Methodology and Case Study for Use of Seismic Margin Assessments in Quantitative Risk-Informed Decision Making

A Revision to EPRI Report 1003121

Technical Report

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REPORT SUMMARY

This report provides a methodology for utility engineers to develop quantitative risk information such as seismically induced core damage frequency from an existing seismic margin evaluation in support of risk-informed decision making. The report includes an example case study utilizing this methodology.

Background

In 1995, the NRC issued a Policy Statement on the use of probabilistic risk assessment (PRA), encouraging its use in all regulatory matters. In response to this new direction, the NRC is changing the governing regulations to interject Risk-Informed and Performance-Based (RI/PB) criteria. The formal transition to a risk-informed regulatory framework is expected to be incremental. Risk-informed regulations will be phased in and will gradually replace existing deterministic and prescriptive requirements. The recent Regulatory Guide 1.200 discusses suggested acceptable approaches for using PRA results for risk-informed activities. The description of these approaches is relatively high level, and the Regulatory Guide only describes the basic NRC position on general topics relating to PRA methods and technical adequacy. In order to derive the maximum benefit from this transition to risk informed regulations/applications, the nuclear industry would benefit from developing methods and procedures to both influence and implement these new regulations in a cost-effective manner. These risk-informed criteria require consideration of all internal and external event risks. In general, the three major events in this regard will be internal events, fire, and earthquakes. On-going industry programs address the “risk-informed” process for internal events and fire: this EPRI program is designed to address the last of the three major events, earthquakes.

Objectives

To demonstrate that the results from Seismic Margin Studies can be used to support a variety of risk informed applications in conformance with the NRC PRA Policy Statement and the subsequent Regulatory Guides that discuss its implementation; to evaluate several technical approaches for developing quantitative risk information from existing Seismic Margin Assessment (SMA) studies; to recommend the most appropriate approach for performing SMA conversions; to perform an example case study using SMA results to develop quantitative risk information in terms of Core Damage Frequency (CDF) values,

Approach

The investigators evaluated several methods for converting SMA results to a format applicable for quantitative RI/PB purposes. They compared CDF values from an existing seismic PRA with estimated CDF values generated from a conversion of seismic margin assessment results. They performed the comparison on the Catawba plant, which has both the seismic PRA and SMA needed for the comparison. In addition, the investigators conducted sensitivity studies by

changing various seismic PRA and SMA parameters such as the seismic hazard, the component High Confidence of a Low Probability of Failure (HCLPF) levels, the random failure rates, and the human action failure rates in order to gain insights into the methods under investigation.

Results

The report outlines the basic PRA and SMA methodology as well as several conversion methods and summarizes the specific risk results from the Catawba plant, both for the seismic PRA and the SMA. The various methods proposed for converting SMAs to PRA-type risk measures were reviewed and evaluated based on these Catawba results. Risk estimates were developed from the plant SMA and compared to the seismic PRA results. The report provides guidance on the appropriate simplified computational procedure to use SMA results for RI/PB applications, along with recommendations for further refinements.

EPRI Perspective

EPRI has issued a variety of reports relating to the development and application of seismic methods in support of RI/PB applications. However, these methods have not been used widely in the nuclear industry with respect to seismic design, plant modification, or operational decision making. To date, the primary application of these methods has been to perform seismic Individual Plant Examination for External Events (IPEEE) evaluations in response to NRC requests. Seismic Margin Assessments and seismic PRAs of varying levels of sophistication have been the primary tools for these evaluations. The application of RI/PB methods to seismic issues involves several professional engineering disciplines. For example, engineers in structural/seismic engineering, systems, operations, and PRA disciplines often perform IPEEE assessments collectively. As methods are improved and simplified, it will be necessary to integrate these efforts and bring the processes into the mainstream of plant engineering and operations. There is a need to cross train utility engineering departments so that each discipline understands and properly coordinates and supports the role of the others. This report provides simplified methods to obtain quantitative risk information from an SMA. A future EPRI report will address qualitative risk information derived from an SMA. The contents of this report will also assist in integration and cross training within the nuclear industry.

Keywords

Seismic PRA
Seismic Risk
Seismic RI/PB
Seismic Margin
Earthquakes
Seismic Hazard

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1 INTRODUCTION AND PURPOSE

1.1 Background

The U.S. Nuclear Regulatory Commission (NRC) is currently focusing on the use of "Risk-Informed" assessments as the technical basis for future decisions. Consequently, the NRC is changing the governing regulations to interject Risk-Informed and Performance-Based (RI/PB) criteria (Reference 1). The formal transition to a risk-informed regulatory framework is expected to be incremental. Risk-informed regulations will be phased-in and will gradually replace existing deterministic and prescriptive requirements. The NRC is fully committed to the philosophy of an expanded use of practical Probabilistic Risk Assessment (PRA) methods and insights for all regulatory activities in the future.

In order to derive the maximum benefit from this transition, the nuclear industry must develop methods and procedures to both influence and implement these new regulations in a cost-effective manner. These risk-informed criteria require consideration of all internal and external event risks. In general, the three major events in this regard will be internal events, earthquake, and fire. Internal events and fire have on-going industry programs to address the "risk-informed" process, and this EPRI program is designed to address the last of the three major events, earthquakes.

The industry conducted a major program to address the issue of seismic risk to each of the U.S. nuclear plants in response to the NRC's Individual Plant Examination of External Events (IPEEE) program (Reference 2). Seismic events have been found via the IPEEE program to be important contributors to the calculated Core Damage Frequency (CDF) for several U.S. plants. In fact, the contributions to total CDF from seismic events can, in some cases, approach (or even exceed) that from internal events. Accordingly, EPRI has initiated a Program to develop and enhance seismic RI/PB methods and procedures. An overall strategic plan has been developed in Reference 3. The fundamental premise of this EPRI program is to provide utilities with the appropriate set of tools with which to respond to a wide range of potential seismic RI/PB applications. A separate study has already been completed to provide the methodology and the application guide for using SPRA results for RI/PB purposes. The use of Seismic Margin Study results for RI/PB type applications can fall within either a *quantitative* type assessment wherein risk information in the format of the CDF would be derived, or within a category which might be characterized as *qualitative* type assessments where risk insights from the SMA would be used to address the RI/PB application. This report addresses the quantitative type of assessment only. EPRI plans on addressing the qualitative type assessments using SMA results in a later study within this program.

1.2 Purpose

Currently there are 103 operating nuclear plant units in the U. S. All of them were reviewed for seismic vulnerabilities under the IPEEE program, which was documented in submittals to the NRC. The NRC has reviewed the seismic IPEEE submittals and has issued a compilation of the perspectives gained from the program (Reference 4). Similarly, a parallel assessment of the seismic IPEEE submittals was conducted by EPRI (Reference 5). Under the guidelines of the IPEEE program (Reference 2), a plant could utilize either a seismic PRA (SPRA) or a Seismic Margin Assessment (SMA) methodology, as identified in Reference 2, to accomplish the required review. A few plants conducted independent seismic review programs. While all plants were reviewed under the program, the scope and level of review was graded, in general, according to the seismic hazard of the site. Each plant was placed within the following bins in descending order of hazard:

- SPRA
- Full Scope SMA
- Focused Scope SMA
- Reduced Scope SMA

Full and focused scope plants required the review to be conducted at a minimum assigned seismic level exceeding the plant seismic design basis. The primary difference between full and focused scope was the effort devoted to review of relay control circuits. Some plants were required to use an SPRA for the program, while plants with the lowest seismic hazard conducted reduced scope reviews at the level of the plant seismic design basis. Some plants chose to modify the level of review according to the revised guidance provided in Reference 6. The methods used to respond to the IPEEE program are summarized in Table 1-1.

Approximately 58% of the operating plants in the U.S. performed seismic margin assessments in response to the IPEEE program. These margin assessments contain a significant amount of information that can be valuable to RI/PB-type applications, but criteria and methodology need to be developed to extract and recast the information so that it can be used directly in these applications. A primary goal of the EPRI RI/PB Program is to allow plants to utilize these seismic margin results for RI/PB applications without the need for the full development of an additional new seismic PRA.

Several technical papers exist with suggested approaches on how to address this transformation of the seismic margin results into PRA-type summary results. The purpose of this report is to evaluate these potential methods that have been discussed within the industry, identify the major issues associated with the application of these approaches, and to make recommendations on how best to use seismic margin assessment results in quantitative risk applications. The level of effort required to implement the transformation of SMA results can vary depending on the plant vintage, design type, design conservatism and plant specific situations. In some cases, the level of effort may be significant.

Since the scope of the SMAs performed for IPEEE was more limited than the scope of the seismic PRAs, some risk-informed applications cannot be unilaterally supported by those SMAs. One particular limitation can be specifically identified. The systems analysis aspect of a typical SMA (e.g., those conducted for IPEEE) contemplates only the evaluation of success paths that would prevent a core-damage accident sequence. Thus, based on such an SMA study alone, there is no explicit way to separate those core-damage accident sequences that might lead to a "large early release" from other sequences that would not lead to such a release. Hence, for situations where SMA's are to be utilized for risk-informed applications related to Large Early Release Frequency (LERF), additional enhancements may need to be undertaken (e.g., review of additional structures and systems for seismic capacity levels, review of accident sequence logic to determine if seismic events cause additional LERF accident sequences, consideration of the potential seismic effects to the human error probabilities used in the internal events analyses, etc.). Guidance on specific LERF considerations related to seismic risks are contained in the EPRI report "Seismic Probabilistic Risk Assessment Implementation Guide" (Reference 23).

Table 1-1
IPEEE Program Seismic Methods

IPEEE Bin	Method	No. of Units
Reduced Scope	Site Specific	4
	EPRI SMA	13
	SPRA	2
Modified Focused Scope	EPRI SMA	3
Focused Scope	EPRI SMA	35
	NRC SMA	2
	SPRA	22
	EPRI SMA and SPRA	1
	Simplified SPRA	2
Full Scope	EPRI SMA	7
	SPRA	6
SPRA Required	SPRA	6
Total Units		103

2 SEISMIC PRA RESULTS AND CDF SENSITIVITY TO FAILURE MODES AND SEISMIC HAZARD

2.1 Introduction

For the IPEEE Program, approximately 38% (39 units) of the US operating commercial nuclear power plants chose a seismic PRA (SPRA) as the assessment methodology option to identify the seismic vulnerabilities beyond the plant design basis.

Seismic PRAs conducted under the IPEEE program consisted of a Level 1 analysis plus a qualitative containment performance analysis. For a SPRA, the seismic risk is usually measured by the mean Core Damage Frequency (CDF) resulting from accident initiators caused by seismic events (more correctly, this seismic CDF value should be referred to as the seismic contribution to the overall plant CDF). First a seismic hazard estimate is made for the plant site and defined by the annual frequency of exceedance for a selected ground motion parameter, such as Peak Ground Acceleration (PGA). To account for the variability in ground motion, this information is presented as a set of hazard curves. For each critical structure, system, or component (SSCs), a seismic fragility curve, or conditional probability of failure taken as a function of the ground motion quantity, is used to define the risk. The fragility curves are also presented as a set of functions to account for variability of SSC failure. These conditional failure probabilities are then used in a Boolean logic expression, which is generated from a systems analysis using event and fault trees for the plant equipment systems, to obtain the overall conditional core damage probabilities. This result is then convolved (in the mathematical sense) with the seismic hazard function to obtain the CDF.

The mean seismic CDF reported (References 4 and 5) for the group of operating plants evaluated in the IPEEE program using SPRAs range from 1.9E-7 to 5.9E-5 per year (using EPRI hazard or site-specific hazard estimates). Approximately 13% of the operating units have mean seismic CDF values less than 1.0E-6 per year while approximately 20% of the operating units have CDF values in the range 1.0E-6 to 1.0E-5 per year. The remaining 67% of the operating units with SPRAs have mean seismic CDF values in the range 1.0E-5 to 6.0E-5 per year.

Review of typical seismic PRAs often reveals a tiered pattern for the dominant cut sets or accident scenarios in terms of contribution to the overall seismic CDF. The first tier consists of two or three cut sets that each contributes approximately 10-20% of the overall CDF. The second tier consists of several cut sets that each contributes 1-10% of the overall CDF, followed by the third tier with the majority of cut sets that each contribute < 1% of the overall CDF. The third tier collectively accounts for approximately 20% of the overall CDF.

In this report, the Catawba Nuclear Station is used as an example for application of the various methods proposed for conversion of SMA results for RI/PB applications. The Catawba units are somewhat unique, in that both a full Level I SPRA (Reference 9) and a SMA review (Reference 14) were conducted. The SPRA was used for the IPEEE final submittal for the Catawba units while the SMA was conducted as an early demonstration of the EPRI SMA methodology. Thus the SPRA results for Catawba can be directly compared to the converted SMA results.

2.2 Typical Seismic PRA Characteristics

The three major steps in a seismic PRA are seismic hazard determination, seismic fragility determination, and systems analysis.

2.2.1 Seismic Hazard Analysis

Seismic PRAs conducted under the IPEEE program have used the results of seismic hazard studies (References 7 and 8) performed either by EPRI or Lawrence Livermore National Laboratory (LLNL). All but two plants reported CDF values based on both hazard estimates. For some sites that were not studied by either EPRI or LLNL, or both, utilities performed site-specific hazard analysis. Both EPRI and LLNL hazard studies are considered “state-of-the-art” studies which fully depict the uncertainties in seismic hazard analysis (the so-called “epistemic” variabilities).

Typically, however, the mean hazard curve has been utilized in seismic PRAs to obtain a point estimate of CDF since it adequately served the IPEEE goal of searching for severe accident vulnerabilities. Figure 2-1 compares some of the mean hazard curves obtained from the EPRI seismic hazard study for Central and Eastern U. S. (CEUS) sites. As can be noted, the Catawba site falls within the upper third of the CEUS sites in terms of seismic hazard. In contrast, the Braidwood site is near the lower boundary of CEUS hazard (ignoring the very low seismicity sites in Texas and Louisiana). Figures 2-2 and 2-3 compare the mean hazard curves obtained from the EPRI and LLNL studies (References 7 and 8) for the Catawba and Braidwood sites. As can be noted, both hazard functions defined for the Catawba site are comparable, however, the hazard functions defined for the Braidwood site differ by a substantial factor.

It should be noted that the spectral shape used to define the amplification characteristics of the site ground motion is also an important consideration. Many of the plants that performed SPRAs for the IPEEE program used the Uniform Hazard Spectra (UHS) provided in the EPRI or LLNL seismic hazard reports. The NRC has observed that the UHS for the Eastern U. S. sites do not have much energy content in the low frequencies associated with buildings flexible equipment and piping response. Some plants, however, used broadband, smooth spectral shapes, such as those presented in NUREG/CR-0098 (Reference 11), modified for site conditions. The Catawba SMA utilized a Sequoyah 84^{th} percentile spectral shape anchored to 0.3g pga. The seismic PRA fragilities are based on a NUREG/CR-0098 median amplification spectral shape. These spectral shapes are very similar and have similar amplification of pga. Consequently, the SPRA results for Catawba can be compared to SMA based risk results without adjusting for the reduced energy content of a UHS in the low frequency range.

2.2.2 Seismic Fragility Evaluation

In SPRAs, the seismic fragility evaluation is performed for critical failure modes of structures, systems and components, for both structural failure and equipment functional failure. The calculation of seismic fragility parameters such as median capacity and variabilities is based on plant-specific data supplemented, as needed, by earthquake experience data, fragility test data and generic qualification test data. The seismic fragility evaluation includes the findings of a detailed walkdown of the plant focusing on the anchorage, lateral seismic support, and potential systems interactions. Fragilities are based on realistic seismic response level that the SSCs experience at their failure levels. Depending on the site conditions and response analysis methods used in the plant design, realistic seismic response may be obtained by an appropriate combination of scaling, new analysis, and new structural models.

In many seismic PRAs conducted under the IPEEE program, a limited number of component specific fragility calculations have been performed. Most components were either screened out using 0.3g PGA walkdown screening criteria or were screened by a generic analysis based upon design criteria. For the Catawba SPRA, however, component specific fragilities were computed for most SSCs modeled in the analysis of plant systems and accident scenarios. Table 2-1 provides a summary (Reference 9) of the median fragilities, variability, and resulting High Confidence of Low Probability of Failure (HCLPF) seismic capacity values (also referred to as the capacity for which we are 95% confident of less than a 5% chance of failure) for the critical components identified for the Catawba units.

2.2.3 Systems Analysis

Seismic PRA systems models consist of interrelated event and fault trees which identify important seismic-caused initiating events that can lead to core damage or large early release. In addition, other important failures that can contribute significantly to CDF or LERF, such as nonseismic-induced equipment failures and human errors are included.

In order to minimize the effort of system modeling and quantification, some seismic PRA analysts chose to screen the components comparing their seismic capacity to the specified IPEEE Review Level Earthquake (RLE). In a few instances, where all the components were screened out, the seismic CDF could not be calculated. To overcome this situation, some analysts have used the concept of “surrogate” elements. The basis and approach for incorporating surrogate elements in a seismic PRA is discussed in Reference 16. The concept is to account approximately for the contribution to CDF from the components that are screened out during the walkdown and screening phase of a seismic PRA. Hence, the potential failures of several components (that might normally be excluded from the event and fault trees of a seismic PRA) are represented by the failure of a single surrogate element or sometimes, several surrogate elements inserted in different fault trees whose top event leads to a node in the seismic event tree. A review of a number of seismic PRAs has shown that the screening was not done at a sufficiently high threshold and the surrogate element or elements were found to be a dominant risk contributor. In these cases the surrogate “masks“ the true dominant contributors to seismic risk and prevents the calculated CDF from being a measure of sensitivity to changes in seismic risk due to changes in hardware procedures.

Treatment of human actions during and following a large earthquake has substantially varied among the seismic PRAs performed as part of the IPEEE. In some seismic PRAs, simplified operator error probabilities have been developed. Some others have used simple scaling factors on internal event error rates based on the importance of the human actions, size of the earthquake, or other factors. In some cases, when operator error probabilities have been applied, they have also masked the seismic failures that should dominate the seismic CDF. In general, since operator error probabilities are not well known, NUREG-1407 recommended that sensitivity studies be done to reveal the relative importance of seismic failures and their impact on operator actions. However, these sensitivity studies were not conducted for most SPRAs performed in the IPEEE program.

The approaches used to include nonseismic-induced equipment failures in seismic PRAs also varied considerably in the IPEEE program. In general, the treatment of nonseismic equipment failures employed the same assumptions as used in the internal event PRA of each respective plant. While there is a lack of consistency for the treatment of surrogate elements, human factors, and nonseismic equipment failures in the various SPRAs conducted under the IPEEE program, there is a general consensus that such effects are important contributors to risk which must be addressed in some fashion in order to insure realistic seismic CDF estimates.

For the Catawba seismic PRA, a surrogate element was not utilized since fragilities were provided for most critical SSC failure modes. However, since not all equipment items were modeled, one could speculate that a surrogate element or elements should be included to represent equipment that is not specifically modeled. The NRC sponsored technical review of the Catawba SPRA (Reference 10) indicates that the treatment of nonseismic failures and human actions was judged to be satisfactory, however, the review notes that justification of the recovery and error probabilities was not provided in the submittal. Table 2-2 lists the primary nonseismic basic event probabilities used in the Catawba PRA (Reference 9) for operator error and nonseismic equipment failures.

2.3 Evaluation of Seismic PRA Results for Catawba

In this study we intend to compare the seismic CDF value determined in the Catawba seismic PRA with estimated CDF values derived from SMA results by alternate simplified ways. In addition, some of the fragility parameters in the seismic PRA and the HCLPF values in the SMA have been altered to test the sensitivity of the seismic PRA/SMA risk computation comparison. The SMA was performed in the 1986-89 time period as a trial plant application of the EPRI Seismic Margin Methodology just after plant startup. The Catawba SPRA (Reference 9) was conducted in the mid-1990's using more recent data as well as the final as-built conditions of the plant, thus the SPRA was used for the IPEEE submittal. We have made use of the seismic fragilities, the seismic hazard curve, and systems models (i.e., Boolean equations) as reported in the Catawba IPEEE submittal to conduct studies of sensitivity to changes in selected fragilities. It should be noted that these variations do not reflect the actual conditions at Catawba. The following is a summary of the Catawba SPRA results and the analysis perturbations considered.

The final results of the Catawba system analysis yields a set of accident scenarios or "cut sets" that each lead to a core damage state and are initiated by a seismic event. Table 2-3 provides an ordered list of the 18 dominant seismic event sequences identified from the system analysis of a

Catawba unit. The Si are the seismic initiated events listed in Table 2-1, while the OPi and Ei represent the operator error and nonseismic failure events listed in Table 2-2. The "*", used in Table 2-3, denotes an intersection of events which yields the particular accident scenario. Core damage is then defined as the union of the accident scenarios or the Boolean expression:

CD = S5+S8*S9+OP1*(S6+S11)+OP2*S17+
S1*[S2+S3+S4+S7+S10+S15+S16+OP1*(S12+S13+S14)+E1*E2+E2*E3+
E1*E4] + (Remainder of Cut sets)

In the above expression, "+" denotes a union of events. Given that $P_{CD/a}$ represent the conditional core damage probability computed using the conditional event probabilities combined in accordance with the above Boolean expression, then the core damage frequency is given by the convolution integral:

$$\mathrm{CDF} = -\int_0^\infty (P_{CD/a})(dH/da)\, da$$

where H is the hazard function and a is the independent PGA value used to define both the hazard function and the core damage probability. The actual process of computing this expression is normally accomplished in a SPRA by Monte Carlo-type simulations which account for the variability of both the hazard and the conditional failure probability. One simplification that is often used, is to compute the CDF of each cut set or accident scenario and then approximate the union of sequences by the sum of the individual CDF values. In this manner, the approximate contribution of each cut set to the overall seismic CDF can be determined. The actual Catawba SPRA considered 98 sequences of which the first 18, shown in Table 2-3, accounted for approximately 80% of the overall mean seismic CDF. The remaining 80 cut sets contributed less than 1% each to the overall CDF but the aggregate sum accounts for approximately 20% of the CDF. As can be noted in Table 2-3, the electrical and control system failures dominate the Catawba seismic CDF. Cut set 7 listed in Table 2-3, which represents the failure of the normal AFW source, the Condensate Storage Tank (CST), *and* the failure of a motor control center (MCC) to transfer to an alternate feed water source (the service water pond at the Catawba site) is the only event sequence in the top 18 cut sets involving a non-electrical and control related component.

Reference 9 indicates (as does the above Boolean) that the CDF is dominated by the seismically initiated loss of offsite power (LOSP) in conjunction with seismic failures of the diesel generators support systems, and by the loss of offsite power in conjunction with failure of the AC power systems. The non-seismic failures of the diesels and their support systems are less significant contributors to the seismic CDF.

To facilitate comparison of any SMA derived risk results, the sensitivity of the SPRA model to conditions assumed in the SMA risk derivation must be understood. In performance of a SMA, it is assumed that offsite power is not available wherein, in SPRAs, the limited seismic capacity of offsite power is modeled. The sensitivity of LOSP would logically depend on the seismicity at the site and it is desirable to examine this sensitivity. SMAs do not directly take into account random failures, human errors or the success of human recovery actions. Therefore, it is also desirable to determine the sensitivity of the SPRA results to these parameters.

The Catawba SPRA and the SMA indicate that there are no outstanding weaknesses in the plant. There are many components essential to safety that have the same seismic capacity as derived from the qualification test data. Reference 19 suggests that in such cases, a SMA based risk assessment should be conducted using, as a minimum, some level of risk modeling to capture the interaction between systems and components that can lead to core damage. In cases where a single component or a few weaker components, govern the CDF, a very simple SMA based risk assessment can be performed by using the max/min theory as suggested in Reference 18. In the max/min theory, the probability of failure of the weakest component in a success path defines the probability of failure of the success path. For two or more success paths, the maximum success path capacity is used to develop the probability of failure leading to core damage. With this in mind, it was desirable to modify some fragilities in the SPRA and some HCLPFs in the SMA to determine the sensitivity of the comparison of SPRA and SMA derived risk results.

In order to determine the sensitivity of the various event sequences to the hazard input, component fragilities, and random and human error event probabilities, Duke Power agreed to rerun the SPRA model (using the original simulation method) used for the IPEEE submittal with certain perturbations to the hazard input, component fragilities, and nonseismic event probabilities. Table 2-4 summarizes the results of the additional Duke Power simulations using the Catawba hazard. Table 2-5 summarizes the additional simulations using the Braidwood hazard. The baseline CDF for the Catawba hazard was computed as approximately 1.5E-5 per year which compares to the 1.6E-5 per year value reported in the IPEEE submittal (in general, a slightly different CDF will result each time the Monte Carlo type simulation is run).

As can be noted from Case 1 in Table 2-4, the most significant change in CDF occurred when the Loss of Off Site Power (LOSP) was taken as a near certainty; i.e., essentially assumed to have failed. For this assumption (same as used in a SMA), the CDF increased by a factor of 2.4, to a value of 3.6E-5 per year. This is an important comparison CDF value to be used for Catawba SMA conversion studies since SMAs are conducted assuming LOSP has a probability of 1.0. For case 10 where the LOSP fragility was increased by a factor of 1.5, a 22% reduction in CDF was realized. Overall, a range of LOSP capacity from essentially zero to 1.5 times the base case results in a CDF range of a factor of about 3.1.

SMA studies do not explicitly consider nonseismic failures wherein, these random failures and unavailabilities due to maintenance are explicitly modeled in SPRAs. Case 2 was conducted to determine the sensitivity if nonseismic failures, except those of the diesel generators, are removed. The result was a 12% decrease in calculated CDF. In Case 11, the nonseismic failure rates are doubled and the core damage increased by only 2%. For Catawba, the sensitivity to CDF from nonseismic failures is relatively small and unless there are essential components with high unavailability, ignoring these nonseismic failures in an SMA based risk assessment is likely not a significant deficiency. This conclusion may not be the same, however, for some BWRs with steam driven single train HPCI and RCIC systems where the reliability of the pumps is marginal in some cases.

Human errors, such as failure to align the auxiliary feedwater pumps to an alternate source of water in case of failure of the condensate storage tank, are modeled in SPRAs. In SMAs, these human actions, if required in a success path, are assumed to be accomplished. Case 3 was conducted assuming that there was a 100% success rate for human actions. This resulted in a 24% decrease in the computed core damage frequency. A separate case for doubling the human error failure rate was not conducted. However, we can estimate the effect from Table 2-4 by examining Case 12 vs. Cases 7 and 11. Doubling the human error failure rate would appear to increase CDF by about 22%. These cases indicate that ignoring human errors can have a moderate impact on results.

In SMAs, relay chatter that requires specific operator actions for recovery is usually assumed to be a failure. In full scope SPRAs, relay chatter consequences and recovery actions are usually modeled. In the Catawba base case, the recovery of relay chatter was assumed to have a 0.1 failure rate. Cases 4 and 5 were run for a failure rate of 0.05 and 0.25. The ratio in CDF between the two cases is about 1.42. The difference between the base case and the 0.25 failure rate case is about 27%. Note that events S6, S11, S12 and S13 in Table 2-1 show low capacity for relay chatter, thus, in this case, the results are sensitive to the recovery rate. In an SMA based risk assessment that assumes that relay chatter is not recoverable results derived from low relay chatter HCLPF values could be conservative by a significant amount.

The Catawba SPRA did not show any weak links in the capacity of safety related components other than the four relay chatter events. In some SPRAs a single weak link is shown to be a very large contributor to CDF. Cases 6 through 9 were conducted assuming lower capacity of selected safety related components, some in conjunction with random and human error event probabilities. In Case 6 the RWST HCLPF capacity is set at the 0.15g SSE value whereas all other HCLPF capacities except for relay chatter, are much higher. Note that the 0.1g HCLPF for the normal AFW source is not a dominant basic event since the dedicated seismic source of AFW is the pond. This lower capacity for the RWST only results in a 5% increase in CDF, likely due to the fact that the probability of LOCA or loss of primary coolant pump seal cooling is low, thus there is a low probability of demand for the RWST. In SMAs, one of the shutdown paths must be capability of mitigating a small break LOCA and the RWST is essential for this path. A SMA based risk assessment, assuming that a SBLOCA had occurred, could be very pessimistic.

In Case 7, three essential electrical components have their HCLPF capacity reduced to 0.15g. In the case of failure of the main control board, operator action to go to the alternate shutdown panel can mitigate the consequences, hence the other two failures are more significant. In this case, the CDF increases by 75% demonstrating the sensitivity of some components to CDF as opposed to the RWST in Case 6.

Case 8 combines 100% success of human action with the case 7 failures. The incremental difference between Case 7 and Case 8 is about 15%. This is likely due to the assumed 100% probability of the operator to go to the alternate shutdown panel, given loss of the main control board. Other human successes also would contribute some. With weaker components in the model, the results appear less sensitive to human errors.

Case 9 combines the three weak components of Case 7 with the deletion of random failures (except the random failure of the diesel generators). There is only about a 5% difference between the two cases indicating that the effects of random failure are much less when there are weak components in the model.

Next, the effect of site hazard was considered. The CDF was recomputed using the hazard function for the Braidwood site which, as shown in Table 2-5, yielded a baseline CDF of approximately 2.2E-06, for the Catawba SPRA model. This is the effect of postulating a Catawba unit at the Braidwood site; it is not a CDF for a Braidwood unit, which would, in general, have a different SPRA model. The ratio of computed CDF for the Catawba base case to that for the Braidwood base case is about 6.7. The objective of conducting the parametric studies for a lower hazard was to compare the sensitivity of the parameter variations to the Catawba hazard case.

Changes in LOSP frequency were not conducted for the Braidwood hazard. For the much lower hazard case, the availability of offsite power would be greater and we would expect a larger spread between the base case and the zero capacity case.

In Case 1, the lower RWST capacity increased the calculated CDF by 9% instead of the 5% increase for the Catawba base case. This is not significant and is likely the result of LOSP having less of an effect on the total CDF wherein other low capacity components contribute more.

Case 2 shows that the lowering of the HCLPF capacity of three safety related components to 0.15g increases the calculated CDF by 82% compared to the 75% increase for the Catawba hazard. Again, this difference in CDF increase is not significant and likely results from the lower contribution from LOSP, thus more sensitivity to other components.

In Case 3, nonseismic failures, except those of the diesel generators are removed. The decrease in calculated CDF is 13% as compared to 12% for the parallel case for the Catawba hazard. A similar conclusion can be reached for Case 4 where 100% success of human actions is assumed. The small difference in CDF reduction relative to the base case is 23% as opposed to 24% for the Catawba hazard. These small differences are insignificant.

Cases 5 and 6 explore a range of failure rates for operators to recovery relay chatter. The base case is modeled with a 0.10 failure rate. If the rates are changed to 0.05 and 0.25 the change in CDF is –7% and 35% respectively. The CDF rates for the Catawba hazard are –10% and 27% respectively. The changes as a result of hazard are not significant although the range in calculated CDF is about 45%, which is significant.

These perturbations of component fragilities and random and human error event probabilities indicate that, for a given hazard, the absolute values of calculated CDF can vary as much as a factor of 3 for LOSP alone. The difference between the base case and the typical SMA 100% LOSP failure assumption is shown to be a factor of about 2.4. By itself, this may be too conservative for many applications. However, when combined with other assumptions regarding nonseismic failures, human error and human recovery actions, the difference could be less but still significant.

Table 2-1
Catawba Seismic Component Fragilities

Event	Description	Median Capacity, g	β_r	β_u	HCLPF, Capacity, g
S1	Loss of off-site power	0.30	0.25	0.50	0.09
S2	Neutral ground resistor failure	0.91	0.08	0.58	0.31
S3	DG control panel failure	0.86	0.08	0.55	0.31
S4	DG engine control panel failure	0.82	0.30	0.31	0.30
S5	4KV switchgear failure	1.32	0.34	0.55	0.31
S6	600V MCC chatter	0.53	0.28	0.42	0.17
S7	Inverter failure	1.58	0.28	0.72	0.30
S8	Normal AFW source water fails	0.40	0.30	0.52	0.10
S9	600V MCC failure	1.32	0.24	0.64	0.31
S10	Load sequencer failure	1.28	0.08	0.58	0.43
S11	4KV switchgear chatter	0.67	0.28	0.46	0.20
S12	Inverter chatter	0.61	0.28	0.44	0.19
S13	DG engine control panel chatter	0.97	0.08	0.93	0.18
S14	DG control panel chatter	0.71	0.08	0.48	0.28
S15	125V DC panelboard failure	1.48	0.30	0.36	0.50
S16	120V AC panelboard failure	1.48	0.30	0.36	0.50
S17	Main Control Boards fail	0.96	0.37	0.39	0.27

Table 2-2
Catawba NonSeismic Basic Event Probabilities

Event	Description	Probability
OP1	failure to recover from relay chatter	1.00E-01
OP2	failure to go to alternate Control Panel	1.00E-01
E1	DG 1A fails to run	1.10E-01
E2	DG 1B fails to run	1.10E-01
E3	DG 1A in maintenance/testing	4.30E-02
E4	DG 1B in maintenance/testing	4.30E-02

Table 2-3
Catawba Dominant Cut Sets

Cut Set	Event Sequence	Percent of CDF	Cumulative Percent
1	S1*S2	13.1	13.1
2	S1*S3	12.5	25.6
3	S1*S4	9.0	34.7
4	S5	7.4	42.0
5	S6*OP1	5.7	47.7
6	S1*S7	5.4	53.2
7	S8*S9	4.5	57.7
8	S1*S10	4.2	61.9
9	S11*OP1	3.5	65.4
10	S1*E1*E2	3.2	68.6
11	S1*S12*OP1	2.7	71.3
12	S1*S13*OP1	2.6	73.8
13	S1*S14*OP1	1.7	75.5
14	S1*E3*E2	1.3	76.8
15	S1*E1*E4	1.3	78.1
16	S1*S15	1.2	79.3
17	S1*S16	1.1	80.4
18	S17*OP2	1.1	81.5

Table 2-4
Catawba SPRA Model Sensitivity to Catawba Hazard

Catawba Hazard															
Case	**Catawba IPEEE PRA Model**	**LOSP Set to Low Capacity (1)**	**LOSP Set to High Capacity (2)**	**Remove Nonseismic Failure (3)**	**Double Nonseismic Failure Rates**	**100% Human Actions Success**	**Double Human Action Failure Rates**	**Recovery of Chatter at 0.05 Failure Rate (4)**	**Recovery of Chatter at 0.25 Failure Rate (4)**	**RWST at 0.15g HCLPF**	**4 kV Switchgear at 0.15g HCLPF**	**MCC at 0.15g HCLPF**	**Main Control Board at 0.15g HCLPF**	**Resulting CDF Per Year**	**% Change in CDF**
Base	**X**	·		·		·		·	·	·	·	·	·	**1.46E-05**	**----**
1	**X**	✓		·		·		·	·	·	·	·	·	**3.55E-05**	**143%**
2	**X**	·		✓		·		·	·	·	·	·	·	**1.28E-05**	**-12%**
3	**X**	·		·		✓		·	·	·	·	·	·	**1.11E-05**	**-24%**
4	**X**	·		·		·		✓	·	·	·	·	·	**1.31E-05**	**-10%**
5	**X**	·		·		·		·	✓	·	·	·	·	**1.86E-05**	**27%**
6	**X**	·		·		·		·	·	✓	·	·	·	**1.53E-05**	**5%**
7	**X**	·		·		·		·	·	·	✓	✓	✓	**2.56E-05**	**75%**
8	**X**					✓					✓	✓	✓	**2.22E-05**	**52%**
9	**X**			✓							✓	✓	✓	**2.43E-05**	**66%**
10	**X**		✓											**1.14E-05**	**-22%**
11	**X**				✓									**1.49E-05**	**2%**
12	**X**				✓		✓				✓	✓	✓	**2.91E-05**	**99%**

1. HCLPF = 0.01g.
2. LOSP High = 1.5 x LOSP Base Case.
3. *Nonseismic failures associated with diesel generators remain; all other nonseismic failures removed.
4. Base Case is 0.10 failure to recover.

Table 2-5
Catawba SPRA Model Sensitivity to Braidwood Hazard

Braidwood Hazard											
Case	**Catawba IPEEE PRA Model**	**Remove Nonseismic Failure (1)**	**100% Human Actions Success**	**Recovery of Chatter at 0.05 Failure Rate (2)**	**Recovery of Chatter at 0.25 Failure Rate (2)**	**RWST at 0.15 HCLPF**	**4 kV Switchgear at 0.15g HCLPF**	**MCC at 0.15g HCLPF**	**Main Control Board at 0.15g HCLPF**	**Resulting CDF Per Year**	**% Change in CDF**
base	X									2.17E-06	----
1	X					✓				2.37E-06	9%
2	X						✓	✓	✓	3.94E-06	82%
3	X	✓								1.88E-06	-13%
4	X		✓							1.67E-06	-23%
5	X			✓						2.02E-06	-7%
6	X				✓					2.92E-06	35%

1. Nonseismic failures associated with diesel generators remain; all other nonseismic failures taken out
2. Base case is 0.10 failure to recover.

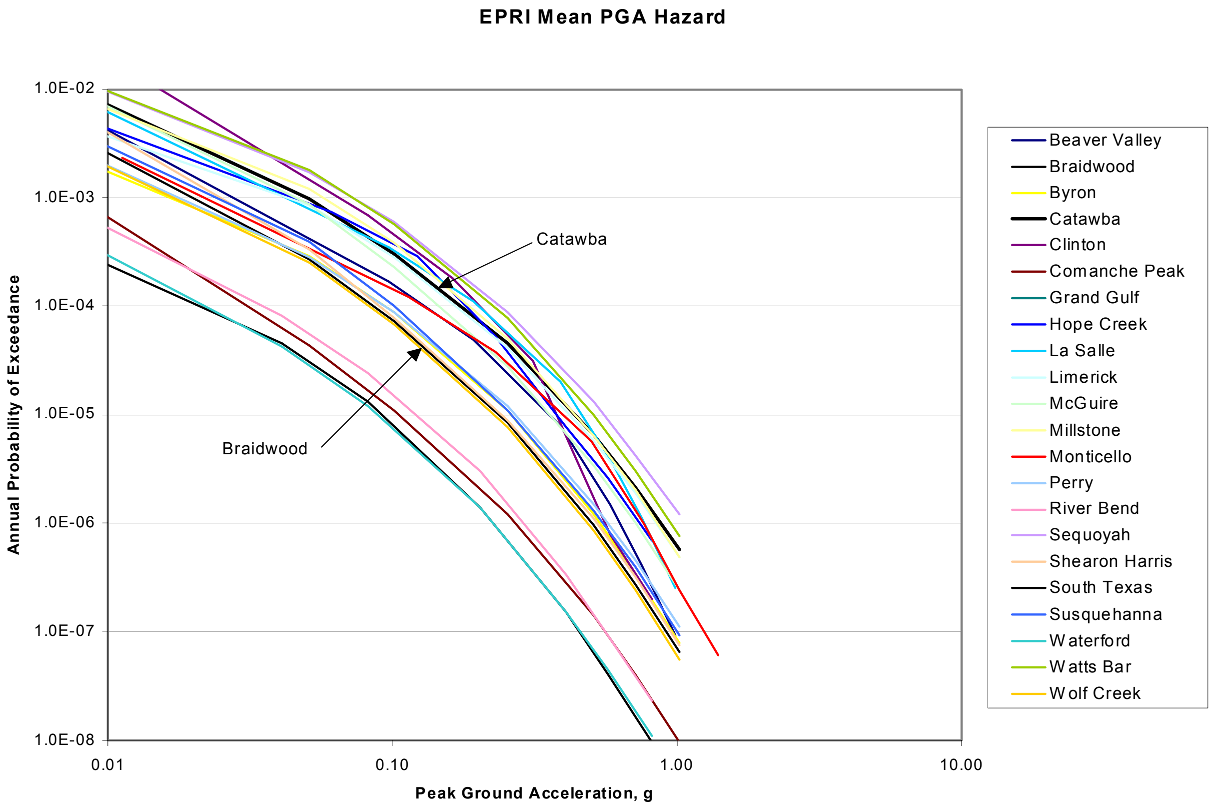

Figure 2-1
EPRI Mean PGA Hazard Curves

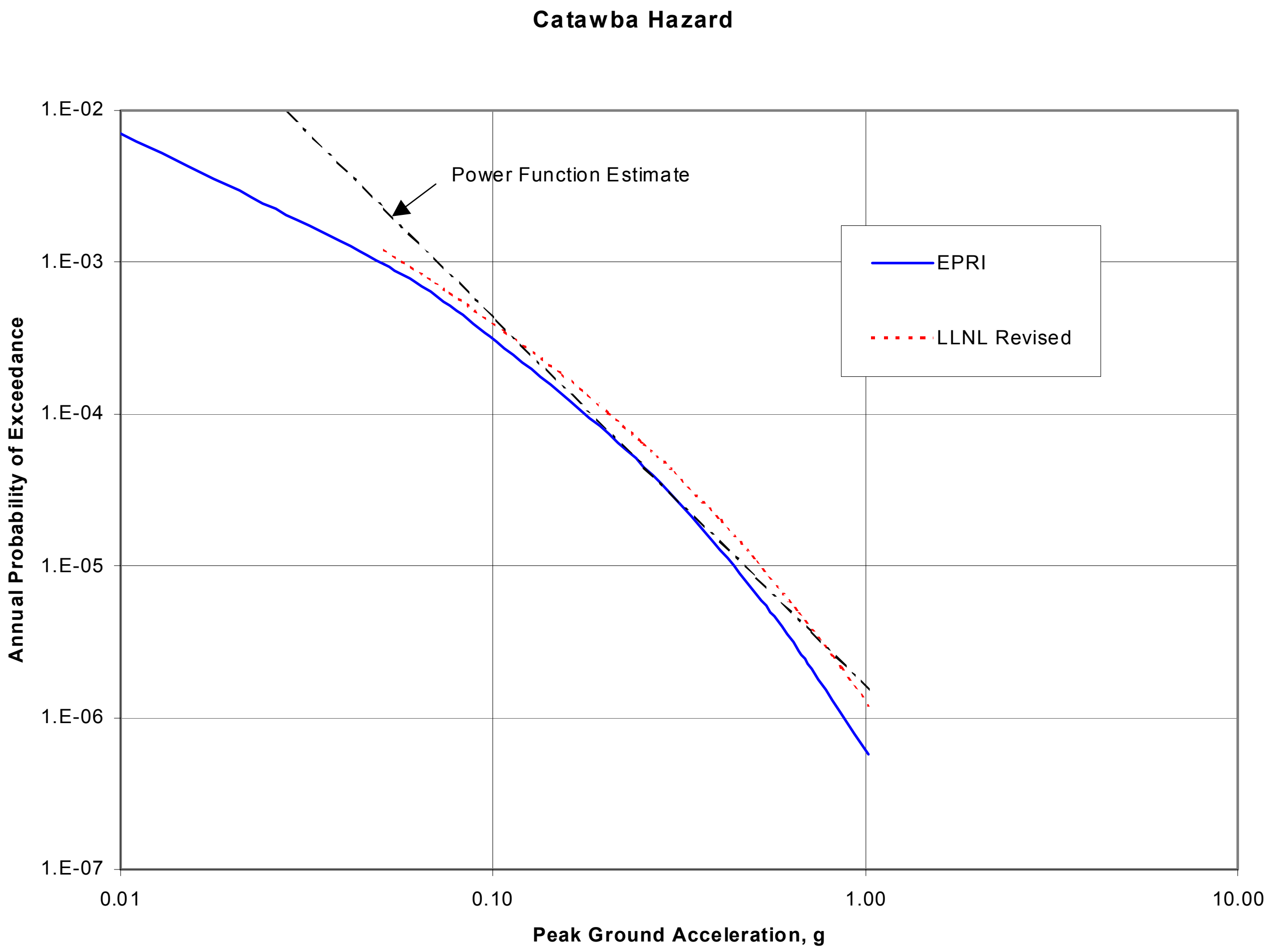

Figure 2-2
Catawba Mean PGA Hazard Curves

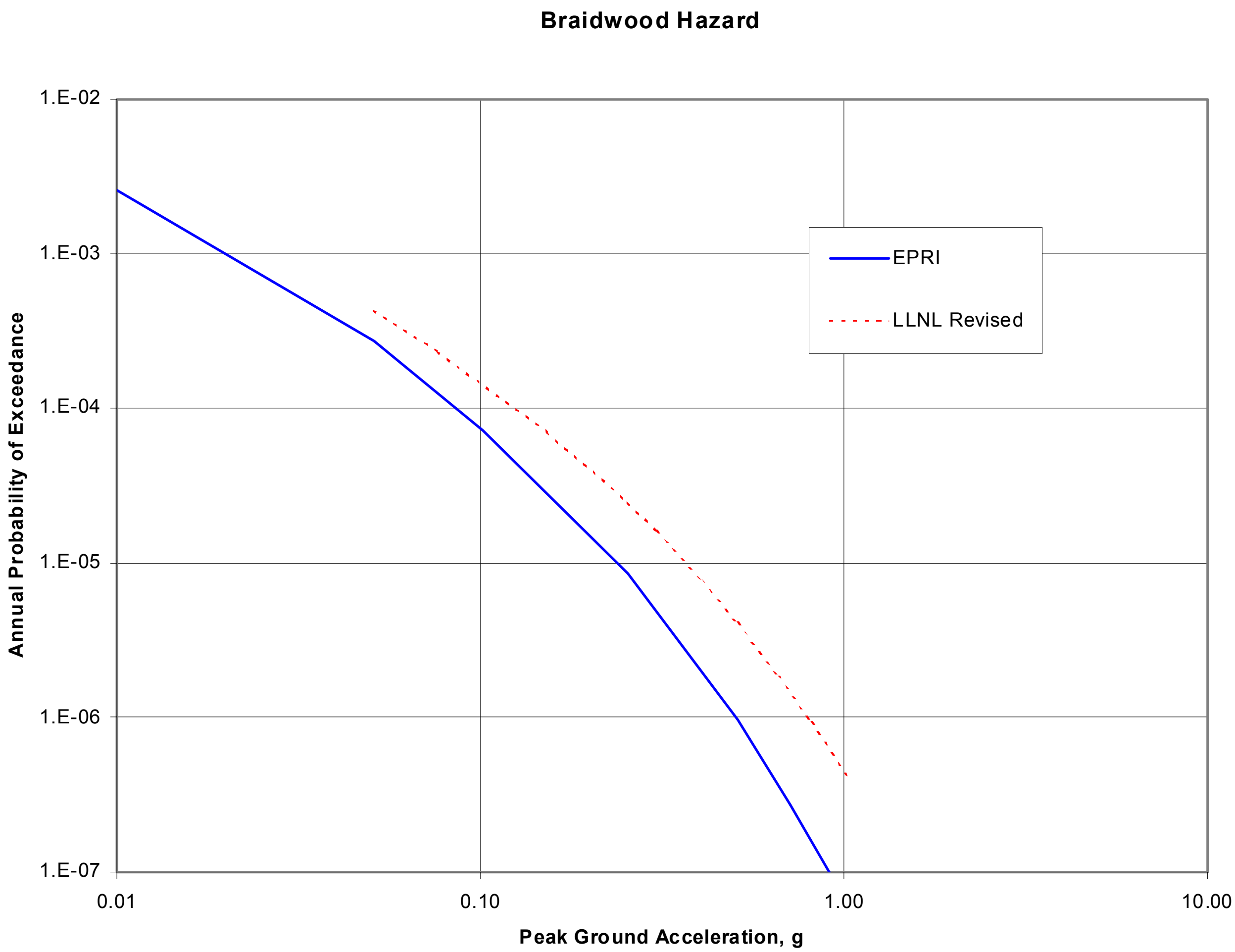

Figure 2-3
Braidwood Mean PGA Hazard Curves

3
SEISMIC MARGIN ASSESSMENT RESULTS

3.1 Background

Under the Individual Plant Examination of External Events (IPEEE) program, about 58% (61 units) of the operating nuclear power plants in the U.S. chose to perform an SMA as the assessment methodology for identifying the seismic vulnerabilities beyond the plant design basis.

The goal of a SMA is to obtain a High Confidence Low Probability of Failure (HCLPF) seismic capacity of a plant based on either the success path approach (EPRI Method) or a simplified plant Boolean (USNRC method). Ninety-seven per cent of the plants that chose the SMA option for the IPEEE program response used the EPRI method (Reference 12) of identifying two primary success paths for plant shutdown following the seismic event. Either of the methods results in an estimate of the plant level HCLPF expressed as the scaled Zero Period Acceleration (ZPA) or Peak Ground Acceleration (PGA) of a Reference Level Earthquake (RLE) response spectrum shape, which is always interpreted as being at the 84% non-exceedance probability level (NEP). The HCLPF for each component within a success path was estimated using the Conservative Deterministic Failure Margin (CFDM) method or screened-out, based on walkdown evaluations, as having a HCLPF greater than “X” g where “X” in most cases was the default level of 0.3g of the RLE assigned by the USNRC. According to Reference 12, the plant level HCLPF capacity is the lowest component HCLPF capacity in the more rugged success path. In most cases, IPEEE submittals did not report plant HCLPF values greater than the RLE level of 0.3g. For those plants which were placed in the Reduced Scope IPEEE bin (low seismic hazard), the plant HCLPF was only reported as being greater than the SSE.

The RLE selection guidance provided in Reference 12 emphasizes the use of a broadband spectrum shape, thus all SMAs are based on broadband, smooth spectral shapes, such as those presented in NUREG/CR-0098 (Reference 11) modified for site conditions. Reference 12 uses the Conservative Deterministic Failure Margin (CDFM) approach to estimate the 1% probability of failure levels for SSCs which are taken as equivalent to a HCLPF fragility level. For capacities of equipment based on qualification testing, Reference 12 takes the HCLPF capacity as the achieved test response spectrum (TRS) divided by a factor of 1.2.

As noted in the preceding section, both a full Level I SPRA and a SMA review were conducted for the Catawba units. Herein, the key SMA results (Reference 14) for the Catawba units are presented.

3.2 Typical SMA Characteristics

The EPRI SMA approach is based on identifying two primary success paths for plant shutdown following a seismic event taken as the RLE and then reviewing the equipment associated with each path for the effects of the RLE. There is considerable guidance provided in Reference 12 concerning the systems review required in order to select the two success paths. A primary assumption of the SMA system evaluation procedure is that offsite power is not available. While many of the discussions of the SMA approach emphasize the independence of the two success paths, they fail to note that this independence is for the frontline components of the dual paths. Each path is dependent upon the same support systems, particularly the on-site power sources. Reference 12 carefully notes that the two trains of on-site power are to be considered as one system which is in series with both frontline paths. Figure 3-1, shows the success path logic diagram for the Catawba SMA (Reference 14). As can be noted, the common support systems are identified as AC power, DC power, Service water, component cooling water, HVAC, and in the case of Catawba, the RPS and ESFAS are considered as support systems. Reference 12 requires that each SMA document the dependencies between the frontline and support systems in the form of a matrix of dependencies. A similar matrix of dependences is also required to be documented between support systems. The identification of these dependencies is the critical information that must be reviewed if a SMA is converted for RI/PB applications.

Since the control systems of current operating plants are relay logic based, the review of relays for seismic effects (chatter) can be an important factor in determining the overall seismic performance of a plant. However, only plants designated in the IPEEE program as Full Scope had full relay reviews conducted at the RLE level for SMAs. Plants designated as Focused Scope or Reduced Scope were assumed to have either design basis qualification of the relays or relays evaluated in the resolution of the A-46 issue. The implicit assumption was that plants with either a lower hazard or higher design basis would resolve any relay chatter based control anomalies using operator action. Focused Scope plants were required only to screen for "bad-actor" relays in the SMA review of IPEEE systems. No additional review of relays was required for SMAs conducted for Reduced Scope plants.

Catawba is designated as a Focused Scope plant. As a EPRI SMA demonstration plant, however, a full scope relay evaluation was conducted. The following section includes a summary of the Catawba SMA relay chatter results.

3.3 Evaluation of SMA Results

The plant HCLPF values reported (References 4 and 5) for the group of operating plants evaluated in the IPEEE program using SMAs range from design basis SSE to > 0.30g. The Full Scope plants had reported HCLPF values from 0.27g to > 0.30g (0.27g: 2 units, 0.28g: 1 unit, and > 0.30g : 4 units). The Focused Scope plants had reported HCLPF values from 0.16g to > 0.30g (0.16g: 2 units, 0.20-0.24g : 11 units, 0.25-0.29g : 9 units, and > 0.30g :18 units). The Reduced Scope plants were evaluated (except for 1 unit with a reported HCLPF of 0.26g) at their respective SSE levels and should not be interpreted as HCLPF values.

In this report, the seismic margin results for Catawba are to be used to derive estimated CDF values. The SMA results are fully documented in Reference 14 as a trial plant application of the

EPRI Seismic Margin Methodology. The following is a summary of the Catawba SMA results in terms of component HCLPF capacity values.

The overall conclusion of the Catawba SMA was that the plant HCLPF exceeded 0.30g. Walkdown screening of the Catawba SSCs was accomplished using the 0.3g PGA bin given in the Reference 13 screening evaluation tables. Since Reference 14 indicates that the Catawba RLE has a peak spectral acceleration (5% damping) of approximately 0.8g, the revised walkdown screening evaluation criterion given in Reference 12 is also satisfied.

Table 3-1 identifies the active electrical and control panels for which the Catawba SMA required verification of function. Reference 14 used the plant specific qualification test data to establish the SMA HCLPF capacity for each active electrical panel. An early version of the EPRI SMA guidance (Reference 13) was used for the Catawba SMA, which used the criteria TRS/1.3 as the test based HCLPF value. In general, all tests except for the Diesel Control panels, Nuclear Service Water Cabinet, Solid State Protection System (SSPS) panels, MCCs, and for the Area Termination Cabinets (ATC) were conducted at sufficient levels to demonstrate that 1.3 x the average RLE demand at each cabinet location was approximately equal to the TRS. For the tested panels, it was judged that the structural integrity of the panel was demonstrated by the testing level equivalent to 1.3 x the average RLE demand, however the function of the mounted control relays could not be verified at the 1.3 x RLE level. Therefore, a relay specific circuit evaluation was undertaken to ascertain if relay chatter could be tolerated. For the Diesel Panels and Nuclear Service Water Cabinet, the relay evaluation concluded that relay chatter would not affect controlled terminal devices. For the MCCs, a less conservative demand analysis indicated that the functional capacity of the MCCs could be verified at the 1.3 x RLE level, however, for the SSPS and ATC panels a certain small group of relays could not be screened in the circuit evaluation. Thus, for SSPS and ATC panels, the HCLPF was estimated as > 0.25 g. Since the Catawba SMA was a demonstration of the methodology, no further effort was undertaken to continue the relay evaluation on a contact-by-contact basis. It should be noted that at the time of the Catawba SMA conduct, the "clipping" evaluation method for comparing narrow-band test data and narrow-band demand had not been fully developed. This clipping procedure, which was incorporated in Reference 12, would have made the Catawba relay evaluation much easier.

The SMA HCLPF values for the active electrical and control panels are included in Table 3-1. The indicated HCLPFs have been factored by the ratio 1.3/1.2 (1.3/1.2 x 0.3 g ≈ 0.33 g) in order to make values reported in Reference 14 consistent with the guidance of Reference 12. The test level for the Diesel Generator Sequencer Panel was at 1.5 x RLE. The Station Batteries and Diesel Generator Battery Set do not appear in Table 3-1 since they were screened out based on test data at > 1.7 x RLE level. (The Diesel Generator Battery Set was actually replaced as a result of the SMA, since the original installation did not meet walkdown screening requirements).

Table 3-1
Catawba SMA Results

Component	SMA HCLPF Capacity, g	
	Integrity	Chatter
Station Battery Inverters	0.33	-
Station Battery Chargers	0.33	-
600V Switchgear	0.33	-
MCC	0.33	-
DG Sequencer	0.38	-
DG Engine Control Panel	0.33	NA
DG Generator Control Panel	0.33	NA
DG Neutral Grounding Panel	0.33	-
Auctioneering Diode Panels	0.33	-
Nuclear Service Water Cabinet	0.33	NA
4160V Switchgear	0.33	-
Main Control Boards	0.33	-
Solid State Protection System	0.33	0.27
Area Termination Cabinets	0.33	0.27

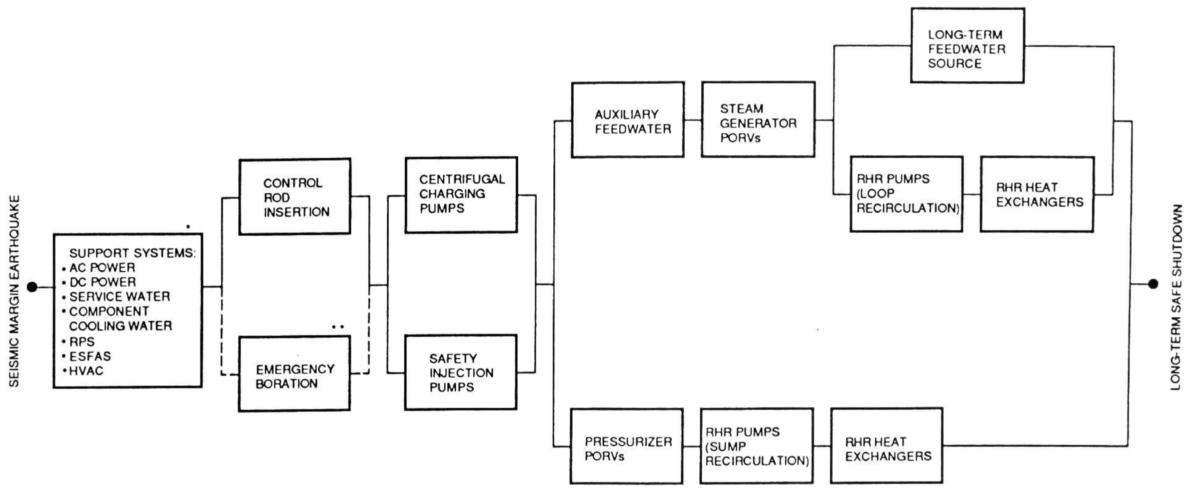

Figure 3-1
Catawba Success Path Logic Diagram

4
APPROACHES FOR OBTAINING CDF FROM SMA

4.1 Introduction

In the process of conducting an SMA, each plant chose two success paths for safe shutdown. One of the paths was required to mitigate a small break LOCA. If a component of a path had a HCLPF less than the RLE, then the HCLPF of that component became the defacto plant HCLPF. The SMA documentation at each plant contains a significant amount of information that can be used in developing risk estimates in RI/PB applications. The transformation of SMA results into PRA-type summary results has been discussed within the nuclear industry, and several technical papers and reports have been published which attempt to provide a general approach. Reference 3 identified four margin conversion approaches that have been presented. Recently, Reference 19 has been published which recommends one of these methods and provides simplified example applications. The following is a summary of the approaches that have been identified.

4.2 Review of Methods

The first method considered was developed by M.K. Ravindra and R.C. Murray as a part of the trial plant review of the NRC seismic margin methodology (Reference 15). In the NRC SMA approach, conservative screening procedures are utilized to focus on a set of components for which fragility functions are estimated, and Boolean expressions are derived for plant accident sequences. Using these expressions, a procedure is outlined where the plant level fragility curves are developed and convolved with the plant hazard curves to obtain the core damage frequency (CDF). For the EPRI margin approach, it is suggested in Reference 15 that simple Booleans be developed to represent the event "success" of the selected path and then inversed to obtain the failure expressions. The component fragilities are then developed using the HCLPF estimates. The fragility curves are convolved in accordance with the Boolean logic to yield the plant level fragility curves which are, in turn, convolved with the plant hazard curves to yield the CDF. For the EPRI margin method, it is postulated that the probability of success initially obtained for the two success paths is a lower bound on the actual success probability since there may be many more plant success paths not considered in the approach. Reference 15 emphasizes that it is important to establish whether the margin study considered all support system dependencies in the success path. It is pointed out that both EPRI and NRC margin methods focus on the margin against core damage and, thus, suffer from not considering different plant damage states. It is also suggested that it may be necessary to consider screening the components at a higher review earthquake level than the target RLE level for the plant in order that the screening level fragility does not mask the dominant contributors to CDF.

The second approach reviewed was developed by J.W. Reed and R.P. Kennedy, and is documented in Reference 16. This reference provides guidelines for obtaining various median component fragilities using deterministic CDFM estimates such as used in SMAs. HCLPF values used in a margin study are based on the assumption of 84% NEP earthquake structure response and, thus, are not the median based HCLPFs to be used to develop component fragility curves. Thus, the methodology within Reference 16 suggests that the margin HCLPF be denoted as $HCLPF_{84}$ and be converted to a median- based $HCLPF_{50}$ using the relation, $HCLPF_{84} = \exp(\beta_{rs})(HCLPF_{50})$, where β_{rs} is taken within the value range of 0.2 to 0.3 which results in factors of $\exp(\beta_{rs})$ ranging between 1.22 and 1.35. The approach recommends estimating the CDF from the SMA based plant $HCLPF_{50}$ by representing the mean hazard curve in the form of a power function (a power function plots as a straight line for log-log axes; see Figure 2-2 for a power function hazard representation). In this case, the convolution can be expressed in closed form and the sensitivity of the resulting CDF for various values of composite variability, β_c, examined. The basis for this representation is further discussed and expanded upon, in Reference 18 (the fourth approach considered below). The development of the plant level HCLPF from the safe shutdown paths and safe shutdown equipment list (SSEL) is not addressed.

The third approach considered for seismic margin conversion was developed by R. Sherry and is documented in Reference 17. This approach also represents the mean hazard curve in the form of a power function and obtains a closed-form solution for the CDF in terms of the plant SMA HCLPF and the hazard function parameters. Unlike the prior method discussed above, only the margin HCLPF value is used directly. The reported SMA HCLPF for five example plants was then used to demonstrate the procedure. The composite variability, β_c, was taken as 0.32 based on the review of several seismic PRAs. In addition, the effect of random failures and human errors were considered by factoring the SMA HCLPF by a value of 0.78, based on review of seismic PRA results. It is stated that this method is intended to yield a CDF accurate to only an order of magnitude.

The fourth seismic margin conversion approach was developed by R.P. Kennedy and is documented in Reference 18. This fourth approach expands upon the approach outlined in Reference 16 (the second approach noted above) and develops a method called the "Hybrid Method". Here the term Hybrid refers to a procedure intermediate between a SPRA and SMA. This approach requires some modeling of the plant systems to obtain Boolean expressions that represent plant damage states. The fragilities for the components in the Boolean expressions are convolved to obtain the mean plant damage state fragility that is then convolved with the mean hazard function to obtain a point estimate (mean value) of the CDF.

Using the EPRI Margin methodology, the plant damage state probability is represented in simplified Boolean form as:

$$DS = SP1 * SP2 \qquad \textbf{Eq. 4-1}$$

where SP1 is the failure rate of Success Path 1 and SP2 is the failure rate of Success Path 2. Reference 18 also recommends that nonseismic failures (random and human error) be included in the Boolean expressions. In simplified form this can be expressed as:

$$DS = (SP1+R1)*(SP2 + R2)*R3 \qquad \textbf{Eq. 4-2}$$

Here, R1 represents a random or human error failure of success path 1, R2 represents a random or human error failure of success path 2 and R3 represents a failure to recover from seismic, random or human error failure of both success paths. These random and human error failures and operator recovery actions are usually obtained from the internal events PRA model. Human error and recovery actions are then suitably modified to account for operator stress from the seismic event.

The estimation of fragilities is obtained by appropriate modification of CDFM margin capacities. The so-called CDFM capacity and HCLPF capacity are equivalent and represent a 1% failure probability capacity, or $C_{1\%} \approx C_{CDFM} \approx C_{HCLPF}$. The median capacity, $C_{50\%}$, can be directly estimated as, $C_{50\%} = C_{1\%}e^{2.326\beta_c}$, where β_c is the composite variability. A default value of $\beta_c = 0.4$ is recommended. Guidance is also provided to select a screening level that will yield a surrogate element that is not a dominant contributor to the seismic risk. In the case of a surrogate element, a composite variability value of $\beta_c = 0.3$ is recommended. This results in a median capacity of two times the HCLPF value.

A simplified Hybrid Method is also outlined in Reference 18 which is intended to be a procedure for obtaining quick manual estimate of seismic risk results. First, the EPRI SMA component CDFM capacities are determined. Then the simplified method uses damage state Boolean cut sets coupled with the HCLPF max/min procedure to obtain plant damage HCLPF capacity. In the max/min procedure, the lowest HCLPF capacity component in a safe shutdown path represents the capacity of the path (weakest link in the chain concept). The success path with the higher HCLPF capacity represents the plant HCLPF capacity. Next, the plant damage HCLPF capacity is used with an assumed composite variability ($\beta_c = 0.3$) to obtain a CDF estimate using a simple closed-form convolution procedure. Random failures and human factors are considered by reducing the affected component HCLPF values (similar to Reference 17) both before and after the application of the max/min procedure to the two success paths to obtain a modified damage state fragility.

R.J. Budnitz and M.K. Ravindra, (Reference 19), have used a simplified application of the Hybrid method of Reference 18 to two case studies using actual plant SPRA and SMA results to develop "pseudo-CDF" values. Based on the generic insights gained from the two case studies, Budnitz and Ravindra have identified the characteristics of a methodology for deriving SPRA-type results from an SMA. Three general categories of plants with an existing SMA are proposed:

1. The plant HCLPF capacity is found to be less than the RLE, and the SMA has explicitly identified one or more SSCs on the Success Paths having HCLPF capacities less than the RLE.

2. The plant HCLPF capacity is found to be higher than the RLE, and all of the SSCs on the Success Paths are screened out with HCLPF capacities equal to or greater than the RLE.

3. A Reduced Scope SMA was performed, using the plant's SSE as the RLE.

The first step in all three categories is to convolve the mean hazard function defined for the plant site (References 7 and 8) and the mean plant fragility curve estimated from the plant $HCLPF_{50\%}$ and a generic variability value, taken as $\beta_c = 0.3$, to obtain a best estimate of the CDF (this is the simplest application of the Hybrid method). If the mean CDF is less than 1.0E-6 per year, then the seismic contribution to plant risk is judged to be low from a risk-informed perspective

(Reference 20). According to Reference 20, for risk informed applications, a change in CDF of less than 1E-6 per year resulting from a change in the plant condition will be considered as acceptable. Thus, one can imply that, as long as the plant level HCLPF results in a CDF of less than 1E-6 per year and any change in a plant component condition results in a HCLPF equal to or greater than the plant level HCLPF, the change in risk is within acceptable limits.

If the calculated seismic risk is larger than 1.0E-6 per year, a more "realistic" plant model is needed. For Group 1 plants, Budnitz and Ravindra suggest that a mini-SPRA be conducted using approximately 20 SSCs identified in the SMA, selected by the ranking of the HCLPF values. The steps are as follows:

1. Determine the family of component fragility curves including full variability due to randomness, $\beta_{R,}$ and uncertainty, β_U, in accordance with Reference 16.

2. Using a transient event tree from the internal event PRA, identify the cut sets that contain the seismic failures identified in the success paths of the SMA. The cut sets are grouped into two categories; those that lead directly to core damage and those that have nonseismic failures and human actions associated with the cut sets.

3. Using the seismic fragilities and the nonseismic unavailabilities and human errors, quantify the cut sets to obtain damage state conditional probabilities of failure.

4. Convolve the family of seismic hazard curves with the family of damage state fragility curves to obtain the seismic induced CDF. The seismic CDF is the sum of the CDF contribution of each cut set.

Plants in Group 2 would identify a set of SSCs that have HCLPF capacities marginally higher than the RLE, or those known to be important from past SPRAs. Group 3 plants would need to identify a set of SSCs that have HCLPF capacities marginally higher than the SSE. Both of these Groups would then proceed to conduct a mini-SPRA in the same manner as suggested for Group 1 plants.

4.3 Summary and Issues

4.3.1 Summary

Of the four approaches for SMA conversion reviewed above, the Hybrid method outlined in Reference 18, as applied in Reference 19, appears to be the most practical approach presented for this focused purpose. In general, all of the approaches have common threads or important considerations that can be identified.

- Reasonable estimates of fragility curves can be developed from component HCLPF values calculated in an SMA by applying a generic value of the composite variability ßc.
- Booleans of the plant failure sequences need to be developed to properly combine the component fragilities with nonseismic failures and human errors to develop the conditional probability of CDF for identified cut sets.

- Point estimates (mean values) of risk are obtained by convolution of the mean hazard curve with the mean fragility curve for the cut sets.
- The simplified approaches which avoid consideration of plant specific Booleans are useful as screening tools for deciding which plants need to consider a more focused approach for CDF estimation based on SMA results.

Budnitz and Ravindra suggest that plants with SMAs be categorized into three Groups: Plant $HCLPF_{84}$ < RLE, Plant $HCLPF_{84} \geq$ RLE, and Plant $HCLPF_{84}$ = SSE. If the plants do not meet a simple screening level CDF based on a point estimate, they suggest that the more traditional SPRA procedures be used on a limited sub-set of SSCs, identified trough a systems review, in order to obtain a realistic CDF value.

4.3.2 Issues

In discussing the EPRI margins method, all of the approaches place undue emphasis on the independent nature of the dual success path approach. If one identifies SUP as the failure of support systems such as power, service water, etc. and FLi as the failure of front line systems of a selected success path, we may write the Booleans for each success path failure, SPi, as:

$$SP1 = SUP + FL1 \qquad \textbf{Eq. 4-3}$$

$$SP2 = SUP + FL2 \qquad \textbf{Eq. 4-4}$$

The plant damage state, or core damage Boolean, is then the intersection of the two success path failure states

$$CD = SP1*SP2 = (SUP + FL1)*(SUP + FL2) = SUP + FL1*FL2 \qquad \textbf{Eq. 4-5}$$

where the support system, SUP, is common to each success path due to common cause and system dependencies. Unless the support systems have high component HCLPF capacities, the support system failures will usually dominate the risk. This result is confirmed in a number of SPRAs. In general, for plants with the SMA HCLPF based on two success paths, it will be necessary to consider realistic Booleans that include the support systems.

In this regard, the two plant case studies considered in Reference 19, represent two extremes of SMA results. In case 1, a very high screening level was used (0.5g) for a plant designated as requiring an SMA at an RLE of 0.3g. The plant HCLPF was then 0.5g which resulted is a very low point estimate CDF of 2.5E-7 per year. In this case, the SPRA that was developed by the utility from the internal event PRA and the SMA HCLPFs showed that the fragility for the surrogate element contributed 60% of the CDF. Using the above simplified Boolean the max/min theory would have a surrogate associated with the support system and the predicted CDF would have been underestimated but would have been within a factor of 2 of the CDF predicted with a more complete model.

In case 2 of Reference 19, the Surry NPP was studied. The lowest HCLPF capacity was for the turbine building. Its failure goes directly to core damage. In the IPEEE SPRA this failure contributed 36% to total calculated core damage. LOSP and nonseismic failure of the diesel generator contributed 32% of the calculated CDF. Surry is unique in that each unit has one seismically qualified diesel generator with one swing diesel shared by both units. The swing diesel power routing goes through a nonseismically qualified building so in effect, a single diesel is available for each unit and random failure becomes more important. The emergency feed water tank and the component cooling water surge tank had HCLPFs below the 0.3g screening level and their failures contributed 26% of the calculated CDF. The plant level HCLPF was calculated to be 0.155g, not much lower than that of the lowest capacity structure. A fragility representing the 0.3g screening level resulted in an unconditional failure rate of about 1.2E-6 but a surrogate element or elements representing the screening level was not included in the IPEEE SPRA model. The calculated CDF was 8.2E-6 per year. Adding in a surrogate would increase the CDF to about 9.4E-6 per year.

In the Reference 19 study, a simple Boolean consisting of the 0.19g HCLPF structure and the 0.26g HCLPF component cooling water surge tank represented the plant response to earthquakes. The simple model did not include the availability of offsite power but ignored the surrogate. Failure of the emergency feed water tank does not go directly to core damage so ignoring it with a HCLPF less than 0.3g was not significant. For fragilities based on an assumed ßc of 0.3, the Reference 19 calculated CDF was 1.1E-5 and was 7.6E-6 per year for an assumed ßc of 0.4. The SPRA CDF including the surrogate contribution is between these values.

In this case, most of the CDF resulted from two components in the SPRA and in the SMA model. A simple max/min analysis derived from the Reference 19 results would result in a CDF of 7.9 E-6 per year for an assumed ßc of 0.3. This is about 84 % of the CDF calculated in the SPRA if the contribution of a surrogate element had been added. In this case, if one only wants to estimate the level of CDF, a simplified hybrid max/min method, as discussed in Reference 18, provides a close approximation. This should be the general case if one or two low capacity components are singletons. However, in this case, since the screening level resulted in a seismic induced failure rate greater than 1E-6 per year, the change in CDF for any change in a component capacity that was included in the screening cannot be identified and the risk informed goal in Reference 20 for less than a 1E-6 per year change in CDF cannot be demonstrated without some further analysis.

In most cases SMAs resulted in all components having a HCLPF equal to or greater than 0.3g. Most components were screened at 0.3g so there are few reported HCLPFs greater than 0.3g. Consequently, the surrogate element for screened out components would likely have a failure rate greater than 1E-6 per year and a reasonable representation of the plant response to earthquakes would be required to estimate seismic induced CDF and changes in CDF. In order to make risk informed arguments as outlined in Reference 20, a reasonably accurate estimate of seismic CDF (e.g., conservatisms involved in considerations of the surrogate element for screened out components may need to be removed) may be required to demonstrate that risk from all internal and external events would meet the 1E-4 per year requirement. In general, the screening of equipment may need to be revisited if a surrogate element representing the screened-out components has a seismic induced failure rate greater than 1E-6 per year.

5
APPLICATION OF CONVERSION APPROACHES

5.1 Introduction

Reference 18 suggests that there is no need to use Monte Carlo type simulations to propagate uncertainty in both hazard and fragility in determination of seismic risk. Point estimates (mean values) of seismic risk are considered sufficient. This was the approach recommended in NUREG-1407 for IPEEE SPRAs. Fragility curves for components to be included in the plant systems model are recommended to be calculated using the SMA HCLPF values and assuming a generic composite uncertainty. This, in effect, results in a best estimate or mean fragility curve. Mean CDF can be calculated by convolution of the mean hazard function with the mean plant level fragility. Reference 19 suggests that as a first step a simple point estimate, based on the minimum HCLPF level, can be conducted as a useful screening tool. If the CDF from the minimum HCLPF, assumed to be the screening HCLPF, is less than 1E-6 per year, then the CDF from any element is very small and further evaluation is not necessary. This is the max/min concept of success path logic. If this simple screen results in greater than 1E-6 per year failure rate, more realistic methods are suggested for determination of risk. Reference 19 suggests that a mini SPRA systems model be developed but that a full uncertainty analysis by Monte Carlo type simulations be conducted. Since such simulations require specialized software (several are available), this report will not further discuss this recommendation, but rather focus on point estimate computation methods, since these can be accomplished with readily available spreadsheet software.

The following subsections first discuss convolution methods followed by an application of the convolution methods for a Boolean expression developed from Catawba SMA information. The CDF developed in this manner is compared to that developed in the IPEEE SPRA. Parametric analyses are also conducted for the SMA based model to compare to the parametric cases conducted in Section 2 using the IPEEE SPRA model.

5.2 Simplified Convolution

Given that $P_{CD/a}$ represent the conditional core damage probability computed using the conditional event probabilities combined in accordance with a plant system Boolean expression, then the core damage frequency is given by the convolution integral:

$$CDF = -\int_0^{\infty} (P_{CD/a})(dH/da)\, da \qquad \textbf{Eq. 5-1}$$

where H is the hazard function. In this form, the integral appears to be simple, however, the $P_{CD/a}$ term represents the total probability of the accident sequences combined as indicated in the plant Boolean. For example, if the core damage Boolean represents the union of two independent component failures, or CD = A + B, where A represents the failure of A with mean fragility function $P_{A/a}$ (the conditional probability of failure of A given the PGA value a), and B represents the failure of B with mean fragility function $P_{B/a}$, then the probability of A or B failing is given by the algebraic equation

$$P_{CD/a} = P_{A/a} + P_{B/a} - P_{A/a}P_{B/a} = 1 - (1- P_{A/a})(1- P_{B/a}) \qquad \textbf{Eq. 5-2}$$

In general, the functions $P_{A/a}$ and $P_{B/a}$ are integrals, thus the function $P_{CD/a}$ can only be represented as a discrete numerical function. The term dH/da is the derivative of the mean hazard function, which is defined in References 7 and 8 as a function of a=PGA using only seven to ten tabular values. Clearly the derivative is an approximation based on the curve used to fit the tabular values. However, the core damage frequency can also be expressed by the alternate convolution integral form:

$$\mathrm{CDF} = \int_0^\infty H\,(dP_{CD/a}/da)\,da \qquad \textbf{Eq. 5-3}$$

Now, we note the derivative of a lognormal fragility function may directly written as be

$$dP_{A/a}/da = \left[1/(\sqrt{2\pi}\,\beta_c a)\right] e^{-t_0^2/2} \qquad \textbf{Eq. 5-4}$$

where t_0 =[ln(a/â)]/β_c and â is the median component capacity.

Next consider a plant Boolean which is a union of several singleton component failures, such that

$$\mathrm{CD} = \bigcup_{i=1}^{n} A_i \qquad \textbf{Eq. 5-5a}$$

$$P_{CD/a} = 1 - \prod_{i=1}^{n}(1 - P_{A_i/a}) \qquad \textbf{Eq. 5-5b}$$

Now

$$dP_{CD/a}/da = \sum_{j=1}^{n} \left[(dP_{Aj/a}/da)\prod_{k \neq j}^{n}(1 - P_{A_k/a})\right] \qquad \textbf{Eq. 5-5c}$$

If we note that for small $P_{A/a}$ the product $\prod_{k \neq j}^{n} (1 - P_{A_k/a}) \approx 1$, then we may write

$$CDF \leq \sum_{i=1}^{n} \int_{0}^{\infty} H\,(dP_{A_i/a}/da)\,da = \sum_{i=1}^{n} \Delta CDF_i \qquad \textbf{Eq. 5-6}$$

Now, if we use Equation (5-4) as a weighting function for the hazard, we may directly compute each ΔCDF_i integral by simple numerical quadrature of the weighted hazard function. Thus, the contribution of each component failure to the CDF is easily found by entering the hazard values into a spread sheet as a function of a, then computing the weighted hazard for each value, and then simply numerically integrating the resulting integrand as a function of a. If one wishes to determine the effect of the union (usually a 20-30% decrease in CDF), the actual value of the product terms, $\prod_{k \neq j}^{n} (1 - P_{A_k/a})$, can be determined as an additional weighting factor prior to numerical integration. It should be noted that a similar development can be shown for a doubleton and unions of singletons and doubletons. Equation (5-4) is also the basis of the closed form expression for ΔCDF found in the literature (References 16 and 17) assuming the hazard function can be represented as a power function. See Reference 18 for a complete derivation of the closed form point estimate of ΔCDF.

5.3 Application of Methods

According to Reference 19, the first step in converting a SMA to risk format is to obtain a point estimate of the CDF using the reported SMA $HCLPF_{84}$. This is an application of the max/min method wherein the plant level HCLPF is defined by the lowest HCLPF determined in the SMA for the highest capacity success path. In cases, such as for Catawba, where all component HCLPFs are 0.3g or greater, the plant $HCLPF_{84}$ = 0.3g. Reference 16, indicates that for a value of β_{rs} = 0.2, or $e^{\beta_{rs}}$ = 1.22, we may estimate

$$HCLPF_{50} = HCLPF_{84}/1.22$$

Table 5-1 provides an example of the spreadsheet convolution of Equation (5-3) using the EPRI hazard function for Catawba (Reference 7) for a plant $HCLPF_{84}$. = 0.3g and β_c = 0.3. The columns denoted If(a) and I*f(a) represent different numerical quadrature methods with If(a) being the result of assuming the integrand is log-linear function while I*f(a) is the application of the trapezoidal rule. We will take the average of these numerical quadrature results as the CDF. For the Catawba plant, the CDF based on the SMA HCLPF = 0.3g is, CDF ≈ 9.7E-6 > 1.0E-6 per year. If all one is interested in is an approximation of the CDF, this value is about 2/3 of the 1.46E-5 per year value calculated from the complete seismic PSA model (Refer to the base case in Table 2-4). For risk informed applications, a reasonable representation of the plant response to seismic events needs to be developed. We note that since 0.3g is the SMA screening level for the Catawba units, this would also be the contribution of a surrogate element to the Catawba CDF. It is of interest to know what level of earthquake is the most significant contributor to the overall seismic risk. Figure 5-1 provides plots of the function If(a) and the Hazard function for

Catawba. This type of comparison indicates the range of acceleration that contributes most to the CDF convolution.

Table 5-2 compares the estimated CDF, based on $HCLPF_{84}$. = 0.3g and $\beta_c = 0.3$, computed for the Catawba, Braidwood, and River Bend EPRI Hazard. The effect of Sherry's (Reference 17) estimation procedure using a HCLPF reduction factor for human errors (HE = 0.78) is also shown in Table 5-2 for the Catawba site. Referring to Figure 2-1, we note that only the very low-seismicity US plants will meet the screening level CDF value of 1.0E-6 for a SMA plant HCLPF = 0.3g. Since the low-seismicity plants were Reduced Scope in the IPEEE program, the SMA HCLPF was reported as the SSE. It is therefore unlikely that any US plant with a SMA HCLPF ≤ 0.3g will meet the target screening level CDF value of 1.0E-6 per year suggested in Reference 19 or as implied by Reference 20 to determine ΔCDF for risk informed applications. Since the CDF point estimate based on the Catawba SMA HCLPF did not meet the 1E-6 per year threshold, we must make a more detailed estimate of the Catawba CDF.

Since all of the SSCs on the Success Paths are screened out or have HCLPF capacities equal to or greater than the RLE (except for some relays which have a high rate of recovery for chatter effects), the Catawba SMA falls into Category 2 of Reference 19. In this case, we need to identify a set of SSCs that have HCLPF capacities marginally higher than the RLE that are known to be important from past SPRAs. Upon examination of the reported SMA HCLPFs in Table 3-1, we note that all of the components listed are part of the support systems power supplies or are control systems. Thus, in accordance with the simplified Boolean in Equation 4-5, it is evident that the CDF will be totally dominated by these electrical and control components. Without delving into the internal event PRA to understand how these electrical and control support system components interrelate with plant system functions, we can take a conservative approach that all of the components in Table 3-1 are singleton cut sets. In a specific plant model, this would not necessarily be the case and this assumption is conservative. In the case of relay chatter in the SSPS cabinets and the Area Termination Cabinets, a reasonable rate of recovery can be assumed. The failure to recover from relay chatter was taken as 0.1. likewise, the loss of the main control board can be compensated for by the operators going to the alternate shutdown panel. The rate of operator failure in this case is also taken as 0.1. Nonseismic failures and other human error failures were not considered in the SMA based risk assessment. The parametric studies shown in Table 2-4 using the SPRA model indicate that the results are not particularly sensitive in this case to these nonseismic failures. The modeling of the nonseismic failures would also have entailed a considerable effort to review the internal event model and develop accident sequences containing the SMA components and associated nonseismic failures.

Based upon the above logic, the following Boolean sequence of events was developed.

$$\begin{aligned} CD = {} & INV+BCH+600AC+MCC+DGSEG+DGECP+DGCP+DGNP+ADP \\ & +NSWC+4160AC+SSPS+ATC+OP2{*}MCB+OP1{*}SSPCc+OP1{*}ATCc \\ & +Surrogate \end{aligned} \qquad \textbf{Eq. 5-7}$$

The median fragility for each component event was computed as:

$$Am = HCLPF_{50}\, e^{2.326\beta c} C_I \qquad \textbf{Eq. 5-8}$$

where C_I is the capacity increase factor defined in Reference 16 for test based fragility. Note that all of the component HCLPFs in Table 3-1 are based on the achieved test level. Review of the Catawba SMA indicated that there was conservatism in the demand used for the SMA evaluations of some components, but a revised demand was used to define the HCLPF for some of the lowest capacity components (relay chatter). Therefore, the demand conservatism factor for the fragility was taken to be unity.

The seismic hazard was convolved with each of the fragilities of the components in the Boolean. Table 5-3 lists the fragilities and seismic failure probabilities for the above Boolean. Simply summing the CDF contribution for each event, we obtain a CDF estimate (upper bound) for the Catawba units based on the SMA HCLPFs as CDF = 5.9E-5 per year which includes the surrogate element contribution. The surrogate failure rate of 9.7 E-6 per year is about 16% of the calculated CDF. The CDF calculated by this simple and conservative SMA risk based representation of the plant response to seismic events is about 4 times the CDF calculated in the IPEEE SPRA as shown in Table 2-4.

Table 5-4 lists the CDF derived for the Catawba SMA Boolean using the Braidwood seismic hazard. The calculated value of 9.08 E-6 per year is about 4.2 times the 2.17 E-6 per year base case shown in table 2-4 for the Braidwood hazard. This ratio is slightly higher for Braidwood, probably because the success of offsite power is not modeled and for lower seismic hazard sites, the success of offsite power can be more helpful.

In examining the results of parametric studies shown in table 2-4, some of this difference is logical. Table 2-4 is reproduced as Table 5-5 with comparable results, where applicable, for parametric studies conducted for the SMA based risk analysis.

Referring to Table 5-5, the SPRA CDF for case 1, assuming complete LOSP, was 3.55E-5 per year, thus the estimated SMA CDF of 5.9E-6 per year is approximately 65% greater than the base case SPRA CDF. This reduces the ratio between the SMA based risk assessment CDF and the SPRA CDF to 1.66. If LOSP were to be explicitly modeled in the SMA based risk model the SMA based risk estimate would reduce and the ratio of SMA based CDF to SPRA CDF, with LOSP modeled in each case, would likely be similar. A similar case for the Braidwood hazard was not conducted. We would expect a larger difference since the probability of LOSP is less for the lower seismic hazard site and the SPRA would capture this success wherein the SMA based risk assessment is based on off site power not being available. According to Reference 18, this difference (a ratio of 1.66) between SPRA and an SMA based risk assessment is within the expected range of accuracy for point estimates. This case though would indicate that in a SMA based risk assessment, the availability of offsite power should be modeled. This would be especially true for plants in low seismicity regions and for plants that have two or more sources of offsite power such as say a 110kV and a 230kV source that are independent of each other.

An observation that was not quantified is that the ratio of the median capacity to HCLPF capacity in the SPRA is generally greater than the ratio arising from using the conservative methods in Reference 18 for estimating fragility functions from HCLPF values. As shown in Reference 18, the choice of ßc for estimating median capacity from HCLPF values is not particularly sensitive in determining failure rates, the failure rate is more sensitivity to HCLPF than median, but many things can add up to make a more pronounced difference between simple modeling and more rigorous modeling. More detailed development of fragilities that dominate seismic risk is a recommendation in Reference 18 and should be considered in SMA based risk assessments where benefit can be gained to achieve the final goal in a risk informed application.

In Table 5-5, the base case and cases 2, 10 and 11 for the SMA based risk assessment are the same since the nonseismic failures are not modeled. Case 1 is also the same as the base case since LOSP is not added to the SMA. The only human action, except for relay recovery, is for the operators to go to the remote shutdown panel in the event of failure of the main control boards. Consequently for Case 3 there is little difference from the base case in the calculated SMA based CDF. A similar conclusion is reached for Case 4 in Table 5-6 for the Braidwood hazard. There are only two components in the SMA based risk model with relay chatter and recovery action. Varying the recovery rate in Cases 4 and 5 for the Catawba hazard has little effect on the computed SMA based CDF wherein in the complete SPRA model, the changes are more significant. The same conclusion can be drawn from Cases 5 and 6 in Table 5-6 for the Braidwood hazard.

As is the case for the SPRA model, cases were conducted for assumed SMA HCLPF values of 0.15g for the RWST and for selected electrical and control components qualified by test. Table 5-7 shows the seismic failure rates for 0.15g components subjected to the Catawba and the Braidwood seismic hazards. Note that the capacity increase factor, CI, is applied to components qualified by test as opposed to those such as the RWST that is qualified by analysis.

In Case 6 of Table 5-5 for the Catawba hazard, if failure of the RWST is assumed to lead to core damage and its HCLPF capacity is set to 0.15g, the SMA based CDF increases by 41% as opposed to only a 5% increase in the SPRA model analysis. This is because; the RWST would normally not be a singleton and is obviously not one in the Catawba SPRA. In this case, the simplifying assumptions made in the SMA based analysis are obviously not correct and are too conservative. The ratio between the SPRA and SMA analysis results is about 5.4 for this case. This would indicate that the combination of excluding offsite power success and improperly modeling success path components that are not singletons in the SMA based risk analysis can significantly over predict the risk.

In Case 7, assuming that three electrical and control components have HCLPFs of 0.15g increases calculated CDF by 75% in the SPRA model wherein the increase in CDF in the SMA based risk model is 55%. The difference is likely due to the fact that the string of 14 events in the SMA based risk model that lead directly to core damage is overly conservative and increasing the failure rate of two of them that lead directly to core damage has less effect than in the SPRA where specific modeling of plant response is conducted. In this case, the components selected for an assumed reduced capacity are more sensitivity in the SPRA model. For the Braidwood hazard, the analogous case is Case 2 and the increases in the SPRA and SMA based CDF are 82% and 69%, respectively. The gap is closer here, likely because the weaker components for a low hazard site are more dominant to the calculated risk.

Cases 8 and 9 in Table 5-5 combine low capacity components with 100 % success of human actions and 0 % nonseismic failures (Cases 3 and 2). In Case 8, the change in CDF is greater in the SPRA since there is only one human action modeled in the SMA based risk assessment (operator goes to remote shutdown panel after failure of the main control boards). Case 9 for the SMA based risk assessment is the same as Case 7 (nonseismic failures are not addressed in the

SMA based evaluation), so no change is calculated. These cases were not evaluated for the Braidwood hazard.

5.4 Issues and Conclusions

It is unlikely that any US plant with a SMA HCLPF of 0.3 g developed by screening will meet the screening level CDF value of 1.0E-6 suggested in Reference 19, or implied in Reference 20 to show less than 1E-6 per year change in CDF. For reduced scope plants, the SMA HCLPF was usually defined as the SSE level and the same conclusion likely holds true. Consequently, we anticipate all plants with SMAs may require additional effort and some systems modeling to obtain justifiable CDF values and to demonstrate CDF differences for RG 1.174 risk informed changes to the licensing basis or other risk informed applications. We believe that the simplified hybrid method produces justifiable CDF values for risk informed applications.

If all one wants to do is demonstrate approximate CDF, the screening level may be adequate. From the cases examined in Reference 19 and the Catawba evaluation conducted herein, a simple hybrid max/min analysis such as described in Reference 18, may accomplish this within a factor of two. If a more accurate and detailed evaluation is desired or required for RG 1.174 risk based evaluations, the plant response to seismic events should be modeled in more detail taking into account the availability of offsite power and including any significant nonseismic failures and human actions. The example presented in this study shows the excess conservatism that can result from just assuming that any component in the safe shutdown path can lead directly to core damage.

Most SMA studies have utilized RLE response spectra shapes which are representative of past Western United States (WUS) 84% NEP earthquake motions with low-frequency (3-8 Hz) emphasis while the current Uniform Hazard Spectra (UHS) estimated in References 7 and 8 for Central and Eastern US (CEUS) sites have high-frequency (18-25Hz) emphasis. These CEUS UHS spectra are much less damaging to structures and equipment due to the much lower velocity and displacement associated with them. Development of component fragilities based on the site specific UHS for use in risk informed applications could make a significant difference in the calculated CDF results. The Catawba SPRA was based on a similar spectra shape as used for the SMA. Therefore, the results can be directly compared. In general, procedures need to be developed to convert the HCLPF capacities based on low frequency broad band RLE's to capacities based on the UHS which is representative of the hazard defined for a given CEUS plant site. This has been done in some instances for rock sites where the modal response of structures is governed by the dynamic characteristics of the structure. For soil sites, the response of a structure to an earthquake is much more complicated and, per References 12 and 16, scaling is not recommended for soil sites. However, depending upon the risk informed applications, approximate scaling may be justified.

Table 5-1
Catawba CDF based on SMA HCLPF(84) = 0.3g and β_c = 0.3

C_I (1)	HCLPF(84) (2)	BETAC (3)	Am (4)
1	0.3	0.3	0.494

n	a(g) (5)	t_0 (6)	PF/a (7)	d(PF/a)/da (8)	H (9)	f(a) (10)=(8)(9)	Δ a (11)	If(a) (12)	CUM CDF (14)	I*f(a) (13)
1	0.005	-15.247	0.000E+00	8.576E-49	1.600E-02	1.372E-50				
2	0.051	-7.572	1.854E-14	9.244E-12	9.700E-04	8.967E-15	0.046	4.99E-18	4.99E-18	2.06E-16
3	0.102	-5.262	7.153E-08	1.270E-05	3.000E-04	3.810E-09	0.051	1.50E-11	1.50E-11	9.71E-11
4	0.150	-3.974	3.535E-05	3.298E-03	1.348E-04	4.447E-07	0.048	4.45E-09	4.47E-09	1.08E-08
5	0.200	-3.015	1.285E-03	7.059E-02	7.432E-05	5.246E-06	0.050	9.73E-08	1.02E-07	1.42E-07
6	0.255	-2.207	1.365E-02	4.566E-01	4.500E-05	2.055E-05	0.055	6.15E-07	7.16E-07	7.07E-07
6	0.330	-1.346	8.918E-02	1.629E+00	2.187E-05	3.563E-05	0.075	2.06E-06	2.78E-06	2.11E-06
7	0.400	-0.705	2.405E-01	2.594E+00	1.278E-05	3.316E-05	0.070	2.41E-06	5.18E-06	2.41E-06
8	0.510	0.103	5.411E-01	2.595E+00	6.500E-06	1.687E-05	0.110	2.64E-06	7.83E-06	2.74E-06
9	0.600	0.647	7.412E-01	1.798E+00	3.758E-06	6.757E-06	0.090	9.98E-07	8.82E-06	1.07E-06
10	0.714	1.225	8.897E-01	8.803E-01	2.100E-06	1.849E-06	0.114	4.30E-07	9.25E-06	4.89E-07
11	1.019	2.414	9.921E-01	7.087E-02	5.700E-07	4.039E-08	0.306	1.45E-07	9.40E-06	2.89E-07
12	1.200	2.957	9.984E-01	1.397E-02	3.139E-07	4.387E-09	0.181	2.93E-09	9.40E-06	4.04E-09
13	1.500	3.701	9.999E-01	9.396E-04	1.388E-07	1.305E-10	0.300	3.63E-10	9.40E-06	6.78E-10
							Sum	**9.40E-06**		**9.97E-06**

(1) $C_I \equiv$ Capacity Increase Factor (used for test based fragility)
(2) $HCLPF_{84} \equiv$ HCLPF value based on 84% NEP earthquake structure response
(3) β_c = BETAC $\equiv$ composite lognormal variability
$HCLPF_{50} = HCLPF_{84}/1.22$
(4) $\hat{a} = Am = [HCLPF_{50}][e^{2.326\beta c}][C_I]$
(5) a(g) $\equiv$ peak ground acceleration
(6) $t_0 = [\ln(a/\hat{a})]/\beta_c$ (fragility parameter)
(7) PF/a = Conditional probability of failure for given value of a(g) (cumulative normal distribution of t_0)
(8) $d(PF/a)/da = \left[1/(\sqrt{2\pi}\,\beta_c a)\right]e^{-t_0^2/2}$
(9) H = EPRI Mean Hazard (function of a(g))
(10) f(a) = H[d(PF/a)/da]
(11) $\Delta a_n = a_n - a_{n-1}$
$k_n = [\log(a_n) - \log(a_{n-1})]/(\Delta a_n)$

(12) $If(a) = \Delta CDF_n = \int_0^\infty f(a)\,da \approx [f_{n-1}/(2.3026k_n)][e^{\{2.3026kn(\Delta an)\}-1}]$

(13) $I^*f(a) = \Delta CDF_n = \int_0^\infty f(a)\,da \approx [\tfrac{1}{2}(f_n + f_{n-1})(\Delta a_n)]$

(14) $CDF_n \approx \sum_{i=1}^{n} \Delta CDF_i$

Table 5-2
CDF based on SMA HCLPF(84) = 0.3g

Site	CDF
Catawba (HE=0.78)	1.8E-05
Catawba (HE=1.0)	9.7E-06
Braidwood (HE=1.0)	1.6E-06
River Bend (HE=1.0)	2.7E-07

Table 5-3
CDF based on Approximate Catawba SMA Boolean Using Catawba Hazard

Event	Description	$HCLPF_{84}$ (g)	BETAc	$HCLPF_{50}$ (g)	Median Fragility, Am (g)	Convolution (x 1.0E-6)	Action	ΔCDF (x1.0E-6)
INV	Station Battery Inverter Fails	0.33	0.4	0.27	0.75	3.74	-	3.74
BCH	Station Battery Charger Fails	0.33	0.4	0.27	0.75	3.74	-	3.74
600AC	600V Switchgear Fails	0.33	0.4	0.27	0.75	3.74	-	3.74
MCC	MCCs Fail	0.33	0.4	0.27	0.75	3.74	-	3.74
DGSEQ	DG Sequencer Fails	0.38	0.4	0.31	0.87	2.42	-	2.42
DGECP	DG Engine Control Panel Fails	0.33	0.4	0.27	0.75	3.74	-	3.74
DGCP	DG Generator Control Panel Fails	0.33	0.4	0.27	0.75	3.74	-	3.74
DGNP	DG Neutral Grounding Panel Fails	0.33	0.4	0.27	0.75	3.74	-	3.74
ADP	Auctioneering Diode Panels Fail	0.33	0.4	0.27	0.75	3.74	-	3.74
NSWC	Nuclear Service Water Cabinet Fails	0.33	0.4	0.27	0.75	3.74		3.74
4160AC	4160V Switchgear Fails	0.33	0.4	0.27	0.75	3.74		3.74
MCB	Main Control Boards Fail	0.33	0.4	0.27	0.75	3.74	x OP2	0.374
SSPS	Solid State Protection System Fails	0.33	0.4	0.27	0.75	3.74	-	3.74
ATC	Area Termination Cabinets Fail	0.33	0.4	0.27	0.75	3.74	-	3.74
SSPSc	Solid State Protection System Chatter	0.27	0.4	0.22	0.62	6.72	x OP1	0.672
ATCc	Area Termination Cabinets Chatter	0.27	0.4	0.22	0.62	6.72	x OP1	0.672
	Plant Surrogate	0.3	0.3	0.25	0.49	9.69	-	9.60
Event	Description	Probability					Total CDF	58.71
OP1	Operator Fails to Reset Chatter	0.1						
OP2	Operator Fails to go to Alternate CP	0.1						

Table 5-4
CDF based on Approximate Catawba SMA Boolean using Braidwood Hazard

Event	Description	$HCLPF_{84}$ (g)	BETAc	$HCLPF_{50}$ (g)	Median Fragility, Am (g)	Convolution (x 1.0E-6)	Action	ΔCDF (x1.0E-6)
INV	Station Battery Inverter Fails	0.33	0.4	0.27	0.75	0.572	-	0.572
BCH	Station Battery Charger Fails	0.33	0.4	0.27	0.75	0.572	-	0.572
600AC	600V Switchgear Fails	0.33	0.4	0.27	0.75	0.572	-	0.572
MCC	MCCs Fail	0.33	0.4	0.27	0.75	0.572	-	0.572
DGSEQ	DG Sequencer Fails	0.38	0.4	0.31	0.87	0.355	-	0.355
DGECP	DG Engine Control Panel Fails	0.33	0.4	0.27	0.75	0.572	-	0.572
DGCP	DG Generator Control Panel Fails	0.33	0.4	0.27	0.75	0.572	-	0.572
DGNP	DG Neutral Grounding Panel Fails	0.33	0.4	0.27	0.75	0.572	-	0.572
ADP	Auctioneering Diode Panels Fail	0.33	0.4	0.27	0.75	0.572	-	0.572
NSWC	Nuclear Service Water Cabinet Fails	0.33	0.4	0.27	0.75	0.572		0.572
4160AC	4160V Switchgear Fails	0.33	0.4	0.27	0.75	0.572		0.572
MCB	Main Control Boards Fail	0.33	0.4	0.27	0.75	0.572	x OP2	0.057
SSPS	Solid State Protection System Fails	0.33	0.4	0.27	0.75	0.572	-	0.572
ATC	Area Termination Cabinets Fail	0.33	0.4	0.27	0.75	0.572	-	0.572
SSPSc	Solid State Protection System Chatter	0.27	0.4	0.22	0.62	1.09	x OP1	0.109
ATCc	Area Termination Cabinets Chatter	0.27	0.4	0.22	0.62	1.09	x OP1	0.109
	Plant Surrogate	0.3	0.3	0.25	0.49	1.59	-	1.59
							Total CDF	9.08

Event	Description	Probability
OP1	Operator Fails to Reset Chatter	0.1
OP2	Operator Fails to go to Alternate CP	0.1

Table 5-5
Catawba SPRA And SMA MODEL Sensitivity for Catawba Hazard

Case	Catawba IPEEE PRA Model	LOSP Set to Low Capacity (1)	LOSP Set to High Capacity (2)	Take Out Nonseis-mic Failure (3)	Double Nonseis-mic Failure Rates	Take Out Human Actions	Double Human Action Failure Rates	Chatter at 0.05 Failure Rate (4)	Chatter at 0.25 Failure Rate (4)	RWST at 0.15g HCLPF	4 kV Switchg ear at 0.15g HCLPF	MCC at 0.15g HCLPF	Main Control Board at 0.15g HCLPF	Add Surrogate	Resulting CDF Per Year	% Change in CDF	SMA Based Risk Assessment	% Change in CDF	Ratio of SMA/ SPRA
Base	X	·		·		·		·	·	·	·	·	·		1.46E-5	---	5.87E-5	0	4.02
1	X	✓		·		·		·	·	·	·	·	·		3.55E-5	143%	5.87E-5	0	1.65
2	X	·		✓		·		·	·	·	·	·	·		1.28E-5	-12%	5.87E-5	0	4.59
3	X	·		·		✓		·	·	·	·	·	·		1.11E-5	-24%	5.70E-5	-3%	5.13
4	X	·		·		·		✓	·	·	·	·	·		1.31E-5	-10%	5.80E-5	-1%	4.43
5	X	·		·		·		·	✓	·	·	·	·		1.86E-5	27%	6.07E-5	3.4%	3.26
6	X	·		·		·		·	·	✓	·	·	·		1.53E-5	5%	8.28E-5 (5)	41%	5.41
7	X	·		·		·		·	·	·	✓	✓	✓		2.56E-5	75%	9.08E-5 (6)	55%	3.55
8	X					✓					✓	✓	✓		2.22E-5	52%	8.89E-5	51%	4.00
9	X			✓							✓	✓	✓		2.43E-5	66%	9.08E-5	55%	3.74
10	X		✓												1.14E-5	-22%	5.87E-5	0	5.15
11	X				✓										1.49E-5	2%	5.87E-5	0	3.94
12	X				✓		✓				✓	✓	✓		2.91E-5	99%	9.27E-5	58%	3.19
13	X	✓												✓	≈4.5E-5	308%	5.87E-5	0	1.3

1. HCLPF = 0.01g
2. LOSP High = 1.5 x LOSP Base Case.
3. Nonseismic failures associated with diesel generators remain; all other nonseismic failures taken out
4. Base case is 0.10 failure to recover.
5. Failure rate for 0.15g HCLPF component added directly to base case.
6. Failure rate increased to reflect lower HCLPFs of these components.

Table 5-6
Catawba SPRA and SMA Sensitivity for Braidwood Hazard

Braidwood Hazard

Case	Catawba IPEEE PRA Model	Remove Nonseismic Failure (1)	100% Human Actions Success	Recovery of Chatter at 0.05 Failure Rate (2)	Recovery of Chatter at 0.25 Failure Rate (2)	RWST at 0.15 HCLPF	4 kV Switchgear at 0.15g HCLPF	MCC at 0.15g HCLPF	Main Control Board at 0.15g HCLPF	Resulting CDF Per Year	% Change in CDF	SMA Based Risk Assess-ment	% Change in CDF	Ratio SMA/SPRA
base	X	·	·	·	·	·	·	·	·	2.17E-6	----	9.08E-6	-	4.18
1	X	·	·	·	·	✓	·	·	·	2.37E-6	9%	1.14E-5[3]	26%	4.81
2	X	·	·	·	·	·	✓	✓	✓	3.94E-6	82%	1.53E-5[4]	69%	3.88
3	X	✓								1.88E-6	-13%	9.08E-6	0	4.83
4	X		✓							1.67E-6	-23%	8.51E-6	-6%	5.10
5	X			✓						2.02E-6	-7%	8.97E-6	-1%	4.44
6	X				✓					2.92E-6	35%	9.41E-6	4%	3.22

1. Nonseismic failures associated with diesel generators remain; all other nonseismic failures taken out.
2. Base case is 0.01 failure to railroad recover.
3. Failure rate for 0.15g component added directly to base case.
4. Failure rate increased to reflect lower HCLPFs of these components.

Table 5-7
Delta CDF for $HCLPF_{50}$ = 0.15 g

Hazard	CI	HCLPF_84	BETA_C	A_m	HCLPF_50	ΔCDF (x1.0E-5)
Catawba	1.1	0.183	0.4	0.418	0.15	1.90
Catawba	1.0	0.183	0.4	0.380	0.15	2.41
Braidwood	1.1	0.183	0.4	0.418	0.15	0.35
Braidwood	1.0	0.183	0.4	0.380	0.15	0.45

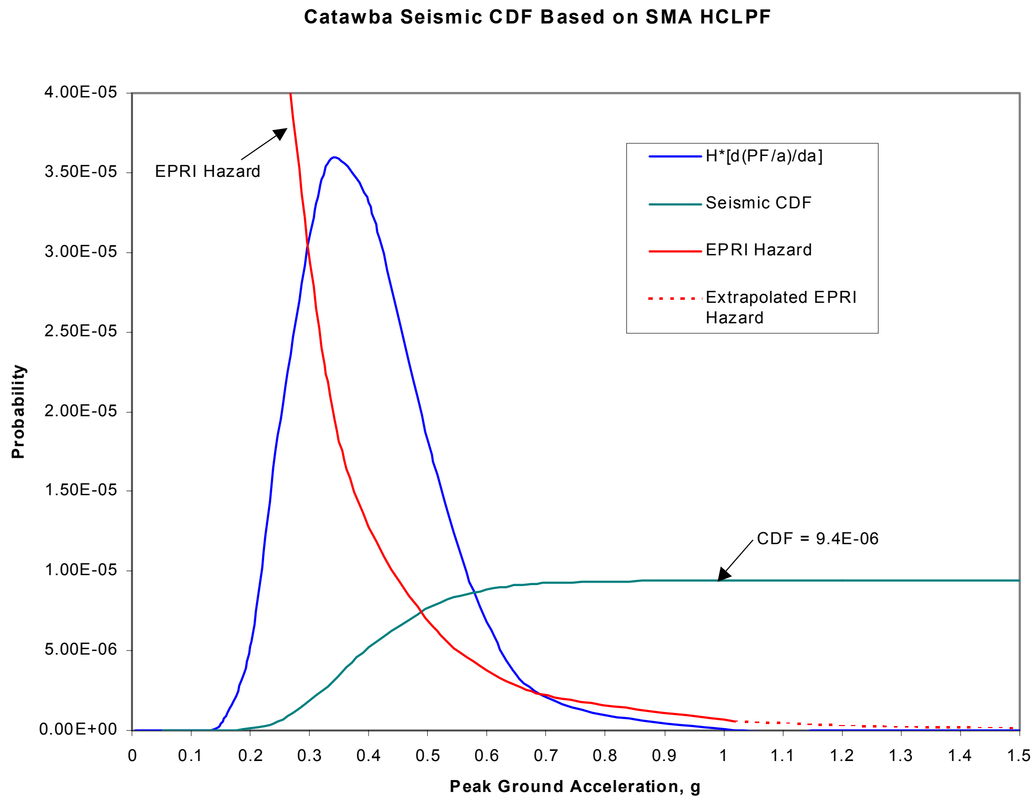

Figure 5-1
Example CDF Convolution

6
CONCLUSIONS AND RECOMMENDATIONS

SMAs were conducted for the majority of the operating NPPs for IPEEE. The objective of the IPEEE program was to search for seismic vulnerabilities beyond the plant design basis. The current guidance from the NRC is that risk-informed decisions will be made using both quantitative results and qualitative insights. Existing SMAs performed under the IPEEE program contain information that can be used to generate both quantitative results and qualitative risk insights. The focus of this project was to develop methods which allow for the generation of quantitative seismic risk results based on converting the SMA results to estimates of the core damage frequency.

The objective of this study was to evaluate several available methods to utilize SMA results for seismic RI/PB applications requiring quantitative estimates of risk and to perform a case study of the most appropriate methodology. The SPRA and SMA results from the Catawba Plant (Catawba is one of the very few nuclear plants with both an SMA and an SPRA) was used for the case study. The general conclusions, Catawba-specific conclusions, and any associated recommendations from this study are provided below:

6.1 General Conclusions and Recommendations

(1) The magnitude of CDF for a plant with an SMA can be estimated by simplified procedures with reasonable accuracy. Several approaches (Sherry Method, Simplified Hybrid Method, Hybrid Method and Budnitz/Ravindra recommendations) were reviewed as a part of this study. Predicted CDF results relative to the SPRA results varied from conservative to unconservative. In addition to estimating the plant risk in terms of the overall CDF, risk-informed applications also require the ability to measure changes in risk caused by changes in components and actions. Since the simplified hybrid method of Reference 18 and the Sherry method of Reference 17 are based on a single lower bound fragility representing the weakest link in the safe shutdown paths, modified by generic factors to account for random failures, they are generally not suitable for most risk informed applications where change in overall risk due to a change in a component fragility must be quantified. The hybrid method of Reference 18 and the similar method recommended by Budnitz and Ravindra, Reference 19, require that the plant response to earthquakes be modeled, at least in a simplified form, in order to perform a mini SPRA. The primary difference in the two recommendations is that the hybrid method is based on achieving a point estimate using the mean value of the seismic hazard and the composite fragility curves for the components that are modeled, whereas Budnitz and Ravindra recommend that the uncertainty in the fragility curves and hazard curves be propagated through the simplified SPRA model using a Monte Carlo simulation process. In this study we have focused on obtaining a point estimate as this was the requirement for IPEEE and most plants that conducted a SPRA for

IPEEE only developed a point estimate. Thus, the hybrid method appears to be the most practical approach for using SMA results in risk informed applications.

(2) The simplified SPRA model can be developed from the information in the internal event PRA model that identifies the accident sequences and cut sets of the plant response to transient initiating events, and considering the capacity ranking of component HCLPFs. Boolean equations can then be derived to represent the plant response to earthquakes. Nonseismic failures and human actions (successes and failures) can be added to the Boolean logic as appropriate. A point estimate of CDF can then be obtained using spread sheet methods for convolving the composite fragility curves in accordance with Boolean logic to obtain a plant level fragility and by convolving the mean seismic hazard with the plant level fragility function.

(3) In most SMAs, components in the safe shutdown paths were screened out at 0.3g HCLPF. If a fragility function is developed from this screening level, in all but the lowest seismic hazard sites, the probability of failure of screened out components exceeds 1E-6 per year. One of the requirements of RG 1.174 for risk informed applications is that the change in CDF should be less than 1E-6 per year. If the screening level results in greater than 1E-6 per year failure rate, the RG 1.174 requirement cannot be demonstrated by the simple SMA screening criteria. Note that this is also an issue with SPRAs, since most SPRAs relied on some form of screening that in many cases resulted in greater than 1E-6 per year failure rate for screened out components and these screened out components were not specifically modeled.

(4) Following the recommendations in NUREG-1407, containment performance evaluation in most seismic SMAs has been done in a qualitative manner in terms of a seismic walkdown screening. Components of containment systems did not typically receive critical review within the SMA. Since LERF is an important measure in future RI/PB decisions, methods for estimation of LERF for plants with SMAs will likely need to be addressed in a similar manner by developing simple models to represent containment isolation and bypass. This should be address in a future phase of the EPRI RI/PB program.

(5) There are three basic categories of plants with existing SMAs (an EPRI SMA is assumed; only two plants have used the NRC SMA method):

   a. Plants with HCLPF capacity less than the RLE, where the SMA has explicitly identified one or more SSCs on the Success Paths having HCLPF capacities less than the RLE.

   b. Plants with HCLPF capacity higher than the RLE, where all of the SSCs on the Success Paths are screened out with HCLPF capacities equal to or greater than the RLE.

   c. Plants that performed a Reduced Scope SMA, using the plant's SSE as the RLE.

As part of the IPEEE program, almost all plants reported a HCLPF ≤ 0.3g. A simple screening criterion has been proposed in an ANS study (Reference 19) to determine if the plant risk is sufficiently low from a risk-informed perspective to justify no additional effort. However, it is unlikely that any U.S. plant with a SMA HCLPF ≤ 0.3g will meet such a simple initial criteria (CDF value of 1.0E-6 per year) suggested in Reference 19. If such a simple screening criterion is to be effective in helping utilities quantify overall seismic risk and changes in seismic risk, further research within the next phase of the EPRI program will need to be conducted in order to provide guidance on modifying the screening level HCLPF by taking into account the difference in structural and equipment response to the probabilistically defined Uniform Hazard Spectra of References 7 and 8 vs. the NUREG/CR-0098 standard spectral shape specified for IPEEE.

(6) While the focus of the EPRI SMA method has been to evaluate two success paths for safe shutdown, most SPRAs have shown that it is the support systems, that are common to all shutdown success paths, which typically contribute most to the seismic risk. Consequently, in many cases, there is effectively no redundancy in shutdown paths which makes the simplified plant response modeling much easier.

(7) Relay chatter HCLPF values were not required to be computed for plants within the “focused” or “reduced” scope bins within the IPEEE program. In order to estimate the CDF value for plants that conducted focused or reduced scope margin studies, some method to estimate these HCLPF values for critical relays (or consideration of operator action for reset) may need to be established. Examples of data sources for establishing relay capacities include vendor data, relay GERS data (EPRI reports) and SQURTS testing data.

6.2 Catawba Study Conclusions and Recommendations

In this study we have used the results of the Catawba IPEEE and the SMA. The SMA was performed in the mid-1980’s as a trial plant application of the EPRI Seismic Margin Methodology. The IPEEE was done in the early 1990’s using the more recent data, as well as the as-built conditions of the plant. We have made use of the seismic fragilities and seismic margin results, as well as the seismic hazard curve reported in these studies. Further, we have employed the systems models (i.e., Boolean equations) reported in the Catawba IPEEE submittal to study sensitivity to parameter variations. Our purpose herein was to examine if comparable seismic risk results could be derived from the SMA. Therefore, we accepted the fragilities, hazard curve, and the systems models and safe shutdown paths and associated component HCLPFs as they are and did not review them. For the purposes of studying the sensitivity of the seismic risk results, both in the SPRA and in the SMA based risk approach, we did change some of the fragility parameters and modeling assumptions in an attempt to converge the results of the two methods and examine sensitivities. These perturbations do not reflect the actual conditions at Catawba.

Several critical issues, with respect to the methods reviewed for utilizing SMA results for RI/PB applications, were addressed in this study. These issues included the following:

- Accuracy of CDF estimates in comparison to SPRA results
- Treatment of Nonseismic Failures
- Treatment of Human Actions
- Treatment of Loss of Offsite Power (LOSP)

Conclusions and recommendations are summarized below:

(1) The Catawba seismic PRA resulted in a computed seismic CDF of 1.46E-5 per year. Using a conservative simplification of the modeling of plant response to earthquake and the "Hybrid" method of Reference 18 for developing fragilities from HCLPFs and calculating a point estimate of CDF, a CDF value of 5.9E-5 per year was obtained. Thus, for the Catawba model and Catawba seismic hazard, the Hybrid method gives very conservative CDF results about a factor of 4 greater than the more detailed SPRA. There are many reasons for this large amount of conservatism, the most significant being that the SPRA takes credit for the limited capacity of offsite power wherein, the SMA based evaluation does not. The difference in the two methods reduces to a factor of 1.66 when LOSP is taken as a certainty in the SPRA model. Another major reason may be that the simplified SPRA risk based model conservatively combines all failure rates of support system components assuming OR gates wherein, some may be only associated with certain plant functions that have redundancy. With more plant specific modeling and inclusion of the success of offsite power, the SPRA and the Hybrid method would converge to a much narrower range.

(2) The Sherry Method delineated in Reference 17 is a very simplified approach for estimating CDF from a margin study. The simplified hybrid method of Reference 18 is likewise very similar. The estimated CDF using the Reference 17 methodology for Catawba was 1.8E-5 per year. This estimate is remarkably close to the SPRA CDF of 1.46E-5. However, a surrogate element was not included in the SPRA. As shown in Table 5-1, the 0.3g screening level results in a surrogate element failure rate of 9.97 E-6. If a surrogate element, used to represent screened out components, were placed in the Catawba SPRA, the surrogate element contribution relative to the base case SPRA plus surrogate contribution would be 40%. If the HCLPF of the surrogate element is modified in accordance with the Reference 17 Sherry method or Reference 18 simplified hybrid method to account for random failures, similar results would occur and the SPRA results, including the addition of a surrogate, would be under predicted by the simplified SMA based risk methods but still within a factor of about 1.5. This raises a question in the SPRA modeling regarding the masking of the SPRA results due to the screening level being too low. Since the Sherry method and the simplified hybrid method are not considered suitable for determining the change in risk from plant changes, they should not receive any further attention. The hybrid method is the recommended approach.

(3) Nonseismic failures within the Catawba SPRA were shown to have on the order of a 12% effect on the CDF. The only nonseismic failure that has been excluded from this calculation is the diesel generator system random failure to start and continue running. The diesel generator random failure rate has been shown to be more significant at some plants, especially those with higher seismicity or with non-standard configurations such as three diesels shared by two units. We recommend that a plant-specific nonseismic failure rate for diesel generators be included in the Boolean equations generated to estimate the CDF. However, the success of offsite power must also be included to reduce the probability of demand on the diesel generators. For all other nonseismic failures, this 12% effect is relatively negligible to the overall risk. Several of the methods reviewed (References 18 and 19) suggest that since it cannot be generically shown that these nonseismic failures have an insignificant effect on the CDF risk, then a fairly explicit characterization of the nonseismic failure risk in the internal events PRA results may be required.

The results of the Catawba study point to the conclusion that nonseismic failures (excluding the diesels) did not contribute significantly and, thus, would not have required explicit characterization. In order to generalize this conclusion to other plants, further review of this issue on additional sample plants is being recommended. The goal would be to either demonstrate nonseismic failures do not need to be included in the simplified risk model or to establish a set of simple screening criteria to identify situations, like Catawba, where the nonseismic failure risks are not very important to the overall CDF estimation.

(4) The effect of human actions within the Catawba SPRA were shown to be on the order of a 24% effect on the seismic CDF. This 24% effect is not considered significant when estimating the seismic CDF and, for Catawba, would not warrant a methodology where an explicit characterization of the human factors risk based on the internal events PRA results would be warranted. However, because of the evolution of NPP designs, some plants may have some unique human factor issues that cannot be ignored. Some screening guidance on when to include human reliability could be developed by studying the results of other SPRAs in a later phase of the EPRI RI/PB program.

(5) Seismic PRAs explicitly consider seismically induced LOSP effects in the CDF calculation. Since the offsite power has some level of inherent capacity with regard to earthquakes, the CDF level would be higher (conservative) if we assumed no offsite power seismic capacity. As shown in the parametric studies summarized in Table 5-5, by excluding the availability of offsite power in the Catawba SPRA, the ratio of SMA based CDF to SPRA based CDF narrows from a factor of 4 to a factor of 1.66. Conversely, if the availability of offsite power were included in SMA-based risk assessments, SMA and SPRA CDF would converge in a similar manner. Further exploration of this LOSP contribution is being recommended in later phases of the EPRI RI/PB program.

(6) Similar comparisons using the much lower Braidwood seismic hazard resulted in essentially the same conclusions. For lower seismic hazard sites, the availability of offsite power can have a more significant beneficial effect on the calculated CDF. For higher seismic hazard sites, the assumption in SMAs that offsite power is not available should be closer to the condition in SPRAs with a higher rate of LOSP.

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