	DW BARRIERS FAIL TO PREVENT DEBRIS FROM CONTACTING SHELL
		1.00E+00	2SIPH-SUMPOV-F--	MELT OVERFLOWS SUMP
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1C	ACCIDENT CLASS IC
		1.00E+00	RCVL2-IC-041	ACCIDENT SEQUENCE IC-041
		1.00E+00	RCVSEQ-ATW6-27	ACCIDENT SEQUENCE ATW6-27
		6.80E-01	SCRAM-C-%G7	SEISMIC FRAGILITY FOR %G7: RPV Internals (Scram)
		7.24E-01	S-DCBY2-C-%G7	SEISMIC FRAGILITY FOR %G7: Unit 3 125 VDC Battery (551 TB)
		1.00E+00	SRX07_2DCOP-ALTRES1H--	S-HEP G7: FAILURE TO SWITCH TO RESERVE DC POWER
		1.00E+00	SRX07_2RXRX-FRECINJH--	S-HEP G7: OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		1.00E+00	SRX07_BDCOP-ALTCHRGH--	S-HEP G7: FAILURE TO SWITCH TO ALTERNATE BATTERY CHARGER
		1.00E+00	SRX07_BDCOPLOADSHEDH--	S-HEP G7: FAILURE TO SHED 125V DC LOAD (UNDER SBO CONDITIONS)
		1.00E+00	SRX07_BDCOPPORTCHRGH--	S-HEP G7: FAILURE TO ALIGN PORTABLE BATTERY CHARGERS
8	3.54E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.90E-01	2GVPH-INERT--X--	CONTAINMENT INERTED; VENTING NOT REQUIRED
		1.00E+00	2RXPH-CRDSYS-F--	CRD SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FWSYS--F--	FEEDWATER SYSTEM UNAVAILABLE
		1.00E+00	2SIHU-RCVR---H--	FAILURE TO RECOVER A WATER SYSTEM
		1.00E+00	2SIPH-BARRIS-F--	DW BARRIERS FAIL TO PREVENT DEBRIS FROM CONTACTING SHELL
		1.00E+00	2SIPH-SUMPOV-F--	MELT OVERFLOWS SUMP
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1C	ACCIDENT CLASS IC
		1.00E+00	RCVL2-IC-041	ACCIDENT SEQUENCE IC-041
		1.00E+00	RCVSEQ-ATW6-27	ACCIDENT SEQUENCE ATW6-27
		6.80E-01	SCRAM-C-%G7	SEISMIC FRAGILITY FOR %G7: RPV Internals (Scram)
		7.24E-01	S-DCBY2-C-%G7	SEISMIC FRAGILITY FOR %G7: Unit 3 125 VDC Battery (551 TB)

Table 5.5-6 DRE Unit 2 Top 10 Seismic LERF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
		1.00E+00	SRX07_2ACOP-28-29T-H--	S-HEP G7: FAILURE TO X-TIE BUSES 28 & 29 (TRANSIENT)
		1.00E+00	SRX07_2ACOP-U2U3EDGH--	S-HEP G7: CROSS TIE AC POWER FROM 34-1 TO 24-1 WHEN 34-1 IS POWERED FROM EDG 3
		1.00E+00	SRX07_2RXOP-ALTINJ-H--	S-HEP G7: OP FAILS TO ALIGN ALT. INJ. SOURCES IN LEVEL2
		1.00E+00	SRX07_2RXRX-FRECINJH--	S-HEP G7: OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		1.00E+00	SRX07_BDCOPALT-BATDH--	S-HEP G7: FAILURE TO ALIGN ALTERNATE BATTERY GIVEN DUAL UNIT LOOP
		1.00E+00	SRX07_BDCOP-ALTCHRGH--	S-HEP G7: FAILURE TO SWITCH TO ALTERNATE BATTERY CHARGER
		1.00E+00	SRX07_BDCOPPORTCHRGH--	S-HEP G7: FAILURE TO ALIGN PORTABLE BATTERY CHARGERS
9	3.54E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.90E-01	2GVPH-INERT--X--	CONTAINMENT INERTED; VENTING NOT REQUIRED
		1.00E+00	2RXPH-CRDSYS-F--	CRD SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FWSYS--F--	FEEDWATER SYSTEM UNAVAILABLE
		1.00E+00	2SIHU-RCVR---H--	FAILURE TO RECOVER A WATER SYSTEM
		1.00E+00	2SIPH-BARRIS-F--	DW BARRIERS FAIL TO PREVENT DEBRIS FROM CONTACTING SHELL
		1.00E+00	2SIPH-SUMPOV-F--	MELT OVERFLOWS SUMP
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1C	ACCIDENT CLASS IC
		1.00E+00	RCVL2-IC-041	ACCIDENT SEQUENCE IC-041
		1.00E+00	RCVSEQ-ATW6-27	ACCIDENT SEQUENCE ATW6-27
		6.80E-01	SCRAM-C-%G7	SEISMIC FRAGILITY FOR %G7: RPV Internals (Scram)
		7.24E-01	S-DCBY2-C-%G7	SEISMIC FRAGILITY FOR %G7: Unit 3 125 VDC Battery (551 TB)
		1.00E+00	SRX07_2ACOP-28-29T-H--	S-HEP G7: FAILURE TO X-TIE BUSES 28 & 29 (TRANSIENT)
		1.00E+00	SRX07_2ACOP-U2U3EDGH--	S-HEP G7: CROSS TIE AC POWER FROM 34-1 TO 24-1 WHEN 34-1 IS POWERED FROM EDG 3
		1.00E+00	SRX07_2RXOP-ALTINJ-H--	S-HEP G7: OP FAILS TO ALIGN ALT. INJ. SOURCES IN LEVEL2
		1.00E+00	SRX07_2RXRX-FRECINJH--	S-HEP G7: OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		1.00E+00	SRX07_BDCOP-ALTCHRGH--	S-HEP G7: FAILURE TO SWITCH TO ALTERNATE BATTERY CHARGER
		1.00E+00	SRX07_BDCOPLOADSHEDH--	S-HEP G7: FAILURE TO SHED 125V DC LOAD (UNDER SBO CONDITIONS)
		1.00E+00	SRX07_BDCOPPORTCHRGH--	S-HEP G7: FAILURE TO ALIGN PORTABLE BATTERY CHARGERS

Table 5.5-6 DRE Unit 2 Top 10 Seismic LERF Cutsets [52]

#	CUTSET PROB	EVENT PROB	EVENT	DESCRIPTION
10	3.54E-07	7.41E-07	%G7	Seismic Initiating Event (0.8g to <1g)
		9.90E-01	2GVPH-INERT--X--	CONTAINMENT INERTED; VENTING NOT REQUIRED
		1.00E+00	2RXPH-CRDSYS-F--	CRD SYSTEM UNAVAILABLE
		1.00E+00	2RXPH-FWSYS--F--	FEEDWATER SYSTEM UNAVAILABLE
		1.00E+00	2SIHU-RCVR---H--	FAILURE TO RECOVER A WATER SYSTEM
		1.00E+00	2SIPH-BARRIS-F--	DW BARRIERS FAIL TO PREVENT DEBRIS FROM CONTACTING SHELL
		1.00E+00	2SIPH-SUMPOV-F--	MELT OVERFLOWS SUMP
		9.78E-01	OSP-C-%G7	SEISMIC FRAGILITY FOR %G7: Offsite Power
		1.00E+00	RCVCL-1C	ACCIDENT CLASS IC
		1.00E+00	RCVL2-IC-041	ACCIDENT SEQUENCE IC-041
		1.00E+00	RCVSEQ-ATW6-27	ACCIDENT SEQUENCE ATW6-27
		6.80E-01	SCRAM-C-%G7	SEISMIC FRAGILITY FOR %G7: RPV Internals (Scram)
		7.24E-01	S-DCBY2-C-%G7	SEISMIC FRAGILITY FOR %G7: Unit 3 125 VDC Battery (551 TB)
		1.00E+00	SRX07_2ACOP-28-29T-H--	S-HEP G7: FAILURE TO X-TIE BUSES 28 & 29 (TRANSIENT)
		1.00E+00	SRX07_2ACOP-U2U3---H--	S-HEP G7: FAILURE TO X-TIE U2 TO U3 AC
		1.00E+00	SRX07_2RXOP-ALTINJ-H--	S-HEP G7: OP FAILS TO ALIGN ALT. INJ. SOURCES IN LEVEL2
		1.00E+00	SRX07_2RXRX-FRECINJH--	S-HEP G7: OPERATOR FAILS TO RECOVER INJECTION BEFORE RPV MELT
		1.00E+00	SRX07_BDCOPALT-BATDH--	S-HEP G7: FAILURE TO ALIGN ALTERNATE BATTERY GIVEN DUAL UNIT LOOP
		1.00E+00	SRX07_BDCOP-ALTCHRGH--	S-HEP G7: FAILURE TO SWITCH TO ALTERNATE BATTERY CHARGER
		1.00E+00	SRX07_BDCOPPORTCHRGH--	S-HEP G7: FAILURE TO ALIGN PORTABLE BATTERY CHARGERS

5.6 SPRA Quantification Uncertainty Analysis

A parametric uncertainty assessment of the DRE SCDF and SLERF is performed using the EPRI UNCERT (Ver. 4.0) software [67] in combination with the EPRI ACUBE (Ver. 2.0) software [68]. UNCERT is a Windows based program that uses CAFTA generated cutsets and PRA databases as inputs to quantify the parametric uncertainty distribution of a group of cutsets.

Probability distribution types and associated distribution statistics are assigned to each of the basic events. These distributions are entered into the CAFTA database for the SPRA. In addition, Type Code information (stored in the TC Table within the CAFTA "rr" database) is used to account for the state of knowledge dependence among correlated input distributions. UNCERT randomly samples from each of the input distributions, in conjunction with the Type Code database, and interfaces with the ACUBE algorithm at each sample to compute the best estimate result of the CAFTA cutset file. The results are stored and the input distributions are sampled many additional times. As the sample trials are completed, the UNCERT software algorithm processes the trial results to form a probability distribution of the sampled SCDF and SLERF result.

A Monte Carlo (or Latin Hypercube sampling algorithm) evaluation can be performed using correlated or uncorrelated probability distributions to represent the inputs for the basic events. The probability density distribution describing the uncertainty in a component failure probability is characterized as a state of knowledge about an assumed fixed value, the same state of knowledge (i.e., the same distribution) may in fact underlie many distinct basic events. For example, the knowledge of the failure rate of one particular motor operated valve is typically based on experience with MOVs in various plant systems. Therefore, the various basic events that involve the failure of a motor operated valve are all in fact estimated from a single "state of knowledge" distribution. Therefore, basic events based on common data are mapped to a single data variable to ensure proper state of knowledge correlation in the parametric sampling process. This is performed by assigning an appropriate Type Code to each unique basic event.

Distribution information is assigned to all basic events in the cutset files, except for those that are intended to be modeled as constants. The sampling covers both non-seismic variables in the cutsets as well as seismic variables. The distribution sampling of the seismic hazard intervals and fragilities are summarized below.

The seismic hazard interval initiating events are sampled using the sampling equations provided by the FRANX software; for example, the equation for the %G1 seismic interval is as follows:

IF(@POINTCALC=1,9.38E-05*@AVAIL ,INVLOGN(5.94E-05, 4.8151, W)*@AVAIL)

This equation uses a logical IF statement as a switch to determine whether the point estimate mean frequency should be returned or whether a sampling capability equation should be returned for use in parametric uncertainty sampling.

If the @POINTCALC variable (added to the SPRA Type Code database) is set by the analyst to a value of one (1) (the base value used in SPRA quantification runs) the initiator frequency equation returns the point estimate mean value (9.38E-05 in the example above) multiplied by the availability factor. If the @POINTCALC Type Code is set by the analyst to 0 (or any value other than 1) the initiator equation invokes the CAFTA INVLOGN (inverse lognormal) function that will be used in parametric uncertainty sampling of cutsets using the UNCERT software.

The INVLOGN function takes three arguments, the median frequency, the frequency error factor, and the sampled lognormal percentile. The third argument of the INVLOGN function, the W Type Code variable, is used to ensure the state of knowledge correlation sampling of the seismic hazard intervals. This variable makes sure that during sampling that each of the hazard interval initiators is being sampled from the same hazard percentile. During UNCERT parametric uncertainty sampling the W Type Code variable (a Uniform distribution variable) is randomly sampled between 0.01 and 0.99. A W Type Code sample instructs the INVLOGN function to return the associated lognormal percentile; for example, a W Type Code sample value of 0.23 during UNCERT analysis will instruct the INVLOGN function to calculate the 23% percentile of the seismic interval initiators.

The seismic fragility basic events are sampled using the fragility sampling equation provided by the FRANX software:

$$f = \Phi\left[\frac{\ell n\left(\frac{a}{A_m}\right)}{\beta_R}\right]$$

Where:

Φ is the standard Gaussian cumulative distribution.

a is the peak ground acceleration level.

A_m is the median seismic capacity of the component.

β_R is the parameter that accounts for random variability in the ground acceleration capacity.

Like the hazard interval initiators, the fragility basic events use CAFTA equations to implement the fragility model concept. For example, the format of a SLC pump fragility basic event equation for the %G6 hazard interval is as follows:

IF(@POINTCALC=1, cummlogn_med('S-SLPM1-AM', EXP(1.645* 'S-SLPM1-BC'), '@%G6'),cummlogn_med('S-SLPM1-AM', EXP(1.645*'S-SLPM1-BR'),'@%G6'))

This equation uses a logical IF statement as a switch to determine whether the point estimate mean fragility should be returned or whether a sampling capability equation should be returned. If the @POINTCALC Type Code is set by the analyst to a value of one (1) (the default value) the fragility equation returns the point estimate mean fragility using the typical β_C version of the fragility mathematical model. If the @POINTCALC Type Code is set to 0 (or any value other than 1) the fragility equation invokes the CAFTA CUMMLOGN_MED function that will be used in parametric uncertainty sampling and will employ the β_R version of the fragility mathematical model with the β_U defining the distribution of the A_m during the sampling.

SCDF Uncertainty

Parametric sampling of the DRE SPRA SCDF was performed on the base SCDF cutset file using the UNCERT Latin Hypercube sampling option, ACUBE BDD value of /c=16000 cutsets (which produces 100% BDD at each pass), and 20,000 samples. The resulting spread of the SCDF is often characterized by the Range Factor of the resulting sampling distribution (calculated as the SQRT(95%/5%)). For the DRE SPRA SCDF, the range factor is approximately 5.4:

- SCDF 95%: 2.76E-05/yr
- SCDF 50%: 5.15E-06/yr
- SCDF 5%: 9.36E-07/yr

This uncertainty range factor on SCDF is reasonable and generally reflective of the uncertainty of the hazard curve (the dominant hazard intervals are %G5 thru %G8 and each of these has an error factor in the 5 to 7 range).

SLERF Uncertainty

Parametric sampling of the DRE SPRA SLERF was performed on the base SLERF cutset file using the UNCERT Latin Hypercube sampling option, ACUBE BDD value of /c=7000 cutsets (which produces ~99% BDD at each pass), and 20,000 samples. The resulting spread of the SLERF is often characterized by the Range Factor of the resulting sampling distribution (calculated as the SQRT(95%/5%). For the DRE SPRA SLERF, the range factor is approximately 6.85:

- SLERF 95%: 1.47E-05/yr
- SLERF 50%: 2.24E-06/yr
- SLERF 5%: 3.13E-07/yr

This uncertainty range factor on SLERF is reasonable and generally reflective of the uncertainty of the hazard curve (the dominant hazard intervals are %G5 thru %G8 and each of these has an error factor in the 5 to 7 range). The uncertainty spread in SLERF is similar to that of SCDF because many of the dominant accident scenarios comprising SCDF proceed directly to SLERF or with very few additional failures.

Completeness Uncertainty

The SPRA should be of sufficient scope and level of detail to support the risk-informed decision under consideration.

Overall Scope: The overall scope of the SPRA is reasonably defined in terms of the following:

- Metrics used to evaluate risk
- Plant Operating States (POSs) for which the risk is to be evaluated
- Types of hazard groups and initiating events that can potentially challenge and disrupt the normal operation of the plant and, if not prevented or mitigated, would eventually result in core damage, a release, and/or health effects.

The following discussions are implemented for the DRE SPRA.

- The risk metrics used are SCDF and SLERF. This is typical of SPRAs and consistent with industry PRA standards and the SPID [2].
- The Plant Operating State (POS) is limited to at-power; this is consistent with the SPID requirements. The DRE SPRA does not model postulated seismic-induced accidents during shutdown or during power transition states.
- The SPRA addresses the entire (i.e., well beyond design basis) seismic hazard curve (PGA-based). Separate SPRA models are not explicitly built to model different spectral hazard curves. This is a typical SPRA modeling approach (i.e., PGA hazard curve used).
- The SPRA covers the typical spectrum of seismic-induced initiating event states (e.g., seismic-induced LOOP, seismic-induced LOOP-LOCA, seismic-induced LOOP-ATWS, seismic-induced key building failures, etc.) as well as seismic-induced secondary hazards.

Level of Detail: A number of decisions made by the analyst determine the level of details included in an SPRA. These decisions include, for example, the structure of the event trees, the mitigating systems that should be included as providing potential success for critical safety functions, the structure of the fault trees, and the screening criteria used to determine which failure modes for which SSCs are to be included.

The level of details needed is that detail required to capture the effect of an application (i.e., the SPRA model needs to be of sufficient detail to ensure the impact of the application can be assessed).

The level of detail in the system fault tree models, accident sequence models, human reliability analysis, and data of the SPRA models is effectively the same as the detailed at-power PRA models used as input to development of the SPRA.

The completeness of the DRE SPRA is sufficient for most risk applications, typical of full-scope SPRAs and consistent with the SPID.

5.7 SPRA Quantification Sensitivity Analysis

Candidate sensitivity cases for the DRE SPRA model were identified consistent with the methodology provided in NUREG-1855 [24] and performed for the DRE FPIE PRA model. The selection process for the sensitivity cases is documented in Appendix I of the DRE SPRA Quantification Notebook [52]. Twenty-two sensitivity cases have been identified in the following PRA element categories:

PRA Element	Description
IE	Probabilistic Seismic Hazard Analysis (PSHA) (Cases 1a, 1b, 1c)
AS	Seismic LERF evaluation (Cases 2a, 2b) and GTR/FLEX evaluation (Cases 2c, 2d, 2e)
SC	Evaluating credit of Long Term IC Operation in selected accident sequences (Case 3a)
SY	Operability of equipment for seismic induced accident sequences (SSC Fragilities) (Cases 4a, 4b, 4c, 4d, 4e and 6)
HR	HRA Evaluation under seismic event (Cases 5a, 5b, 5c)
QU	SPRA logic model quantification approach adjustments (Cases 7a, 7b and 7c)
<Various>	Combination of cases 2a, 5b, 7b and 7c (CASE 8)

Table 5.7-1 provides a summary of the sensitivity cases performed. The seismic PRA model has been used to provide insights and feedback on the degree of

seismic safety enhancement (seismic risk reduction) that can be achieved by potential SPRA model enhancements.

In addition, Tables 5.7-2 and 5.7-3 provide the SCDF and SLERF truncation sensitivity cases, respectively, to support the selection of truncation limits for the base SPRA model quantification. Quantification truncation sensitivities to establish adequate model results convergence were performed and evaluated as part of the peer review.

The truncation information provided in Tables 5.7-2 and 5.7-3 contain conservative results with footnotes referring to the various conservatisms. The truncation test results in Tables 5.7-2 and 5.7-3 are summarized with respect to overall SCDF/SLERF as opposed to hazard interval.

Sensitivity Case 1a: Assume the 84% Upper Bound of Seismic Hazard Curve

This sensitivity was performed by revising the seismic hazard interval initiating event frequencies of the SPRA to use the 84% Upper Bound of the Dresden Seismic Hazard Curve, instead of the Mean hazard curve. The SCDF and SLERF increases by 65% and 68%, respectively. This sensitivity demonstrates that changes to the initiator frequency (Hazard) can have a significant impact on results. This sensitivity is for illustration; industry approaches and expectations are to use the Mean hazard curve for point estimate model quantification runs.

Sensitivity Case 1b: Assume the 16% Lower Bound of Seismic Hazard Curve

This sensitivity was performed by revising the seismic hazard interval initiating event frequencies of the SPRA to use the 16% Lower Bound of the Dresden Seismic Hazard Curve, rather than the Mean hazard curve. The SCDF and SLERF decreases by 80% and 82%, respectively. This sensitivity demonstrates that changes to the initiator frequency (Hazard) can have a significant impact on results. This sensitivity is for illustration; industry approaches and expectations are to use the Mean hazard curve for point estimate model quantification runs.

Sensitivity Case 1c: Assume the EPRI 1989 Seismic Hazard Curve

This sensitivity was performed by replacing the seismic initiating event frequencies of the base quantification [6] with the DRE EPRI NP-6395D (1989) PGA seismic hazard curve [74]. The EPRI NP-6395D hazard curves were typically used in IPEEE program seismic PRA studies (for those utilities that performed an SPRA for the IPEEE). It is understood that the plant specific fragility calculations developed for the DRE SPRA model and used for this sensitivity case are not based on the same seismic hazard input used to develop the EPRI 1989 seismic hazard curve. Therefore, there is a potential disconnect between the seismic hazard frequencies and the seismic fragilities in this sensitivity case. However, for the purposes of evaluating the potential impact of using different mean hazard frequencies from a hazard curve, this potential disconnect between the hazard curve and the fragilities is not explicitly evaluated. The SCDF and SLERF decreases by 76% and 81%, respectively. This sensitivity demonstrates that changes to the initiator frequency (Hazard) can have a significant impact on results. This sensitivity is for illustration; the DRE SPRA development and quantification uses the latest seismic hazard curve (which is the curve used in the base quantification of this risk assessment as developed in [6]).

Sensitivity Case 2a: Conditional SLERF Probability of 0.15 for Short Term SBO

This sensitivity case is based on a separate, more detailed investigation to determine if there are potential conservatisms in the treatment of assigning LERF end states for both the FPIE PRA and SPRA models. This sensitivity case supports potential options to reduce the calculated SLERF value of 2.9E-06/yr for the DRE baseline SPRA model. The discussion below is based on a review of NUREG/CR-7110 [69] and supplemental MAAP runs performed for a separate evaluation [70]. The conclusions from this separate evaluation [70] are as follows:

"An investigation into the assumptions related to the likelihood of unmitigated short term SBO scenarios with no RPV makeup at time=0 leading to LERF resulted in the following insights.

- *The likelihood of experiencing a SORV during the core melt progression process is assessed as being quite high (i.e., 95% likelihood). The presence of a SORV has a dramatic influence on the potential source terms as much of the fission products are swept to the suppression pool prior to vessel failure and subsequent liner melt-through.*
    - *MELCOR and recent MAAP5 runs indicate that the time to vessel failure may be longer than previously anticipated, and the time that the fission product releases exceed the threshold value for being characterized as large in SORV scenarios could be extended for a significant amount of time, and may not occur at all (at least within the first 48 hours).*
    - *The recent evacuation time estimates for DRE indicate that the time to evacuate 100% of the population out to the Emergency Planning Zone (EPZ) is shorter than in previous analysis (i.e., 6.5 hours at most, compared to more than 8 hours previously).*
    - *If [Steam Line Rupture] SLR occurs, then the likelihood of a large and early release increases dramatically since the fission products are released directly to the drywell in this scenario and do not get the benefit of being transported to the suppression pool . High Pressure scenarios are also assessed as being more likely to lead to a large and early release.*
    - *The conditions required for a SLR in [Short Term Station Blackout] STSBO scenarios was examined in the SOARCA study. Conditions for SLR would only occur if the SORV seized partially open such that*

enough depressurization would occur to preclude other SRV openings, but at the same time keep the RPV pressure high enough to enable a SLR. This is assessed as fairly unlikely, and a 10% likelihood value is assigned.

A Monte Carlo analysis was used to estimate the overall likelihood of LERF combining the inputs above. The results show that the likelihood is about 11.6%. This likelihood is dominated by the assumptions related to a SLR occurring."

Given the discussion above, however, a bounding value can also be derived. It is assessed that the likelihood of an SORV leading to conditions that would not be LERF is very high and that the condition that would be LERF is very low. A 5% bounding value is applied to the specific SORV scenario contributors as a conditional probability that the SORV scenario leads to LERF. For the SLR and High-Pressure scenarios, LERF cannot be precluded so these scenarios can be conservatively assumed to be LERF. The results show that the bounding analysis is about 19.3%, or approximately 20%.

This sensitivity case does not re-assign various Level 2 LERF functional accident sequence end states to non-LERF end-states (given that specific sequences may indeed be calculated as LERF or non-LERF depending upon different phenomena assumptions and the random combinations of various assumptions). Instead, the SPRA uses a probability adjustment to the 2OPPH-NOLOCA-F--, "LOCA NOT INDUCED VIA HIGH TEMP, HIGH PRESSURE, OR SORV", basic event in the Level 2 PRA (in the OP node, "RPV DEPRESSURIZED POST CORE DAMAGE", of the CET logic)to simulate the effect on SLERF calculated results from phenomena assumptions for post-core damage stuck open relief valve scenarios. This sensitivity study reduces the probability of the event 2OPPH-NOLOCA-F-- from a value of 0.25 (i.e., the base case) to 0.15 to simulate the effect of the best estimate results of the References [69, 70] studies. The results for this sensitivity study are shown below. The effect on the calculated SLERF is a minor ~6% reduction and no reduction to the calculated SCDF as this is a Level 2 release analysis topic.

Sensitivity Case 2b: Assume Seismic Events >0.5g Result in SLERF

This sensitivity was performed by estimating the impact on SLERF when assuming that all seismic events with magnitude >0.5g result in sufficient delay in the evacuation time such that they are modeled as leading directly to the SLERF end state. The nominal magnitude of 0.5g selected for this sensitivity is a representative value to reflect 2-3x the DRE SSE and intended to reflect that offsite infrastructure (e.g., roads, electric power, stop lights) may be disrupted. This sensitivity case is performed by assuming that all SCDF contributors >0.5g (i.e., %G5, %G6, %G7, %G8) are assumed to be equal to SLERF. The SLERF increases significantly by nearly a factor of 1.7 for this sensitivity case.

Sensitivity Case 2c: Contribution from Seismic-Induced Transient Scenarios

Seismic induced transient events (i.e., no seismic-induced loss of offsite power) are explicitly modeled in the Dresden SPRA. This sensitivity removes the SIET-001 sequence from the DRE SPRA logic model to provide an estimate of the risk contribution from seismic-induced transients. SCDF and SLERF decrease by 1.4% and 0.03%, respectively. The relatively minor contribution of seismic induced transient events is expected given the low median capacity (0.3g) of offsite power.

Sensitivity Case 2d: Remove Credit for FLEX Strategies

This sensitivity case evaluates the impact of FLEX strategies on SCDF/SLERF. FLEX strategies are built into the Dresden FPIE PRA and SPRA and are included in the SPRA base quantification. The Dresden FLEX strategies are implemented, per procedure, in response to an Extended Loss of AC Power (ELAP) condition and are designed and trained to be implemented within t=2 hrs after the initiating event (design and training assumes 1 hr for declaration of an ELAP condition and an additional 1 hr to implement FLEX strategies). The FLEX logic is built into the PRA with these constraints (e.g., FLEX not credited if any AC power is available). The SCDF remains unchanged from the base case while the SLERF increases 0.2% as a result of this sensitivity case.

This sensitivity illustrates the minor benefit to the calculated seismic risk profile from FLEX. The primary reason behind the minor risk reduction benefit from FLEX is that an ELAP declaration is required before FLEX strategies can be credited in the PRA model. The prerequisite for an ELAP declaration is no operable diesel generators. Dresden has a total of five diesel generators split into three relatively diverse sets (EDG2/EDG3, EDG2/3, SBODG2/SBODG3) from both a seismic fragility perspective as well as system dependencies. As such, accident sequences eligible for benefit from FLEX are already of such low frequency that they are very minor contributors to overall SCDF/SLERF.

Sensitivity Case 2e: Enhance Credit for FLEX Strategies

This sensitivity case evaluates potential future modeling refinements and FLEX procedure enhancements that could allow more credit to be taken for FLEX strategies. Enhancements included in this sensitivity include:

- Removed ELAP prerequisite for use of FLEX strategies.
- Removed successful load shedding requirement for use of FLEX strategies.
- Added credit for successful operation of the IC to provide adequate heat removal and delay core damage long enough for FLEX strategies to be implemented.
- Credited FLEX RPV injection in post-containment challenge scenarios.

The combined effect of these FLEX enhancements results in a SCDF and SLERF decrease of 6.1% and 1.4%, respectively. The risk reduction benefit is greater but still a limited benefit given an initial injection source requirement, an hour needed to implement FLEX cannot mitigate some of the more severe scenarios (e.g., Medium and Large LOCAs, ISLOCAs, LOCA breaks outside containment, unmitigated ATWS and seismic-induced failures of key structures).

Sensitivity Case 3a: Long Term Operation of the IC

The base FPIE PRA Dual Unit LOOP (DLOOP) event tree sequence structure conservatively assumes that long term operation of the IC is not viable for the full 24-hr mission time of the PRA even with successful long term makeup to the IC shell (i.e., success at event tree node “ICM”). The DLOOP event tree structure requires another RPV makeup source and decay heat removal system even with initial successful operation of the IC. This conservative success criteria assumption is maintained in the SPRA base quantification given that the SPRA is built upon the FPIE PRA single-top model. This sensitivity case investigates the impact of this success criteria conservatism on the calculated SCDF and SLERF results.

This sensitivity case is approached by manually deleting accident sequences with success of HPCI, IC, and IC shell makeup, and sequences with failure of HPCI but successful IC shell makeup from the SPRA single-top model.

Both SCDF and SLERF remained mostly unchanged as a result of this sensitivity case, decreasing by less than a percent. This sensitivity study illustrates that the conservative assumption in the base model regarding long term IC operation is not significant to the SPRA quantification.

Sensitivity Case 4a: Eliminate Modeling Uncertainty (β_U) in SSC Fragility Probabilities

This sensitivity case assesses the effect of assuming perfect knowledge of the SSC fragility characterization. Fragility modeling uncertainty is a critical impact on the calculated risk metric. For this sensitivity case, the contribution from modeling uncertainty (βu) is eliminated (i.e., no epistemic uncertainty is assumed to exist for all the fragilities) in the seismic-induced failure probability calculations used in the SPRA. The seismic-induced failure probabilities are calculated assuming only the irreducible aleatory uncertainty (βr) exists; this is a theoretical asymptotic assumption of perfect analysis knowledge and quality. For this sensitivity case, the SCDF and SLERF significantly decreases by 62.3% and 55.8%, respectively. This sensitivity is performed for illustrative purposes; future revisions of the SPRA may involve fragility calculation refinements but epistemic uncertainty will always be present and significant.

Sensitivity Case 4b: Improve Am for Normal Offsite AC Power

The fragility for normal offsite AC power is based on an industry generic value of Am=0.3g for the DRE SPRA model. Given the high-risk contribution from seismic induced loss of offsite power events, enhancements to the offsite AC power seismic capacity would reduce the calculated seismic risk profile. However, the ceramic insulators are often a limiting failure mode for offsite AC power. In addition, significant work has been performed at some locations in an attempt to demonstrate significant improvement in the generic value used, without success. For this sensitivity case, offsite AC power seismic capacity is increased an assumed 50% to Am=0.45g. The SCDF and SLERF decreases by 15.2% and 12.9%, respectively.

Sensitivity Case 4c: Decrease Am for Normal Offsite AC Power

This sensitivity evaluates the impact of decreasing the Am for normal offsite AC power by a factor of 2 to 0.15g. The SCDF and SLERF increases by 11.9% and 7%, respectively.

Sensitivity Case 4d: Improve RPV Internals Fragility from Am=0.75g to Am=1.25g

This sensitivity case evaluates the impact of an improved fragility for the RPV Internals fragility group (based on the core shroud fragility calculation) which is modeled in the SPRA to result in a seismic-induced failure to SCRAM. The median capacity for fragility group SCRAM is increased from the base case Am=0.75g to Am=1.25g. Am=1.25g is selected in this sensitivity case based on judgment and to reflect that other US BWR SPRAs commonly show reactor internals fragilities with medians above 1.0g. SCDF and SLERF decreased by 5.5% and 37.5%, respectively. While this sensitivity tends to show that pursuing a greater median capacity for the RPV Internals might be worthwhile to reduce SLERF, it has been determined that a significant improvement is not cost beneficial as discussed in Section 6 of this report.

Sensitivity Case 4e: Fragility Uncorrelation of SBO DG Batteries and Switchboards

The SBO DG batteries 6A (U2) and 7A (U3) are conservatively correlated in the base model to reduce quantification time, although the fragility analysis determined sufficient differences in the failure mode capacities to warrant their modeling as uncorrelated fragilities. The same is true for 125 VDC switchboards 6A and 7A supporting the SBO DGs. This sensitivity case evaluates the impact on the calculated SCDF and SLERF results if these two fragilities groups are split into four uncorrelated fragilities. SCDF decreased by 0.8% and SLERF decreased by 0.2% as a result of this sensitivity case indicating that the modeling assumption to correlate these fragilities in the base model is not significant. The base SPRA model maintains the assumed full correlation of these two fragility groups to facilitate the model quantification.

Sensitivity Case 5a: Remove Credit for Relay Chatter Recovery Actions

Case 5a evaluates the impact on the calculated risk results if relay chatter recovery actions are not credited in the SPRA quantification process. This sensitivity was implemented by adjustments to rules in the post-processor recovery file to set the values of all the modeled relay chatter recovery HEPs to 1.0. SCDF and SLERF increased by 0.4% and 0.04%, respectively. This sensitivity demonstrates that eliminating the credit for relay chatter recovery actions in the SPRA results in a negligible increase in the calculated results for both SCDF and SLERF.

Sensitivity Case 5b: Improve Credit for Relay Chatter Recovery Actions

Case 5b decreases HEPs for all relay chatter recovery events by a factor of 2 to evaluate the impact of improving credit for relay chatter recovery actions. The relay chatter HEPs are incorporated into the SPRA quantification process with HRA dependency values equal to the assumption of Low Dependence (rather than create many new dependent HEP groups for the SPRA to incorporate these chatter response actions). Reducing these values by a factor of 2 is intended to reduce some of that modeling conservatism (for those cutsets where a relay chatter response HEP is the only HEP in the cutset) as well as to postulate reductions in relay chatter response error calculations. The effect of this reduction is small with approximately 1% decrease for both SCDF and SLERF. The reduction by a factor 2 for the relay chatter error rates in the SPRA was selected based on judgment. Regardless of the precision of the error rate reduction factor assumed here, based on review of the FV estimates provided in Sections 5.4 and 5.5 of this report, the maximum reduction in overall SCDF and SLERF is likely no more than 2-3% even if the relay chatter response HEPs were greatly reduced.

Sensitivity Case 5c: Removal of SHRA Adjustments for All HEPs

This sensitivity case was designed to measure the impact of applying seismic adjustments to the FPIE PRA based post-initiator human error probabilities propagating through the SPRA accident sequences. This sensitivity study removes these SHRA adjustments for independent (both screening and detailed seismic HEP calculations) as well as dependent HEPs. The SRX01# through SRX08# seismic versions of post-initiator HEPs are revised to use the FPIE PRA based HEP values and the dependent combinations are recalculated using the FPIE PRA based values. As expected, the SCDF and SLERF both decreased, SCDF by 11.6% and SLERF by 10.4%. However, the assumption that HEP values would not be impacted by a large magnitude seismic event is not reasonable.

Sensitivity Case 6: Incorporate Selected Seismic-Fires

The Dresden SPRA investigation into the potential for postulated seismic-induced fires identified seven SSCs for further consideration (refer to Appendix C31 of Fragility Report EXDR025-REPT-005, Rev. 0) [21]:

- MCC 20-2 (unanchored, fragility calculated as Am=1.34g)
- MCC 20-3 (unanchored, fragility calculated as Am=1.34g)
- MCC 25-1 (unanchored, fragility calculated as Am=1.34g)

- MCC 27-1 (unanchored, fragility calculated as Am=1.34g)
- MCC 30-2 (unanchored, fragility calculated as Am=1.34g)
- H2 Seal Oil cooler vacuum tank @TB-EL534-G/H-31/33 (fragility calculated as Am=0.93g)
- H2 Seal Oil cooler vacuum tank @ TB-EL538-F/G-50/52 (fragility calculated as Am=0.93g)

All of the above items were screened from explicit incorporation into the SPRA based on convolution of the above calculated fragilities with the Dresden hazard curve and use of conditional ignition probabilities from EPRI 3002012980 [76] and then assuming that the postulated ignition directly results in a core damage event. Each of the unanchored MCCs resulted in a convolved bounding CDF of 8.5E-09/yr using the above approach and each of the H2 seal oil cooler tanks resulted in a bounding CDF 4.9E-09/yr frequency. Taken collectively and not applying fragility correlation to the seismic-induced fire scenarios, one may state the bounding CDF contribution is (5 x 8.5E-09/yr) + (2 x 4.9E-09/yr) = 5E-08/yr. This would be a bounding estimate given the assumption of CCDP=1.0.

Reviewing SPRA CCDPs for fragility scenarios representative of the fire-induced effects of each of these postulated fire sources (as determined from the Dresden Fire PRA) shows that the CCDPs are much less than 1.0. For example, the H2 seal oil cooler vacuum tanks are located in the TB south area modeled by fragility groups TB-S-HI. Assuming a postulated fire for a H2 seal oil cooler vacuum tank conservatively results in loss of all equipment in the TB south area results in a weighted average CCDP (and considering the effect of Boolean addition if merged with other cutsets) less than 0.1. If this weighted average estimate is applied to the above scenarios the total risk contribution would be approximately 5E-08/yr * 0.1 = 5E-09/yr and this would even be on the conservative side. SLERF would be expected to be lower and an estimate of 1E-09/yr is reasonable for SLERF for the purposes of this sensitivity discussion. Based on this, incorporation of these scenarios into the PRA model quantification would have no impact on the results.

<u>Sensitivity Case 7a: Incorporate Fragility Complement Logic</u>

The Dresden Seismic Initiating Event Tree (SIET) was developed to include nodal “success logic” on event tree success branches. When converting the SIET to the fault tree format, a flag event (S-NO-SUCC-FLAG) is included to control the propagation of success logic through the model. This success logic is turned off for the base quantification (user preference to minimize the numerous complement fragilities that would appear in every cutset). This sensitivity case evaluates the impact on the calculated SCDF/SLERF when the fragility success (complement) logic is turned on for the SPRA model quantification. SCDF decreased by 3.2% while SLERF decreased by 16.7%. These results are inappropriately conservative (i.e., overstated) due to the inability to quantify SPRA logic model at lower truncation limits.

Given that the DRE SPRA quantification process uses a BDD algorithm (i.e., the EPRI ACUBE software) to process the cutsets, the final results would be expected to be effectively the same regardless of explicit model of accident sequence nodal success complements as long as the model can be quantified at sufficient low truncation and 100% level of BDD is reached in both cases. The use of an explicit modeling of fragility success logic is a modeling technique that can assist the ACUBE software in computer memory management and ability to solve large problems at lower truncation. However, in the case of the DRE SPRA logic model this sensitivity could not be processed at sufficiently low enough truncation to show the true SCDF and SLERF result and the sensitivity SCDF and SLERF are conservatively high.

This sensitivity is performed for illustrative purposes; fragility complement logic does not provide a quantification processing benefit to the DRE SPRA logic model and the SPRA is quantified with fragility complement logic turned off.

Sensitivity Case 7b: Increase Number of Seismic Hazard Intervals

This sensitivity study is to illustrate the change in calculated results if more (thinner) hazard intervals are used to perform the SPRA.

Eight (8) seismic hazard ground motion intervals are used in the Dresden SPRA (as discussed in the S-IE notebook) [47]. The selection of the number of discrete ground motion intervals is a balance between modeling complexity and adequately representing the convolution of the seismic hazard and plant structure, system, and component (SSC) fragility curves. Past industry seismic risk studies have shown that 6-8 ground motion intervals are typically sufficient to adequately perform the discrete convolution quantification. More recent SPRAs (i.e., NTTF 2.1 Seismic era) are often using 8-10 hazard intervals; some more than that.

This sensitivity study uses the two most dominant hazard intervals from the base quantification (i.e., the %G5 and %G6 intervals, from 0.5g thru 0.8g and comprising 50% of the base SCDF) to perform a sensitivity quantification. Instead of the two %G5 (0.5g–0.6g) and %G6 (0.6g-0.8g) intervals, this sensitivity quantification uses 6 intervals over this range, each with a 0.05g interval width. These two intervals are selected for illustration of the effect on the calculated SCDF and SLERF if finer interval widths were used because focusing on two intervals in the majority of the risk profile is sufficiently illustrative and does not require extensive reconstruction of the SPRA model files to revise seismic HRA re-assignments (i.e., the SHEP recovery file can be used as is for this sensitivity).

The results show that the combined base SCDF over the range 0.50g -> 0.80g (i.e., 2.96E-06/yr for G5 and G6) reduces to 2.73E-06/yr if six hazard intervals are used over this range instead of the base calculation of two hazard intervals. Applying this ~8% reduction in calculated results over the %G1 thru %G7 portion of the hazard curve (at ~1.0g and greater, i.e., the %G8 interval, the base result would not change as the CCDP is already ~1.0 and this portion of the hazard curve calculated results would change non-significantly with more and thinner intervals beyond this point) results in an estimated sensitivity total SCDF of 5.45E-06/yr (a reduction of 6.7% from the base total SCDF of 5.84E-06/yr).

The SLERF results show that the combined base SLERF over the range 0.50g -> 0.80g (i.e., 1.21E-06/yr for G5 and G6) reduces to 1.01E-06/yr if six hazard intervals are used over this range. Applying this ~17% reduction in calculated results over the %G1 thru %G7 portion of the hazard curve results in an estimated sensitivity total SLERF of 2.54E-06/yr (a reduction of 11.4% from the base total SLERF of 2.87E-06/yr).

Sensitivity Case 7c: Extend the G8 Hazard Interval

This sensitivity case was designed to evaluate the impact on overall SCDF, SLERF, and risk importance measure results by extending the final G8 hazard interval farther to the right in the hazard curve. The base case Dresden SPRA model groups all seismic initiators greater than 1.0g PGA into the G8 hazard interval using a representative ground motion of 1.1g to calculate SSC seismic failure probabilities for that interval. Given that the G8 interval is assumed to lead directly to core damage and large early release (i.e., CCDP/CLERP for the G8 interval is 1.0), extending the G8 interval by adding additional hazard intervals beyond 1.0g would not have a significant impact on SCDF/SLERF.

This sensitivity is performed by adding five additional hazard intervals ranging from 1.0g to 1.5g with width 0.1g, and one final open-ended interval for ground motions greater than 1.5g. As expected, extending the G8 interval resulted in a < 1% increase in SCDF and a 2.5% decrease in SLERF. SLERF experienced a decrease in frequency because the base case assumes any ground motion greater than 1.0g leads directly to large early release while this sensitivity case credits mitigation of large early release. SSC FVs were generated for this sensitivity case to evaluate the impact of extending the G8 interval on risk importance measure results. Because the G8 hazard interval is a significant risk contributor to SCDF and SLERF and the failure probabilities for components at > 1.0g ground motion are close to 1.0, including the risk contribution from the extended G8 intervals resulted in reduced FVs for most SSCs. No SSCs crossed the risk significance threshold as a result of this sensitivity, while several SSCs decreased below the threshold. This sensitivity demonstrates that the base case assumption that the G8 interval leads directly to core damage and large early release is not overly conservative and does not skew risk importance measures.

Additionally, this sensitivity shows that CLERP = 1.0 near ground motions of 1.5g. The table below shows the base frequencies and CLERPs compared to the sensitivity CLERPs for the extended G8 intervals.

Table 7c-1: Sensitivity Case to Extend the G8 Hazard Interval [52]**Extended G8 Hazard Interval Range (g)**	**SLERF (/yr)**	**Initiator Frequency (/yr)**	**CLERP**
1.0 to 1.1	1.78E-07	2.10E-07	8.76E-01
1.1 to 1.2	1.37E-07	1.54E-07	9.23E-01
1.2 to 1.3	1.05E-07	1.14E-07	9.57E-01
1.3 to 1.4	7.98E-08	8.43E-08	9.80E-01
1.4 to 1.5	6.06E-08	6.31E-08	9.96E-01
1.5+	2.77E-07	2.83E-07	1.00E+00

Sensitivity Case 8: Combine cases 2a, 5b, 7b, and 7c

This sensitivity case is a combination of sensitivity topics with reasonable alternatives and/or expected revisions in the future. This sensitivity case combines the changes made to the model in sensitivity cases 2a, 5b, 7b, and 7c. Prior to the sensitivity quantification, the following changes were made:

- Probability of 2OPPH-NOLOCA-F-- reduced from 0.25 to 0.15
- Recovery flag file edits from sensitivity case 5b implemented

As both of these changes result in decreased SCDF and SLERF cutset frequencies, the existing base case cutsets for hazard intervals G1-G7 were used as a starting point to make these changes, along with the extended G8 interval cutsets created in case 7c. After the ACUBE processing of these altered cutsets, the individual SCDF contributions from each hazard interval (except the extended %G8) were decreased by ~8% to simulate division of the hazard curve into additional seismic hazard intervals as demonstrated in sensitivity case 7b. The resulting SCDF for this combined sensitivity case is calculated to be 5.38E-06/yr (7.8% reduction from the base case SCDF).

Similar adjustments were made to the SLERF cutsets, and the individual SLERF contributions from the G1-G7 hazard intervals were decreased by ~17% to simulate division of the hazard curve into additional seismic hazard intervals as demonstrated in sensitivity case 7b. The resulting SLERF for this combined sensitivity case is calculated to be 2.30E-06/yr (20% reduction from the base case SLERF).

Table 5.7-1 Summary of DRE SPRA Sensitivity Cases [52]

Sensitivity Case #(1)	Description	CDF (/yr)	Delta CDF (/yr)	% Delta CDF	LERF (/yr)	Delta LERF (/yr)	% Delta LERF
Base Case	Base Case (Unit 2)	5.84E-06	N/A	N/A	2.87E-06	N/A	N/A
Case 1a	Use 84% upper bound limit from DRE Seismic Hazard curve.	9.66E-06	3.82E-06	65.5%	4.81E-06	1.95E-06	67.9%
Case 1b	Use 16% lower bound limit from DRE Seismic Hazard curve.	1.16E-06	-4.68E-06	-80.2%	5.03E-07	-2.36E-06	-82.5%
Case 1c	Use 1989 EPRI Hazard Curve	1.40E-06	-4.44E-06	-76.0%	5.39E-07	-2.33E-06	-81.2%
Case 2a	Apply conditional SLERF probability of 0.15 for Short Term SBO to address potential more realistic evaluation of Level 2 phenomena.	N/A	N/A	N/A	2.70E-06	-1.69E-07	-5.9%
Case 2b	Assume that seismic events >0.5g have a significant impact on evacuation and any core damage events result in LERF.	N/A	N/A	N/A	4.81E-06	1.94E-06	67.8%
Case 2c	Remove contribution from seismic-induced transients	5.75E-06	-8.40E-08	-1.4%	2.87E-06	-7.41E-10	-0.03%
Case 2d	Remove credit for FLEX strategies	5.84E-06	0.00E+00	0.00%	2.87E-06	4.35E-09	0.2%
Case 2e	Enhance credit for FLEX strategies	5.48E-06	-3.55E-07	-6.1%	2.83E-06	-3.99E-08	-1.4%
Case 3a	Credit long term operation of the IC	5.80E-06	-3.20E-08	-0.6%	2.87E-06	-3.78E-10	-0.01%

Table 5.7-1 Summary of DRE SPRA Sensitivity Cases [52]

Sensitivity Case #(1)	Description	CDF (/yr)	Delta CDF (/yr)	% Delta CDF	LERF (/yr)	Delta LERF (/yr)	% Delta LERF
Case 4a	Eliminate Modeling Uncertainty (β_U) in SSC Fragility Probabilities	2.20E-06	-3.64E-06	-62.3%	1.27E-06	-1.60E-06	-55.8%
Case 4b	Improve fragility of offsite AC power to Am = 0.45g	4.95E-06	-8.88E-07	-15.2%	2.50E-06	-3.70E-07	-12.9%
Case 4c	Decrease fragility of offsite AC power to Am = 0.15g	6.53E-06	6.96E-07	11.9%	3.07E-06	1.99E-07	7.0%
Case 4d	Improve SCRAM fragility from Am = 0.75g to Am = 1.25g.	5.52E-06	-3.20E-07	-5.5%	1.79E-06	-1.07E-06	-37.5%
Case 4e	Fragility uncorrelation of SBO DG batteries	5.79E-06	-4.55E-08	-0.8%	2.86E-06	-6.25E-09	-0.2%
Case 5a	Remove credit for rely chatter recovery actions	5.86E-06	2.21E-08	0.4%	2.87E-06	1.12E-09	0.04%
Case 5b	Improve credit for relay chatter recovery actions	5.76E-06	-7.40E-08	-1.3%	2.84E-06	-2.59E-08	-0.9%
Case 5c	Remove seismic adjustments from HEPs (use FPIE HEP values)	5.16E-06	-6.78E-07	-11.6%	2.57E-06	-2.99E-07	-10.4%
Case 6	Incorporate selected seismic-fires	5.84E-06	<5E-9	<0.1%	2.87E-06	<1E-9	<0.03%
Case 7a	Incorporate fragility complement logic	5.65E-06	-1.88E-07	-3.2% (conservative)	2.39E-06	-4.79E-07	-16.7% (conservative)
Case 7b	Increase number of seismic hazard intervals	5.45E-06	-3.88E-07	-6.7%	2.54E-06	-3.27E-07	-11.4%

Table 5.7-1 Summary of DRE SPRA Sensitivity Cases [52]

Sensitivity Case #(1)	Description	CDF (/yr)	Delta CDF (/yr)	% Delta CDF	LERF (/yr)	Delta LERF (/yr)	% Delta LERF
Case 7c	Extend G8 hazard interval	5.84E-06	7.49E-10	0.01%	2.80E-06	-7.07E-08	-2.5%
Case 8	Combine cases 2a, 5b, 7b, and 7c	5.38E-06	-4.57E-07	-7.8%	2.30E-06	-5.69E-07	-19.8%

Notes to Table 5.7-1:

(1) The sensitivity study SCDF and SLERF quantifications use the same truncation levels (per hazard interval) as the Base Case. This is reasonable for the purposes of sensitivity studies and is typical practice given that truncation levels are typically set at a level that already challenges computer memory and computational speed. The truncation level convergence test (i.e., < +~5% per decade decrease in truncation level) used in the Base Case quantifications if specifically re-confirmed for each of the sensitivity cases is expected to produce the same truncation levels for most of the sensitivity studies.

Table 5.7-2 SCDF Quantitative Truncation Investigation Summary [52]

Seismic Hazard Interval	Hazard Interval Frequency	Base SCDF and Base Truncation					Truncation Test - Truncation Reduced One Decade					Hazard Interval Truncation Selected
		Base SCDF	% BDD(1)	% of Total SCDF	Base Truncation	Base CCDP	Sensitivity SCDF	% BDD(1)	Sensitivity Truncation	% Change (Total SCDF)	Note	
%G1	9.38E-05	2.13E-09	100%	0%	1.00E-12	2.35E-05	2.38E-09	100%	1.00E-13	0.00%	Note (2)	**1.00E-12**
%G2	2.27E-05	4.33E-08	100%	1%	1.00E-10	1.98E-03	5.04E-08	100%	1.00E-11	0.12%	Note (2)	**1.00E-10**
%G3	8.23E-06	3.05E-07	100%	5%	1.00E-10	3.84E-02	3.33E-07	100%	1.00E-11	0.48%	Note (2)	**1.00E-10**
%G4	3.71E-06	8.77E-07	100%	15%	5.00E-10	2.45E-01	1.00E-06	100%	5.00E-11	2.11%	Note (3)	**5.00E-10**
%G5	2.00E-06	1.23E-06	100%	21%	1.00E-08	6.36E-01	1.43E-06	100%	1.00E-09	3.51%	Note (2)	**1.00E-08**
%G6	1.85E-06	1.73E-06	100%	30%	1.00E-08	9.71E-01	1.79E-06	100%	1.00E-09	0.89%	Note (4)	**1.00E-08**
%G7	7.68E-07	7.39E-07	100%	13%	1.00E-07	9.97E-01	7.41E-07	100%	1.00E-08	0.03%	Note (2)	**1.00E-07**
%G8	9.41E-07	9.08E-07	100%	16%	1.00E-07	1.00E+00	Note (5)	Note (5)	1.00E-08	Note (5)	Note (5)	**1.00E-07**
	Total SCDF	**5.84E-06**										

Notes to Table 5.7-2:

(1) % BDD refers to the percentage of the total SCDF result from the ACUBE software calculated using the BDD algorithm. 100% BDD indicates the entire cutset file was processed using BDD and thus the SCDF value is the true Boolean total result of the cutset file.

(2) Dropping an additional decade in truncation for this hazard interval results in <5% increase in total SCDF.

(3) As part of the truncation study attempts were made to achieve 100% BDD for the G4 hazard interval SCDF truncation sensitivity case. The ACUBE run attempting full BDD on the 64-bit version of the software on a multi-core server machine with 254 GB of RAM failed due to inadequate memory. Multiple runs were then performed for lower levels of BDD to identify the asymptotic trend toward the true total SCDF value. 100% BDD could not be achieved but plotting the ACUBE SCDF for this hazard interval as a function of %BDD indicates based on visual inspection that the hazard interval true SCDF value asymptotically reaches the value shown in the 'Sensitivity SCDF' column. The result is that dropping an additional decade in truncation for this hazard interval results in <5% increase in total SCDF.

(4) As part of the truncation study attempts were made to achieve 100% BDD for the G6 hazard interval SCDF truncation sensitivity case. The ACUBE run attempting full BDD on the 64-bit version of the software on a multi-core server machine with 254 GB of RAM failed due to inadequate memory. However,

the CCDP for G6 is already close to 1.0 (0.971). The increase of 0.89% for dropping truncation by one order of magnitude is calculated assuming a CCDP of 1.0 for this hazard interval, i.e., SCDF sensitivity result for G6 equals the initiator frequency with the availability factor (1.79E-06/yr). The result is that dropping an additional decade in truncation for this hazard interval results in <5% increase in total SCDF.

(5) CCDP is 1.0 at this hazard interval because the %G8 hazard interval initiator is modeled as leading directly to core damage; a lower truncation (if sufficient computer capacity existed to quantify) would not change the results for total SCDF.

Table 5.7-3 SLERF Quantitative Truncation Investigation Summary [52]

Seismic Hazard Interval	Hazard Interval Frequency	Base SCDF and Base Truncation					Truncation Test - Truncation Reduced One Decade					Hazard Interval Truncation Selected
		Base SLERF	% BDD(1)	% of Total SLERF	Base Truncation	Base CLERP	Sensitivity SLERF	% BDD(1)	Sensitivity Truncation	% Change (Total SLERF)	Note	
%G1	9.38E-05	2.10E-11	100%	0%	1.00E-13	2.33E-07	3.30E-11	100%	1.00E-14	0.00%	Note (2)	**1.00E-13**
%G2	2.27E-05	4.47E-09	100%	0%	1.00E-11	2.04E-04	4.96E-09	100%	1.00E-12	0.02%	Note (2)	**1.00E-11**
%G3	8.23E-06	4.54E-08	100%	2%	1.00E-11	5.71E-03	5.25E-08	100%	1.00E-12	0.25%	Note (3, 4)	**1.00E-11**
%G4	3.71E-06	1.51E-07	100%	5%	1.00E-10	4.22E-02	1.60E-07	100%	1.00E-11	0.31%	Note (3, 4)	**1.00E-10**
%G5	2.00E-06	4.00E-07	100%	14%	1.00E-10	2.07E-01	Note (5)	100%	1.00E-11	1.36%	Note (5)	**1.00E-10**
%G6	1.85E-06	8.11E-07	100%	28%	1.00E-08	4.54E-01	8.59E-07	100%	1.00E-09	1.67%	Note (2)	**1.00E-08**
%G7	7.68E-07	5.47E-07	100%	19%	1.00E-07	7.39E-01	Note (6)	100%	1.00E-08	2.67%	Note (6)	**1.00E-07**
%G8	9.41E-07	9.08E-07	100%	32%	1.00E-07	1.00E+00	Note (7)	Note (7)	1.00E-08	Note (7)	Note (7)	**1.00E-07**
	Total SLERF	**2.87E-06**										

Notes to Table 5.7-3:

(1) % BDD refers to the percentage of the total SCDF result from the ACUBE software calculated using the BDD algorithm. 100% BDD indicates the entire cutset file was processed using BDD and thus the SCDF value is the true Boolean total result of the cutset file.

(2) Dropping an additional decade in truncation for this hazard interval results in <5% increase in total SLERF.

(3) Hazard intervals G3 and G4 were unable to be quantified at the indicated sensitivity truncation level due to computer memory limitations. To produce representative cutsets for G3 and G4 at the sensitivity truncation levels, the G2 hazard interval (which is grouped with the G3 and G4 hazard intervals in the same SHRA bin) was quantified at a truncation of 5E-13 for G3 and 1E-14 for G4, then the resulting G2 cutset files were updated with initiator frequencies and seismic failure probabilities corresponding to G3 and G4.

(4) As part of the truncation study attempts were made to achieve 100% BDD for the G3 and G4 hazard interval SLERF truncation sensitivity case. The ACUBE run attempting full BDD on the 64-bit version of the software on a multi-core server machine with 254 GB of RAM failed due to inadequate memory. Multiple runs were then performed for lower levels of BDD to identify the asymptotic trend toward the true total SLERF value. 100% BDD could not be

achieved but plotting the ACUBE SLERF for these hazard intervals as a function of %BDD indicates based on visual inspection that the hazard interval true SLERF value asymptotically reaches the value shown in the 'Sensitivity SLERF' column for each interval. The result is that dropping an additional decade in truncation for this hazard interval results in <5% increase in total SLERF.

(5) Hazard interval G5 was unable to be quantified at the indicated sensitivity truncation level due to computer memory limitations. A similar approach to Note 3 was attempted using the G6 cutsets and updating failure probabilities to G5 values, but the maximum achievable %BDD of 40% represented an overly conservative estimate of the true SLERF for the G5 interval. Instead, the G5 interval was quantified at a truncation half a decade higher than the base truncation (5E-10/yr) and half a decade lower (5E-11/yr). The total increase in SLERF caused by the increase in G5 SLERF between truncations of 5E-10/yr and 5E-11/yr is 1.37%. This indicates that the base truncation level of 1E-10/yr is acceptable as the increase in total SLERF is expected to decrease with decreasing G5 truncation level. The 1.37% increase in G5 SLERF between truncations 5E-10/yr and 5E-11/yr is conservatively used to represent the G5 interval truncation sensitivity total SLERF increase in Table 5.7-3.

(6) Hazard interval G7 was unable to be quantified at the indicated sensitivity truncation level due to computer memory limitations. Instead, the G5 interval was quantified at a slightly higher truncation (3E-07/yr) than the base truncation and one decade lower (3E-08/yr). The total increase in SLERF caused by the increase in G7 SLERF between truncations of 3E-07/yr and 3E-08/yr is 2.67%. This indicates that the base truncation level of 1E-07/yr is acceptable as the increase in total SLERF is expected to decrease with decreasing G7 truncation level. The 2.67% increase in G7 SLERF between truncations 3E-07/yr and 3E-08/yr is conservatively used to represent the G7 interval truncation sensitivity total SLERF increase in Table 5.7-3.

(7) CLERP is 1.0 at this hazard interval because the %G8 hazard interval initiator is modeled as leading directly to SLERF; a lower truncation (if sufficient computer capacity existed to quantify) would not change the results for total SLERF.

5.8 SPRA Logic Model and Quantification Technical Adequacy

The DRE SPRA risk quantification and results interpretation methodology were subjected to an independent peer review against the pertinent requirements in the ASME/ANS PRA Standard [4].

The peer review assessment, and subsequent disposition of peer review findings, is described in Appendix A, and establishes that the DRE SPRA seismic plant response analysis is suitable for this SPRA application.

6. Conclusions

A seismic PRA has been performed for DRE in accordance with the guidance in the PRA Standard [4] and the SPID [2]. The Seismic PRA shows that the point estimate seismic CDF is 5.8E-06 per year (yr) for Unit 2 and is 5.8E-06 per yr for Unit 3 [52]. The seismic LERF is 2.9E-06/yr for Unit 2 and is 2.8E-06/yr for Unit 3 [52]. Uncertainty, importance, and sensitivity analyses were performed. Sensitivity studies were performed to investigate critical assumptions, evaluate the risk impact to variations in the critical assumptions, and identify potential areas to consider for the reduction of seismic risk. These sensitivity studies demonstrated that the baseline model results were consistent with the modeling and the assumptions incorporated into the SPRA model.

One of the risk insights from the DRE SPRA results is that the fragility for RPV Internals has a high contribution to SLERF (i.e., U2 SLERF FV =0.65 and U3 SLERF FV = 0.66). The RPV Internals are calculated to have a relatively low Am = 0.75g due to the identified governing failure mode of the upper and lower clamps on the core shroud tie rods. Seismic failure of the RPV internals is modeled to prohibit successful insertion of the control rods into the reactor (SCRAM), resulting in an ATWS scenario. The SPRA modeling assumptions are conservative for this issue (e.g., failure of the core shroud clamps is assumed to result in instantaneous failure of welds due to rapid expansion of previously identified weld indications and sufficient failure of the shroud and core geometry that leads to failure to scram). Despite the high SLERF FV contribution to the base DRE SPRA model, sensitivity evaluations for potential improvements in the RPV internals fragility (i.e., Sensitivity Case 4d) indicate that the estimated quantitative risk benefits would not justify the cost for the necessary structural improvements to the core shroud. Sensitivity Case 4d determined that increasing the Am for the core shroud tie rod failure from 0.75g to 1.25g would decrease SLERF by approximately 37.5%. However, modification to the core shroud tie rods would require a significant design effort and significant work within the reactor vessel, as well as inherent plant risk associated with implementing the modifications. In addition, while the core shroud tie rod failure currently controls the fragility of the RPV internals, there are several other components with Am values at or below 1.0g. Thus, to achieve an Am of 1.25g for the RPV internals failure would require modification to several internal RPV components. Raising the Am of the core shroud tie rod failure only would lead to a smaller improvement in SLERF because the controlling Am would still be less than 1.0g. A sensitivity study was not performed with an assumed Am of 1.0g but raising from 0.75g to 1.0g to this value would be expected to have approximately one-half the risk benefit seen in sensitivity case 4d. Therefore, it is expected that the cost and effort to perform these modifications to the RPV internals, as well as the inherent plant risk associated with performing these modifications, would not justify the relatively small reduction in SLERF or risk benefits gained.

The Seismic PRA as described in this submittal reflects the as-built/as-operated Seismic PRA freeze date of May 4, 2018 [75]. Appendix A provides a discussion of the peer review assessment performed for the SPRA. It also contains a list and subsequent disposition of peer review findings. There are no significant plant changes that are not included in the model which would have an adverse or significant impact on the results. Reference section A.9 for additional information. Further, no seismic hazard vulnerabilities were identified, and no plant actions have been taken or are planned given the insights (including final SCDF and SLERF values) from this study.

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8. Acronyms

ADS	Automatic Depressurization System
ANS	American Nuclear Society
AOV	Air Operated Valve
ARI	Alternate Rod Insertion
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram (also ATWT, Anticipated Transient Without Trip)
BDD	Binary Decision Diagram
BE	Best Estimate
BOC	Break Outside Containment
BOP	Balance of Plant
BWR	Boiling Water Reactor
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CCSW	Containment Cooling Service Water
CDF	Core Damage Frequency
CENA	Central and Eastern North America
CET	Containment Event Tree
CEUS	Central and Eastern United States
CDFM	Conservative Deterministic Failure Margin
CI	Criticality Importance
CLERP	Conditional Large Early Release Probability
CRD	Control Rod Drive
CS	Core Spray
CST	Condensate Storage Tank
DG	Diesel Generator
DGCW	Diesel Generator Cooling Water
DLOOP	Dual Loss of Offsite Power
DRE	Dresden Nuclear Power Station
ECCS	Emergency Core Cooling System
ECD	Electrical Chatter Device
EDG	Emergency Diesel Generator
ELAP	Extended Loss of AC Power
EPRI	Electric Power Research Institute
EPZ	Emergency Planning Zone

ERV	Electromatic Relief Valve
ESEL	Expedited Seismic Equipment List
ESEP	Expedited Seismic Evaluation Program
FEM	Finite Element Model
FIRS	Foundation Input Response Spectra
FLEX	Diverse and FLEXible Coping Strategies
F&O	Fact and Observation
FP	Fire Protection
FPS	Fire Protection System
FPIE	Full Power Internal Events
FSAR	Final Safety Analysis Report
FV	Fussell-Vesely
GERS	Generic Ruggedness Response Spectra
GIP	Generic Implementation Procedure
GMC	Ground Motion Characterization
GMRS	Ground Motion Response Spectra
GTR	General Transient
H2	Hydrogen (H_2)
HCLPF	High-Confidence-of-Low-Probability of Failure
HCTL	Heat Capacity Temperature Limit
HCVS	Hardened Containment Vent System
HDPR	Horizontal Direction Peak Response
HEP	Human Error Probability
HF	High Frequency
HI	Human Interaction
HLR	High Level Requirement
HPCI	High Pressure Coolant Injection
HRA	Human Reliability Analysis
HROI	Hazard Range of Interest
HVAC	Heating, Ventilation and Air Conditioning
Hz	Hertz (unit)
IA	Instrument Air
IC	Isolation Condenser
ICPH	Isolation Condenser Pump House
ID	Identification
IPEEE	Individual Plant Examination for External Events
IPSF	Integrated Performance Shaping Factor
ISLOCA	Interfacing Systems LOCA

ISRS	In-Structure Response Spectrum
LB	Lower Bound
LERF	Large Early Release Frequency
LF	Low Frequency
LLOCA	Large LOCA
LMSM	Lumped Mass Stick Model
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
MAFE	Mean Annual Frequency of Exceedance
MCC	Motor Control Center
MCR	Main Control Room
MCUB	Minimum Cut Upper Bound
MLOCA	Medium LOCA
MOV	Motor Operated Valve
MSA	Mitigating Strategies Assessment
MSL	Mean Sea Level
MYBP	Million Years Before Present
N2	Nitrogen
NEI	Nuclear Energy Institute
NEDO	New Energy and Industrial Technology Development Organization
NGA	Next Generation Attenuation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTTF	Near Term Task Force
OPS	Operations
OSP	Offsite Power
PEER	Pacific Earthquake Engineering Research
PFM	Potential Failure Modes
PGA	Peak Ground Acceleration
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PRT	Peer Review Team
PSHA	Probabilistic Seismic Hazard Analysis
RB	Reactor Building
RBCCW	Reactor Building Closed Cooling Water
RG	Regulatory Guide

RLME	Repeated Large Magnitude Earthquake
RM	Risk Management
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RW	Radwaste
RWCU	Reactor Water Cleanup
SEWS	Seismic Evaluation WorkSheet
SBCS	Standby Coolant System
SBO	Station Blackout
SBODG	Station Blackout Diesel Generator
SCDF	Seismic Core Damage Frequency
SCRAM	Safety Control Rod Axe Man
SEI	Structural Engineering Institute
SEL	Seismic Equipment List
SFP	Spent Fuel Pool
SFR	Seismic Fragility Element Within ASME/ANS PRA Standard
SHA	Seismic Hazard Analysis Element Within ASME/ANS PRA Standard
SHEP	Seismic Human Error Probability
SIET	Seismic Initiating Event Tree
SLC	Standby Liquid Control
SLERF	Seismic Large Early Release Frequency
SLR	Steam Line Rupture
SLOCA	Small LOCA
SOARCA	State of the Art Reactor Consequence Analysis
SORV	Stuck-Open Relief Valve
SoV	Separation of Variables
SPC	Suppression Pool Cooling
SPID	Screening, Prioritization and Implementation Details
SPR	Seismic PRA Modeling Element Within ASME/ANS PRA Standard
SPRA	Seismic Probabilistic Risk Assessment
SQUG	Seismic Qualification Utility Group
SPRAIG	Seismic PRA Implementation Guide
SR	Supporting Requirement
SRT	Seismic Review Team
SRV	Safety Relief Valve
SSC	Structure, System and Component; Seismic Source Characterization
SSHAC	Senior Seismic Hazard Analysis Committee

SSEL	Safe Shutdown Equipment List
SSI	Soil Structure Interaction
STSBO	Short Term Station Blackout
SV	Safety Valve
SW	Service Water
SWEL	Seismic Walkdown Equipment List
TB	Turbine Building
TBCCW	Turbine Building Closed Cooling Water
TBV	Turbine Bypass Valve
UB	Upper Bound
UHRS	Uniform Hazard Response Spectra
UHS	Ultimate Heat Sink (i.e. Normal Heat Sink)
USACE	U S Army Corps of Engineers
USI	Unresolved Safety Issue
V/H	Vertical/Horizontal acceleration ratio
Vs	shear wave velocity
VSR	Generator Excitation Start Relay
WW-DW	Wet Well – Dry Well
ZPA	Zero Period Acceleration
βc	Composite logarithmic standard deviation
βr	Randomness logarithmic standard deviation
βu	Uncertainty logarithmic standard deviation

Appendix A

Summary of SPRA Peer Review and Assessment of PRA Technical Adequacy for Response to NTTF 2.1 Seismic 50.54(f) Letter

A.1. Overview of Peer Review

The DRE PRA was subjected to an independent peer review against the pertinent requirements in Part 5 of the ASME/ANS PRA Standard [4]. The peer review assessment [23], and subsequent disposition of peer review findings, is summarized here. The scope of the review encompassed the set of technical elements and supporting requirements (SR) for the SHA (seismic hazard), SFR (seismic fragilities), and SPR (seismic PRA modeling) elements for seismic CDF and LERF. The peer review therefore addressed the set of SRs identified in Tables 6-4 through 6-6 of the SPID [2].

The information presented here establishes that the SPRA has been peer reviewed by a team with adequate credentials to perform the assessment, establishes that the peer review process followed meets the intent of the peer review characteristics and attributes in Table 16 of RG 1.200 R2 [13] and the requirements in Section 1-6 of the ASME/ANS PRA Standard [4], and presents the significant results of the peer review.

The DRE SPRA peer review was conducted during the week of January 14, 2019 at the Exelon Generation offices in Warrenville, IL. As part of the peer review, a walk-down of portions of DRE Units 2 & 3 was performed on January 15, 2019 by 4 members of the peer review team who have the appropriate walkdown training.

A.2. Summary of the Peer Review Process

The peer review was performed against the requirements in Part 5 (Seismic) of Addenda B of the PRA Standard [4], using the peer review process defined in NEI 12-13 [5]. The review was conducted over a four-day period, with a summary and exit meeting on the morning of the fifth day.

The SPRA peer review process defined in [5] involves an examination by each reviewer of their assigned PRA technical elements against the requirements in the Standard to ensure the robustness of the model relative to all of the requirements.

Implementing the review involves a combination of a broad scope examination of the PRA elements within the scope of the review and a deeper examination of portions of the PRA elements based on what is found during the initial review. The supporting requirements (SRs) provide a structure which, in combination with the peer reviewers' PRA experience, provides the basis for examining the various PRA technical elements. If a reviewer identifies a question or discrepancy, that leads to additional investigation until the issue is resolved or a Fact and Observation (F&O) is written describing the issue and its potential impacts and suggesting possible resolution.

For the review of the SHA, a team of two peer reviewers was assigned, one having lead responsibility. For the review of the SFR, a team of three peer reviewers was assigned, one having lead responsibility. For the review of the SPR, a team of four peer reviewers was assigned, one having lead responsibility. One of the SPR reviewers also served as the entire team's lead. In addition, there was one observer for SFR area as well as observers from the USNRC. The NRC observers did not submit questions to the SPRA team.

For each SR reviewed, the responsible reviewers reached consensus regarding which of the Capability Categories defined in the Standard that the PRA meets for that SR, and the assignment of the Capability Category for each SR was ultimately based on the consensus of the full review team. The Standard also specifies high level requirements (HLR). Consistent with the guidance in the Standard, capability Categories were not assigned to the HLRs, but a qualitative assessment of the applicable HLRs in the context of the PRA technical element summary was made based on the associated SR Capability Categories.

As part of the review team's assessment of capability categories, F&Os are prepared. There are three types of F&Os defined in [5]: Findings, which identify issues that must be addressed in order for an SR (or multiple SRs) to meet Capability Category II; Suggestions, which identify issues that the reviewers have noted as potentially important but not requiring resolution to meet the SRs; and Best Practices, which reflect the reviewers' opinion that a particular aspect of the review exceeds normal industry practice. The focus in this Appendix is on Findings and their disposition relative to this submittal.

A.3. Peer Review Team Qualifications

The members of the peer review team were [23]:

Team Lead

The Team Lead was Mr. Paul Amico of Jensen Hughes. Mr. Amico also served as one of the reviewers of the technical elements associated with SPR. Mr. Amico has 40 years of experience in the performance and management of domestic and international programs related to risk assessments and their application in nuclear power plants. He has been involved with seismic PRA for more than 35 years and is active in development of seismic PRA standards and in performance of seismic PRAs.

SHA

The SHA Lead was Dr. Glenn Rix of Geosyntec. Dr. Rix has over 30 years of experience in geotechnical earthquake engineering and engineering seismology, particularly in the central and eastern US (CEUS), and in seismic hazard assessment and risk mitigation. Dr. Rix was assisted in the hazard review by Dr. Annie Kammerer of Annie Kammerer Consulting. Dr. Kammerer has more than 19 years of experience in integrated seismic hazard and risk evaluations and performance-based risk-informed engineering. She is the lead of the seismic hazard working group for the ASME/ANS external event PRA and a member of the working group for the ANS SHA Standard ANSI/ANS 2.29-2008. She is also the author of the current NRC guidance for performing PSHA.

SFR

The SFR Lead was Mr. Gregory Hardy of Simpson, Gumpertz and Heger (SGH). Mr. Hardy has 37 years of experience in structural mechanics engineering with emphasis on probabilistic risk assessments, earthquake experience data based studies, finite element analysis, seismic margin studies and vibration testing for equipment qualification. Mr. Hardy was assisted by Mr. Wen Tong and Dr. Ram Srinivasan. Mr. Tong has over 41 years of experience in seismic evaluations of structures and equipment and seismic risk assessments. Dr. Srinivasan, an independent consultant currently assisting TVA with the SPRA at Watts Bar, Sequoyah and Browns has 45 years of experience.

SPR

The SPR Lead was Mr. Lawrence Mangan of First Energy Nuclear Operating Company. Mr. Mangan has 10 years of experience in developing and maintaining PRA models for the Perry Nuclear Power Plant. He participated in two previous internal events PRA peer reviews. He also co-authored NUREGs related to reliability modeling of digital control systems for nuclear power plants. Mr. Mangan was assisted by Mr. Thomas John, Dr. Hongbing Jiang as well as by Mr. Paul Amico. Mr. John has 28 years of experience in developing seismic PRAs as well as in related activities. Dr. Jiang has 12 years of experience in development seismic PRAs as well as in related activities.

In addition to the reviewers listed, the team was assisted by several working and non-working observers. Working observers included Mr. Samer El-Bahey of Jensen Hughes, 12 years of experience including 7 years PRA experience. Non-working observers included Mr. Shilp Vasavada, Mr. Keith Tetter and Mr. Wesley Wu from the NRC. Some of the observers from the NRC attended portions of the peer review.

The peer review team members met the peer reviewer independence criteria in NEI 12-13 [5]. None of the peer review team members had any involvement with the DRE elements under review as documented in the peer review report

A.4. Summary of the Peer Review Conclusions

The review team's assessment of the SPRA elements is summarized as follows. Where the review team identified issues, these are captured in peer review findings, for which the dispositions are summarized in the next section of this appendix.

SHA

As required by the PRA standard, the frequency of occurrence of earthquake ground motions at the site was based on a probabilistic seismic hazard analysis (PSHA). The seismic source characterization (SSC) inputs to the PSHA are based on the Central and Eastern U.S. (CEUS) regional SSC model published in NUREG-2115 (i.e., the "CEUS-SSC" model). The ground motion characterization (GMC) inputs to the PSHA are based on an updated CEUS ground motion model published by EPRI [34]. The seismic hazard analysis for the DRE site also accounts for the effects of local site response for those structures, systems, and components that are not founded on hard rock; site response analyses were performed to calculate Ground Motion Response Spectra (GMRS) and Foundation Input Response Spectra (FIRS) at two elevations: 472.5 ft and 515 ft.

For DRE, both the SSC and GMC portions of the PSHA were developed as a result of a Senior Seismic Hazard Analysis Committee, Level 3 methodology (SSHAC, Level 3). In the case of the GMC, a SSHAC level 2 analysis was performed to update a prior Level 3 study. These studies satisfy the requirements of the PRA Standard related to the method of conduct of the PSHA, as well as addressing several individual requirements related to data collection, data evaluation and model development, and quantification of uncertainties supporting HLR-A to HLR-D.

In the implementation of the CEUS-SSC model for the DRE site, all distributed (i.e., background) seismic sources in the CEUS-SSC model were included in the PSHA calculations. By including these seismic sources in the analysis, the contributions of "near-" and "far-field" earthquake sources to ground motions at the DRE site were considered. In addition, an effort was made to identify local seismic sources that may not have been included in the regional model, but none were identified.

The CEUS-SSC described only includes earthquakes through 2008. For developing the PSHA at DRE, the analysts developed an updated seismicity catalog that was quantitatively assessed to ensure that (1) assumptions regarding the distribution of the maximum magnitude are not violated and (2) no new data exists that undermines the rate of seismicity of sources in the CEUS-SSC model important to the seismic hazard at the DRE site. In addition, a separate seismicity catalog of non-tectonic (human-induced) earthquake was compiled and evaluated. It was concluded that an additional hazard analysis was not required for these sources.

The PSHA results are provided over an appropriately wide range of spectral frequencies and annual frequencies of exceedances. Uncertainties on the rock hazard are quantified, analyzed and reported, as required in the standard [4]. The lower-bound magnitude chosen for the analysis is consistent with standard practice. The results include fractile and mean hazard curves, median and mean uniform hazard response spectra, and deaggregation results by magnitude and distance and by seismic source.

The seismic hazard analysis for the DRE site included a site response analysis for structures, systems and components not founded on hard rock. GMRS and FIRS were developed for two profiles corresponding to outcrop elevations of 472.5 ft and 515 ft. As part of the characterization of the site, historical, site-specific compression-wave velocity measurements were used to estimate shear wave velocity profiles used in the site response analysis. The analysis includes the effects of site topography, surficial geologic deposits, and site geotechnical properties on ground motions at the site.

Both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources, ground motion models, and site response analyses. Epistemic uncertainty is represented by six shear wave velocity profiles (including two different depths to hard rock) and two sets of modulus reduction and damping curves. Aleatory variability is represented by 60 random realizations of each profile, including random variations in shear wave velocity and modulus reduction and damping curves. In general, the parameters selected to model each type of uncertainty are consistent with values recommended in the SPID [2]. Correlation between properties is modeled when appropriate.

The later sections of this Appendix provide a summary of the Facts and Observations (F&Os) identified by the Peer Review Team that were classified as Findings. The Appendix also provides a resolution for each of these “findings”.

SFR

As required by the PRA Standard, three principal elements of fragility analysis process are covered by the SFR assessment of DRE SPRA:

1) Seismic Response analysis,
2) Plant Walkdown, and
3) Fragility analysis calculations.

For the Reactor-Turbine (RB-TB) Building complex that houses most of the SEL items, seismic response analyses were performed for two reference earthquake (RE) levels; GMRS and 3xGMRS. The RB-TB concrete structures were determined to be in an "uncracked" condition at the GMRS ground motion input level. At the 3xGMRS level most of the RB-TB structure elements were determined to be "cracked" and thus, stiffness and damping values corresponding to this cracked condition were used in the seismic response analysis. For the SBO building and the Crib House the DRE ground motion UHS corresponding to a mean annual frequency of exceedance (MAFE) of 1E-05/yr was used as the reference earthquake.

New 3D Finite Element Models (FEM) were developed for the RB-TB, SBO Building and Crib House. The models were generated consistent with current industry practice using the guidelines provided in ASCE/SEI 4-16 [22]. Structural responses, primarily In-Structure Response Spectra (ISRS) were generated using best estimate structural and soil properties consistent with the hazard levels. Five sets of time histories for the different hazard levels were used for most of the structural response analyses. In generating the ISRS and other structural response parameters, effects of soil-structure interaction (SSI) were included. The Dresden structures are founded on competent rock (shear wave velocity > 4,000 fps) and thus the effects of SSI were primarily due to the spatial incoherency of the ground motion. Industry accepted Incoherency (Abrahamson) Model for hard rock was used in the SSI analyses.

The procedures in EPRI NP 6041-SL [12] were followed for conducting the seismic walkdowns. The walkdown review credited the existing walkdown findings from the previous seismic programs, e.g., A-46/IPEEE, NTTF 2.3 Seismic, and ESEP. A walk-by was performed for the SEL items that were included in the walkdowns performed as part of these previous programs. Walk-by notes were documented in tabulated format. For those items that were not in the scope of the previous programs, a detailed walkdown review was performed and findings were documented in a project-specific SEWS form. Separate walkdowns were performed for seismic/fire interactions, seismic/internal flooding interactions, and containment performance related components. The walkdown team consisted of minimum of two Seismic Capability Engineers (SCE) along with Probabilistic Risk Assessment (PRA) or system engineers with assistance by plant operators.

Credited operator's pathways for required actions outside of the Main Control Room were walked down for seismic interactions and similar issues. Potential seismically induced flooding and fire sources were identified during the walkdowns and evaluated for their likelihood to impact the SEL items. Credible seismic induced flood and fire sources were identified during the walkdowns for further evaluation. Other potential seismic interactions were also identified during the walkdowns and documented on the SEWS forms.

Potential failure modes of the SSCs were reviewed and documented in the seismic walkdown report [31] and corresponding SEWS. The walkdown review conclusions on the equipment and its anchorage meeting the EPRI NP-6041-SL Table 2-4 [12] screening criteria were the bases of developing the representative seismic fragilities. As-installed configurations collected from the walkdowns were used for developing equipment-specific seismic fragilities for risk-contributing SSCs.

The Seismic Fragility Report [21] summarizes the fragility evaluation results for the failure modes of interest for the Dresden SSCs. The fragility parameters for risk significant SSCs used for the risk quantification were based on the Conservative Deterministic Failure Margin (CDFM) method and/or the Separation of Variables (SoV) method. The fragility analyses were performed in three steps. In the initial step, representative fragilities were developed based on site-specific information related to the SSCs. Fragility values were calculated based on simple methods and scaling from design basis calculations, A-46, ESEP and/or IPEEE evaluations. An initial quantification of the Dresden SPRA model was performed and risk significant SSCs were identified. For those risk significant items from the initial quantification, enhanced fragility calculations were performed based on calculating the HCLPF value using the Conservative Deterministic Failure Margin (CDFM) approach and applying generic uncertainties from the SPID EPRI report to obtain median fragility. A second risk quantification was performed using these enhanced fragilities and resulted in a revised set of dominant risk contributing SSCs. Following this second risk quantification, fragilities for a subset of the dominant risk contributors were further refined to include a more plant specific characterization of the uncertainty portion (Beta U) of the overall fragility variability (Beta R remained generic based on the SPID) and were used in the final risk quantification.

As noted earlier, the RB-TB structural response analysis was performed at two RE levels, GMRS and 3xGMRS. The fragility evaluations were likewise performed at the same two levels, Low capacity components were governed by the GMRS and the high capacity components were governed by the 3xGMRS case, and for intermediate capacity components, a weighted average of the two levels was used. For components located in the remaining DRE structures such as the SBO, Crib House, etc., only one reference earthquake level (corresponding to a MAFE of 1E-05/yr) was used for the seismic response analyses. This MAFE of 1E-05/yr corresponds to a level of earthquake at which only a small portion of the risk (both SCDF and SLERF) is shown to exist. As such, the peer review team recommends the effects of a more realistic reference earthquake be assessed for SSCs (particularly for the dominant risk contributors) in these structures.

In summary, the seismic response analyses used detailed finite element models, multiple time histories specific to the site-specific hazard and used appropriate SSI methods. Seismic walkdowns were performed using the appropriate methods and with appropriately trained seismic capability engineers. The walkdowns focused on the key elements of differential displacements, seismic interaction, anchorage, load path and failure modes used for the fragility analyses. The fragilities generally were increased in detail as the dominant risk contributors were identified as part of successive risk quantifications. As such, the peer review team assessed that the seismic fragility analysis generally meets the applicable Capability Category II supporting requirements of the ASME/ANS RA-Sb Addendum B. The later sections of this Appendix provide a summary of the Facts and Observations (F&Os) identified by the Peer Review Team that were classified as Findings. This Appendix also provides a resolution for each of these "findings".

SPR

As required by the PRA Standard, the logic model appropriately includes seismic initiating events and other failures including seismic-induced unreliability and unavailability failure modes, based on the Full Power Internal Events (FPIE) model, and human errors. The seismic PRA model was developed by modifying the FPIE PRA model to incorporate specific aspects of seismic analysis that are different from the FPIE. The seismic PRA model integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify CDF and LERF.

The screening of SSC fragilities from the model was found to be generally acceptable. However, the screening is not confirmed in the quantification process. Additionally, documentation of the fragility screening refinement process was located in a number of locations; the PRT issued observations to increase the traceability of the process and to ensure that potentially significant risk contributors were not inappropriately screened. A number of sensitivities were performed to understand the impact of the various modeling and screening assumptions. In these aspects, the quantification of the Dresden SPRA is judged to meet the PRA Standard.

A number of deviations from realism were identified in the Dresden Seismic PRA. On an individual basis these deviations are all minor and do not significantly affect the risk profile. However, the cumulative impacts are unclear. Reduction in these conservatisms will improve the capability of the SPRA model for risk-informed applications.

The review team concluded that the DRE seismic PRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify CDF and large early release frequency. The seismic PRA analysis was documented in a manner that facilitates applying and updating the SPRA model. Facts and observations identified as findings and SRs graded as Not Met are discussed in the following section along with a resolution for each.

A.5. Summary of the Assessment of Supporting Requirements and Findings

Table A-1 presents a summary of the SRs graded as Not Met or less than Capability Category II, and the disposition for each. Table A-2 presents summary of the Finding F&Os that have not been closed through an NRC accepted process, and the disposition for each. As indicated in Table A-2, all Finding F&Os have been addressed or dispositioned, along with the one (1) SR graded as Not Met.

Table A-1: Summary of SRs Graded as Not Met or Capability Category I for Supporting Requirements Covered by the DRE SPRA Peer Review

SR	Assessed Capability Category	Associated Finding F&Os	Disposition to Achieve Met or Capability Category II
SHA			
[None]	Not Met	N/A	N/A
[None]	CC-I	N/A	N/A
SFR			
[None]	Not Met	N/A	N/A
[None]	CC-I	N/A	N/A
SPR			
SPR-E3	Not Met	25-11	Associated F&O has been resolved. SR is judged to be met.
[None]	CC-I	N/A	N/A

A.6. Summary of Technical Adequacy of the SPRA for the 50.54(f) Response

The set of supporting requirements from the ASME/ANS PRA Standard [4] that are identified in Tables 6-4 through 6-6 of the SPID [2] define the technical attributes of a PRA model required for a SPRA used to respond to implement the 50.54(f) letter. The conclusions of the peer review discussed above and summarized in this submittal demonstrates that the DRE SPRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2 [13] as clarified in the SPID [2].

The main body of this report provides a description of the SPRA methodology, including:

- Summary of the seismic hazard analysis (Section 3)
- Summary of the structures and fragilities analysis (Section 4)
- Summary of the seismic walkdowns performed (Section 4)
- Summary of the internal events at power PRA model on which the SPRA is based, for CDF and LERF (Section 5)
- Summary of adaptations made in the internal events PRA model to produce the seismic PRA model and bases for the adaptations (Section 5)

Detailed archival information for the SPRA consistent with the listing in Section 4.1 of RG 1.200 Rev. 2 is available if required to facilitate the NRC staff's review of this submittal.

The DRE SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, May 4, 2018 [75] (i.e., the revision date of the DRE FPIE PRA model of record used as the starting basis of the SPRA final logic model). There are no permanent plant changes that have not been reflected in the SPRA model.

A.7. Summary of SPRA Capability Relative to SPID Tables 6-4 through 6-6

The Owners Group performed a full scope peer review of the DRE internal events PRA and internal flooding PRA that forms the basis for the SPRA to determine compliance with ASME PRA Standard, RA-S-2013 [4] and RG 1.200 [13] in the week of October 31 through November 4, 2016 [17]. This review documented findings for all supporting requirements (SRs) which failed to meet at least Capability Category II. All of the internal events and internal flooding PRA peer review findings that may affect the SPRA model have been addressed.

The Owners Group performed a peer review of the DRE SPRA in the week of January 14, 2019 [23]. The results of this peer review are discussed above, including resolution of SRs not assessed by the peer review as meeting Capability Category II, and resolution of peer review findings pertinent to this submittal. The peer review team expressed the opinion that the DRE seismic PRA model is of good quality and integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify CDF and large early release frequency. The general conclusion of the peer review was that the DRE SPRA is judged to be suitable for use for risk-informed applications.

- Table A-1 provides a summary of the disposition of SRs judged by the peer review to be not met, or not meeting Capability Category II.
- Table A-2 provides a summary of the disposition of the open SPRA peer review findings.
- Table A-3 provides an assessment of the expected impact on the results of the DRE SPRA of those SRs and peer review Findings that have not been fully addressed.

Table A-3 Summary of Impact of Not Met SRs		
SR #	**Summary of Issue Not Fully Resolved**	**Impact on SPRA Results**
SPR-E3	The screening criteria utilized in the final quantification of the Dresden SPRA model is based on an Am of 0.8g. However, there is not a clear quantifiable justification presented for screening SSCs above 0.8g. It is unclear what the contribution to risk would be for components at 0.8g.	The fragility level screening criteria for the DRE SPRA was finalized after the peer review and the bases for the determination of the screening level were documented using the final SCDF and SLERF results. This documentation is in Appendix C of the SPRA Fragility Modeling report [50]. A final screening level of 1.0g, PGA HCLPF was selected and used in the final DRE SPRA. The determination of the appropriateness of the 1.0g HCLPF screening level used a quantitative sensitivity study of the final SCDF and SLERF models to demonstrate that a 1.0g HCLPF fragility group modeled directly as SCDF and SLERF would meet the FV < 5E-03 criterion for non-risk significant. Additional SSCs up to the final fragility screening level of HCLPF = 1.0g were explicitly included in the SPRA model or dispositioned as not required to be included. As a result of these changes alone (i.e., when not accounting for the risk impact of various other SPRA fragility and modeling enhancements), the SCDF and SLERF increased slightly due to the inclusion of additional SSC fragility failures. Based on the resolution discussed above and the work performed, Exelon considers this SR to be “met” at CC II.

A.8. Identification of Key Assumptions and Uncertainties Relevant to the SPRA Results.

The PRA Standard [4] includes a number of requirements related to identification and evaluation of the impact of assumptions and sources of uncertainty on the PRA results. NUREG-1855 [24] and EPRI 1016737 [25] provide guidance on assessment of uncertainty for applications of a PRA. As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

- Parametric uncertainty was addressed as part of the DRE SPRA model quantification (see Section 5 of this submittal).
- Modeling uncertainties are considered in both the base internal events PRA and the SPRA. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the DRE SPRA technical elements are noted in the SPRA documentation that was subject to peer review, and a summary of important modeling assumptions is included in Section 5.
- Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application. No specific issues of PRA completeness were identified in the SPRA peer review.

A summary of potentially important sources of uncertainty in the DRE SPRA is listed in Table A-4.

Table A-4 Summary of Potentially Important Sources of Uncertainty

PRA Element	Summary of Treatment of Sources of Uncertainty per Peer Review	Potential Impact on SPRA Results
Seismic Hazard	The DRE Seismic PRA peer review team noted that both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources. In addition, uncertainties in each step of the hazard analysis were propagated and displayed in the final quantification of hazard estimates for the DRE site.	The seismic hazard reasonably reflects sources of uncertainty.
Seismic Fragilities	The DRE SPRA peer review team had no issues with SFR related sources of uncertainty treatment.	As discussed in Section 5.7, sensitivity studies are identified and quantified related to a number of analysis areas. Some of the

Table A-4 Summary of Potentially Important Sources of Uncertainty		
PRA Element	**Summary of Treatment of Sources of Uncertainty per Peer Review**	**Potential Impact on SPRA Results**
		sensitivity studies are related to SSC fragility topics. Section 5.7 discusses the sensitivity case results.
Seismic PRA Model	The DRE SPRA peer review team had no issues with SPRA sources of uncertainty treatment and noted that the sources of uncertainty are discussed in Appendix I of the SPRA Quantification report. Appendix I of the SPRA Quantification report considers the various technical aspects of the SPRA development to identify key modeling uncertainties to investigate with sensitivity studies.	As discussed in Section 5.7, sensitivity studies are identified and quantified related to the following analysis areas: • PSHA • Level 1 and Level 2 Accident Sequence analysis • SSC Fragilities • Seismic HRA • SPRA quantification approaches Section 5.7 discusses the sensitivity case results.

A.9. Identification of Plant Changes Not Reflected in the SPRA

The DRE SPRA reflects the as-built and as-operated plant as of the cutoff date for the SPRA, May 4, 2018 [75] (i.e., the revision date of the DRE FPIE PRA model of record used as the starting basis of the SPRA final logic model). All modifications to the plant prior to the cutoff date that have an impact on the seismic PRA model have been included in the model. This includes implementation of FLEX and the hardened containment vent system (HCVS).

No permanent changes to the plant following the cutoff date with a potential significant impact on the SPRA results have been excluded from the analysis. Various revisions to procedures that became available after the cutoff date were used in final seismic HRA work for FLEX and HCVS HEPs.

Table A-2: Summary of Finding F&Os and Disposition Status

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-B9	19-2	Only a single internal flood event has been added to the model, despite the fact that unscreened flood scenarios could occur in various locations in the plant.	Combining all flood scenarios into a single basic event, even if it leads directly to core damage, can undercount the total contribution from internal flood (it generates a single cutset for all floods rather than a cutsets for each flood). This is essentially equivalent to screening out seismic failures without a defensible screening criterion. Furthermore, the piping may not be the weak point for each scenario; there is also the need to consider heat exchangers, etc., and to address all the ways the fluid could be released from the system due to a seismic failure (all the fluid boundary equipment failures)	Provide a disposition of all the internal flood scenarios, while considering boundaries such as heat exchangers. Alternatively, use of walkdown information and subsequent analysis information that shows that the system that is the source in each area was identified, walked it down, and screened everything except the piping, to justify that piping is the weak point. For any flood scenario that remains unscreened, add a basic event to the Seismic model. It is acceptable to assume each flood goes directly to core damage if they are not risk-significant, but in the case that one or more are they should be modeled realistically considering the actual flood impacts.	Updated seismic flood sections of Walkdown Report [31] (e.g., Section 3.6.1 and 4.1.5, Appendix C2) to clarify that the piping is the weak point for all seismic induced internal flood scenarios identified for fragility consideration. Consistent with the concept of fragility screening level and the suggested resolution of the F&O, all postulated seismic induced flood scenarios with fragilities below the 1.0g HCLPF fragility screening level are explicitly included in the SPRA logic model (those with fragilities estimated above 1.0g HCLPF are not included in the SPRA). All the seismic-induced flood scenarios with fragilities below the fragility screening level (and included in the SPRA logic model) are piping systems equipped with

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					Victaulic couplings that represent the weak point in the piping. Ten such seismic-induced internal flooding scenarios were included in the SPRA (i.e., fragility group IDs DP1 through DP10 with HCLPF=0.54g based on Victaulic Coupling fragility). Seismic fragility mapping of internal flooding consequences into the SPRA model was performed and varied depending on the location of the Victaulic couplings (e.g., RB, TB, Crib House). Seismic fragility groups DP1 through DP4 are modeled as leading directly to core damage because the Victaulic couplings are located in the RB and subsequent flooding is assumed to fail all ECCS equipment in the RB basement; this is conservative (i.e., termination of the flood or use of other mitigation alternatives is not credited) but each of these RB floods

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					are non-risk significant. The other seismic-induced floods in the TB and Crib House are mapped to fail equipment that would be inundated based on information from the internal flooding analysis. SPRA Methods Notebook [46] and SPRA Fragility Modeling Notebook [50] were updated to clarify methodology and to discuss the incorporation of the additional seismic induced flood scenarios into the SPRA model.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-A1	19-3	Although the documents state that a systematic process was used to identify the potential seismically induced flood sources, the peer review team cannot determine that this is true. No reasoning is provided as to why most of the flood scenarios from the internal flood PRA are screened other than that the ones from the internal flood model are screened because a distributed piping surrogate is used. In the end, all that is given is the final answer of what flood sources are considered credible.	Since there is no way to reproduce the final list of flood sources based on the discussion of the process used, it could be that viable and potentially risk-significant flood sources have been screened inappropriately. Also, there is no identification of the actual seismic failures that could lead to the floods identified. It was noted in responses to questions that walkdowns that evaluated the potential for floods were conducted, and those walkdowns appear to have been very thorough, but it is not possible in the short time available to relate the walkdown results to the conclusion that the probability of the flood scenarios can be adequately represented by a generic piping fragility.	Perform a systematic review of the internal flood sources/scenarios to assure that they have been appropriately screened from the PRA. Specify and justify the screening criteria used. The following is one possible approach. 1) For each of the internal floods identified in Table 3-1 of DR-PRA-020.003 and each system identified in Section 3.4 of DR-PRA-020.005 (it is noted there is likely overlap between the two), review the drawings to identify possible seismic failure that could lead to the flood. 2) Review the walkdown sheets to assure that all the areas associated with these floods were walked down and that they support a conclusion that the 'weak point' for each flood scenario is the piping and that a generic fragility is appropriate. As needed (if this is not conclusive), conduct walkdowns to assess the ruggedness/capacity	This F&O is on the same topic as F&O 19-2 (i.e., provide additional details on the seismic-induced internal flood disposition process and incorporate seismic-induced internal floods into the SPRA consistent with the final fragility screening level). F&Os 19-2 and 19-3 are both addressed. Consistent with the F&O Suggested Resolution, an updated disposition summary table of postulated seismic-induced internal flood sources was documented containing the requested disposition information (applicable walkdown sheets, applicable fragility calculation, consideration of fragility weak points, and whether scenario is included in the SPRA consistent per the fragility screening level). The updated seismic-induced internal flooding disposition summary table is included in the SPRA

SR	F&O	Description	Basis	Suggested Resolution	Disposition
				associates with these failures to identify the 'weak point' for the associated flood scenario. 3) Use appropriate screening criteria to determine if the flood can be screened from the model. This could be determination that the 'weak point' is sufficiently rugged, that the impact of the flood on SPRA equipment is minimal, that the maximum CDF/LERF associated with the flood is not significant based on an estimate of the 'weak point' fragility, or other such criteria. Note that in the end this MAY turn out to be simply a documentation issue, but the peer review team cannot reach that conclusion at this time. Per the requirement of SPR A-5 and B-9, unscreened flood scenarios should be added to the SPRA model.	Methods Notebook [46]. The modeling of the seismic-induced flooding scenarios in the SPRA is discussed in the SPRA Event Tree Notebook [48] from a sequence progression perspective and in the SPRA Fragility Modeling Notebook [50] from the perspective of fragility mapping. Updated seismic flood sections of Walkdown Report [31] (e.g., Sections 3.6.1 and 4.1.5, Appendix C2) to clarify that the piping containing Victaulic couplings is the weak point for all seismic induced internal flood scenarios identified for fragility consideration.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-A2	19-4	The initiating event 'reactor trip' (no LOOP) is not included explicitly in the SPRA logic model even though it leads to a plant shutdown.	A quantitative sensitivity analysis is documented that estimates the realistic contribution from seismic-induced transients without loop of offsite power (LOOP) to be <1% of SCDF. Past and current SPRAs often do not explicitly incorporate seismic-transients into the SPRA logic model due to low risk significance. The EPRI SPRA Implementation Guide also discusses this point. However, it may be useful for future information or risk applications to add seismic-transient sequences explicitly into the SPRA.	Incorporate the contribution of non-LOOP sequences into the SPRA model. It is not necessary to perform extensive fragility analysis of the power conversion system - this can still be assumed to fail since there are many reasons why this could happen even in a small earthquake. Performing limited estimation of fragility of some BOP systems (e.g., representative fragility for plant air) could be considered so that these could be credited for a more realistic estimate of the potential contribution of non-LOOP sequences, but this is NOT required in order to resolve the finding. Assuming that all BOP systems still fail in the event of LOOP success is still an acceptable resolution.	Seismic induced transient scenarios (e.g., non-LOOP) were explicitly added to the SPRA model. Seismic induced transient scenarios are modeled by transferring SIET sequence SIET-001 to the GTR (i.e., General Transient) event tree to propagate quantification in the SPRA model. Limited estimation of fragility of some BOP systems (i.e., fragilities for IA and TBCCW modeled with fragility groups S-IAS and S-TBCCW, respectively) were added into the SPRA model to support the modeling of seismic induced transient scenarios [48; 50].
SFR-B1	19-5	The screening reported in Table 6 of the fragility report contains the results from the screening that was conducted based on the seismic fragilities developed. The basis for that screening was not documented in a clear manner that could be followed by the peer	Some of the SSCs that were screened in Table 6 of the fragility guide make sense to the peer reviewers (e.g. valves that are not required to change state). However, other SSCs that were screened were based on judgments and should be supported by justification. Examples of SSCs that were	Establish a screening level commensurate with the risk results obtained. Develop justification/documentation for the screening of the SSCs in Table 6 of the fragility guide. Develop a bounding type seismic capacity/fragility argument/calculation for the less obviously rugged SSCs	Further developed the justifications of the rugged components and documented these justifications in Table 10 of Fragility Analysis Report [21]. Note that Table 6 of the fragility report was changed to Table 10 in later revisions.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		reviewers. Some of the screened SSCs were not supported by the experience of the peer review team.	screened for the Dresden based on judgment that the peer review team feels that a more specific justification for that screening would be required include the FLEX equipment (diesel generator, tied down equipment, etc.), buried fire water piping, buried piping and drywell torus.	(such as those listed in the basis section) being screened to justify the screening.	For the components initially judged to be rugged but not supported by the experience of the peer review team (e.g., LPCI heat exchangers), fragilities were developed. Developed bounding type calculations for the less obviously rugged SSCs and added to SEWS to justify the ruggedness judgement. FLEX components were walked down, and fragilities were developed for the FLEX equipment [21]. As discussed in Appendix C of the SPRA Fragility Modeling report [50], a screening level of 1.0g was established based on a review of the results of the PRA quantification. SSCs with a fragility below this screening level were added to the SPRA model as discussed in [50].

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-B9	19-6	Seismic fire sources requiring review and assessment that were identified during the review of plant SSCs and the walkdown. These are listed in Table 4-1 of DR-PRA-020.005 (there are approximately 40). Ultimately, these were assessed, and it was determined that they could be screened. However, the basis for the screening of each is not identified.	The peer review team cannot determine if the screening is appropriate, and so cannot determine if significant contributors have been eliminated from the model. The PRA team, in answer to a question, provided the screening criteria used, but did not indicate which was applied to each source or how to find the basis for application.	Review the approximately 40 SSCs listed in Table 4-1 that are identified as potential seismic-fire sources and provide the specific basis for screening. In the case where a calculation was used, provide a cross-reference to the location. In the case where a walkdown determination was used, identify the SEWS where the assessment can be found. Note that this may be affected by the resolution of Finding 19-5. Note also that, ultimately, this MAY turn out to be a documentation-only finding, but the peer review team cannot reach this conclusion at this time.	Consistent with the F&O Suggested Resolution, an updated disposition summary table of postulated seismic-induced internal fires was documented containing the requested disposition information (applicable walkdown sheets, applicable fragility calculation and whether scenario is included in the SPRA consistent per the fragility screening level). The updated seismic-induced internal fire disposition summary table is included in the SPRA Methods Notebook [46]. All postulated seismic-induced fires were dispositioned as not requiring explicit modeling in the SPRA logic model, based on convolution of the calculated fragilities with the Dresden hazard curve and use of conditional ignition probabilities from EPRI 3002012980 [76] and then assuming that the

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					postulated ignition directly results in a core damage event. Updated seismic fire sections of Walkdown Report [31] (e.g., Sections 3.6.2 and 4.1.6, Appendix C3) to clarify the seismic fire walkdown information to support the screening process.
SPR-B1	19-8	Most significant HFEs were analyzed using detailed HRA, but a number of them that were revealed as significant late in the quantification analysis were not analyzed in detail due to a lack of time.	Back-reference SR HR-G1 requires that in order to meet CC-II it is necessary to 'PERFORM detailed analyses for the estimation of HEPs for significant HFEs. USE screening values for HEPs for non-significant human failure basic events.' Since there are admittedly some significant HFEs that did not receive detailed analysis, it is necessary to issue an F&O.	Complete the process of performing detailed HRA on significant HFEs.	After resolving a number of peer review F&Os for fragility and SPRA modeling enhancements, the results of an interim quantification (i.e., Quantification #4) identified five (5) risk-significant operator actions that needed detailed HRA. An additional six (6) FLEX related operator actions were not risk significant, but detailed HRA was performed for completeness. Following implementation of the detailed HRA for the additional total of 11 operator actions and resolution of additional peer review F&Os, another

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					interim quantification (i.e., Quantification #5) identified one (1) additional risk-significant operator action (i.e., Initiate Containment Heat Removal During ATWS Event) that needed detailed HRA. Following implementation of the detailed HRA for the one (1) additional operator action, the Final SPRA results (i.e., Quantification #6) support that all risk significant operator actions are based on detailed seismic HRA. The Seismic PRA HRA Notebook [49] was revised to document the additional detailed HRA for the 12 operator actions.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-A2	19-10	The SPRA includes earthquakes from the OBE at 0.10g to an acceleration level up to ~1.1 g. The basis to stop at 1.0 g is that the conditional probability of core damage begins to approach 1.0 range.	The top interval contributes 16% to the total and stopping at this point could change the risk profile and importance.	As a minimum, perform a sensitivity study using more bins at the top end to show whether there is a significant change in the risk profile.	Sensitivity Case 7c was performed to extend the %G8 interval from a single interval for >1.0g to add five additional hazard intervals ranging from 1.0g to 1.5g with width 0.1g, and one final open-ended interval for ground motions greater than 1.5g. For the Base Case SPRA model, the %G8 interval contributes to 16% of SCDF and 32% of SLERF. Given that the Base Case models the %G8 interval as leading directly to SCDF and SLERF, the %G8 interval is not explicitly included in the Base Case SCDF and SLERF FV calculations. Sensitivity Case 7c supports that nearly all of the SLERF SSC and operator action FV importance measures decreased by approximately 30% when extending the %G8 interval because nearly all of the individual %G8 FV importance measures have small or negligible FV

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					contribution for the additional hazard intervals. The "RPV Internals" SLERF FV increased from 0.65 to 0.67 (i.e., approx. 3% increase) because of the high contribution of Failure to SCRAM scenarios to the extended %G8 results. The OSP SLERF FV remained at 0.999 for both the Base Case and Sensitivity Case 7c. All other SLERF FV values decreased by approximately 30%, which is proportional to the contribution of explicitly accounting for the %G8 in the SLERF FV calculations. Therefore, the risk profile essentially remains the same between the Base Case and Sensitivity Case 7c. Sensitivity Case 7c was performed to evaluate the impact on the SLERF risk profile and not SCDF because the conclusions for the impact on SCDF are estimated to be the same. The Seismic PRA Quantification Notebook

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					[52] was updated to document the assumptions and results of Sensitivity Case 7c.
SPR-C1 SPR-E6	19-11	For determination of the effects of postulated seismic-induced failure of MCR panels the DRE SPRA consulted effects modeled in the DRE Fire PRA for MCR cabinet fires. The DRE SPRA investigated these FPRA-based effects and removed some of these effects as being conservative (as described in Appendix A of the Fragility modeling notebook, DR-PRA-020.005, Vol. 1). However, based on review of the SPRA fragility impacts and discussion with the SPRA team conservatisms still exist in the MCR control panel	Control cabinet effects modeled in FPRA are not in all cases applicable to seismic-induced failures (e.g., FPRA effects involve potential hot shorts whereas SPRA effects are often assumed as open-circuits). One of the conservative fragility mapping effects in the DRE SPRA for the MCR panels is to fail containment isolation signal for a number of the cabinets (for example, fragility group S-INCP04-). Based on investigation of the model and discussion with the SPRA team this mapping inappropriately contributes a significant contribution (on the order of ~5%, estimate based on	Remove the fragility mappings from the MCR cabinet failure scenarios that are inappropriate and overly conservative for the SPRA. The Dresden team provided the Peer Review team with TPJ-12-R in response to a Peer Review question, which included 'Table X' containing a detailed review and disposition of the effects of a seismic failure to the risk-significant panels. This review did note some conservatisms in the current PRA modeling. Updating the PRA model to incorporate the identified revisions, and including Table X in the formal PRA documentation will address this F&O.	Updated SPRA model to remove the fragility mappings from the MCR cabinet failure scenarios that are inappropriate and overly conservative for the SPRA (e.g., failure of Containment Isolation signals in FRANX scenario mapping). SPRA Fragility Modeling Notebook [50] was updated to incorporate the results of a detailed review and disposition of the effects of a seismic failure to the risk-significant panels during the Dresden SPRA peer review.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		fragility mapping used in the DRE SPRA.	judgment and discussions) to the calculated SLERF.		
SFR-C1	19-13	The ASME PRA SFR-C1 states, 'ESTIMATE the seismic responses that the components experience at their failure levels...' This implies that the structural response analysis for the SPRA should be performed at the hazard range that dominates the contribution to risk (SCDF and SLERF). The SPRA team used different Reference Earthquakes (RE) for different buildings. Two RE levels were used for the RB-TB, GMRS and 3xGMRS. For the SBO Building and Crib House, the RE corresponded to the ground motion corresponding to a Mean Annual Frequency of Exceedance (MAFE) of 1E-05/yr.	It is seen in Tables 6-2-1 of PRA Quantification Notebook (DR-PRA-020.006) that the cumulative SCDF is 42% for bin G05 (0.5g to 0.6g) and 71% for bin G06 (0.6g to 0.8g). Thus, the use of RE corresponding to 3xGMRS (PGA of 0.588g) would appear to be appropriate for SSCs within RB-TB. However, it is not evident that for SSCs housed in the SBO Building and Crib House that used a RE corresponding to MAFE of 1E-05/yr. (PGA of 0.37g) is appropriate.	Additional justification through sensitivity analyses or alternate approaches is required in cases where the fragility of risk significant SSCs was derived based on RE of MAFE of 1E-05/yr.	Based on the results of third quantification and Exelon Dresden SPRA Peer-Review, the reference earthquake was updated and now corresponds to 3xGMRS. Most of the top risk contributor components for Dresden SPRA are housed in the RB-TB complex. ENERCON revised the refined fragilities for all components housed in RB-TB complex to be based on the 3xGMRS seismic input and anchored to corresponding PGA of 0.548g [21]. Based on the results of the third risk quantification, there are only a few risk significant components housed in the SBO Building. These components have low seismic fragilities and fail at relatively lower

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					seismic demands such that the use of 3xGMRS as seismic input is not realistic. Therefore, the components housed in the SBO Building are evaluated based on the same seismic input as was used in the fragility evaluations before the peer review, which corresponds to the 1E-05 Hazard Level [21]. The only components housed in the Crib House are the vertical and horizontal pumps and these pumps are not risk significant based on the results of the third risk quantification. Therefore, the components housed in the Crib House are evaluated based on the same seismic input of 1E-05 hazard level FIRS1 as was used in fragility evaluations before the peer review. This approach is consistent with the requirement of the ASME Standard [4] that items be evaluated at

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					seismic levels corresponding to their fragility level [21].
SFR-F1	19-14	Appendix C24 of the Seismic Fragility report contains the fragility derivation for the three larger LOCAs (SLOCA, MLOCA & LLOCA). The fragility is based on generating of the LOCA HCLPF based on a build up of margin existing in the design basis criteria used for piping systems at Dresden. The basis for the factors used in this build up were not adequately justified in the calculation and the peer review team determined some factors were not valid for this particular CDFM calculation.	Generic fragilities for the three LOCA fragilities are documented in the EPRI SPRAIG. These SPRAIG LOCA fragilities are generally felt to be conservative if seismic interactions (falling, differential displacements, etc.) have been precluded based on a walkdown. The Dresden SPRA team stated they have done that review. Current industry practice is to develop more realistic LOCA fragilities if the SPRAIG fragilities contribute appreciably to the seismic risk. The piping supports typically govern the fragility for piping, thus the emphasis of the fragility should be on the supports themselves.	The recommended resolution to this finding would be to develop fragilities or HCLPFs for a bounding sample of piping specific to Dresden. This approach has been taken at several recent SPRAs and EPRI 6041 contains some good description of methods for piping. The alternative approach would be to develop the design basis margin calculation along the lines of the one in Appendix C24, but with a more detailed and defendable CDFM calculation.	More refined design basis margins based on plant specific design standards were developed and Appendix C24 of the DRE Fragility Analysis Report [21] was updated to derive more realistic fragilities. Additional references and bases are documented in Appendix C24 of the DRE Fragility Analysis Report [21].
SFR-C1 SHA-G1	21-1	SFR-C1 states, 'ESTIMATE the seismic responses that the components experience at their failure levels using input earthquake response spectra in three orthogonal directions, anchored to a	Report EXDR025-REPT-001 (Fugro report 160034-PR-01) provides the results of the SHA for use in the subsequent fragility and risk model evaluations. These results include horizontal response	Develop V/H ratios for use with higher ground motion levels used to understand the magnitude of the issue. Evaluate the potential impact to SSI and fragility evaluations, either by performing sensitivity	New V/H ratios consistent with the 1E-05 UHRS and 3xGMRS hazard levels were developed in addition to the already existing V/H ratios consistent with the GMRS. The new V/H ratios

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		ground motion parameter such as peak ground acceleration or average spectral acceleration over a given frequency band, and ENSURE that the spectral shape used bounds the site-specific conditions.' However, the spectral accelerations in the vertical direction are not bound by site-specific conditions in some evaluations performed.	spectra at a variety of MAFE for two control point elevations. There is only one V/H ratio provided in the report, although the V/H changes with magnitude, distance, and PGA. The SSI evaluations and other subsequent evaluations variously used as input hazard results the GMRS, the 10-5 ground motions, and the 3xGMRS horizontal ground motions with the V/H ratio provided in the report. However, the V/H ratio used is only applicable for the GMRS ground motion level and associated deaggregation results. Based on an evaluation performed using information in the PSHA report, the V/H ratios for the 10-5 and 3xGMRS ground motions would be higher than the V/H for GMRS.	analyses or changes to the previous structural response and fragility analyses, as appropriate.	and the corresponding updated FIRS2 ground response spectra at 1E-05 hazard level and 3xGMRS level hazard are provided in the DRE PSHA Report [6]. The building response analyses of RB-TB and SBO Building were updated for the new V/H ratios and documented in [58], which is an update to [14], and in [15]. The refined fragility evaluations for all components housed in the RB-TB and SBO buildings were updated for the new V/H ratios [21]. None of the components in the Crib House are risk significant so fragilities for components in the Crib House were not updated to reflect the changes in V/H ratio.
SHA-I1	21-2	SHA-I1 says, 'DOCUMENT the bases and methodology used for any screening out of the seismic hazards other than	An evaluation of soil failure and slope stability was provided in the official documentation (report EXDR025-REPT-008)	An appropriate process should start with developing/brainstorming a comprehensive list. The	Section 6.1 of ENERCON report [33] was modified to include a discussion on the following secondary

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		vibratory ground motion.' The purpose of SHA-I1 is to ensure that a screening analysis is performed using a structured approach to ensure that all possible secondary (or 'other' seismic-related) hazards are identified and appropriately dispositioned, either by being screened into further evaluation or by being screened out with a documented technical basis.	and generally provided sufficient technical bases to disposition the hazards addressed in the report (liquefaction-induced loss of strength and settlement and slope instability). Separately, an evaluation of available literature was performed to determine if any new local faulting had been identified. However, there is no evidence in the documentation of a structured (or even ad hoc) effort to compile (consider) the broader list of secondary hazards and perform a screening evaluation. There is no information to document the basis for the screening out of hazards. Although the FSAR is a valid source to inform a screening evaluation, the decisions and technical bases in the FSAR should have been reviewed to determine whether they are still valid and if any analysis methods used are still consistent with the state-of-practice.	disposition of hazards could (in some cases) be as simple as reviewing the information in the FSAR to determine that the basis provided still holds and any analysis methods used are still valid in current practice. The continued validity of the technical basis in the FSAR should be discussed, if relied upon. In some cases, evaluations using up-to-date analysis methods should be (or have been) performed. Generally something similar to the response to AMK-02 (with dry settlement added) is sufficient to start to address this finding. The resulting table/memo/report update must be incorporated into the official documentation in some way.	hazards in a tabular format (Table 6.1-1 of [33]): - Soil consolidation and differential settlement - Loss of bearing capacity - Dry settlement - Fault displacement - Tsunami / seiche
SHA-J2	21-4	The noted documentation issues have been identified.	1. Section 4.2 of report EXDR025-REPT-001 (Fugro report 160034-PR-01) states,	1. Correct the report to accurately note PEER as the contractor, not the sponsor.	1. The PSHA Report [6] was corrected to state that

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		These issues should be addressed.	'One project is the Next Generation Attenuation (NGA) models, sponsored by Pacific Earthquake Engineering Research (PEER) Center, which developed ground motion prediction relations for Central and Eastern North America (CENA) through a comprehensive and interactive research program.' This is incorrect. The project was conducted by PEER and was sponsored by the NRC, DOE, and EPRI. 2. Section 6.1 of report EXDR025-REPT-001 (Fugro report 160034-PR-01) states, 'The review concluded that insufficient shear wave velocity data was present at the project site to develop defensible dynamic profiles and that there is significant risk in using the current shear wave velocity interpretation in the absence of deeper shear wave velocity profiles and more robust shallower shear wave velocity profiles.' Some of the uncertainties resulting from the data limitations are addressed	2. Provide some evaluation of sensitivity information based on the quantification of various branches of the logic tree. 3. Revise Section 7.3 to more clearly explain that the expressions based on the SPID were not used to calculate the soil hazard curves. Alternatively, consider deleting these expressions altogether and modify Figures 7-1 and 7-2 so they do not depend on these expressions. 4. Add a figure (and associated description in the text) to compare the UHRS for Profiles P1 through P3 and P4 through P6 as done in the response to Question GJR-04. Alternatively, compare the soil amplification factors for Profiles P1 through P3 and P4 through P6 in Section 7.3 where other sources of epistemic uncertainty (i.e., sensitivity analyses) are presented.	PEER is the contractor and not the sponsor. 2. See response to No. 4 below. 3. The paragraph in Section 7.3 was re-written to explain that the expressions in the SPID were not used to calculate the soil hazard curves but were used in the report solely to develop the figures presented to provide an overall understanding of what amplification functions will look like when combined. However, the logic tree was applied with each MxPy branch implemented separately. 4. The following text was added to the bottom of Section 8.2: "To illustrate the effect of epistemic uncertainty on the depth to hard rock in the site response, the 10-4 UHRS for the M1Py profiles for GMRS/FIRS1 is presented on Figure 8-42. The blue

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			in the logic tree (for example, the deep shear wave velocity profile); However, even though the sources of uncertainty were calculated in the logic tree, an evaluation of the issues through a review of the alternative tree branches is not presented or discussed. 3. The expressions presented in Section 7.3 for the mean and standard deviation of the site amplification factors (based on the SPID) are used only for presenting the site amplification factors in Figures 7-1 and 7-2. Soil hazard curves are subsequently developed by (correctly) maintaining the separation of epistemic and aleatory uncertainties. However, the discussion in Section 7.3 is not sufficiently clear about this distinction. 4. A key source of epistemic uncertainty in the site response is the depth to hard rock. Profiles P1 through P3 use		line is the average of the 10-4 UHRS's for P1 through P3 and the dark red [line] is the average of the 10-4 UHRS's for P4 through P6 (deeper profiles). It is key/important to note that in the actual soil PSHA, the UHRS is developed from the logic tree applied to the individual MxPy soil hazard curves (i.e., weights applied to hazard curves and not to the UHRS)."

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			1,000 +- 300 ft, and profiles P4 through P6 use 5,000 +- 1,500 ft. Although seismic hazard curves for each profile are provided in Figures 8-21 through 8-34, it would be helpful to compare the UHRS for profiles P1 through P3 and P4 though P6 to evaluate the effects of the epistemic uncertainty in depth to hard rock on spectral shapes.		
SFR-G2	21-6	Several areas were identified throughout the fragility analysis documentation which required corrections or additional clarifications.	The following items require corrections/additional clarifications: 1. (EXDR025-REPT-004) Include EXDR025-REPT-001 as the basis for the time history developed for FIRS1 at 1E-05 Hazard Level in the spreadsheet provided by Enercon. 2. (EXDR025-REPT-004) Clarify that the 'live load' of 300 psf used accounts for the equipment mass present during normal operation. Use appropriate terminology to characterize the load, rather than referring to it as 'live load.'	Incorporate the proposed corrections	Incorporate as proposed. 1. Updated Crib House Response Analysis Report [16] to include PSHA Report [6] as the basis for the time history FIRS1 at 1E-05 Hazard Level. 2. Updated Crib House Response Analysis Report [16] to clarify the live load of 300 psf used in the crib house response analysis. 3. Updated Fragility Analysis Report Appendix C15 [21] to fix the typo.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			3. EXDR025-REPT-005 Page C15-04 states 'The fragility parameters associated with the anchorage and functional failure are referenced to the Peak Ground Acceleration (PGA) of 0.183g corresponding to the GMRS (Ref. 3.5).' 0.183g (corresponding to PGA of GMRS/FIRS1) should be revised as '0.385g' (corresponding to PGA of 1E-05 FIRS1) 4. EXDR025-REPT-005 Appendix C15 Page C15-10 states that 'The HCLPF capacity is reported in terms of 5% damped Peak Ground Acceleration (PGA) of the Reference Earthquake i.e. plant Ground Motion Response Spectra (GMRS) at 1E-05 Hazard Level. A GMRS PGA of 0.385g is considered for the seismic fragility evaluation of the components in the SBO building.' The 'GMRS at 1E-05' should be revised as '1E-05 FIRS1'. 5. EXDR025-REPT-006 Walkdown notes for valves states 'vulnerability in all three directions'. It is not clear what		4. Updated Fragility Analysis Report Appendix C15 [21] to fix the typo. 5. Updated Walkdown Report [31] to clarify the definition of the term ‘vulnerability’ used in the SEWS. This walkdown note is intended for ‘seismic vulnerability’.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			is meant by that. Need to clarify what type of vulnerability.		
SFR-G1	21-7	Several areas were identified throughout the fragility analysis with insufficient documentation.	Documentation of the fragility analyses need to be improved as noted below to facilitate peer reviews, PRA applications, and upgrades: 1. Bases (not just judgment) for the assumed fundamental frequency of SSCs. 2. Screening out of Unit 1 TB should include all potential failure modes, including failure of the TB superstructure and the consequence on Unit 2 SSCs. 3. Additional bases for the selection of the assumed ground motion input of 1.5xGMRS for the Radwaste Building supporting the above ground SW piping 4. Additional documentation is required demonstrating that the GERS caveats were met for the chatter sensitive relays	SFR Documentation needs to be supplemented with the proposed items noted in the 'Basis'.	1. Fragility calculations included in Appendix B and Appendix C of fragility analysis report [21] were updated to provide the basis of the fundamental frequency of the SSCs. 2. The basis of the screening of Unit 1 TB was prepared and documented in SC Solutions' letter dated 9/11/2019 [81]. The potential failure modes reviewed includes (1) Building pounding, (2) Structural failure of Unit 1 TB that could impact Unit 2/3 TB, including control room, and (3) Failure of Unit 1 TB steel superstructure that could impact Unit 2/3. 3. For Fragility Analysis Report [21], Attachment C29-1 to Appendix C29 was created which provides further justification for the

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			5. Additional documentation is required to justify the assumption of a lower bound frequency of 10 Hz assumed in the fragility analysis of the A8 Type MOV's		seismic fragility of the Service Water Piping, developed using 1.5 * (3XFIRS1/GMRS or Reference Earthquake) as the seismic demand, and to account for in-structure amplification at the location of the pipe support. 4. Relay calculations (Appendices B26 and C26 of Fragility Analysis Report [21]) were revised to clarify the GERS caveat verification result. 5. Valve calculation (Appendix B8 of Fragility Analysis Report [21]) was revised to provide the justification of the frequency range of interest used in the fragility calculation.
SFR-A2	22-2	SFR A2 requires that the seismic fragilities be realistic (median and uncertainties). The intent of this requirement is to require that the dominant risk contributors should be	The Standard requires the use of realistic fragilities, both the median and the uncertainties. While the term 'realistic' has not been specifically defined, it has been interpreted in many SPRAs to mean that no	Realistic uncertainties for the dominant risk contributors to SCDF and SLERF should be developed. This could be accomplished along the lines of the refined hybrid calculations performed for two of the	New Separation of Variables (SoV) calculations were prepared to develop more realistic fragilities for selected top risk contributors, based on the

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		required to be realistic (median and uncertainties). Representative generic fragilities were developed for the first quantification of the Dresden SPRA. For the important risk contributors, hybrid fragilities were developed for future risk quantifications. These hybrid fragilities incorporated the HCLPFs (which were specifically calculated) and conservative generic fragilities selected from the SPID. Two of the dominant risk contributors were further refined to develop more specific uncertainties for the final risk quantification. The remaining dominant risk contributors should be reevaluated to demonstrate they are realistic.	intentional/known conservatism or unconservatism exists. The peer review team feels this realistic criterion should only be applied to the dominant risk contributors.	dominant risk contributors as part of the work to support the final risk quantification.	review results of the third risk quantification. The detailed discussion of SoV Method is added in Fragility Analysis Report [21].
SFR-F1	22-3	SFR-FI requires that fragilities incorporate all of the appropriate fragility parameters necessary to develop the median capacity as well as the uncertainties. One of the fragility parameters is the Horizontal	Current practice is to include the HDPR factor in the fragility and/or HCLPF calculations.	Review the dominant risk contributors to assess the impact of including the HDPR factors. This could result in an update of those dominant risk contributor fragilities or a sensitivity study to justify that	For the CDFM calculations included in Appendices B and C of the Fragility Analysis Report [21], and new CDFM and SoV calculations in Appendix D of the fragility analysis report, the seismic

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		Direction Peak Response (HDPR) which accounts for the use of the geomean hazard for the horizontal directions of the seismic response. The CDFM calculations for the Dresden SPRA did not specifically incorporate these HDPR factors in the hybrid fragilities.		changes to the fragilities are negligible.	demands used for the anchorage and functional fragility evaluations were adjusted using a Horizontal Direction Peak Response (HDPR) factor to account for the use of same input (Uniform Hazard Response Spectra (UHRS) for the Reference Earthquake (RE)) in both horizontal directions.
SFR-F2	22-4	This SR requires that component seismic-fragility parameters such as median capacity and variabilities be based on plant-specific data or, if necessary, on earthquake experience data, fragility test data, and generic qualification test data. The seismic fragility of MCC 29-7, which is one of the dominant risk contributors, does not meet this requirement.	The following issues were noted during the walkdown and fragility peer reviews of MCC 29-7. -The MCC has modifications to its top by attaching heavy steel cable trays that are heavily loaded. The functional capacity was calculated based on NP-6041 Table 2-4. Appendix F page F-7 of NP-6041 states that Table 2-4 screening is valid if the tributary weight of external attachment is less than about 100 pounds per bay. It is believed that the tributary weight of the cables and the support stands is heavier than 100 lbs per bay.	-In light of the discussion on this topic, justify the applicability of using Table 2-4 of NP-6041 for estimating functional HCLPF capacity of the MCC with the additional weight at the top. Investigate the extent of condition of using NP-6041 for other similar MCC with additional weight at the top. -Update the existing seismic fragility calculation to include the bay that is mounted on a steel base frame. Furthermore, evaluate the potential failure mode of seismic-induced uplift of the concrete base which is glued to the structural floor slab.	Calculations were updated as suggested. - The extent of condition review performed for all MCC calculations (Appendices A1, C1.1 and C1.2 of Fragility Analysis Report [21]). The MCC calculations are updated to include the justification for the caveat. -MCC calculations are updated to consider the effect of the added bay. MCC Calculations are updated to evaluate the bonding stress between concrete pad and floor slab.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			- An additional bay was added to the original MCC which consisted of three bays. The added bay is welded to a steel base frame constructed of steel channel sections. The base frame was not filled with concrete as the other three bays were. However, the frame was anchored to the floor slab with two bolts at one end of the base frame. The current fragility calculation assumes all four bays are founded on a concrete pad. -The failure mode of the concrete pad glued to the floor slab should be evaluated as part of the load path evaluation.		
SFR-F1	22-5	HCLPF capacity of the Crib House was estimated using the generic screening-level seismic capacity of Table 2-3 of EPRI NP-6041-SL. The Crib House meets the caveats of the first column of Table 2-3, hence a generic capacity of 0.8g (5% damped spectral acceleration at ground) was applicable. The HCLPF	Since seismic demand of the Crib House is represented by the DRE GMRS/FIRS1, the HCLPF capacity of the Crib House should be anchored to the GMRS/FIRS1.	The seismic fragility calculation of the Crib House should be updated. Furthermore, other seismic fragility calculations using this similar approach to estimate HCLPF capacity should be reviewed to ensure that an extent of condition does not exist for this situation.	In Appendix A25.4 of Fragility Analysis Report [21], the seismic input for the structural fragility evaluation of the Crib House is the FIRS1/GMRS. Consistent with this input the fragilities are anchored to the GMRS level PGA of the control point (i.e. PGA of GMRS/FIRS1). Therefore,

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		capacity of the Crib House was estimated as the product of the ratio of this generic capacity (i.e., 0.8g) to the 5% damped peak spectral acceleration of the GMRS/FIRS1 and the PGA value of 1E-05 mean UHRS at Elevation 472.5 feet. Use of PGA value of 1E-05 mean UHRS is incorrect.			as recommended by the peer review team, the PGA used for the fragility evaluation for all components in the Crib House was updated to the PGA of GMRS/FIRS1 (0.183g).
SFR-C1	24-1	The ASME PRA SFR-C1 states, 'ESTIMATE the seismic responses that the components experience at their failure levels...' This implies that the structural response analysis for the SPRA should be performed at the hazard range that dominates the contribution to risk (SCDF and SLERF). The fragility team used different approaches for different buildings. For the Reactor/Turbine building, they based the reference earthquake used for the fragility on the HCLPF for the individual SEL components within that structure.	The fragilities for SSCs located in the Reactor/Turbine building are based on using reference earthquakes that range from the GMRS to three times the GRMS. Individual SSCs are assigned responses within that range based on the SSC HCLPF calculated. The use of an SSC HCLPF as a measure of where the PRA risk contribution exists was a concept that the peer review team knows was postulated in the past, but more recent reviews of this concept resulted in the understanding that higher earthquake levels exceeding the HCLPF actually contribute most to the risk. As a result, the resulting response used for	Establish a more realistic basis for determining the appropriate probability of failure level for assessing the reference earthquake to be used for the SSCs in the RB-TB. Assess the impact to any projected changes to the calculated fragilities based on the more realistic reference earthquake assignment.	Based on the results of the third quantification (the quantification reviewed by the peer review team), the reference earthquake corresponds to 3xGMRS. Most of the top risk contributor components at DRE site are housed in RB-TB complex. ENERCON revised the refined fragilities for all of the components housed in RB-TB complex to be based on the 3xGMRS seismic input and anchored to the respective PGA of 0.548g. This included all components important to seismic risk. Additional SoV calculations developed for post-peer review risk

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			these fragilities may not be realistic and unconservative. The background for this finding is documented in question GSH16. The response to this question identified the basis for the use of the HCLPF, and concluded 'We agree that the use of something higher than the HCLPF to determine the seismic input of interest based on the weighted average approach is likely more appropriate, given the current state of knowledge, and we would have used something different if we were implementing this approach at the current time.'		quantifications were also prepared, based on the 3xGMRS seismic input for all components housed in RB-TB complex [21]. Based on the results of the third quantification, there are only a few risk significant components housed in the SBO Building. These components fail at relatively lower fragilities (lower than the PGA of 3xGMRS) and this results in the fact that using 3xGMRS as seismic input is not realistic for these components. Therefore, the components housed in the SBO Building continue to be evaluated based on the same seismic input of 1E-05 hazard level FIRS2 as used in fragility evaluations before the peer review [21]. The only components housed in the Crib House are the vertical and horizontal pumps and these pumps are not risk

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					significant based on the results of third quantification. Therefore, the components housed in the Crib House continue to be evaluated based on the same seismic input of 1E-05 hazard level FIRS1 as used in fragility evaluations before the peer review [21].
SFR-D1	24-3	The potential failure modes of the Unit 2/3 Chimney (SEL Item #5708) were not evaluated for impact on nearby SEL SSCs.	Section 4.7 of the Fragility Report (EXDR025-005, pg. 18/92) states in part: 'It was determined that there is likely no credible failure mode of the off-gas stack that could seal and prevent an adequate containment venting path, and the failure of the stack would have no significant impact on the SPRA model since the hardened containment vent would continue to work as designed and the stack would not impact any structure or component on the SEL due to the distance from the stack to these items. Therefore, it was determined that an evaluation of the stack was not required.' However, it was observed during the January 15	The seismic fragility of the chimney needs to be determined and an assessment made of the failure mode(s) on nearby SSCs.	A new fragility calculation for U1 Chimney and for the U2/3 Chimney (Appendix D23 of Fragility Analysis Report [21]) was developed. This fragility and the consequences of failure were evaluated for inclusion into the SPRA model.

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			walkdown by the PRT that the above-ground service water piping from the Crib House to the power block is in proximity to the Units 2/3 chimney. Furthermore, the Crib House itself is within the zone of influence of the chimney.		
SFR-D1	24-4	The seismic fragility of Dresden Lock and Dam is based on an incomplete consideration and analysis of potential failure modes. As a result, the PRT cannot assess the potential implications of the failure of the dam to plant risk.	Per the USNRC Interim Staff Guidance on Guidance For Assessment of Flooding Hazards Due to Dam Failure (ML13151A153), the seismic failure of concrete dams should consider the following items: - Seismic analysis of concrete dams should include assessment of ground shaking, surface displacement, and forces due to water in the reservoir - Both structural and foundation failure modes should be considered - Foundation liquefaction/deformation potential should be considered - Structural failure modes considered should take into account the unique concerns for the type of dam in question.	The information in SL-81-1 should be reviewed and used to inform a more complete analysis of potential failure modes (PFMs) of Dresden Lock and Dam. The information may be used to screen out some PFMs and/or provide the appropriate engineering properties of the foundation materials to perform simplified analyses of other PFMs.	The potential failure modes associated with the components of the lock and dam complex were determined in conjunction with U.S. Nuclear Regulatory Commission Report [82]. Appendix C28 of Fragility Analysis Report [21] is revised to address all applicable potential failure modes. The fragility initially provided was based on an evaluation of the mode judged to be governing. Based on an evaluation of other failure modes, it was determined that this failure mode does govern, and the fragility was unchanged.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			The seismic analysis of Dresden Lock and Dam assumed that the out-of-plane failure of the piers is the controlling failure mode without due consideration of other potential failure modes, most notably those that may impact the foundation of the dam. Although very limited information about the dam is available publicly, the results of a detailed foundation investigation of the dam are presented and discussed in USACE Waterways Experiment Station Miscellaneous Paper SL-81-1, which is among the documents provided for peer review.		
SFR-F1	24-5	The SR requires justification for the use of generic betas (provided in the SPID document) as being applicable for the plant specific conditions.	The fragility analysis of most of the SSCs included in the Dresden SPRA was based on the CDFM approach. Based on the HCLPF values generated from the CDFM approach generic beta values provided in the SPID document were used. The peer review team noted that in at least one instance (2A Standby Liquid Control Accumulator) is high up in a structure and is not an active	Perform an extent of condition using a sampling approach and demonstrate that the SPID criteria were met in using the generic beta values.	For the components with fragilities based on the generic Betas from the SPID criteria, an extent of condition review was performed to validate the beta values used in the fragility calculations. Beta values that were determined to be inappropriate were updated. Table 10 in Fragility Analysis Report

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			component. Thus, the use of a generic beta-c of 0.45 is not appropriate. In response to a PRT question, the SPRA team agreed with the above finding.		[21] reflected the corrected beta values. Note that Table 6 of the fragility report was changed to Table 10 in later revisions.
SFR-F1	24-6	The SR requires justification for the use of generic fragility for any SSC as being appropriate for the plant. Several groups of SSCs were assigned to be rugged with a HCLPF of 2.0 g without justification for that decision.	A large group of the SSCs in Group 5-R, 6-R, 7-R, 9-R, 10-R, and Group 21 heat exchangers were considered rugged and assigned a HCLPF of 2.0g without an adequate justification. Per a response to a question, the Dresden SPRA team stated that portable and FLEX items were assigned an arbitrary HCLPF of 2.0g.	Justify the use of 2.0g for rugged components and the basis for inclusion of components within this rugged class. For SSCs with functional failure modes, provide adequate justification (e.g. calculations) why these components would not be governed by a functional failure. Provide the documentation that demonstrated that a seismic interaction could not have governed these fragilities at levels below that assigned fragility level.	Additional bounding type calculations were prepared to support the judgement of the ruggedness for the applicable components. The bounding calculations were prepared and added as Appendix A32 of Fragility Analysis Report [21].
SFR-F1	24-7	The benchboard anchorage HCLPF was calculated using a fundamental frequency of ZPA in the horizontal Front to back directions which is not realistic. The benchboard functional HCLPF was calculated using a sine beat test crediting amplification factors that are unrealistic.	EXDR025-RPT-005 Appendix C calculates the fragility of the control group benchboards group C20-1, C20-2, and C20-3. The seismic demand in the F/B direction was taken at ZPA. Per EPRI TR-102180, the horizontal natural frequency of benchboards is at a lower bound of 11Hz. Per the NEDO - 10678 seismic qualification	Revise the anchorage and functional HCLPF to account for the appropriate frequency range of interest and appropriate functional capacity.	Control room benchboards have a large footprint on the concrete floor slab and are typically rigid. During the DRE seismic walkdowns, it was observed that these benchboards consisted of added stiffeners, which increased the stiffness of the panels in the front to

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			report, a resonance search was conducted for the benchboards and was measured to be equal to 12.6Hz It is understood from the report that the functional capacity was based on a 1.5g seismic input and not a sine beat test.		back direction. Per EPRI TR-102180 [83], the lower bound horizontal natural frequency of the benchboards is 11Hz. However due to the presence of added stiffeners, based on industry experience with similar benchboards, it is judged that the frequency of the stiffened panel in the front to back direction is well above 20Hz. Thus, rigid natural frequencies are used for estimating the seismic demands for the benchboards in the front to back direction. Appendix C20 of Fragility Analysis Report [21] was updated to provide further justification for the frequency used in the fragility evaluation. For the benchboard functional fragility evaluation, a thorough review of the NEDO report showed that the test conducted was a sine sweep test. Additionally, it is confirmed from Table 3.4 of the NEDO report that

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					the maximum floor acceleration capacity for the benchboards was 1.8g. Consequently, this 1.8g capacity was scaled per the procedures listed in EPRI NP-5223-SL for a sine sweep test and Section 8.4 of Appendix C20 of fragility analysis report was updated to reflect the revised capacity.
SFR-F1	24-8	Seismic fragility of the contaminated condensate storage tanks (CST) was evaluated using the response spectra at the node (location) of the CST that were calculated from the Reactor Building-Turbine Building SSI response analysis. The EPRI NP-6041-SL Appendix H approach was followed for the tank evaluation. The tank HCLPF capacity, governed by yielding of the bolt chair top plates, was obtained by multiplying the calculated factor of safety to the PGA of the SSI response spectra at the CST location. Use of this PGA is incorrect.	The SSI response spectra at the location of the CST were generated from the SSI response analysis of the Reactor Building-Turbine Building finite element model. The input ground motion for the RB-TB SSI analysis is the site-specific GMRS/FIRS1 defined at elevation 472.5 feet which is the control point of the input motion. Thus, the HCLPF capacity should be anchored to PGA of the GMRS/FIRS1, not the PGA value of the SSI response spectra.	The seismic fragility calculation of the CST should be updated to reflect the corrected HCLPF anchored to the PGA of the GMRS/FIRS1. Furthermore, seismic fragilities of the other surface-founded SSCs in the yard which used response spectra generated from the RB-TB SSI response analysis should be reviewed for similar issue.	CST calculation (Appendix B21.3 of Fragility Analysis Report [21]) was updated to use correct PGA. Calculations for other surface-founded structures were reviewed and no additional occurrences of using an incorrect PGA were identified.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-A5	25-1	No fragilities were developed for Instrument Air and TBCCW, yet these systems are credited in the SPRA model. Justification for this was provided such that a failure of the HEP to restore these systems following a LOOP is the dominant contributor to loss of these systems, and the modeling of seismic fragilities would contribute little to the model. This was confirmed via a sensitivity study for TBCCW. However, excluding these fragilities could skew other model insights, such as the importances of the associated HFEs to restore the systems following a LOOP, as well as the sensitivity study on the potential contribution of a seismic event with no LOOP.	This is a Finding as the cumulative impact of this modeling with other findings could have an impact on the results that could rise to the level of significance.	Add fragility estimates or assume a failure to the IA and TBCCW systems. It is not the intent of this F&O that detailed fragility calculations be required to resolve this F&O, the sensitivity studies performed demonstrate that conservative estimates should be fine, but some treatment of these systems is needed.	Fragilities for IA (fragility group S-IAS with Am=0.3g) and TBCCW (fragility group S-TBCCW with Am=0.3g) systems were added into the SPRA model. Use of Am=0.3g value is based on fragility data provided by the DRE SPRA Fragility team [21]. The RM SPRA Fragility Modeling Notebook [50] was updated to include these SPRA model fragility groups.
SPR-A5	25-2	Fragilities for FLEX components, including FLEX buildings, were developed but not included in the model. Justification for this was provided such that a	This is a Finding as the cumulative impact of this modeling with other findings could have an impact on the results that could rise to the level of significance.	Include the fragilities calculated for FLEX components in the model, consistent with, and if required by, the fragility screening level criteria and the	Fragilities for the FLEX A building (fragility group S-FLXBA with Am=1.73g) and the FLEX Diesel Generator (fragility group S-FLXDG with Am=1.73g based on

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		failure of the HEPs to align FLEX provisions were the dominant failure modes, and the modeling of seismic fragilities would contribute little to the model. This was confirmed via a sensitivity study. However, excluding these fragilities could skew other model insights, such as the importances of the associated HFEs.		fragility calculation refinements requested by other F&Os.	the fact that it is stored in the FLEX A building) were added into the SPRA model. Use of Am=1.73g value is based on fragility data provided by the DRE SPRA Fragility team [21]. The RM SPRA Fragility Modeling Notebook [50] was updated to include these SPRA model fragility groups.
SPR-B4	25-3	The DR-PRA-020.005 Seismic Fragility Notebook Table B-2, page B-35, Record ID #'s 431 and 404 are not included in the SPRA model, with the note 'Not explicitly modeled due to high fragility for NSSS LOCA piping.' This is potentially non-conservative, as at the ground motion levels where a LOCA may occur, chatter of these relays would likely have occurred as well, inhibiting the plant's response to the LOCA.	This is a Finding as relay chatter influencing the plant response to a LOCA may be inappropriately screened.	Do not screen relays on the basis of a high NSSS fragility when the components affected by the relay chatter may be required to respond to the LOCA.	Fragilities for the relay chatter events impacting LPCI injection valves during LOCA events (i.e., fragility groups S-CH414 and S-CH431) were added into the SPRA Model. Although the system impacts were modeled in a conservative manner, the FV for these relay chatter events is non-risk significant (i.e., CDF and LERF FV < 7E-04). RM SPRA Fragility Modeling Notebook [50] was updated to include these SPRA model fragility groups.

SR	F&O	Description	Basis	Suggested Resolution	Disposition
SPR-F1	25-5	Some statements made in the Fragility Modeling Notebook DR-PRA-20.005 appear to be incorrect and lend to confusion. Correcting these statements will enhance the documentation.	In the Fragility Modeling Notebook DR-PRA-20.005: 1. Table B-1 contains entries with the Correlation column stating: 'Location unknown. Not correlated.' The intent of this statement is confusing as correlation considers more than location information, and some of the ECDs with this statement (0595-115A, 0595-116A, 0595-116B) are the same model type (12HFA151A2H) and may be in similar cabinets. This appears to be strictly a documentation issue, as the ECDs' with this comment are either screened from the model due to high fragility, or if modeled appear to be appropriately correlated or uncorrelated based on all the typical correlation rules (same model type, same or similar host cabinet, similar calculated fragilities, etc.) 2. Table B-2 ECD 3-7826-CR39B (Record #476) has the PRA Modeling Note: 'U3 SPRA uses fragility group S-CH462 as a surrogate for relay ID 476. Am	Revise the documentation as suggested.	The Dresden Fragility Modeling Notebook [50] was updated as follows: 1. Given that the in-cabinet relay location is not known, seismic correlation decision-making will not be based on the location for these relays. The text in the Correlation column of Table B-1 of the Fragility Modeling Notebook [50] was revised from *'Location unknown. Not correlated.'* to *'In-cabinet relay location unknown. Given that the in-cabinet relay location is not known, the seismic correlation for these relays in the SPRA model is based on other correlation factors per the relay chatter correlation methodology (e.g., identical relay make and model number, and located in the same or correlated host cabinets).'* 2. Table B-2 has been updated to reflect the following:

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			increased from 0.44g to 0.52g to model unavailability of MCC 39-7 (conservatively assume not recoverable). Upon comparison with the actual FRANX database, this appears to be a typo and the Am was decreased from 0.52g to 0.44g as this component is correlated with another component with Am of 0.44g. Similar instances were identified for other table entries, such as ECD 3-7838-7-CR3871(#465). Please revise the table to appropriately state the relay fragilities were decreased. 3. Some relays on Table B-2 appear to be solid state relays (for example, 187-DG3, Record #160). This is not evident in table B-2, instead one must go to the Fragility Notebook EXDR025-REPT-005 and search for the relay itself. This is further challenged as an initial fragility value is reported here (Am of 0.56g in Appendix B-26), and later analysis indicates it is solid state and recommends screening it out. Table B-2 should be updated to clearly		Unit 2 SPRA Model SPRA fragility group S-CH462 models ECD 462 (unavailability of MCC 29-7) in the Unit 2 specific FRANX file with Final Am = 0.93g. SPRA fragility group S-CH451 models correlated failure of ECD 451 and 461 (unavailability of MCC 29-7) in the Unit 2 specific FRANX file with Final Am = 1.04g. Unit 3 SPRA Model SPRA fragility group S-CH462 used as a surrogate to model Unit 3 ECD 476 (unavailability of MCC 39-7) in the Unit 3 specific FRANX file with Final Am = 0.47g. SPRA fragility group S-CH451 models correlated failure of ECD 465 and 475 (unavailability of MCC 39-7) in the Unit 3 specific

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			identify what relays are solid state and can thus, be screened from the model. 4. Table B-2 (should be Table B-1) of the Seismic Fragility Modeling Notebook contains entries for ECD's TS-2391-02A (should be TS-2391-02A & -02C) and TS-2391-02A (should be TS-2391-02B & -02D), which under the Notes states, 'Temperature switches are rugged...' However, Table B-2 provides a calculated fragility Am = 0.47g, and these components are in the SPRA model under fragility group S-CH101. Recommend removing the statement in table B-1 that those components are considered rugged, since evidently the fragility analysis concluded they are not. 5. The LPCI HXs (SEL Item #'s 624, 625, 5379, and 5333) were mapped to Fragility Group S-LIHX1. It is noted that the Unit 2 LPCI HXs (SEL Item #'s 624 and 625) have the Fragility Note 'Screen Out' while the Unit 3 HXs do not. Update the		FRANX file with Final Am = 0.47g. 3. The data provided in Appendix B26 of Fragility Analysis Report [21] represents the initial relay chatter fragility data prior to plant specific walkdowns. The cover sheet of Appendix B26 of Fragility Analysis Report identifies that several relay chatter fragilities were updated in Appendix C26 of Fragility Analysis Report, such as the fragilities for the solid state relays. In Appendix C26 of Fragility Analysis Report, there are notes to identify that solid state relays are rugged and can be screened out. Appendix C26 of [21] was updated to clearly identify what relays are solid state and can thus, be screened from the model. 4. For the entries for ECDs TS-2391-02A & -02C and TS-2391-02B & -02D, the complete note in Table B-1

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			table entries to all indicate if the HXs are Screened Out or not. Other Notebooks: 6. In the SEL Notebook (DR-PRA-020.005 Vol. 2), the Diesel Fire Pump 1-G-112A (SEL Item #3137) has the Fragility Note 'Taken Out from SEL,' however it remains on the SEL with disposition F2. Recommend updating the table entry with the appropriate disposition code. 7. Some additional discussion on the treatment of instrumentation, how they were identified based on a review of HFEs credited in the model, and how they are dispositioned from further consideration in the SPRA (i.e., rugged components, more dominant failures such as loss of DC used for HRA binning) would enhance the documentation. 8. The Quantification Notebook and/or Seismic Methods		states: *"Temperature switches are rugged. No relay location information available. Assigned cabinet fragility."* The fragility included in the SPRA model is based on the cabinet fragility and not the temperature switches themselves. The note is judged to be appropriate and no changes to the note are necessary. 5. Table A-1 was updated to remove the text "Screen Out" in the Fragility Notes column for SEL Items #624 and #625. Other Dresden SPRA Notebooks were updated as follows: 6. The SEL Notebook [51] and SPRA Fragility Modeling Notebook [50] were revised to be consistent and to remove text stating that the Unit 1 Diesel Fire Pump 1-G-112A (SEL Item #3137) is "Taken Out from SEL". A fragility

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			notebook could be enhanced on the treatment of the Unit 3 Dresden SPRA model, i.e. How the model is created from the Unit 2 model to build a new Fault Tree that is used with Unit 3 flag files and recovery rules to produce the Unit 3 cutsets. 9. Table B-2 of the DR-PRA-020.005 Seismic Fragility Notebook is incomplete. Beginning on page B-37 for Record ID-#622, the table provides no indication if the items from this point forward are modeled in the PRA, or if not, what the basis is for exclusion. Complete the Table's disposition for screening items from inclusion in the SPRA model.		was not explicitly developed for Diesel Fire Pump 1-G-112A because the Unit 1 DFP is conservatively not credited in the SPRA model. This conservatism has a negligible quantitative risk impact on the SPRA results. 7. Section 3.9 of the SEL Notebook [51] discusses instrumentation items on the SEL. Per the F&O suggested resolution, additional statements regarding the identification and disposition of instrumentation were added into the SEL Notebook [51]. 8. The Seismic Methods Notebook [46] has been updated to document the process for development and quantification of the Unit 3 SPRA model based on the Unit 2 SPRA model. 9. Table B-2 of the Seismic Fragility Notebook [50] was updated to include

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					disposition for all items that are screened from inclusion in the SPRA model.
SPR-B4a	25-10	Some components with a fragility of less than 0.8g were screened from the model. The basis for this screening was typically documented in Table A-1 of the Fragility Modeling Notebook (DR-PRA-020.005 Vol. 1), and these dispositions appear reasonable, although there are some outliers for which a Finding was written. Table A-1 of the Fragility Modeling Notebook D03-0903-0049 (SEL Item #5377), D03-0903-0050 (SEL Item #287), 2-0902-49 (SEL Item #121), 2-0902-50 (SEL Item #123), and D02-83125-2---P06 (SEL Item #1069) are dispositioned with the PRA Note 'Indirect impact (i.e., not directly CDF or LERF). At G6 (i.e., 0.6g to 0.8g) interval, CCDP > 0.9 and CLERP > 0.5. Numerous other fragilities govern and thus this fragility not explicitly	This is a Finding as the basis provided to screen certain components from the final SPRA model may not be appropriate.	Revise the basis to screen the identified SSC fragilities from the model (for example, '1 of 3 uncorrelated redundant power sources') or include them in the SPRA model. For SEL Items 2751 and 2752, identify which fragility is the correct fragility and use that in the model.	The fragility level screening criteria for the DRE SPRA was finalized after the peer review and the bases for the determination of the screening level were documented using the final SCDF and SLERF results. This documentation is in Appendix C of the SPRA Fragility Modeling report [50]. A final screening level of 1.0g, PGA HCLPF was selected and used in the final DRE SPRA. Additional SSCs with fragilities up to the final fragility screening level of HCLPF = 1.0g were explicitly included in the SPRA model or dispositioned as not required to be included. Resolutions to specific issues identified in F&O 25-10 are as follows:

SR	F&O	Description	Basis	Suggested Resolution	Disposition
		modeled.' It is unclear what PRA functions would be affected by a loss of these components, and if other more dominant failure modes for those components are already captured in the PRA. Additionally, 3-83125 (SEL Item #2751) is dispositioned with the PRA Note Duplicate SEL line item, see SEL# 2752.' SEL Item #2752 is reported as having an Am of 0.86g and is included in the SPRA model, however, SEL Item #2751 reports an Am of 0.63g. If this is a duplicate entry it is unclear why a different fragility value is reported and why the higher one is selected for inclusion in the model.			1. D03-0903-0049 (SEL Item #5377), D03-0903-0050 (SEL Item #287), 2-0902-49 (SEL Item #121), 2-0902-50 (SEL Item #123) were explicitly modeled in the SPRA as part of correlated SPRA fragility group S-INCP19, “U2 and U3 ESS Bus PANEL 120/240 VAC ESS SERV DIST PNL” with final Am = 0.78g. It is noted that SEL Item #287 was renumbered to SEL Item #5378 in the Final SEL. 2. D02-83125-2--- P06 (SEL Item #1069) was explicitly modeled in the SPRA as part of correlated SPRA fragility group S-DCBU2, “Unit 2 125 VDC TRAIN A BUSSES (2-83125) - 549 TB” with final Am = 1.2g. 3. For SEL Items #2751 and #2752, SEL Item #2752 was the correct SEL item and the fragility used in the Final SPRA model quantification for the Unit

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					3 125 VDC Battery with final Am = 0.72g. Duplicate SEL Item #2751 was deleted from the final SEL. The fragility disposition commentary in Table A-1 of the SPRA Fragility Modeling report [50] has been revised and addresses the specific items noted in the F&O description.
SPR-E3	25-11	The screening criteria utilized in the final quantification of the Dresden SPRA model is based on an Am of 0.8g. However, there is not a clear quantifiable justification presented for screening SSCs above 0.8g. It is noted that the Reactor Building, whose failure results directly in CDF and LERF, is included in the model and has a negligible contribution to both CDF and LERF. However, the Am for the Reactor Building is 2.35g, which is well above the 0.8g criteria. It is unclear what the contribution to risk would be for components at 0.8g.	The Quantification Notebook Appendix K states that for seismic interval %G6 (i.e., 0.6g to 0.8g), the CCDP > 0.9 and CLERP > 0.5. However, it may be possible for the cumulative impacts of components currently screened to further increase these CCDP and CLERP values.	Perform additional sensitivity studies to confirm that screening of components with Am above 0.8g is appropriate.	The fragility level screening criteria for the DRE SPRA was finalized after the peer review and the bases for the determination of the screening level were documented using the final SCDF and SLERF results. This documentation is in Appendix C of the SPRA Fragility Modeling report [50]. A final screening level of 1.0g, PGA HCLPF was selected and used in the final DRE SPRA. The determination of the appropriateness of the 1.0g HCLPF screening level used a quantitative sensitivity study of the final SCDF and

SR	F&O	Description	Basis	Suggested Resolution	Disposition
					SLERF models to demonstrate that a 1.0g HCLPF fragility group modeled directly as SCDF and SLERF would meet the FV < 5E-03 criterion for non-risk significant. Additional SSCs with fragilities up to the final fragility screening level of HCLPF = 1.0g were explicitly included in the SPRA model or dispositioned as not required to be included.
SPR-A5	26-1	There are some SSCs where the functional and anchorage HCLPFs are relatively close (e.g. within 20%). For the SSCs where the anchorage and functional HCLPFs are close, these individual fragilities of the multiple failure modes is not accounted for in the SPRA.	Modeling of just the governing failure mode in the SPRA fragility groups is typically done (and acceptable) when the governing failure Am is significantly lower (e.g. more than 20%) than the other failure modes. The reason is that the failure probabilities for the other failure modes is typically much smaller than the failure probabilities of the weaker failure mode when the Am is significantly lower. However, when the Am for the different failure modes are relatively close, then the failure probabilities from the different	The Am capacities of the different failure modes for the fragility groups should be reviewed to determine if the fragilities should be combined. This should be done at least for the significant SSCs and those that are near the significance threshold. It is acceptable to not combine the fragilities if the fragility for one of the failure modes is considered to have significant conservative margin such that if it was refined further, the fragilities are no longer be close. Otherwise, the fragilities	If the fragility difference between two failure modes were found to be less than 20%, the combined fragility for a closely spaced failure modes was calculated by the DRE SPRA Fragility team based on guidance in Section 6.8 of EPRI 3002012994 [80] and documented in [21]. For SSCs where a combined fragility was calculated, a note has been added to Table A-1 of the RM SPRA Fragility Modeling Notebook [50]. The

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			failure modes can sum to a higher failure probability. Therefore, for SSCs where the Am for the different failure modes are close, then these failure modes are either explicitly modeled in the SPRA logic model or the Am capacities are combined resulting in a lower Am capacity that represents the combined Am for the combined failure modes.	should be combined or explicitly modeled.	combined fragility data was incorporated into the SPRA model for the associated SPRA model fragility group.
SPR-F2	26-4	Documentation of SSCs screened from the SPRA based on seismic capacity is not clear making it difficult to confirm that SSCs screened from the SPRA are not significant contributors.	The process described in Appendix K provides a good understanding of the phased approach in refining the fragilities modeled in the SPRA. This approach is consistent with industry practice. And the discussions with the Dresden SPR team helped in understanding the refinements made to the SPRA model in terms of grouping the SSCs and using the Am of the weakest SSC as the fragility group capacity. However, in reviewing the tables (e.g. Table A-1) in the Fragility Modeling notebook (DR-PRA-020.005 Vol 1), there were a number SSCs that are listed as not modeled	A section should be added to the SPRA documentation that describes the screening process and to document the following: - A list of SSCs screened from the SPRA based on seismic capacity - What the estimated impact on seismic risk is Confirmation that the SSCs screened are not significant contributors to seismic risk	The fragility level screening criteria for the DRE SPRA was finalized after the peer review and the bases for the determination of the screening level were documented using the final SCDF and SLERF results. This documentation is in Appendix C of the SPRA Fragility Modeling report [50]. A final screening level of 1.0g, PGA HCLPF was selected and used in the final DRE SPRA. The determination of the appropriateness of the 1.0g HCLPF screening level used a quantitative sensitivity study of the final SCDF and

SR	F&O	Description	Basis	Suggested Resolution	Disposition
			and the PRA Modeling Notes are not clear why they are not modeled. Some examples are for SEL items 5289, 5366, 1063, and 5384 are four of approximately 650 SSCs that do not have a note indicating why the SSC is not modeled. It appears to be based on the SSC Am being greater than a certain screening capacity (0.9g), but this is not clearly documented. Some others are mentioned in the F&O for SPR-E3. There are also inconsistencies in Table A-1 such as SEL items 624 and 625 shows Screened Out in the Fragility Notes column but the Model in SPRA column says Yes.		SLERF models to demonstrate that a 1.0g HCLPF fragility group modeled directly as SCDF and SLERF would meet the FV < 5E-03 criterion for non-risk significant. For SSCs screened from the SPRA model based on seismic capacity, a note has been added to Table A-1 of the RM SPRA Fragility Modeling Notebook [50]. Additional SSCs up to the final fragility screening level of HCLPF = 1.0g were explicitly included in the SPRA model or dispositioned as not required to be included. As a result of these changes alone (i.e., when not accounting for the risk impact of various other SPRA fragility and modeling enhancements), the SCDF and SLERF increased slightly due to the inclusion of additional SSC fragility failures.