UCRL--91681
DE85 000851
UCRL-91681
PREPRINT
CONF-8410142--12

The Seismic Safety Margins Research Program - A Concluding Look

G. E. Cummings

This Paper was Prepared for Submittal To

The 12th Water Reactor Safety Research
Information Meeting
Washington, D.C.
October 22-26, 1984

Lawrence Livermore National Laboratory

THE SEISMIC SAFETY MARGINS RESEARCH PROGRAM - A CONCLUDING LOOK*

By

G. E. Cummings
Lawrence Livermore National Laboratory

The Seismic Safety Margins Research Program (SSMRP) was started at LLNL in 1978 with the goal of developing tools and data bases to compute the probability of earthquake - caused radioactive release from commercial nuclear power plants. These tools and data bases were to help the sponsoring agency, NRC, assess seismic safety at nuclear plants. The methodology to be used was finalized in 1982 and applied to the Zion Nuclear Power Station. Results of this application were reported at the last Water Reactor Safety Research Information Meeting. The SSMRP will be completed this year with the development of a more simplified method of analysis and a demonstration of its use on Zion. This simplified method is also being applied to a boiling-water-reactor, the LaSalle Nuclear Power Plant, as part of another NRC sponsored program.

DESCRIPTION OF SSMRP METHOD

There are five steps in the SSMRP method for calculating the seismic risk at a nuclear power plant:

1. Determine the local earthquake hazard.

2. Identify potential accident scenarios for the plant which lead to radioactive release.

3. Determine failure modes for the plant emergency safety systems.

4. Compute failure probabilities of the critical components in the emergency safety systems.

5. Compute probability of radioactive release using information from Steps 1 through 4.

*This work was supported by the United States Nuclear Regulatory Commission under a Memorandum of Understanding with the United States Department of Energy.

A brief discussion of each of these steps is given below and illustrated in Figure 1.

Step 1 - Determine the Earthquake Hazard

The earthquake hazard at a given power plant site is characterized by a frequency plot which gives the probability of occurrence (per year) of earthquakes causing different peak ground accelerations. This curve is derived from a combination of recorded earthquake data, estimated earthquake magnitudes of known events for which no data are available, review of local geological investigations, and use of expert opinion based on a survey of seismologists and geologists familiar with the region in question.

In addition to computing the seismic hazard curve, a number (usually 30) of random synthetic earthquakes are generated by using the data just discussed and a Monte Carlo procedure incorporated in our HAZARD code. These earthquake time histories provide the random ground motion uncertainty inherent in real earthquakes, and are used as input to the building response calculations described below. Each synthetic earthquake is described by three time histories in three orthogonal directions.

Step 2 - Identify Accident Sequences

In the event of an earthquake or other abnormal condition in a power plant, the plant safety systems act to bring the plant to a safe shutdown condition. In this step of the risk analysis process, we identify the possible paths that a reactor system could follow during a shutdown, given that an earthquake-related event has occurred which causes shutdown. These paths are referred to as accident sequences. For the SSMRP analysis of Zion, 315 accident sequences were identified and analyzed.

All the accident sequences result from one or more seismically-induced initiating events (events resulting in immediate shutdown of the plant). For the Zion plant, we considered seven classes of initiating events. Four LOCA's of different severity were considered, and two types of transients. In addition, an initiating event "Reactor Vessel Rupture" was identified which is a LOCA for which the ECCS cannot effectively flood the core.

For each of these initiating events, an event tree is constructed. Each branch of an event tree is an accident sequence. In computing the probability of core melt, we compute the probability of each accident sequence occurring. The sum of the probabilities of all accident sequence leading to core melt is the core melt probability.

Step 3 - Determine Failure Modes of Safety Systems

To determine failure modes for the plant safety systems, we use fault tree methodology. Construction of a fault tree begins by identifying the immediate causes of system failure. Then each of these causes is examined for more fundamental causes, until one has constructed a downward branching tree, at the bottom of which are failures not further reducible, i.e., failures of mechanical or electrical components due to all causes such as structural failure, human error, etc. These lowest order failures on the fault tree are called basic events.

Fault trees are required for each safety system identified on the event trees. For Zion, seven safety systems were modeled. The emergency core cooling system was modeled with fault trees for the Safety Injection System, Charging System, Residual Heat Removal System and the Accumulator System. The emergency core cooling function is provided by different combinations of these systems in the injection and recirculation phases of a LOCA, dependent on break size. The auxiliary feedwater system (AFWS) is of primary importance, and a complete fault tree was developed for this system. All the above systems (except the accumulators) require both electric power and service water, so detailed fault trees were also developed for both these systems.

The basic failure events which resulted after all fault trees were constructed fell into three categories: (1) human and maintenance errors, 533; (2) other random failures, 20; and (3) seismically-induced component failures, 1923. In all, a total of 2476 basic failure events were considered.

Step 4 - Compute Failure Probabilities of Critical Components in the Safety Systems

To compute the failure of critical components and safety systems, it is necessary to have both a measure of the maximum load or acceleration that the component experiences during an earthquake as well as a measure of the load or acceleration level at which it fails. Both the maximum load and the strength at failure are random variables. The strength at failure of the buildings and the mechanical and electrical equipment is never known exactly, for there is usually wide variation in the results of tests to determine their failure characteristics. Uncertainties in material properties, soil layering, wall dimensions and joint connectivity influence the response of the building to an earthquake. All of these uncertainties give rise to uncertainties in calculating the response and onset of failure of each building and component in the power plant. The most important feature of the SSMRP is that these uncertainties are explicitly recognized and propagated through the calculational scheme, so that the result is not a single number, but rather, the statistical probability of the occurrence of core melt and radioactive release.

(a) Response Calculations

The buildings, foundations, major components, and piping systems are all modeled by the finite element method. SSI and structure response were calculated by the substructure approach. Piping analysis was performed by multi-support time history analysis. Models were developed for four buildings and five different piping systems in the Zion power plant analysis. For Zion, responses at over 400 points in the buildings and over 1000 points in the piping systems were computed for each input time history. The computer code, SMACS, was developed to do response calculations for SSMRP.

To incorporate the uncertainties, multiple analyses of the entire power plant are made. In each of these repeated calculations, the magnitudes of the input parameters are varied in a random fashion, and each calculation is performed using a different set of three input time histories. Typically, 30 calculations are made (at each earthquake level) with the result that 30 values of response are computed for each building wall, slab, pipe segment, valve and component. From these 30 values, a statistical distribution of the response of each wall, component, etc., can be constructed. Such distribution functions were determined for the responses of every wall, slab, pipe segment, and electro-mechanical component identified on the fault trees.

(b) Determination of Fragility Functions

Component failure is defined as either loss of operability or pressure boundary integrity. Failure (fragility) is characterized by a cumulative distribution function which describes the probability that failure has occurred given a value of load. Loading may be local spectral acceleration or moment, depending on the component and failure mode under consideration. Contrary to previous work, fragility is related to the appropriate local response, rather than being related directly to the free-field peak ground acceleration.

A data base of the necessary fragility functions was developed. As a first step, all components identified on the fault trees were grouped into 37 generic categories. Fragility functions for each generic category were developed based on a combination of design analysis reports, experimental data and an extensive expert opinion survey. Statistical methods were used to combine data from several sources.

Step 5 - Compute Probability of Core Melt and Radioactive Release

Accident sequence probabilities are calculated to determine radioactive release probabilities. Core melt probability is the sum of the probability of all accident sequences leading to core melt.

(a) Calculation of Cut Set Probabilities

Each accident sequence consists of the statistical union of sets of events (successes or failures of components) which must occur together (in systems analysis terminology, called min cut sets). The Zion accident sequences each contained up to 5000 of these component failure groups and each component failure group (min cut set) was allowed to have up to ten basic events (component failures).

The computer code SEISIM was written expressly to calculate the probability of such component failure groups including all common-cause failures. Given the individual component responses and fragilities (in terms of the means and variances of their distributions) and given the computed correlations between the responses (obtained from the 30 time history response calculations at each earthquake level), SEISIM constructs a multi-variate lognormal distribution for each component failure group, and then uses n-dimensional numerical integration to compute the probability of the component failure group occurring.

(b) Calculations of Probability of Radioactive Release

Once the component failure group probabilities have been computed, the probability of each accident sequence can be found using the expression for the statistical union of independent cut sets, which is an upper bound to the accident sequence probability. Then each accident sequence probability is multiplied by the probability of the earthquake's occurrence and the probability of failure of the containment to obtain the probability of radioactive release. Several different containment failure modes of different severity were identified, ranging from rupture of the containment shell down to leakage of the containment isolation valves. Different containment failure modes are assigned to different accident sequences depending on our understanding of the physical processes involved. One accident sequence can result in one or more containment failure modes.

Finally, accident sequence probabilities are assigned to different release categories to reflect their severity with respect to radioactive release to the surrounding population. These release categories relate to the type and energy content of the radioactive fission product release, as well as the mode and timing of the release. They range from rupture of the top of the containment with a rapid, high energetic release (due to a fuel/water explosion or due to steam overpressure) down to slow melt-through of the containment concrete foundation, which is expected to have the least effect on the surrounding population. The containment failure modes and the release categories are those derived and used in the Reactor Safety Study.

IMPORTANCE AND SENSITIVITY

Results from SSMRP analyses of the Zion Nuclear Power Plant have been reported previously[1,2]. More recently, importance and sensitivity studies have been completed to help in prioritizing and testing the results.

A recent study[3] identified key accident sequences found from the Zion risk study. These are shown in Table 1. Systems found important to seismic risk at Zion where the containment spray and fan cooler systems, the auxiliary feedwater and secondary steam relief systems and the RHR system in LOCA mode.

Another study[4] was done using marginal, importance and sensitivity studies to identify changes in structures, systems, equipment, components and parameters that affect seismic risk, to estimate the effect and rank the changes. This study would be useful if an allocation of available resources to reduce seismic risk or uncertainty were felt to be desirable. Although the results are specific to Zion, they have some generic implications for similar plants of that generation.

Areas found most important to seismic risk by this study are listed below. Some of these areas are now receiving further study.

- Local Site Effects
- Piping Between Buildings
- Piping Fragility
- Crib House Pump Enclosure Roof Fragility
- Base Slab Uplift Fragility

More detailed categories are in our report.

Local site effects refers to a phenomenon that occurs at 20 to 30 sites in the eastern United States including Zion. These sites have relatively shallow soil deposits on crystalline bedrock. The available information from past earthquakes and SSMRP calculations reveals that these sites may simultaneously have accelerations and spectral values at certain frequencies that are amplified by factors of as much as 2 to 10 times that obtained if the special physical features typical of these sites are not considered. It is thus not surprising that this area ranks high.

Previous SSMRP results have identified piping between buildings as important. This piping is important when it is restrained in close proximity at two buildings that have independent soil foundations. The relative motion of these buildings at accelerations greater than the safe shutdown earthquake (SSE) causes high stresses and strains in such piping. The relative motion of buildings is a known problem area from past earthquake data. This area would probably not be as important if the piping supports fail before the piping does or in the case of a rock site.

It is surprising that piping fragility ranks so high. Previous seismic risk analyses, including the SSMRP, have found that only a few piping systems were important in safety systems and then under special circumstances such as piping between buildings. Our result in this study arises because pipe breaks are initiating events for the more severe initiators such as LOCAS as well as the assumption that the possible error or bias in the estimated fragility of all piping in the plant is simultaneously biased high or low. This piping fragility is an important input but piping failures are not necessarily key contributors to risk except between buildings.

Crib house pump enclosure roof fragility ranks high because of (1) the relatively low capacity of this roof at accelerations beyond the SSE due to the detail of the connection between the roof and the supporting walls and (2) the assumption that the collapse of this roof causes the loss of function of all six service water pumps. While this category is specific to Zion, it points out: (1) the importance of connection detail (which is a known problem area from past earthquakes) and (2) the capability of structures to act as common-cause failure contributors, as two generic interpretations of our results.

Base slab uplift fragility refers to the failure of the soil foundation of the reactor building at accelerations beyond the SSE. This category is important because it is assumed to lead to failure of the piping between the reactor building and the auxiliary-fuel-turbine complex at Zion. This category points out the importance of soil and foundation failure which is also a known problem area from past earthquakes.

Finally, there is an important category that is not on the above list: relay chatter and breaker trip of electrical equipment.[5] In the SSMRP seismic risk analyses and thus also in our study, relay chatter and breaker trip were assumed not to lead to loss of function or accident initiation at the levels indicated by fragility test data. If this assumption is not made then relay chatter and breaker trip would have a "significant" effect on the SSMRP risk results and hence on our study. These analyses have not been performed and so we can give no accurate indication of how much "significant" is. However, we estimate the inclusion of these failures would lead to an order of magnitude or more increase in the median annual probability of core melt of 3×10^{-5} that was found. This would be a significant increase. Recent SSMRP results lead us to believe that relay chatter and breaker trip could be a much more important factor than we previously assumed in the SSMRP analyses; thus our inclusion of this category here.

Sensitivity studies relating to soil-structure-interaction effects were conducted in three areas: flexible foundation modeling, structure-to-structure interaction and basemat uplift.[6] The auxiliary-fuel handling-turbine building (AFT) complex was modeled to behave ridgidly and also modeled as a series of rigid segments interconnected by structural elements. It turned out that modeling as rigid provided adequate response predictions at locations of interest.

During an earthquake, the vibration of one structure can affect the motion of an adjacent structure due to through-soil coupling. This phenomena is called structure-to-structure interaction and is of significance when small distances separate adjacent structures on massive structure-foundation systems are involved as at Zion. Sensitivity studies were conducted to determine the effect of including or excluding modeling of this phenomena. It was found that the responses of piping systems running between the containment and AFT complex were significantly affected by structure-to-structure interaction effects. Response increases of up to a factor of 2 were noted. The effect on risk was found less significant, a 20% increase in core-melt frequency.

Another soil-structure-interaction phenomena investigated was the separation of the foundation from the soil during an earthquake. This phenomena is called basemat uplift. The separation of the foundation from the soil may not in itself be a problem but upon resettlement, the potential exists for large soil pressures due to stress redistribution. Soil failure may result leading to increased relative displacement between adjacent structures. Basemat uplift was found to be important and as noted previously, basemat fragility is an important effect when considering earthquakes at nuclear power plants.

SIMPLIFIED METHOD

The basic objectives of the SSMRP simplified seismic PRA methodology are to save time and money but to adequately estimate seismic risk. Several assumptions serve as a point of departure for our development efforts:

- o Systems information about the plant is available and an identification of unique features relating to seismic risk has been made. Simultaneous development of plant logic models for all initiators is the best way to achieve consistency in the calculated risk estimates from the various initiators.

- The seismic hazard models (site specific hazard functions and response spectra) for any eastern United States (EUS) site will be available from the NRC EUS Seismicity Projects.(7) Western U.S. sites will probably have to be treated separately.

- Seismic design data is available for all structures, systems, components and equipment.

In the most simple general perspective of a seismic PRA, three different kinds of data are sought:

- Seismic hazard
- Response and fragility and
- Plant logic

In the seismic hazard models we generally eliminated the need for the development of time histories and rely instead primarily on response spectra. In some cases, it may be prudent to develop time histories for limited site- and plant-specific calibration purposes.

The major change in methodology is in response and fragility models. We generally eliminated the need for detailed time history seismic response analysis and rely instead primarily on calibration factors to provide this information. These calibration factors are based on generic studies such as those performed as part of our development efforts or on a limited re-analysis of the nuclear power plant for which the seismic PRA is being performed. The studies that led to the generic calibration factors we recommend formed the bulk of our efforts.

We have developed selected simplifications in plant logic models. However, we believe full-scope PRA needs will dictate requirements here.

The primary focus of our simplification efforts is in the seismic response of structures, piping systems, components and equipment. The SSMRP detailed seismic PRA method involves detailed response calculations as a means to relate free-field acceleration to fragility based on local response quantities. It is thus logical to focus on response as an area to simplify.

The most important aspect of the response simplification effort is to "calibrate" seismic design data. By calibration we mean: To develop a calibration factor F_c that provides a relationship:

$$R_{BE} = R_D/F_c \qquad (1)$$

between seismic responses used in the plant design, R_D, and a best estimate response, R_{BE}. This calibration factor is a key element in the development of fragility functions. R_D is developed for the design earthquake and thus keys the responses to free-field acceleration at that level.

F_c is in general larger than 1.0 and in many cases it is much larger than 1.0. This is a reflection of conservatisms or margin in the calculational methods of analysis used in nuclear design practice. Calibration factors for shear wall structures are shown in Table 2.

A number of sensitivity studies were performed to obtain the calibration factors F_c in equation (1). They assessed the potential influence of many factors that might vary in the design of the various plants to which the SSMRP simplified seismic PRA methodology might be applied. The factors studied included parameters such as damping as well as alternative methods of analysis that are used in nuclear practice. For example, one study showed the relative unimportance of structural damping - particularly at soil sites. Another study revealed that there was surprisingly little difference in results between over ten different methods of analysis of piping systems.

We compared estimates of core melt probability obtained using the SSMRP detailed and simplified seismic PRA methodology. The results are given in Table 3. As shown in this table, the core melt probability for the simplified method is about four times the base case results from the detailed methodology and the dose is twice as much. The major contributors remain to be basemat uplift and inter-building pipes. The absolute results, although different, are well within the uncertainty bounds and the key contributors remain much the same. Further investigation will be necessary to investigate the difference.

CONCLUSION

SSMRP has been a multi-year program to develop probabilistic analysis techniques to determine the seismic behavior of nuclear power plants. These techniques have been applied to Zion in both a detailed and simplified manner. The simplified method is now being applied to LaSalle. Seismic risk assessments are now commonly being conducted by utilities at plants both in the U.S. and offshore. The SSMRP stands as the most detailed of these type seismic risk assessments and provides a benchmark and resource for further work concerning seismic safety.

FIGURE 1: DESCRIPTION OF SSMRP SEISMIC RISK CALCULATION METHOD - HAZARD, SMACS AND SEISIM ARE THE THREE COMPUTER CODES USED TO DO THE CALCULATION

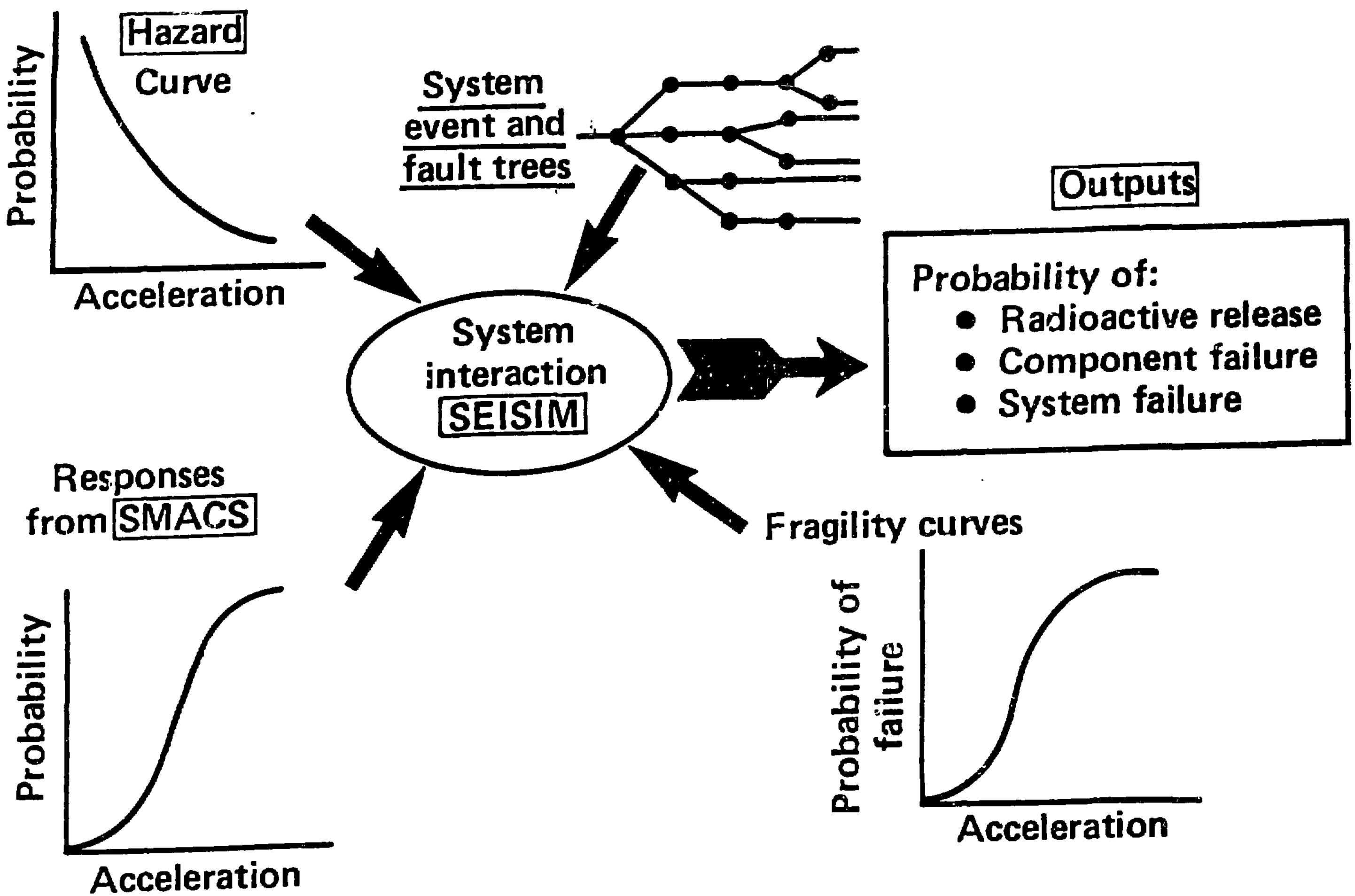

TABLE 1: ZION TOP SEVEN ACCIDENT SEQUENCES (82% of Core-Melt Probability)

Rank	Initiating Event	Accident Sequence	Core-Melt Probability (Per Year)
1	Transient with no PCS (T2)	* $\bar{K}$ L B $\bar{P}$ $\bar{Q}$ $\bar{C}$	1.3e-6
2	Small-Small LOCA (S2)	$\bar{K}$ L C F	4.1e-7
3	Small LOCA (S1)	$\bar{K}$ C $\bar{D}$ $\bar{J}$ F $\bar{H}$	3.4e-7
4	Small LOCA (S1)	$\bar{K}$ C D F	3.2e-7
5	Large LOCA (A)	$\bar{C}$ D $\bar{E}$	2.3e-7
6	Reactor Pressure Vessel Rupture (R)	C F	1.6e-7
7	Large LOCA (A)	C D F	1.3e-7

Systems

K - RPS
L - AFW
B - Bleed & Feed
P - PORV (open)
Q - PORV (close)
C - CSIS & CPCS (inject)
D - ECI
J - core geometry
F - RHR
H - ECR
E - CFCS (recirculation)
PCS - Pwr. Conversion Sys.

* Bar over letter means system success. No bar means system failure.

Table 2: Calibration Factors for Shear Wall Structures

Soil Stiffness Characteristic V_s (fps)	Peak Accelerations Mean	Peak Accelerations Cov	Peak Forces Mean	Peak Forces Cov
3500	1.07	.261	1.23	.178
2000	1.25	.389	1.42	.289
1000	1.47	.500	1.64	.360
500	2.02	.601	2.14	.403

Assumes half-space modeling of the soil with soil density of 130 pcf, Poisson's ratio of 0.4, and soil material damping of 5%.

Table 3: Comparison Between Detailed & Simplified Analysis of Zion

Release Category	Release Probability/Year		Man-Rem/Year	
	Detailed	Simplified	Detailed	Simplified
1	2.9E-8	1.6E-7	0.2	0.9
2	1.4E-6	3.7E-6	6.5	17.8
3	5.4E-7	1.1E-7	2.9	0.6
4	0	0	0	0
5	8.3E-10	0	0	0
6	1.7E-7	6.3E-8	0	0
7	1.5E-6	9.5E-6	0	0.2
TOTALS:	3.6E-6*	1.3E-5*	9.6	19.5

* These totals are equivalent to the core melt probability per year.

REFERENCES

1. P.D. Smith, et al., Seismic Safety Mardgins Research Program Phase I Final Report, Lawrence Livermore National Laboratory, Livermore, CA, UCRL-53021, Vols. 1-10, NUREG/CR-2015, Vols. 1-10 (1982).

2. M.P. Bohn, et al., Application of the SSMRP Methodology to the Seismic Risk at the Zion Nuclear Power Plant, Lawrence Livermore National Laboratory, Livermore, CA, UCRL-53483, NUREG/CR-3428 (1983).

3. C.Y. Kimura, J.E. Wells, Categorization of PWR Accident Sequences and Guidelines for Fault Trees: Seismic Initiators, Lawrence Livermore National Laboratory, Livermore, CA, UCID-20211 (1984).

4. L.L. George, W.J. O'Connell, Importance and Sensitivity of Parameters Affecting the Zion Seismic Risk, Lawrence Livermore National Laboratory, Livermore, CA, (To Be Published).

5. H.E. Lambert, Circuit Breaker Operation and Potential Failure Modes During an Earthquake: A Preliminary Investigation, Lawrence Livermore National Laboratory, Livermore, CA, UCID-20086 (1984).

6. J.J. Johnson, et al., SSI Sensitivity Studies and Model Improvements for the U.S. NRC Seismic Safety Margins Research Program, Lawrence Livermore National Laboratory, Livermore, CA, UCID-20212, NUREG/CR-4018 (1984).

7. D. Bernreuter, et al., Seismic Hazard Characterization of the Eastern United States: Methodology and Interim Results for Ten Sites, Lawrence Livermore National Laboratory, Livermore, CA, UCRL-53527, NUREG/CR-3756 (1984).