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SUMMARY REPORT ON THE SEISMIC SAFETY MARGINS RESEARCH PROGRAM

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Prepared by
G. E. Cummings

Lawrence Livermore National Laboratory
7000 East Avenue
Livermore, CA 94550

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Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
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ABSTRACT

The Seismic Safety Margins Research Program (SSMRP) was a U.S. NRC-funded multi-year program conducted by Lawrence Livermore National Laboratory. Its goal was to develop a complete, fully coupled analysis procedure (including methods and computer codes) for estimating the risk of an earthquake-induced radioactive release from a commercial nuclear power plant. The SSMRP was the first effort to trace seismically induced failure modes in a reactor system down to the individual component level, and to take into account common-cause earthquake-induced failures at the component level.

This report summarizes methods and results generated by SSMRP. The SSMRP method makes use of three computer codes, HAZARD, SMACS and SEISIM to calculate ground motion acceleration time histories, structure and component responses and failure, and radioactive release probabilities. The process starts with development of an earthquake-hazard function and a set of time histories for the earthquake levels of interest. These time histories are used as input for dynamic structural response calculations for plant structures taking into account soil-structure interaction effects. From the response of the structure, the responses of piping and safety system components within the buildings are determined. These are combined with probabilistic failure criteria (fragilities) and plant logic models (event and fault trees) of the plant systems to estimate the frequency of core melt and radioactive release due to earthquakes.

To demonstrate the methodology, an analysis was done of the Zion Nuclear Power Plant. The median frequency of core melt was computed to be 3E-5 per year, with upper (90%) and lower (10%) bounds of 8E-4 and 6E-7 per year. The main contribution to risk came from earthquakes about 2 through 4 times the design basis earthquake level. Risk was dominated by structural and inter-building piping failures and loss of off-site power.

Sensitivity studies were undertaken to test assumptions and modeling procedures relative to soil-structure interaction effects , feed-and-bleed cooling, and structural failures. Assumptions made could have an order-of-magnitude effect on core melt frequency. Also, guidelines were developed for simplifying the SSMRP method, and importance rankings were generated based on the Zion analysis.

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ACKNOWLEDGMENTS

This report is a summary of the work accomplished as part of the Seismic Safety Margins Research Program (SSMRP). It represents efforts of many people, over a number of years, from many different organizations. Limitations on space prevent a listing of all the contributors to SSMRP, and any omissions are for this reason or through unintentional oversight by the author.

A team approach accomplished the objectives of SSMRP and provided the multi-discipline staff composition necessary to provide the best technical product within the time available. The team consisted of a core group of Lawrence Livermore National Laboratory (LLNL) personnel and selected subcontractors and consultants. The work was carried out within the Nuclear Systems Safety Program (NSSP) headed at that time by L.L. Cleland and currently by G.E. Cummings, with personnel drawn primarily from the Mechanical, Electrical, Computation, and Earth Sciences Departments.

The first phase of the Program was started by F.J. Tokarz, leader of the Engineering Mechanics Section of the M.E. Department. P.D. Smith (now with EQE, Inc.), was the SSMRP Program Manager, C.K. Chou was SSMRP Deputy Program Manager, and G.E. Cummings was responsible for Systems Analysis and SEISIM development. Also involved were J.J. Johnson (now with NTS/Structural Mechanics Associates) who was responsible for soil-structure interaction, structural response, and development of the SMACS code, D.L. Bernreuter, responsible for seismic input and development of the HAZARD code; T. Lo, responsible for structures; T.Y. Chuang, responsible for subsystems analysis; R.G. Dong, responsible for fragilities and later SSMRP Deputy Program Manager; M.P. Bohn, fragilities; R. W. Mensing, statistics; and J.E. Wells, systems analysis.

The second phase of the Program was carried out under P.D. Smith as NSSP Associate Program Leader for Seismic and Structural Safety, M.P. Bohn as SSMRP Program Manager, and L. C. Shieh as Zion Phase II Analysis Project Leader. Others involved who have not been previously mentioned include J.C. Chen, soil-structure interaction; L.E. Cover, fragilities; and L. L. George and D. A. Lappa, systems analysis.

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The subcontractors and consultants used on SSMRP are too numerous to present here but a special acknowledgment should be given to the Senior Research Review Group which helped launch and guide SSMRP. Members included S.H. Bush, C.A. Cornell, W.J. Hall (final meetings), R.M. Hamilton, W.W. Hays, N. M. Newmark (deceased), and D.H. Worledge (final meetings). Also, the cooperation of Commonwealth Edison and Sargent and Lundy was much appreciated.

Special thanks should be given to C.S. Aubuchon who supervised the clerical support during most of SSMRP and in other ways helped keep the wheels turning.

Finally, acknowledgment should be given to the many U.S. Nuclear Regulatory Commission staff members who supported and guided SSMRP from its formulation through to completion. These include S. Levine (deceased), then Director of the Office of Nuclear Regulatory Research, who started the Program; J.E. Richardson, now Branch Chief, Mechanical/Structural Engineering Branch; the original NRC Project Manager, C. W. Burger, who followed as Project Manager; and D.J. Guzy, the current Project Manager.

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EXECUTIVE SUMMARY

To assist in its licensing and evaluation role, the NRC funded the Seismic Safety Margins Research Program (SSMRP) at the Lawrence Livermore National Laboratory (LLNL) with the goal of developing tools and data bases to evaluate the risk resulting from earthquake-caused incidents at commercial nuclear power plants. This program began late in 1978, and the methodology was finalized in 1982. This report is a summary report which describes the SSMRP risk assessment methodology and the results of the Zion analysis as well as sensitivity studies, validation tasks, and guidance for simplified application of the method.

Scope of the SSMRP

A nuclear power plant is designed to ensure the survival of safety-related systems, buildings, and equipment for a low probability, safe shutdown earthquake (SSE). The assumptions underlying this design process are deterministic and subject to uncertainty. For example, it is not possible to accurately predict the "worst" earthquake that will occur at a given site so uncertainty in the earthquake definition is always present. Soil properties and dynamic characteristics of structures and subsystems are subject to uncertainty in their definition. Large uncertainty exists in prediction of loss of function and failure. Therefore, to model and analyze the coupled phenomena that contribute to the total plant risk due to seismic events, it is necessary to consider all significant sources of uncertainty. Total risk is then obtained by considering the entire spectrum of possible earthquakes and rationally combining the associated consequences.

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

1. Characterize the seismic hazard.
2. Determine response of structures and subsystems to seismic excitation.
3. Determine fragility functions.
4. Identify accident scenarios.
5. Calculate probability of failure and frequency of radioactive release.

A brief discussion of each of these steps is given below.

Step 1: Characterize the Seismic Hazard.

The earthquake hazard at a given power plant site is characterized by a hazard function which gives the probability of exceedence (per year) of a ground motion parameter, such as peak ground acceleration. This hazard 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 characteristics, and use of expert opinion based on surveys of seismologists and geologists familiar with the region in question.

The frequency characteristics of the earthquakes are required, as well as their likelihoods. Response spectra or time histories are used to define earthquake frequency characteristics. The computer program HAZARD is used to generate the seismic hazard curve and response spectra. From the response spectra, artificial acceleration time histories are generated. For calculational purposes, the hazard curve is discretized into intervals with a set of acceleration time histories of equivalent peak ground acceleration generated for each interval.

Step 2: Determine Response of Structures and Subsystems to Seismic Excitation.

The computer program SMACS calculates soil-structure interaction (SSI) and response of the plant's major structures and subsystems to seismic excitation. Seismic excitation is given by ensembles of acceleration time histories in three orthogonal directions, obtained as described above for Step 1. SSI and detailed structure response are determined simultaneously using the substructure approach to SSI. The response of subsystems is calculated by multi-support time history analysis procedures. Uncertainty is treated explicitly in each of these links of the seismic methodology chain by analyzing an ensemble of free-field acceleration time histories and by varying a discrete number of input parameters of the soil, structures, and subsystems. SMACS performs repeated deterministic analyses, each analysis simulating an earthquake occurrence. The peak responses calculated by SMACS are then used by the computer code SEISIM (see Step 5) to calculate response medians, coefficients of variation and correlation coefficients.

Step 3: Determine Fragility Functions.

Different subsystems, structures, parts of structures, and components have different susceptibilities to failure caused by an earthquake. Susceptibility is specified by a fragility, which is a cumulative probability of failure as a function of loading.

Step 4: Identify Accident Scenarios.

All failures are not equally serious. Following the most likely earthquakes, plant systems will be effective, either permitting the continued operation of the plant or bringing about a safe shutdown. Larger, less probable earthquakes may result in system failures and in an extreme case, core damage and radioactive release would occur.

In this step of the risk analysis, possible accident scenarios are identified during an earthquake-induced shutdown. The consequences from these scenarios vary from minor to severe. Fault trees are used to determine the success or failure of the systems whose success or failure paths make up the accident sequence within the scenario. Each accident sequence is started by an initiating event which defines what systems need to be considered.

Step 5: Calculate Probability of Failure and Frequency of Radioactive Release.

Step 5 combines the results of Steps 1-4 to express plant risk as the frequency of 1) failure of structures and components, 2) failure of a group of structures and components (systems), and 3) core damage and radioactive release.

Each accident sequence consists of the union of sets of events (component failure groups) which must occur together to have system failure. The computer code SEISIM was written expressly to calculate the probability of occurrence of such sets of events explicitly accounting for earthquake-related dependent failures. Given the individual component responses and fragilities (in terms of the means and variances of their distributions), SEISIM computes correlations between the responses from the time history response calculations at each earthquake level, then constructs a multi-variate lognormal distribution for each component failure group, and then uses n-dimensional numerical integration to compute the probability of occurrence of the component failure group.

Once the component failure group probabilities have been computed, the probability of each accident sequence can be found by using the min-cut upper bound approximation. Then each accident sequence probability is multiplied by the probability of the earthquake's occurrence, the probability of the initiating event (which starts the accident sequence), and the probability of each containment failure mode to obtain the frequency of radioactive release. Containment failure modes need to be specified. Typically, these range from rupture of the containment shell to leakage of the containment isolation valves. Different containment failure modes are assigned to different accident sequences according to the physical processes involved. One accident sequence can result in one or more containment failure modes. In turn, the containment failure modes relate to release categories. These release categories relate to the type and energy content of the radioactive fission product release, as well as to the mode and timing of the release.

Results of the Zion Risk Analysis

To demonstrate the SSMRP methodology, a seismic risk assessment was made of the Zion Nuclear Power Plant, a Westinghouse pressurized water reactor owned by Commonwealth Edison and located north of Chicago. Base case, point estimate calculations were made as well as an assessment of uncertainty. The base case is the best estimate of the configuration of the Zion plant and its emergency procedures.

A number of important assumptions were made when carrying out this analysis. These were that: 1) feed and bleed cooling of the core following a loss-of-coolant accident (LOCA) would be an alternative to emergency core cooling system operation, 2) failure of a structure was assumed to result in failure of all components associated with that structure, 3) soil failure under the toe of the containment building following basemat uplift was assumed to fail piping between the containment and auxiliary/fuel/turbine (AFT) building, 4) failure of the steam generator and reactor coolant pump supports was assumed to result in a large LOCA (LLOCA) or an equivalent to a reactor pressure vessel rupture (RVR) if in more than one loop.

A modified Monte Carlo procedure using Latin hypercube design was used to determine the uncertainty in the calculated frequency of core melt per year. Repeated calculations were made of the core melt frequencies for the base case, while varying the median values of all input variables by sampling from distributions of input variable values. Fourteen calculations were performed, with new values for structural responses, fragility curves and hazard curves for each calculation. The median value and uncertainty intervals were inferred from these 14 calculations. The median frequency of core melt for the plant was calculated to be 3.E-5 per year. This value reflects inherent randomness in all the input variables and the hazard curve, as well as modeling uncertainties attributable to lack of exact knowledge of mean values of input variables. The 10-90% uncertainty band on the core melt frequency was found to be about 3 orders of magnitude.

To illustrate the important accident scenarios and to conduct sensitivity studies, point estimates using base case assumptions were calculated. To obtain these point estimates, all input variables were assigned best-estimate values, and a specially constructed median hazard curve was used. Thus, this base-case calculation gives the risk at Zion based on best-estimate values of the input parameters and includes random but not modeling uncertainty. Point-estimate calculations are useful for comparisons between SSMRP calculations using different assumptions or for ranking.

For the Zion analysis the seismic hazard curve was discretized into 6 levels. In terms of both core melt frequency and dose, it was found that earthquake levels 2, 3, and 4 (2 through 4 times SSE level) were dominant and the frequencies and dose were significantly smaller at earthquake levels 1 and 6. The design basis earthquake (SSE) is within level 1.

The dominating initiating events at the three lower earthquake levels are the transients (trip of the reactor not caused by a LOCA). The seismically induced failure causing the transient initiating events is primarily the loss of off-site power by failure of the ceramic insulators at the point where off-site power is brought into the switch yard. At the lower three earthquake levels, risk is dominated by failure of the auxiliary feedwater system (AFWS). The uplift of the containment basemat causes failure of the AFWS pipes, and is also assumed to fail the containment sprays. A second important risk contributor is failure of the roof slab in the crib house service-water pump enclosure room. This roof slab failure is assumed to fail the six service water pumps, which in turn fails the diesel generators due to lack of cooling water. This, in conjunction with loss of off-site power, results in loss of all AC power, and hence loss of both the AFWS and the containment sprays.

LOCAs are the dominant initiating events for the upper three earthquake levels. Risk is due almost entirely to accident sequences, which are caused by the failure of pairs of pipes between the containment and AFT buildings. These pairs of pipes fail because of differential motion between the buildings. Failure of any one of these pipe pair combinations causes failure of both emergency core injection and the RHR systems. Release at the upper earthquake levels is also due to two small LOCA accident sequences which are both the result of loss of emergency core cooling due to uplift and service water pump room roof failures.

In summary, out of the total 9.6 man-rem/year calculated (point-estimate) using base-case assumptions, approximately 6.1 man-rem/year is due to accident sequences caused directly by the uplift and crib house pump room roof failures, and 2.7 man-rem/year is due to failures of pairs of pipes between the containment and AFT buildings. Thus, for the base-case computations of the seismic risk at Zion, the structural failures and the assumptions as to their consequences play a dominant role.

Sensitivity, Validation, and Simplification Studies.

Sensitivity studies were made relative to Zion base-case results to help assess the significance of some of the assumptions made for the Zion analysis. When it was assumed that no structural failures could occur but that "feed and bleed" cooling was possible, the core melt frequency and dose decreased 50%. Assuming that no structural failures could occur and that no "feed and bleed" existed, the core melt frequency increased 250% but the dose decreased 40%. Assuming that structural failures could occur but "feed and bleed" was not possible increased the core melt frequency 300% and the dose 13%. Thus, it can be seen that, depending on the assumptions made concerning the possibility of structural failure and "feed-and-bleed" cooling, the core melt frequency can vary by an order of magnitude, and the dose by 250%.

A number of other cases were analyzed during the Zion analysis to demonstrate the importance of various modeling aspects. Sensitivity studies concerning soil-structure interaction effects found that the local amplification of the free-field motion was important, that the effect of structure-to-structure interaction was important to response but not to risk, and that the assumption of rigid foundations was appropriate. Also, the effect of correlation of responses and fragilities of plant components was found to have a significant impact on risk if dominant risk contributors are pairs of component failures instead of single failures and if structural failures are not dominant.

Following the completion of the Zion analysis, additional studies were undertaken. One study evaluated importance and sensitivity of input parameters affecting Zion seismic risk. Also, rankings of accident sequences, plant systems, responses and fragilities were made. These rankings were developed by use of probability of occurrence, derivatives of probability or risk with respect to changes in means or standard deviation of responses and fragilities, and Vesely-Fussell importance measures. Sensitivities to input parameters which control vibratory loads and the strength of components are important to understand in order to change risk if this is found necessary. When associating the parameters with the components they influence or to physical effects a ranking can be made. The top five in terms of risk importance were found to be: 1) local site effects, 2) piping between buildings, 3) piping fragility, 4) crib house roof fragility, 5) base slab uplift fragility.

One item not on the list was failure caused by relay chatter. This was not considered a failure in the Zion analysis, but a subsequent SSMRP study showed that it could contribute to the Zion seismic risk and needs to be studied further. Also, further studies have shown that the effect of base slab uplift on inter-building piping failure was overstated and that failure of the crib house roof probably does not destroy all the pumps within the building as originally assumed.

Since SSMRP was to develop research tools, its use in individual seismic risk assessments could be cumbersome. To make the method more suitable for individual applications, guidelines were developed to aid in simplifying the analysis. Since the majority of the effort for the Zion analysis related to calculation of building and component responses, the focus of the simplification was in this area. Response factors were determined so that design calculations can be used to determine a best estimate of the responses rather than constructing detailed models of plant structures for plants which are to be analyzed. This simplified method was applied to Zion and compared within a factor of 4 with the core melt frequency of the detailed analysis.

Validation of the SSMRP to date has relied primarily on peer review, comparisons with other methods, expert judgement, and sensitivity studies. One such validation study took a close look at the fragilities of reinforced concrete structures as generated for SSMRP. Since structural failures were shown to be of major importance in the Zion analysis, this appeared to be a fruitful area for investigation. A number of improvements for concrete fragility construction were suggested including use of piecewise linear analysis and approximate inelastic analysis methods to supplement the ductility-based reduced spectrum method used in SSMRP. Other validation studies reviewed assumptions concerning the effects of containment basemat uplift and cribhouse roof failure.

Results of a risk analysis carried out for the utility owner of Zion were compared with results from SSMRP. A comparison of the median core melt frequencies calculated in the two studies showed both values within the uncertainty bands of either study but different by an order of magnitude. The difference was found to be attributable to differences in the seismic hazard and response/fragility data used. Differences in the way the two studies modeled the RHR piping between the containment and AFT building explain some of this difference.

Limitations

SSMRP represents a point in time with respect to seismic risk assessment techniques and data. Although SSMRP accomplishments have been substantial, further work is needed to perfect the techniques or to account for areas less completely covered. More work should be done to adequately model the effect of relay chatter, operator performance, gross design and construction errors, structural failure, and piping and pipe mounting failures. In addition, improvements in uncertainty analysis are needed to better quantify the mean, median, and uncertainty bands. Some of these limitations will be addressed in continuing NRC-sponsored programs in seismic margins, validation, and BWR risk assessment.

Section I: INTRODUCTION

The Seismic Safety Margins Research Program (SSMRP) was started in 1978 at the Lawrence Livermore National Laboratory (LLNL) to develop tools and data bases for computing the probability of earthquake-caused radioactive release from commercial nuclear power plants. The Program was sponsored by the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission.

The interest in further study of seismic safety at nuclear power plants was brought about by recognition of several unique aspects that seismic design must consider. These aspects are that: 1) the maximum size earthquake possible is unknown, 2) strengthening may result in increased risk, 3) little experimental data is available to guide design, and 4) ground shaking compromises the redundancy needed to assure high reliability of safety systems. In addition, picking parameters to assure a conservative design is difficult because of the complex interactions between site, structure, and safety systems.

For these reasons SSMRP was asked to develop a complete, fully coupled analysis procedure (including methods and computer codes) for estimating the probability of earthquake-induced radioactive release and important contributors to that release. Using probabilistic techniques, an overall perspective of relative importance and vulnerability of components, systems, and structures would be obtained to help with the judgments needing to be made with respect to seismic safety.

The goals of the SSMRP were achieved in two phases. In Phase I, the overall seismic risk assessment methodology was developed and assembled. The methodology is embodied in three computer codes: HAZARD, SMACS, and SEISIM. In addition, extensive data bases on earthquake occurrence models and failure data for nuclear power plant components were assembled. An operating pressurized water reactor was selected for demonstration calculations, and fault trees were developed for its essential safety and auxiliary systems. The plant chosen was the Zion nuclear power plant, located about 40 miles north of Chicago. Zion was chosen on the basis of being reasonably typical (in terms of power, systems design, and site conditions) of pressurized water reactors in the 1960's era. The limited demonstration calculations (and Phase I) were completed in February, 1981. A 9-volume final report (Ref. 1) on Phase I was issued subsequently.

The goals of Phase II of the SSMRP were to complete the development of seismic risk methodology and to perform a complete seismic risk assessment of the Zion plant. This risk assessment was not only to compute the frequency of core melt and radioactive release, but also to include an uncertainty analysis on the entire risk assessment process so that uncertainty intervals on the core melt frequencies could be determined. This report was completed in 1984 (Ref. 2).

Since then, SSMRP efforts have been focused on additional sensitivity studies, technology transfer, validation studies, and the development of guidelines for simplified analysis procedures. SSMRP is coming to a conclusion and the tools and data bases are helping the NRC assess seismic safety and margins issues at nuclear plants. Further work continues as part of another NRC sponsored program which calls for the application of SSMRP methods to the La Salle boiling water reactor.

This summary report starts with an overview of the method and the results from its application to Zion. Additional efforts including sensitivity study results, simplified methods description, and validation tasks are then discussed. Finally, SSMRP contributions and limitations, a complete bibliography, and suggested research needs are presented. Results from the LaSalle study will be reported at a later date.

Section 2: OVERVIEW OF METHOD

A nuclear power plant is designed to ensure the survival of safety-related systems, buildings, and equipment in a worst-case ("safe shutdown") earthquake. The assumptions underlying this design process are deterministic. In practice, however, these assumptions are subject to uncertainty. It is not possible, for example, to absolutely predict the "worst" earthquake that will occur at a given site. Soil properties and dynamic characteristics of structures and subsystems are subject to uncertainty in their definition. To model and analyze the coupled phenomena that contribute to the total probability of radioactive release, it is therefore necessary to consider all significant sources of uncertainty. Total seismic risk (as measured by probability of release) is then obtained by considering the entire spectrum of possible earthquakes and integrating their calculated consequences.

There are five steps for calculating the seismic risk at a nuclear power plant (Figure 2.1).

1. Characterize the seismic hazard. This includes the likelihood and magnitude of potential earthquakes and the degree to which they transfer energy through the ground.

2. Determine response of structures and subsystems to seismic excitation including soil-structure interaction effects.

3. Determine fragility functions. A fragility function defines failure probabilities with respect to a given response.

4. Identify accident scenarios.

5. Calculate probability of failure and frequency of radioactive release.

Each of these steps is discussed briefly below. More detail is given in Reference 2.

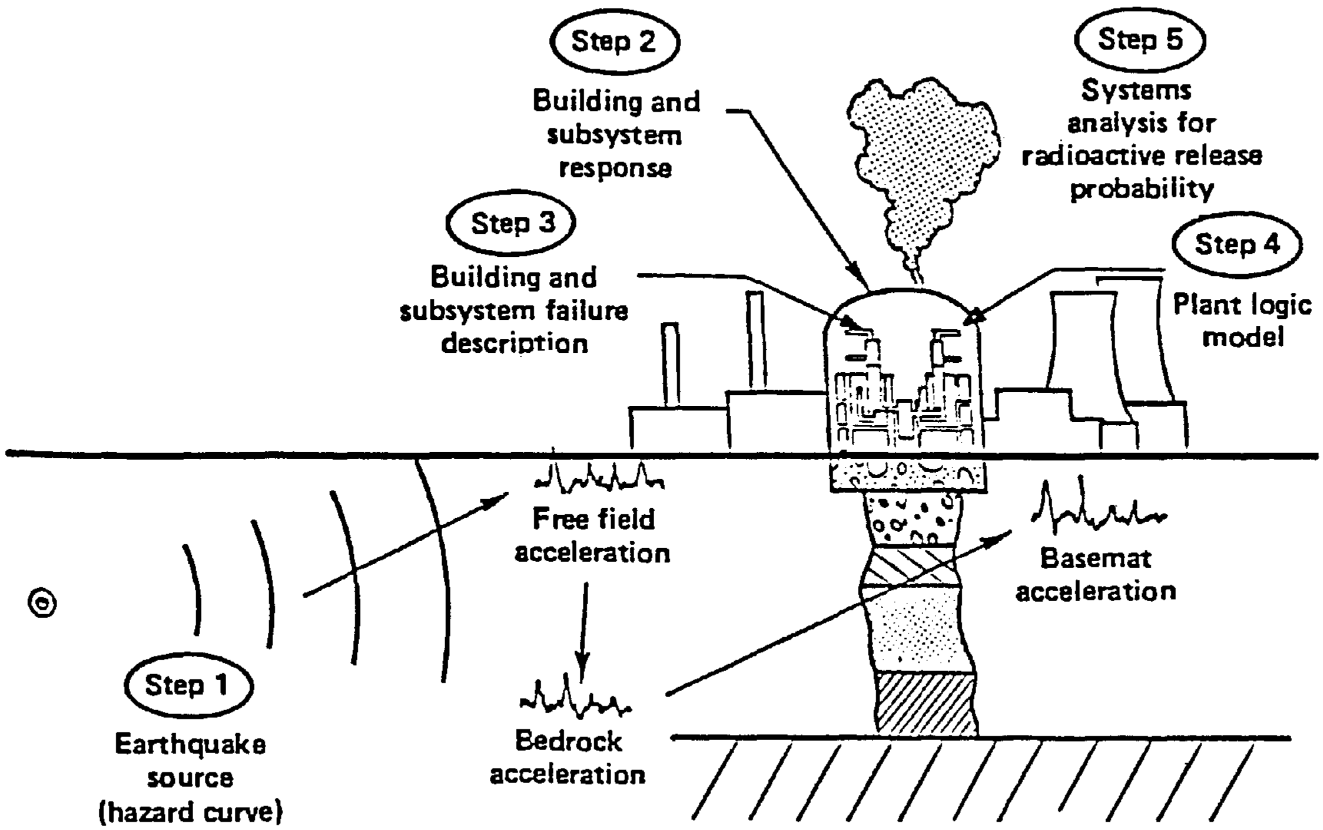


Figure 2.1 A seismic risk analysis methodology.

2.1 Step 1 - Characterize the Seismic Hazard

The earthquake hazard at a given power plant site is characterized by a hazard function giving the annual probability of exceedence of a ground motion parameter, i.e., peak acceleration, peak velocity, uniform hazard response spectrum. Figure 2.2 shows the hazard curves for the Zion nuclear power plant used in the SSMRP. All 14 curves were used in the uncertainty analysis. A specially constructed median hazard curve (marked in Figure 2.2) was used in point-estimate calculations. For a given site, a hazard 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 the seismic hazard curve, descriptions of the corresponding frequency characteristics of the motion are required. Response spectra and/or acceleration time histories usually serve this purpose. For the SSMRP, response spectra were generated in conjunction with the seismic hazard curve using the computer program HAZARD. Artificial acceleration time histories were generated to match these response spectra and for use in Step 2, computing the seismic response of structures and subsystems. Three orthogonal components (two horizontal and one vertical) of acceleration time histories were generated for each earthquake simulation. The seismic hazard curve is discretized into intervals for calculational purposes. For the Zion analysis, six earthquake intervals were specified.

The ground acceleration used in the Zion analysis is referenced to bedrock which at Zion is about 110 ft. below the surface. This was done because of the pronounced local site effect at Zion due to the shallow soil layer of 110 ft. thickness. Table 2.1 relates the rock acceleration to the peak ground acceleration at the surface. Because of the uncertainties involved in propagation of the motion through the soil layer a discrete interval referenced to the rock relates to a broader interval at the soil surface. In either case, the Zion SSE Level is within the lowest interval and intervals increase approximately in multiples of the SSE level, e.g. median acceleration of interval 2 is about twice the SSE level.

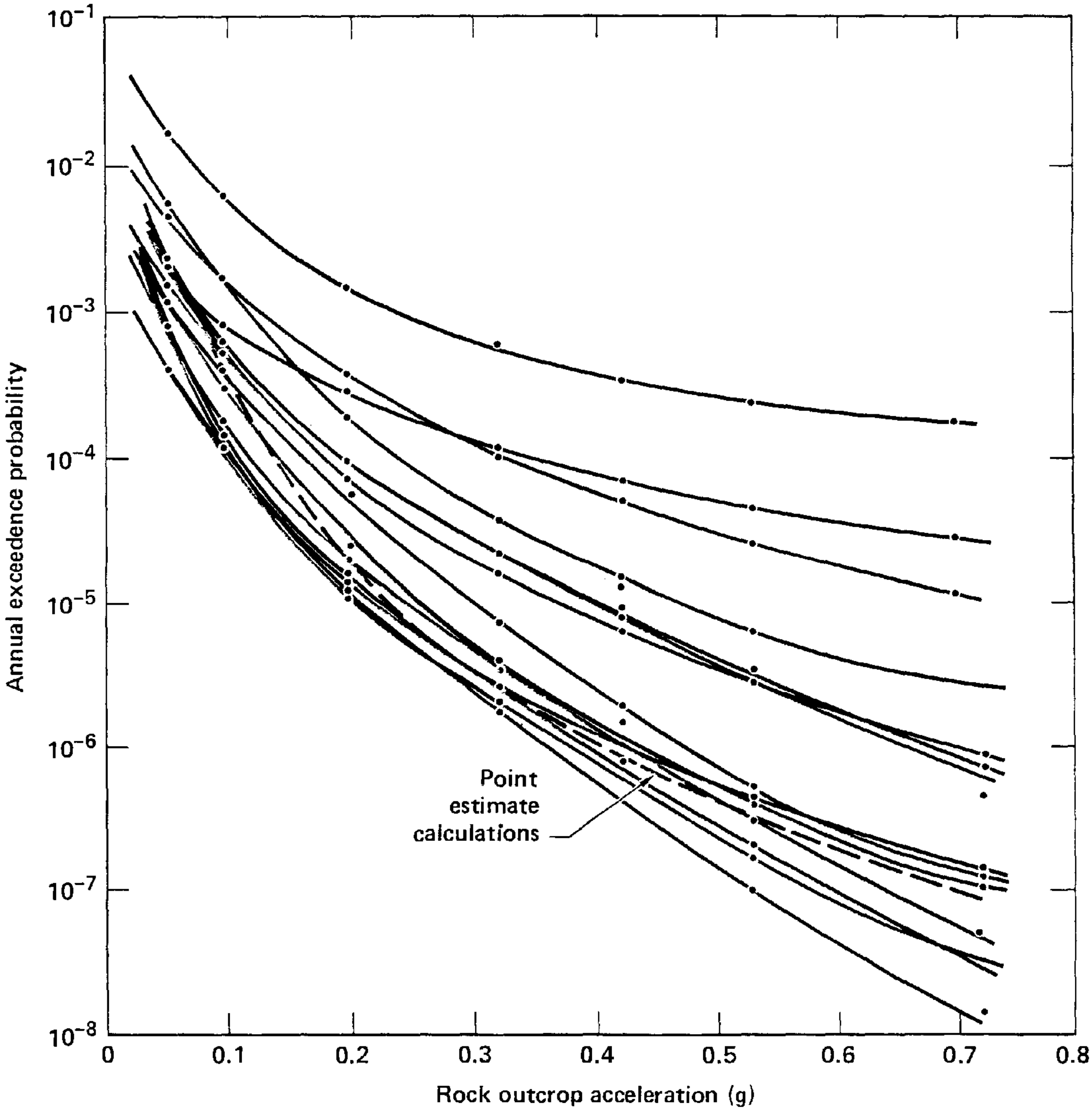


Figure 2.2 Fourteen approximately equally weighted hazard curves for Zion.

Table 2.1: Zion analysis ground acceleration intervals.

Interval No.	Rock Acceleration (g) Range	Median	Peak Ground Acceleration (g) Range	Median
1	0.06 - 0.10	0.07	0.09 - 0.27	0.16
2	0.10 - 0.20	0.12	0.17 - 0.49	0.26
3	0.20 - 0.32	0.24	0.26 - 0.65	0.51
4	0.32 - 0.42	0.35	0.49 - 1.19	0.72
5	0.42 - 0.53	0.46	0.42 - 1.56	0.94
6	0.53 - 0.69	0.58	0.67 - 1.76	1.14

2.2 Step 2 - Determine Response of Structures and Subsystems to Seismic Excitation

For each level of earthquake described by the seismic hazard curve (Step 1), three aspects of seismic response must be characterized to perform the seismic risk analysis: median response, variability of response, and correlation of responses. Seismic responses are required for all structures and components contained in the plant logic models (fault trees and event trees) (Step 4). The three aspects of seismic response are discussed briefly here.

- Median response--the median value of the seismic response to an earthquake is required. In general, the median response differs from the design values because, in the latter case, design analysis procedures, parameter selection, and qualification procedures are conservatively biased. An additional consideration, due to analyzing for the range of earthquakes, is the change in properties of the soil/structure/piping systems which occurs as excitation levels increase. For example, higher excitations lead to lower soil shear moduli, lower structure frequencies, and higher soil and structure damping characteristics. Such changes must be taken into account when determining best-estimate responses.

- Variability of response--variability in seismic response resulting from variations in the earthquake excitation, the physical properties of the soil/structure/piping system, and our ability to model them must be acknowledged and included in the seismic risk analysis to permit calculation of probability and uncertainty intervals.

- Correlation of responses accounts for response coupling with respect to the level of the earthquake and the dynamic characteristics of the system. A large earthquake (large peak acceleration) may cause all responses to be large, whereas a small earthquake produces the opposite effect. Correlation needs to be accounted for to model this correctly. The second source of correlation is due to system response itself. For example, floors within a structure may all experience high values of response for large earthquakes simultaneously due to the dynamic characteristics of the structure itself. Hence, equipment supported on these floors may simultaneously have high response. The importance of response correlation on frequencies of system failure, core melt, and radioactive release depends also on correlations between fragilities and the functional characteristics of the systems themselves.

For the SSMRP, these three aspects of seismic response were determined calculationally using the computer programs SMACS (Seismic Methodology Analysis Chain with Statistics) and SEISIM. A description of the SMACS code operation is given in Figure 2.3. SEISIM is described in Step 5.

SMACS links seismic input with the calculation of soil-structure interaction (SSI), major structure response, and subsystem (equipment and piping) response (Figure 3). Seismic input is defined by ensembles of acceleration time histories in three orthogonal directions. SSI and detailed structure response are determined simultaneously using the substructure approach to SSI. The response of subsystems is calculated by multi-support time history analysis procedures. Uncertainty is treated explicitly in each of these links of the seismic methodology chain by analyzing for an ensemble of free-field acceleration time histories and by varying a discrete number of input parameters in the soil, structure, and subsystem. SMACS performs repeated deterministic analyses, each analysis simulating an earthquake occurrence. By performing many such analyses and by varying the values of the input parameters, we

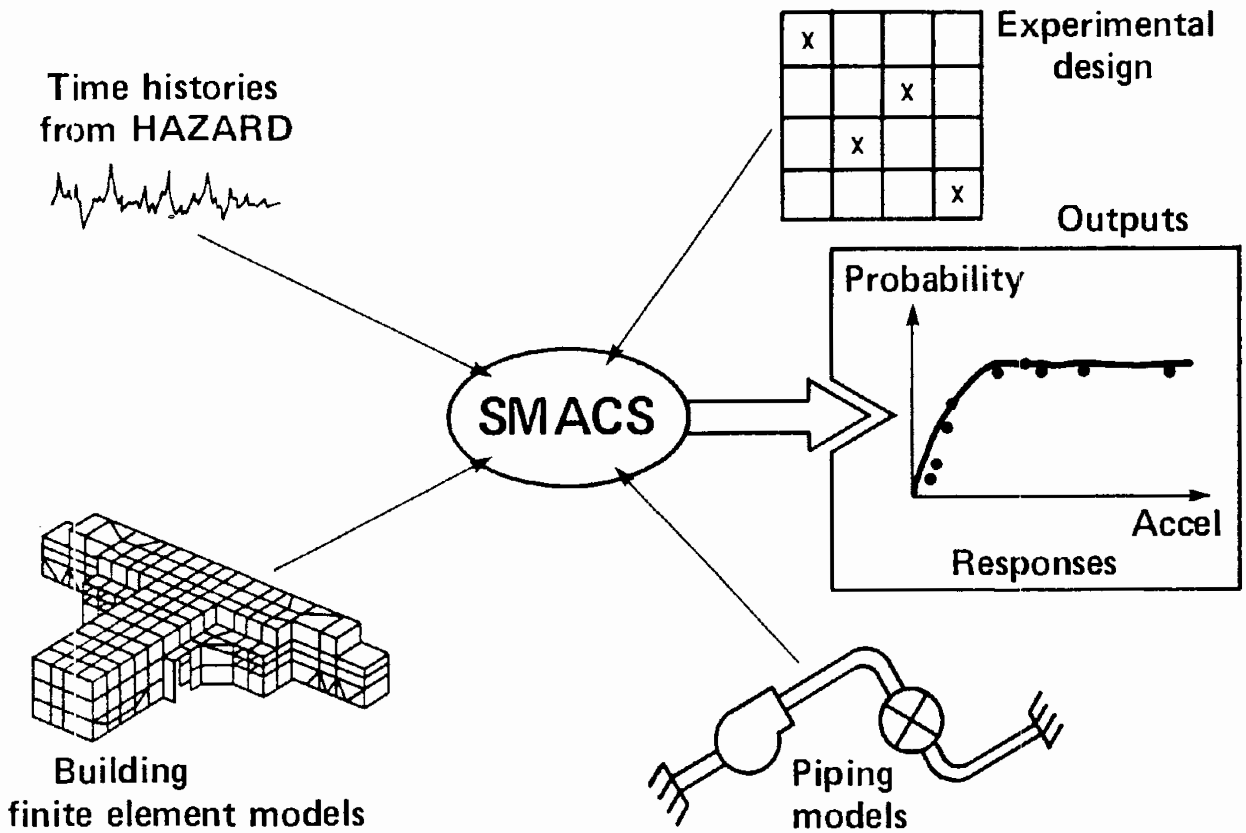


Figure 2.3 Description of SMACS Code Operation.

account for the uncertainty inherent in any deterministic analysis. Parameter values for each simulation were sampled from assumed probability distributions according to a Latin-hypercube sampling procedure. SMACS outputs a vector of peak responses. The response medians, correlation coefficients and coefficients of variation ( ) are calculated in SEISIM.

2.3 Step 3 - Determine Fragility Functions

The failure of structures or structural elements may occur in one of several possible modes. If the structure provides a pressure boundary, then a failure mode is loss of pressure boundary integrity. Structures whose main purpose is to support subsystems and components fail when they no longer provide adequate support. Secondary failures of subsystems and components occur when structural elements collapse causing their failure. An example of secondary failure is collapse of the pump enclosure roof on the service-water pumps. Component failure is defined as either loss of pressure boundary integrity or loss of operability. In all cases, failure (fragility) is characterized by a cumulative distribution function which describes the probability that failure has occurred given a value of loading. In the context of the SSMRP, the loading may be described by local spectral acceleration, local peak acceleration, or an internal force resultant such as moment, depending on the component and failure mode under consideration. Contrary to other studies, SSMRP fragility is related to the appropriate local response, rather than being related directly to the free-field peak acceleration.

Fragilities for structures and large components (such as steam generators) were developed uniquely for the Zion plant configuration. Fragility functions for other Zion components were treated generically. As a first step, all components identified in the fault tree analyses were grouped into 37 categories. Fragility functions for each category were developed from design analysis reports, experimental data, and an extensive expert opinion survey. Statistical methods were used to combine data from the several sources.

2.4 Step 4 - Identify Accident Scenarios

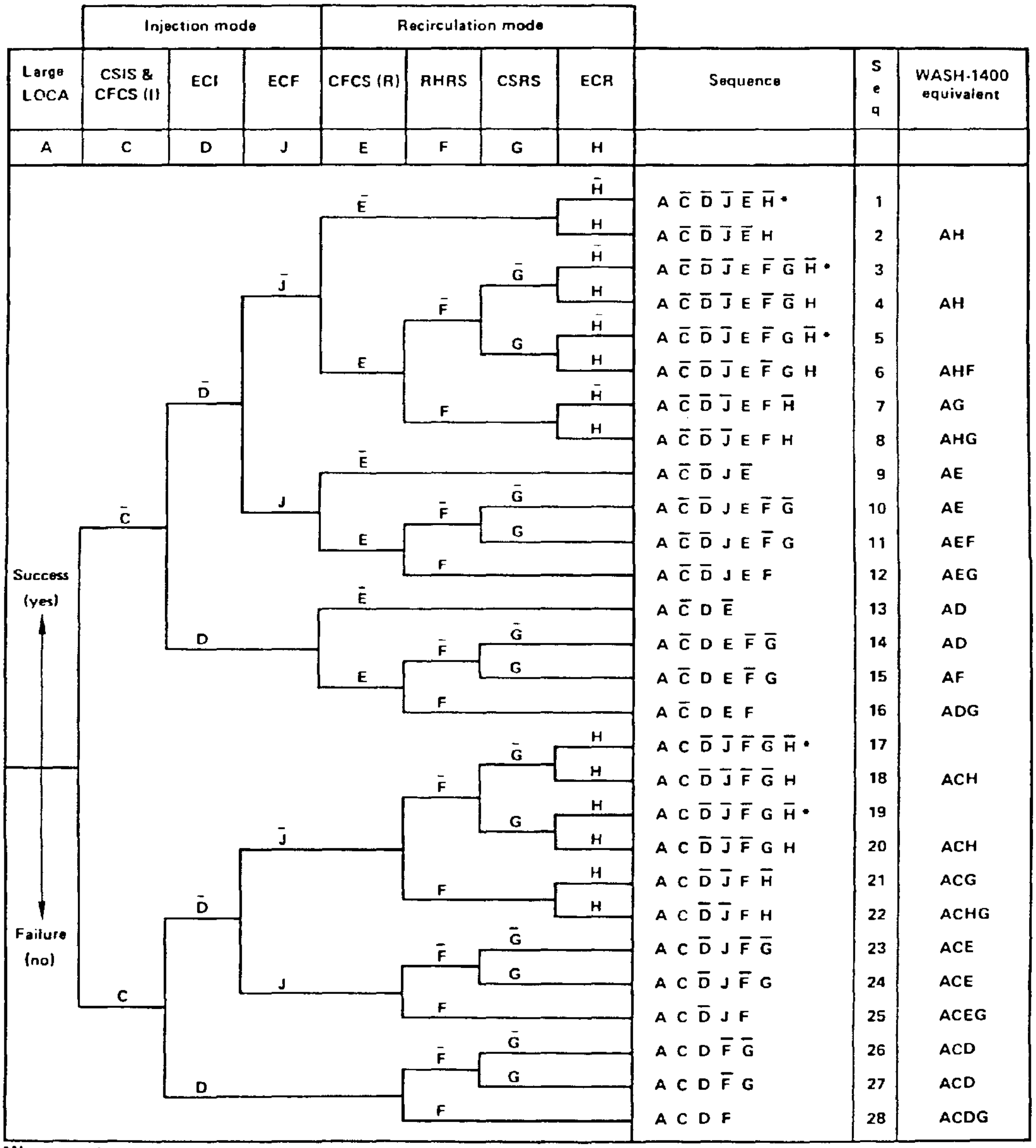
In the event of an earthquake or other abnormal condition in a power plant, the plant safety systems act to bring the plant to a safe shutdown condition. Should the earthquake damage essential components in the plant systems, it is possible that the effectiveness of the systems would be diminished. In an extreme case, it is possible that damage to the fuel rods or core melt could result. In this step of the risk analysis process, the possible paths that a reactor system could follow during a shutdown caused by an earthquake-related event are identified. These paths usually involve an accident initiator and a subsequent failure of one or more plant systems, and are referred to as accident sequences. For the SSMRP analysis of the Zion power plant, 315 accident sequences were identified.

All the identified accident sequences result from one or more seismically induced initiating events (events requiring immediate shutdown of the plant). In the SSMRP analysis of the Zion plant, seven classes of initiating events were identified. Although this number encompasses the possible accidents at Zion, a larger number of initiating events can be prescribed depending on the event-tree/fault-tree construction used. Four loss-of-coolant accidents (LOCAs) of different severity and two types of transients were considered. In addition, an initiating event "Reactor Vessel Rupture" was identified as a LOCA for which the emergency core cooling system (ECCS) cannot effectively flood the core. For this event it is assumed that multiple coolant loop failures occur. Pipe failure probabilities in the reactor coolant loop were calculated to give an estimate of LOCA frequencies. These initiating events were treated as mutually exclusive in the analysis.

For each of the seven initiating events, an event tree is constructed. Each branch of an event tree is an accident sequence. The event tree for the large LOCA initiating event (greater than 6 in. diameter equivalent pipe break) is shown in Figure 2.4. System descriptions for the systems listed across the top (CSIS, ECI, etc.) are given in Reference 2.

In determining failure modes for the plant systems, fault tree methodology is used. This methodology was originally developed in the aerospace industry to identify all the groups of system components whose simultaneous failure would result in failure of the system. Fault trees are used to determine the success or failure of each of the plant systems whose success or failure make up the accident sequence.

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



*No core melt

Figure 2.4 Large LOCA event tree.

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 or human error. These lowest-order failures on the fault tree are called basic events.

Fault trees are required for every system identified on the event trees. For the Zion plant, seven systems were modeled. The emergency core-cooling system was modeled with fault trees for the Safety Injection System (SIS), the Charging System (CHG), the Residual Heat Removal System (RHR), and the Accumulator System (ACC). The emergency core-cooling function is provided by different combinations of these systems in the injection and recirculation phases of a LOCA, and is dependent on break size. The auxiliary feedwater system (AFWS) is of primary importance in risk analysis, 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 support systems.

The size of the resulting fault trees varied considerably, as shown below:

System	Number of basic events	Number of logic gates
ACC	54	17
SIS	242	117
RHR	309	130
CHG	378	172
AFWS	1288	786

The basic events which resulted after all fault trees were constructed fell into three categories:

Seismically induced failures	1905
Human and maintenance errors	583
Other random failures	20

In all, 2508 basic events were considered in the Zion analysis.

2.5 Step 5 - Calculate Probability of Failure and Frequency of Radioactive Release

Step 5 combines the results of steps 1-4. With respect to the seismic hazard at the site in question, Step 5 calculates:

- Structure and component failure probability.
- Probability of failure of a group of structures and components, e.g., a system.
- Frequency of radioactive release.

Each of these elements is discussed briefly here. The computer program SEISIM was written to perform these probabilistic calculations, including treatment of the common-cause nature of earthquakes, as discussed below. The discussion here emphasizes the calculational process to obtain a point estimate of the quantities of interest. Section 2.6 discusses the methodology to obtain uncertainty intervals. As mentioned earlier, the seismic response (Step 2) and systems analyses were performed for discretizations of the seismic hazard curve. The final step in the process is convolution of these conditional results with the seismic hazard curve.

To calculate the probability of failure of structures and components requires input from Steps 1-3. For each discretization of the seismic hazard, the results of the SMACS analysis define the response of the structures and subsystems of interest. Correlation coefficients are calculated and used in succeeding stages of Step 5. The fragility functions of Step 3 are convolved with the response functions to yield the conditional probability of failure (conditional on the occurrence of an earthquake).

Accident sequence probabilities must be calculated to determine radioactive release frequencies. As discussed in Step 4 above, each accident sequence consists of groups of events which must occur together. The failure of each plant system can be represented in terms of cut sets, which are groups of component failures which will cause the system to fail. System cut sets for the accident sequences are combined so that every accident sequence can be expressed in a Boolean expression of the form

$$ACC = C_1 \, C_2 \, C_3 \text{ or } C_4 \, C_5 \text{ or...or } C_i \, C_j \, C_k \ ,$$

in which the C_i are basic events (i.e., failures of individual components) identified on the system fault trees. If at least one of the component failure groups $C_i\ C_j\ C_k$ occurs, then the accident sequence occurs. As previously mentioned, in the Zion analysis a total of 315 accident sequences were considered. These accident sequences contained up to 5000 component failure groups (in systems analysis terminology, called min cut sets). Each of these component failure groups was allowed to have up to ten basic event failures. The calculation then proceeds as follows.

- Probability of failure of a group of structures and components.
  To calculate the probability of an accident sequence (that is, to evaluate the probability of simultaneous, or joint, component failures in a group C_i ... C_k), it is necessary to evaluate the probability of the component-failure groups, Prob($C_i\ C_j$ ... C_k). If these events were independent of one another (so that the failure of one component had no correlation with the failure of any other component), then a well-known law in probability theory states that the probability of several independent events is equal to the product of their individual failure probabilities. That is,

$$\text{Prob}(C_1\ C_2 \dots C_n) = \text{Prob}(C_1)\text{Prob}(C_2) \dots \text{Prob}(C_n).$$

  However, when failure is caused by ground shaking due to an earthquake, all parts of the plant are shaken together. Therefore, the accelerations or loads which the various buildings and components experience are highly correlated. This correlation increases the probability of occurrence of the component failure group. In fact, two components sitting side by side would experience essentially the same floor accelerations, and thus their responses would be said to be "perfectly correlated." Hence, if either component failed, it would be more likely that the other component would also fail. The computer code SEISIM was written expressly to calculate such dependent failures (Figure 2.5). 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 SMACS analyses), SEISIM constructs a multivariate normal distribution for each component-failure group, and then uses n-dimensional numerical integration to compute the probability of the component-failure group occurring.

- Frequency of Radioactive Release.

  Once the component failure group probabilities have been computed, the probability of each accident sequence can be found. To obtain the frequency of radioactive release, the accident sequence probability (which is a measure of the frequency of core melt) is then multiplied by the probability of the earthquake's occurrence and the probability of failure of the containment. Different containment-failure modes are assigned to different accident sequences, depending on the physical processes involved. One accident sequence can result in one or more containment failure modes. For Zion, radioactive releases were sorted into seven different release categories to reflect their severity with respect to the surrounding population. These release categories relate to the type and energy content of the radioactive fission product release, as well as to the mode and timing of the release. They range from rupture of the top of the containment with a rapid, energetic release (due to a fuel/water explosion or steam overpressure) 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 were those derived and used for the pressurized water reactor studied in the Reactor Safety Study (Ref.3). Each of the seven release categories can be assigned a corresponding dose to the public for a PWR at an average site (Ref.4). The total public dose in terms of man-rem per year for each release category was obtained by multiplying the frequency of the release category by the corresponding dose. The frequency of core melt is the sum of release frequencies from all categories.

2.6 Uncertainty Analysis

In seismic risk analysis, it is important to recognize two types of uncertainty--random uncertainty and modeling uncertainty. Random uncertainty is the inherent randomness associated with the events of interest. It is fundamental to the phenomenon being represented and it is also irreducible given present state-of-the-art understanding and modeling of the phenomenon. Modeling uncertainty reflects incomplete knowledge of the models and calculational techniques used to estimate risk. Modeling uncertainty, in many cases, can be reduced within present limits of

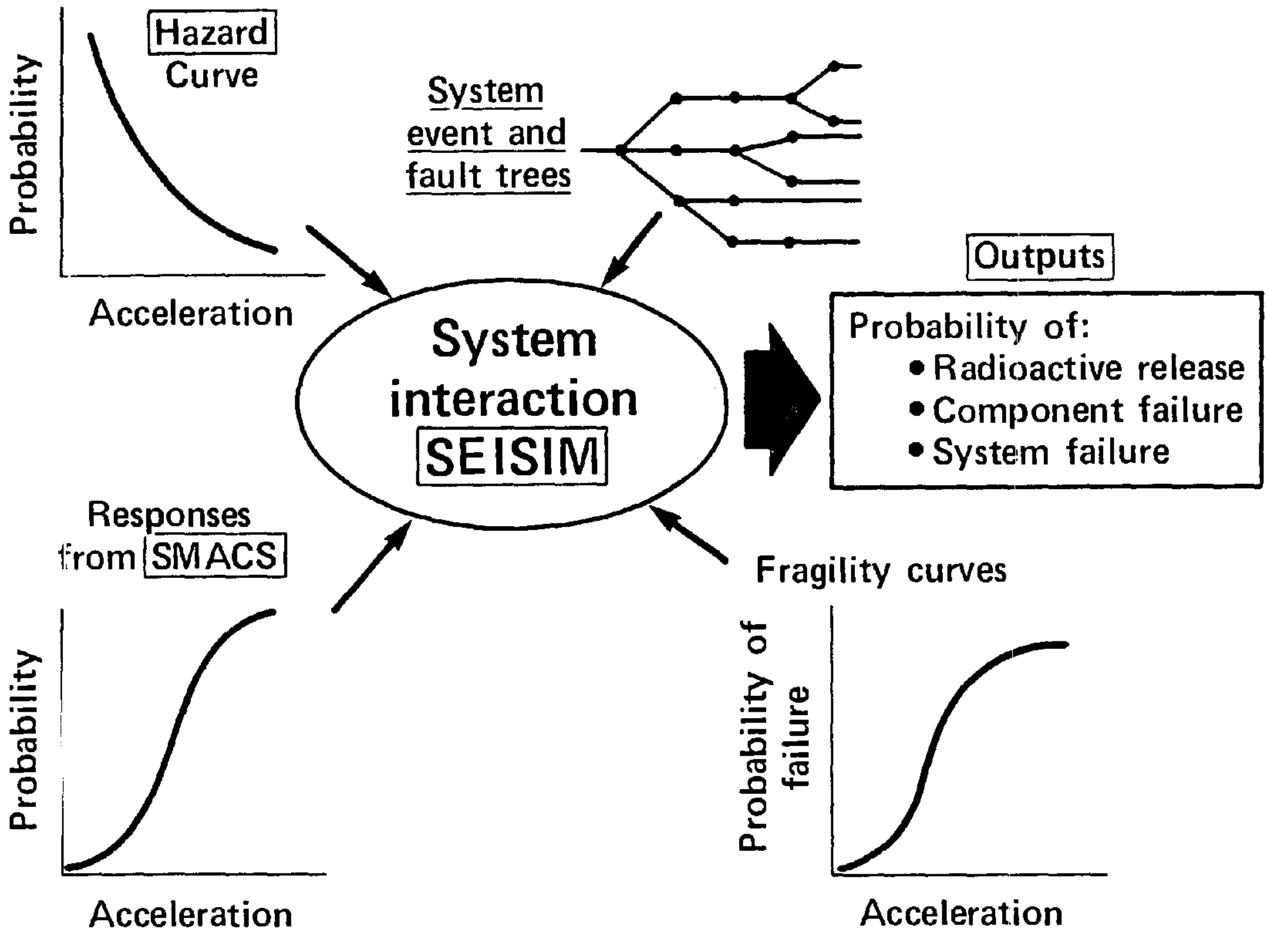


Figure 2.5 Description of SSMRP Seismic Risk Calculation Method – HAZARD, SMACS and SEISIM are the Three Computer Codes Used to do the Calculation.

knowledge by improved analytical models, experiments, etc. Both types of uncertainty exist in all of the elements that comprise the five steps in the SSMRP methodology described earlier. The separation of random and modeling uncertainty and the methodology used to propagate the modeling uncertainty through the probabilistic calculations is denoted "uncertainty analysis."

One goal of the SSMRP uncertainty analysis was to describe the variation in the estimated frequency of radioactive release due to the uncertainties associated with the risk analysis techniques. Perhaps the optimal method for assessing the effect of analysis uncertainties on the estimator of risk would be to analytically propagate the uncertainties through the analysis. Because of the complexity of the analysis, this was not possible. Alternatively, a Monte-Carlo simulation procedure was selected to assess the variation in the output probabilities due to modeling uncertainties.

The procedure to propagate random and modeling uncertainties is a two-loop process; the outer loop treats modeling uncertainty and the inner loop treats random uncertainty. The parameters varied on the outer loop are the seismic hazard function, the medians of structure and soil parameters in the calculations of the seismic responses (resulting from the SMACS analyses), the medians of the fragilities, and the probability of failure of unmodeled systems. For each inner loop calculation, these parameters are fixed. Usually, a simulation analysis such as the outer loop involves a large number of iterations of the basic risk analysis. However, the number of iterations can be reduced by appropriately choosing the sets of inputs to be used for each iteration. The basis of one such method is to choose the values for each input so that the entire range of values for the input is represented in the sample of inputs used. The designs of the outer and inner loops are based on using such a sampling method, called a Latin hypercube design. Using this modified Monte Carlo approach, discrete evaluations of the seismic risk can be performed to span the range of modeling uncertainties in the calculational procedure. From these discrete analyses, median values of release frequences and uncertainty intervals on the median values can be inferred. Fourteen evaluations of the seismic risk were done for the Zion analysis.

Section 3: RESULTS OF THE ZION RISK ANALYSIS

This section presents the results of the base case, point estimate calculations made for the seismic risk analysis of the Zion Nuclear Power Plant. The calculations of the median core melt frequency and the uncertainty intervals are also described. Further details are given in Reference 2. Highlights are shown in Table 3.1.

Table 3.1: Highlights of SSMRP Zion Seismic Risk Analysis

- Median core melt probability calculated as 3.E-5/year.
- Uncertainty on core melt was about 3 orders of magnitude between the 10% and 90% confidence levels.
- Uncertainty in the hazard curve contributed about one-half of the total uncertainty. Uncertainty in fragility contributed most of the rest.
- Risk came from earthquakes in the 2 through 4 times Safe Shutdown Earthquake (SSE) range.
- Risk was dominated by soil and structural failures, failures of piping between buildings, and electrical equipment failures (loss of off-site power).

3.1 Results for the Base Case with Random Uncertainty Only

The base case uses the best estimate of the configuration of the Zion plant and its emergency procedures. A number of important assumptions were made in doing the base-case analysis.

1. It is assumed that "feed-and-bleed" emergency core cooling can be performed after an earthquake. In this procedure, which is employed if the auxiliary feedwater system has failed, the operator makes use of the

emergency safety pumps to pump cooling water to the core. The resulting steam is bled-off through the pressurizer relief valves.

2. The identified structural failure modes are assumed to have the most serious consequences. Two structural failure modes play crucial roles.

   (a) The failure of the roof of the service water pump enclosure (at the top of the crib house) is assumed to fail all six service water pumps beneath it, leading to loss of the emergency AC power diesel generators, due to lack of cooling water.

   (b) The failure of the wall between the turbine building and the auxiliary building is assumed to cause loss of all electrical wiring and control circuits, so both power and control to the containment building are lost.

   Both of these structural failures are assumed to fail the safety injection system (SIS), the charging system (CHG), the containment spray injection system (CSIS), and the containment spray recirculation system (CSRS).

3. Soil failure under the foundation of the containment is assumed to result in sufficiently large rocking motions so as to fail the SIS, CHG, RHR, CSIS and CSRS piping between the auxiliary-fuel-turbine (AFT) building and the containment building.

4. Failure of the vertical column supports under the steam generators and reactor coolant pumps is assumed to result in a double-ended guillotine break of the primary coolant piping, equivalent to a large LOCA. Failure of supports in two different loops is assumed to result in a reactor vessel rupture initiating event.

These assumptions all play crucial roles in the base-case results.

To illustrate the important accident scenarios, point estimate calculations for the base case were done. Only random uncertainty was included. A median hazard curve was used (see Fig. 2.2). All input variables were assigned best-estimate values and response medians and standard deviations calculated. A single set of fragility curves (median curves) was used to calculate point estimate failure, core-melt, and release frequencies. These point estimate calculations are useful for comparison and ranking within the SSMRP analysis and take less time to produce. Median or mean estimators which include modeling uncertainty should be used to compare with other analyses.

Table 3.2 presents the frequencies per year of occurrence of the seven release categories and the radiation dose associated with each release category. Release categories 2 (containment failure due to overpressure) and 7 (melt-through of basemat) have the highest frequencies of occurrence, 1.E-6 each. The man-rem/year dose comes from release categories 2 and 3 (containment failure due to overpressure). The total frequency of core melt is seen to be 3.E-6 per year and the total release is 9.6 man-rem/year. The releases were found to be due primarily to the failures of certain local structural elements and inter-building piping, which resulted in common-cause failures of the safety systems.

Table 3.2: Release categories and frequencies of release per year for the base case (with feed-and-bleed and structural failures), point estimate calculation.

Release	Earthquake Level							man-rem/
Category	1	2	3	4	5	6	Total	yr.
1	3.E-13	4.E-11	3.E-9	1.E-8	8.E-9	6.E-9	3.E-8	0.2
2	1.E-8	3.E-7	6.E-7	3.E-7	1.E-7	8.E-8	1.E-6	6.5
3	4.E-10	4.E-9	1.E-8	4.E-7	8.E-8	1.E-8	5.E-7	2.9
4	0	0	0	4.E-11	5.E-11	6.E-12	9.E-11	0
5	4.E-12	1.E-11	9.E-12	3.E-10	5.E-10	6.E-11	8.E-10	0
6	7.E-16	1.E-13	4.E-12	1.E-7	4.E-8	6.E-9	1.E-7	.03
7	4.E-9	8.E-8	8.E-7	4.E-7	2.E-7	5.E-8	1.E-6	.03
TOTALS	2.E-8	4.E-7	1.E-6	1.E-6	4.E-7	2.E-7	3.E-6	9.6

In terms of both core melt frequency and dose, it was found that earthquake levels 3 and 4 were dominant and the frequencies and dose were significantly smaller at earthquake levels 1 and 6. This indicates that the bulk of the risk is captured in the middle earthquake levels (2 through 4 SSE), and that an adequate range of earthquakes was covered.

It was found that the dominant initiating events at the three lower earthquake levels were the transient initiators. At earthquake level 4, it is primarily the small and small-small LOCAs that are important. At earthquake level 5 the initiating event probabilities are fairly evenly spread over the initiating events and the LLOCA and RVR initiating events have become significant. Finally, at level 6 the dominant initiating events are the RVR and LLOCA events. Thus, we see that the contribution of the more severe initiating events increases as we increase the level of earthquake excitation.

The seismically-induced failure causing the two transient initiating events is primarily the loss of offsite power by failure of the ceramic insulators at the point where offsite power is brought into the switch yard. Failure of the condensate storage tank also contributes. The component failures which cause the LLOCA and the RVR initiating events are the failure of the primary coolant piping due to the failure of the supports of the steam generators and reactor coolant pumps. Without these two failures the initiating events for the RVR and the LLOCA would be significantly smaller. Thus, it was found that it is not failure of the piping which results in a RVR or LLOCA but rather the possibility of failure of the supports of the major components.

At the lower three earthquake levels, both the release category frequencies and the dose are dominated by failure of the auxiliary feedwater system caused by structural failures. The uplift of the containment basemat causes failure of the AFWS pipes, and is also assumed to fail the containment sprays. When the containment sprays fail, the release occurs in release category 2 (80%) and release category 7 (19%). A second contributor is failure of the cribhouse service-water pump enclosure room roof slab. This is assumed to fail the six service water pumps, which in turn fails the diesel generator due to lack of cooling water. This, in conjunction with loss of off-site power, results in loss of all AC power, and hence loss of both the AFWS and the containment sprays.

For earthquake levels 4, 5 and 6, significant contributions are found from release category 3 as well as 2 and 7. For the upper three earthquake levels, the containment sprays are assumed to have failed due to ground shaking alone. (For levels 1, 2 and 3, the containment sprays were assumed to have failed <u>only</u> in those accident sequences caused by the structural failures.) Release category 3 is due almost entirely to small LOCA sequences, which are caused by the failure of pairs of pipes between the containment and AFT buildings. These pairs of pipes fail due to differential motion between the buildings. Failure of any one of these pipe pair combinations causes failure of both emergency core injection and the RHR systems. The release in category 7 at the upper earthquake levels is due to two small LOCA accident sequences which are both the result of loss of emergency core cooling due to uplift and service water pump room roof failures.

In summary, out of the total 9.6 man-rem/year, approximately 6.1 man-rem/year is due to accident sequences caused directly by the uplift and crib house pump room roof failures, and 2.7 man-rem/year is due to failures of pairs of pipes between the containment and AFT buildings. At the three lower earthquake levels, transient accident sequences predominate, while at the upper three earthquake levels, the smaller LOCAs predominate. Thus it is seen that, for the base case computations of the seismic risk at Zion, the structural failures and assumptions as to their consequences play a dominant role.

3.2 Frequency of Core Melt.

A Monte Carlo procedure was used to determine the probability function relating the probability and frequency of core melt release per year. Repeated calculations were made of the core melt frequencies for the base case, while varying the median values of all input variables by sampling from distrbutions of input variable values. Fourteen calculations were performed, with new values for structural responses, fragility curves and hazard curves, for each calculation. The median value and uncertainty intervals were inferred from these 14 calculations. The median frequency of core melt for the plant was calculated to be 3.E-5 per year. This value reflects inherent randomness in all the input variables and the hazard curve, as well as modeling uncertainties attributable to lack of exact knowledge of mean values of input variables. The 10% - 90% uncertainty band on the core melt frequency was found to be about 3 orders of magnitude. Figure 3 shows these results.

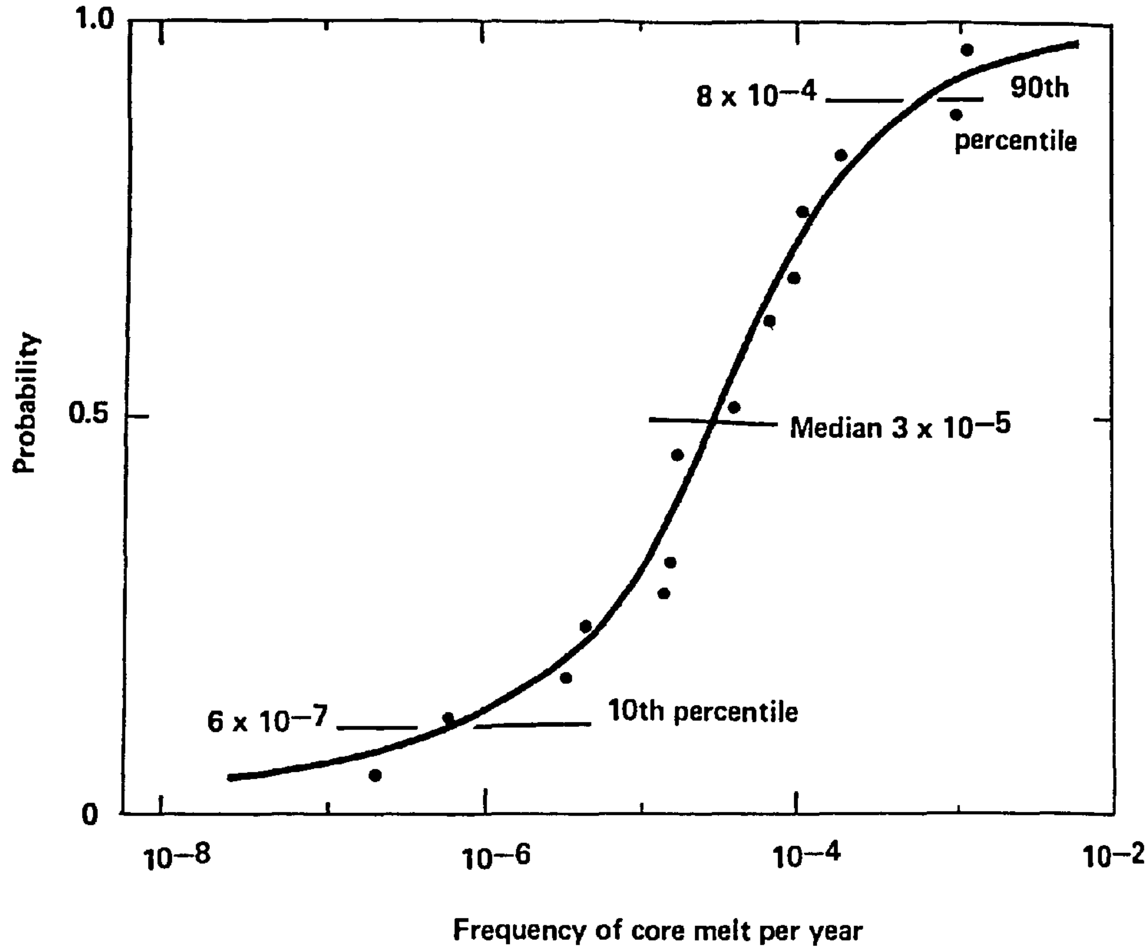


Figure 3. Uncertainty intervals on frequency of core melt. The data points are from the 14 runs described in Section 2.6.

3.3 Sensitivity of Risk Results to Basic Assumptions

To test the fundamental assumptions on which the base-case results were predicated, three additional point estimates of risk at Zion were calculated, with the results shown in Table 3.3. In Case 1, the effects of the structural failures (the service water pump enclosure room roof, auxiliary building shear wall, and soil failure and basemat uplift) were removed, but the "feed-and-bleed" capability was retained. In this case, both the frequency of core melt and radioactive dose decreased by a little more than 50% relative to the base case.

Table 3.3: Comparison of cases designed to test the effects of fundamental assumptions.

	Release Category	Frequency Per Year	Man-Rem/Yr.
Base case (with feed-and-bleed, with structural failures)	Total	3.E-6	9.6
Case 1 (with feed-and-bleed, without structural failures)	Total	1.E-6	4.3
Case 2 (without feed-and-bleed, without structural failures)	Total	9.E-6	6.6
Case 3 (without feed-and-bleed, with structural failures	Total	1.E-5	10.8

In Case 2, both the effects of structural failures and the "feed-and-bleed" cooling capability were removed. For this case, the core melt frequency increased by a factor of 2 1/2 over the base case. This increase in frequency results because with no "feed-and-bleed" cooling capability, the auxiliary feedwater system (AFWS) has no back-up, and thus electrical component failures in the AFWS became significant.

In Case 3, the effects of the structural failures are included, but no "feed-and-bleed" capability is assumed. This results in the highest (point estimate) values of both core melt frequency and radioactive release. The core melt frequency is over 3 times higher than in the base case, and the radioactive release is 13% higher than for the base case.

In summary, it can be seen that, depending upon the assumptions made as to the consequences of the localized structural failures and the credibility of performing "feed-and-bleed" cooling, the core melt frequency can vary by an order of magnitude, and the release can vary by 250%.

A number of other cases were analyzed to demonstrate the importance of various aspects of modeling the Zion plant. These cases showed the local amplification of the free-field motion was quite important. The effect of structure-to-structure interaction on core melt frequency was not important. The assumption of rigid foundations was appropriate. Finally, several cases were analyzed that showed the effects of correlation between responses in the plant due to the common ground shaking and correlation between fragilities. It was found that correlation has little effect if the total risk is due primarily to single failures but can change the risk result by an order of magnitude if the risk is dominated by pairs of failures.

Section 4: ADDITIONAL SSMRP EFFORTS

Following the completion of the Zion Nuclear Power Plant analysis in Phase II of the SSMRP, additional studies were undertaken prior to completion of the Program. These included additional sensitivity and importance studies to test assumptions made in the Zion analysis or to amplify the results. One of the studies ranked the importance of input parameters, systems, structures, and components to core melt probability and risk. Others looked at specific phenomena such as relay chatter and soil-structure interaction.

In another area, guidelines were developed and data generated to allow for a more simplified analysis of seismic risk. To do this, response factors were calculated based on the Zion analysis which allow conversion of design response calculations to best-estimate calculations. Hence, the original design data can be used to give response data for the risk assessment minimizing the need for an expensive generation of structure and piping models.

Finally, some separate studies were performed to help establish the validity of the SSMRP process. These included, among other things, a study of the fragility of reinforced concrete structures and a comparison of SSMRP Zion results with a separate utility seismic risk assessment. A listing of these studies is shown in Table 4.1 and a brief discussion of the results are given in the text that follows.

4.1 Sensitivity Studies

Importance and Sensitivity.

The importance and sensitivity of input parameters affecting the Zion seismic risk are presented in Reference 5. These results can be used to focus attention on key contributors to risk or parameters which most affect risk. Since the results are specific for Zion, generalization to other nuclear power plants must be made with caution. The analyses addressed the following questions:

Table 4.1: SSMRP Phase II Follow-on Studies

Topic	Description	Reference
I. SENSITIVITY STUDIES		
1. Importance and Sensitivity of Parameters Affecting the Zion Seismic Risk.	Presents results as to the importance and sensitivity of structures, systems, equipment, components and input design parameters to seismic risk at Zion.	UCID-20475
2. Sensitivity of Piping Seismic Responses to Input Factors.	Investigated the sensitivity of calculated peak dynamic seismic responses of piping to piping damping values and other input parameters.	UCID-20466
3. Soil-structure Interaction (SSI) Sensitivity Studies and Model Improvements.	Investigates modeling effects on SSI predictions for flexible foundation modeling, structure-to-structure interaction, and basemat uplift.	NUREG/CR-4018
4. Comparison of the Effect of Hazard and Response/ Fragility Uncertainties on Core Melt Probability Uncertainty.	Proposes a method for comparing the effects of the uncertainty in input parameters such as hazard and fragility on the uncertainty in the calculated risk.	UCID-20326

Table 4.1 (Continued)

Topic	Description	Reference
5. Circuit Breaker Operation and Potential Failure Modes During an Earthquake.	Describes potential circuit breaker failure modes during earthquake strong motion due to relay chatter effects.	UCID-20086
II. SIMPLIFIED METHODS DEVELOPMENT		
1. Simplified Seismic Probabilistic Risk Assessment Guidelines.	Provides guidance for executing a seismic risk analysis based on simplification of SSMRP methods.	NUREG/CR-4331
2. Categorization of PWR Accident Sequences and Fault Trees.	Developed a set of dominant accident sequences for PWR's based on the Zion analysis.	UCID-20211
3. Seismic Margin and Calibration of Piping Systems.	Developed calibration factors relating best estimate seismic responses of piping systems to design response values.	UCID-20314
4. SSI Response of a Typical Shear Wall Structure.	Developed calibration factors to relate best estimate response to design values accounting for simplifications in SSI analysis procedures.	UCID-20122

Table 4.1 (Continued)

Topic	Description	Reference
III. VALIDATION STUDIES		
1. Seismic Fragilities of Reinforced Concrete Structures and Components.	The failure and fragility analysis of reinforced concrete structures and elements is evaluated.	NUREG/CR-4123
2. The Effects of Basemat Uplift on Structural Response.	Additional discussion on the effects of basemat uplift on Zion results.	UCID-20446
3. Impact of Assumptions Concerning Containment Failure on Seismic Risk.	Reanalyzed dominant Zion accident sequences to assess the effect of different containment failure mode assumptions.	UCID-20163
4. Damage to Safety Systems from Structural Failures at Zion.	Re-evaluated the consequences of failure of the crib house roof and the common wall between the turbine and auxiliary buildings.	Reference 17
5. Comparison of Utility and SSMRP Seismic Risk Studies at Zion.	Compares results from the SSMRP and utility seismic risk analyses at Zion.	UCID-20464

a. Which inputs contribute substantially to release probability and risk? Which are the important responses, fragilities, plant systems, accident sequences, and accident scenarios? The answers to these questions help focus future effort.

b. What are the rates of change in probabilities and risk when input parameters such as means, standard deviations, and correlations of responses, strengths, and primary input variables change for one category of components, for one type of response, or for one primary input variable? The answers to these questions help decide how much effort should be spent on each improvement.

c. What are the changes in probabilities for discrete shifts in input parameters? The answer to this question indicates the benefit of research or manufacturing change that shifts an input parameter by a finite amount.

Importance and sensitivity were measured by probability of occurrence, derivatives of probability or risk with respect to changes in means or standard deviations of responses and fragilities and Vesely-Fussell importance measures. This latter measure is used to rank components and is the probability of the cut sets which contain the component failure divided by the total failure probability.

Looking at the set of questions in "a" above, it can be asked which scenarios are important, or which elements of scenarios (initiating events, accident sequences, etc.) are important to risk. It turns out that of the 1000-plus scenarios, about 20 comprise more than 90% of the core melt probability and more than 90% of the risk. An accident scenario includes the accident initiating event, accident sequence and containment failure mode.

Among accident sequences, seven (with five different initiating events) comprise more than 90% of the risk and about 80% of the core melt probability. Five of these seven incorporate failures of two systems:

- CSIS and CFCS failure (during injection phase)
- RHR system failure

The auxiliary feedwater and secondary safety relief systems are also important. The top seven accident sequences based on probability of their occurrence are listed in Table 4.2.

Table 4.2: Zion Top Seven Accident Sequences (about 80% 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.

**Base case, point estimate calculation.

The sensitivities to input parameters (those which control vibratory loads on components and control the strength of the components to sustain such loads), when ranked, give another indication of what is important toward being able to change the risk. When associating the parameters with their subject components or physical effects, the following list results. These are the top five in terms of risk importance.

- Local Site Effects
- Piping Layout Between Buildings
- Piping Fragility (or Strength)
- Crib House Pump Enclosure Roof Fragility
- Base Slab Uplift Fragility

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 2 to 10 times those 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 foundations. The relative motion of these buildings at accelerations greater than the safe shutdown earthquake (SSE) causes high stresses and strains in such piping and is a known problem area based on past earthquake data. This area would probably not be so important for a rock site, or if the piping supports fail before the piping. SSMRP studies did not take into account possible failures of piping supports.

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 to risk in plant systems and then under special circumstances such as piping between buildings. Results of this study arise because (a) pipe breaks are initiating events for the more severe initiators such as LOCAS as well as (b) the assumption that the possible bias in the estimated fragility of all piping in the plant is biased high or low. It is reasonable to assume simultaneity on the possible bias. Thus, the result is explained by considering (1) the pervasiveness of piping throughout the plant and (2) the capacity of an earthquake to simultaneously threaten all piping in a plant (and seek out any weaknesses).

Crib house pump enclosure roof fragility ranks high because of (1) it's relatively low capacity 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. It should be noted, however, that subsequent to the importance analysis done in Reference 5, the possibility of crib house roof failure was reanalyzed and found less threatening to the equipment inside (see Section 4.3). It appears unlikely that all six service water pumps would be put out of service given the roof failure. Thus, on reanalysis, this failure may not rank so high.

Base slab uplift fragility refers to the failure of the soil beneath the foundation of the containment building at accelerations beyond the SSE. This category is important because it is assumed to lead to failure of the piping between the containment building and 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. Recent studies have shown that this type of soil failure may not contribute to piping failure as significantly as assumed in Reference 5. Hence, the importance of the uplift fragility would be less (see Section 4.3).

Finally, there is an important category that is not on the above list: relay chatter and circuit breaker trip. In the SSMRP and most other seismic risk analyses, relay chatter was 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 would have a "significant" effect on the SSMRP risk results. These analyses have not been performed and so no accurate indication has been made of how much is "significant." However, an estimate of the inclusion of these failures is that their inclusion could lead to an order of magnitude or more increase in the reported median annual probability of core melt of 3.E-5. The actual effect of relay chatter depends on circuit design and must be analyzed for specific cases. Further details are given in Reference 6.

The rankings do not reflect a study of uncertainty in the seismic hazard function. This is because the focus was on possible seismic research in areas related to the site and plant and because while research on the seismic hazard may reduce the perceived risk or uncertainty in estimates of risk, it will not reduce the actual risk. However, note that the local site effect is closely related to the seismic hazard. The importance of the local site effect found in this study points out the importance of the hazard.

The rankings also do not reflect a study of the possible effects of assumptions on plant logic models. This is because the emphasis was on seeking guidance for research in the traditional technical areas associated with earthquake-related research. Guidance on how to allocate research resources between seismic and other research was outside the scope of the study.

Sensitivity of Piping Responses to Input Factors.

A linear regression analysis was used to estimate piping response (acceleration and moment) sensitivities to input parameters such as damping and frequency (Ref. 7) as calculated by the SMACS computer code. Two sensitivity topics motivated the study. The first was to study the sensitivity of piping response to the mean value of piping damping. The second was to study the sensitivity of piping response to all other earthquake and model input parameters.

It was found that a primary source of the variability in piping response was due to the local site effect caused by the shallow soil layer at Zion. The most important parameter for this variability was soil damping. It also was found that sensitivity results changed from location to location in the plant. In general, variances were greatest for piping moments followed by piping acceleration, structural responses, and free-field responses. Sensitivity of pipe response to pipe damping was usually less than to frequency content of motion. At some locations, e.g. near an anchor, neither was important. Other results can be found in Reference 7.

Soil-Structure Interaction Sensitivity Studies.

Sensitivity studies relating to soil-structure interaction effects were conducted in three areas: flexible foundation modeling, structure-to-structure interaction and basemat uplift (Ref. 8). The auxiliary/fuel handling/turbine building (AFT) complex foundation was modeled to behave rigidly in the SMACS analysis. A comparison of building responses for the assumption of ridgid behavior and flexible behavior showed 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 phenomenon is called structure-to-structure interaction. It is of significance when small distances separate adjacent structures and 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 less significant. There was 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. Separation of the soil and foundation may not in itself be a problem; however, 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, this failure mode is an important effect when considering earthquakes at nuclear power plants. However, more recent studies using non-linear analysis have shown that the effect of basemat uplift on inter-building piping may not be so important as Reference 8 indicates (see Ref. 9).

Effect of Hazard and Response/Fragility Uncertainty.

Reference 10 discusses a proposed method for comparing the effects of uncertainty in seismic risk analysis input parameters on the uncertainty of risk results. The method is used to compare the effects of uncertainties in the seismic hazard and response/fragility descriptions as to their influence on the Zion core melt probability. The method is based on an analysis of the variance of the predicted risk. Uncertainty in the probabilistic inputs a PRA is usually described by associating an "uncertainty" distribution with the characteristics of the probabilistic inputs. Thus, the predicted risk can be treated as a random variable, and the variance is a natural measure of variation or uncertainty. Although the data available does not satisfy all requirements of a valid analysis, application of the methodology suggests that the effect on the uncertainty in the risk prediction of seismic hazard uncertainty and fragility/response uncertainty are comparable.

4.2 Simplified Method

The basic objectives of the SSMRP simplified seismic risk assessment (Ref. 11, 12, 13, 14) methodology are to save time and money and to adequately estimate seismic risk. Several assumptions served as a point of departure for development of this method.

- Systems information about the plant to be analyzed is available and an identification of unique features relating to seismic risk has been made. It should be noted that 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 site will be available. Studies such as the NRC EUS Seismicity Project will provide seismic hazard curves for eastern U.S. sites (Ref. 15). Special studies may be needed of western U.S. sites.

- 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
- Plant logic
- Response and fragility

Simplification of the seismic hazard models is accomplished by eliminating the need for the development of time histories and relying instead primarily on response spectra. In some cases, it may be prudent to develop time histories for limited site- and plant-specific calibration purposes.

Some selected simplifications in plant logic models are suggested. However, complete probabilistic risk assessment needs should dictate requirements here.

The major difference between the detailed and simplified methods is in the generation of response data. The SSMRP detailed seismic risk analysis method involves detailed response calculations as a means to relate free-field acceleration to fragility based on local response quantities. This requires detailed modeling and expensive calculations. It was 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. Calibration means: To develop a response calibration factor, F_R, that provides a relationship:

$$F_R = R_D/R_{BE}$$

between seismic responses used in the plant design, R_D, and a best-estimate response, R_{BE}. R_D is developed for the design earthquake and thus keys the responses to free-field acceleration and design ground response spectra at that level. R_D is assumed a median value.

F_R 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. As an example, Table 4.3 shows response factors to account for conservatisms associated with ignoring soil-structure interaction, i.e., assuming a fixed base structure, for shear walls.

Table 4.3: Response Factors for Shear Wall Structures

Soil Stiffness Characteristic V_s (fps)	FR (Peak Accelerations)	F_R (Peak Forces)
3500	1.07	1.23
2000	1.25	1.42
1000	1.47	1.64
500	2.02	2.14

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%.

Several sensitivity studies were performed to obtain the response factors, F_R. These studies assessed the potential influence of the many factors that might vary in the design of the various plants to which the SSMRP simplified seismic risk assessment method might be applied. The factors studied included parameters like damping as well as alternative methods of analysis that are used in 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 ten different methods of analysis of piping systems.

Point estimates of core melt probability obtained using the SSMRP detailed and simplified seismic risk assessment methodology were compared. The core melt probability for the simplified method was calculated as about four times the base case result from the detailed method and the dose twice as much. The major contributors remain basemat uplift and inter-building pipe failures. The absolute results, although different, are within the uncertainty bounds and the key contributors remain much the same. Further investigation will be necessary to explain the difference.

The simplifications discussed above greatly reduce the complexity of the analysis but does add to the uncertainty in the results. Performing selected response calculations at two different earthquake levels can help minimize this increased uncertainty by providing a basis for extrapolating design calculations to above the SSE level. For instance, in the LaSalle analysis, structural responses are being calculated at two different levels, SSE and 3 x SSE. Piping responses are being determined by applying response calibration factors (F_R) to design values defined at the SSE level. These piping responses are extrapolated to the 3 SSE level using the same slope as found in the structural response calculation.

In general, the simplified method seems to be working well in the LaSalle analysis. It does increase the uncertainty in the result but greatly reduced the amount of effort. This method is useful where accuracy is not quite so important, design calculations and plant logic models are available and generic fragility data is applicable to most structures and components.

4.3 Validation Studies

Validation of SSMRP to date has relied primarily on peer review, comparisons with other methods, expert judgment, and sensitivity studies. In completing the Program, several additional validation tasks were accomplished and are described in what follows.

Fragility of Reinforced Concrete Structures.

Part of SSMRP determined cumulative distribution functions of the probability of failure of critical elements or systems as a function of load levels. These are called fragility functions. Reference 16 evaluates the failure analysis used for reinforced concrete structures or elements. Most critical structural elements associated with potential undesirable accident consequences in nuclear reactor facilities are made of reinforced concrete.

The evaluation emphasizes validation of the idealization and analytical description of the behavior of reinforced concrete elements, but associated analytical steps are also discussed. Low aspect ratio shear walls receive particular attention because they are critical and prevalent in nuclear facilities. The evaluation examined the nonlinear idealization and behavior of low aspect ratio shear walls; the interaction of viscous damping and ductility-based reduced spectra; energy absorption and deterioration levels; the failure mode of the main wall in the north-south direction in the Zion AFT complex, the analysis of roofs and a brief summary of methods of accounting for nonlinear behavior. Detailed findings are listed in Reference 16 but some of the main conclusions are listed below:

1. Static nonlinear (piecewise linear) analysis would generate essential information and is strongly recommended as part of seismic risk or other extreme load risk analysis. Such an analysis would greatly aid in the calculation of system ductilities, in load redistribution after nonlinearities develop, and in the identification of the second line of defense. Most approximate nonlinear analysis methods utilize static force-deflection curves.

2. Approximate inelastic analysis methods are preferable to direct response-history analyses for routine applications because the latter involve too many uncertainties. The ductility-based reduced spectrum method, which was used in the SSMRP, is one such preferable method but other approaches also show potential. These include the substitute structure method, the capacity spectrum method, and the reserve energy technique. The main feature of most approximate methods is the use of static nonlinear force-curves and an elastic spectrum with a shift in the natural frequencies.

3. The failure of the main wall between the auxiliary and turbine buildings of the Zion plant contains an embedded steel frame. The failure of the studs on the frame was assumed to be sudden, resulting in a transfer from shear behavior to flexural action of separated slender walls which would fail at a low load. However, review of the force transfer mechanism and stud behavior has led to a different stress distribution and a model with gradual stiffness loss.

4. The role of nonstructural walls, which was ignored in the fragility analysis because additional stiffness would result in reduced response, needs further

study. Their effect on the response, especially on torsional response, initially and after deterioration, may be significant. Walls also affect damping, especially after cracking.

Review of Effects of Basemat Uplift, Structural Failures and Release Category Assignments.

Three separate studies were undertaken to test certain assumptions made in the Zion Phase II analysis. One was of the effect of basemat uplift and slapdown on the failure of piping running between the auxiliary and turbine building (Ref. 9). The original work (Ref. 8) on this subject indicated that this piping would fail at a median level of 0.7g (measured at the soil surface) because of the resulting differential displacement between the two buildings and that basemat uplift and slapdown was a contributor to this displacement. Upon reanalysis, it appears that this phenomena may not lead to as high a stress in this piping as originally predicted and therefore the original estimate of piping failure probability could be reduced. Only one pipe (RHR SII) was analyzed in Reference 9 and extrapolation of results from this one analysis cannot be made to the other inter-building piping. If results could be extrapolated and pipe failures were shown to be less probable, then the point estimate core melt frequency would be reduced by about 30%.

In a second study, a visit was made to the Zion facility to re-evaluate the potential damage to systems of two types of structural failure (Ref. 17). A limited re-evaluation of the consequences of these structural failures was performed principally through a review of the structural drawings and the load paths available, a detailed walkthrough of the area, and a review of the systems models. The first structural failure concerned the service water pump enclosure roof of the crib house. In the original analysis, failure of this roof was assumed to lead to failure of all six service water pumps resulting in failure of the important system. Upon re-evaluation it appears that assumption was too pessimistic and that at least one pump would survive. One pump is sufficient to assure system success.

The second structural failure concerned failure of the common wall between the auxiliary and turbine building. Upon re-evaluation it appears the original assumption that failure of this wall would result in failure of all equipment above the 592 foot elevation was appropriate. Referring to Table 3.3, it can be inferred that discounting the failures of the two studies (Ref. 9, 17) could reduce core melt probability and risk by a factor of 2 or 3 although it is not proven that either failure can be completely discounted.

A third study was made of the dominant accident sequences found in the Phase II results to see if new source term research would effect the release quantities calculated in Phase II (Ref. 18). Results of this reanalysis showed that the release quantity for the base case would decrease by about a factor of three with the newer release category assumptions.

Comparison of Utility and SSMRP Seismic Risk Studies at Zion.

Seismic risk analyses of the Zion Nuclear Power Plant have been conducted by both Commonwealth Edison (hereafter referred to as "the Utility") and the Seismic Safety Margins Research Program (SSMRP). Although the two analyses used similar methodologies, there are differences in the results. Reference 19 compares the methodologies to identify how certain factors contribute to those differences. The factors considered were the three main parts of the methodologies: data set, systems analysis, and final assembly.

"Data Set" refers to the aggregate of seismic hazard, response, and fragility data. Both studies used expert opinion, historical data, and geological analysis to determine the seismic hazard at the Zion site. The SSMRP calculated local responses explicitly using dynamic analysis and defined fragilities in terms of local response. The Utility did not calculate local responses explicitly. It made use of existing dynamic analyses and engineering judgement to estimate local responses and used these estimates to define component fragilities in terms of the ground motion. Both studies made use of existing data and engineering judgement to estimate component fragilities. However, the SSMRP study included an extensive survey of expert opinion regarding fragilities.

The systems analysis portions of both studies used fault/event tree analysis to model the plant logic and thereby identify the component failures which lead to core melt. The SSMRP employed 7 event trees (one for each of 7 initiating events) and 7 fault trees to model hundreds of core melt sequences. In the seismic risk part of their study, the Utility modeled 3 core melt sequences in a master fault tree containing both the initiating and mitigating portions of the sequences.

For final assembly, both studies obtained component failure probabilities by convolving the component response distribution with the fragility distribution. Both studies also provided estimates of the uncertainty in their calculations. However the Utility employed Discrete Probability Distribution (DPD) arithmetic for this purpose, whereas the SSMRP used simulation.

A comparison of the median core-melt frequencies calculated in the two studies showed both values within the uncertainty bounds of either study but different by an order of magnitude. The conclusion from the comparison was that most of the difference between the two studies arises from differences in the data sets employed. Although there are contributions to the results which cannot be clearly classified as stemming from data sets or systems analyses, the differences in the failure frequencies in dominant contributors are great enough to lead to this conclusion.

In order to help distinguish the effects of the seismic hazard and the response/fragility data used in the two studies, the SSMRP seismic hazard and response/fragility data were combined with a benchmark reproduction of the Utility's results. Both the hazard and response/fragility data contribute to the difference.

Within the fragility/response data used in the two studies, there are three components which deserve mention. They are the inter-building RHR piping, the crib house pump enclosure roof, and the condensate storage tank. Each of these components was identified as a dominant contributor to core melt frequency in the SSMRP study. Of these, by far, the greatest difference in fragilities employed by the two studies is for the RHR piping (SSMRP median fragility of .2g vs. Utility median fragility of 9.3g). This piping runs between the AFT building and the reactor building. It was modeled in the SSMRP as rigidly fixed in each building at points 2.2 ft. apart. Consequently, the response in this pipe segment, due to differential motion of the buildings, was calculated to be greater in the SSMRP than in the Utility study, which, presumably, did not model the piping segments in a like manner. Although the SSMRP local capacity for the pipe segment is in better agreement with the Utility value, the high response calculated for that segment results in a markedly lower failure level in the SSMRP analysis.

In another area, the effect of crib house roof failure on service water pump failure appears to be too conservatively modeled in SSMRP. Both these effects would tend to lower SSMRP core melt estimates.

A final note should be made regarding the role of statistical error. The seismic risk analyses performed by both the Utility and the SSMRP involve samplings from distributions of many random variables. The use of 14 data sampling sets by the SSMRP and the use of 5 data sampling sets during our benchmark of the Utility results represent rather small sample sizes. It would have been desirable to use much larger sampling sets so that greater confidence could be placed in the conclusions.

Section 5: CONCLUSIONS AND LIMITATIONS

The Seismic Safety Margins Research Program was a multi-year program to develop probabilistic analysis techniques to describe the seismic behavior of nuclear power plants. These techniques were applied to the Zion nuclear power plant, a pressurized water reactor, using both detailed and simplified methods. The simplified method is now being applied to the La Salle nuclear power plant, a boiling water reactor. Three computer codes, HAZARD, SMACS and SEISIM, were developed to do the analysis. HAZARD is used to describe the seismic hazard and resulting ground motion, SMACS calculates the building and equipment responses, and SEISIM calculates the risk and importance ranking. In addition, a fragility data base was created for the Zion analysis, part of which has generic usefulness (Ref. 2, 20).

In addition to the development of the methods, computer codes and data bases, other useful results were generated from the Zion analysis. The median annual core melt frequency value (3.E-5/yr.) is in the range of those found from other analyses and the risk (9.6 man-Rem/yr.) was found to be low. Sensitivity studies showed that core melt frequency could vary by an order of magnitude depending on the assumptions made, while the 90% confidence band (see Fig. 3) on the estimate of this frequency was 3 orders of magnitude. The main contributor to seismic risk was found to be from earthquakes 2 through 4 times the design basis earthquake (see Table 3.2), giving some feeling for the amount of margin in the plant design. Risk was dominated by structural, inter-building piping, and electrical equipment failures. More recent studies have called into question some of the assumptions concerning failure of the inter-building piping, but it is felt that such piping should be reviewed in any seismic risk assessment or margin study to make sure piping failures due to relative building displacements are unlikely. It was also found that local site effects need to be considered for seismic risk assessments on shallow soil sites.

Validation studies done following completion of the Zion analysis pointed out the necessity for reviewing assumptions made concerning the role of key contributors to risk. For instance, following such a review, the crib house roof failure was found to have less of an effect on service water pump failure than originally assumed. Another validation study of interest was the comparison of SSMRP with the utility risk study of Zion. The data differences were found to cause different results but the system analysis methods gave equivalent results even though the methods are quite different. Concerning simplification, the simplified method developed in SSMRP will give great savings in response calculations but add to the uncertainty of the analysis results. It is most useful where accuracy is not so important, design calculations and plant logic models are available and generic fragility data is, for the most part, applicable.

Seismic risk assessments are now commonly being conducted by utilities at plants both in the U.S. and overseas. Over twenty have been done to date. The SSMRP stands as the most detailed of this type of risk assessment and provides a benchmark and resource for further work concerning seismic safety. SSMRP methods and codes can be used to perform seismic risk assessments, to investigate relative hardness between plants, designs and locations, to prioritize areas needing further attention, and to suggest areas for further research. Reference 21 is an example of how SSMRP methods have been used to assess risk-benefit for NRC licensing decisions.

SSMRP represents a point in time with respect to seismic risk assessment techniques and data. Continuing work relating to seismic margin is being conducted within the LLNL Seismic Margins Program, whereby a panel of outside experts is reviewing SSMRP and other risk assessments for margin implications. From this work as well as from SSMRP, research needs have been identified. These research needs which define some of the limitations inherent in the SSMRP analyses are highlighted below.

- In the seismic hazard area, new data suggests that unexpected strong high frequency components of ground motion may exist in eastern U.S. earthquakes. An increase in high frequency components may have an impact on relay chatter and breaker trip failure probabilities.

- There is a need for more fragility data particularly for electrical components such as relays. Data is also needed for other components, and such data should more adequately take into account aging effects. Fragility of structures, particularly reinforced concrete structures, needs further study.

- Operator performance and human error was minimally modeled in SSMRP. Very little is known about operator response during or after a strong earthquake. Studies relating to recoverability, operator-instrument interaction, and error rate increases are needed.

- Design and construction errors were not considered in SSMRP, and gross errors could have a significant impact on seismic risk. A review of past experience both within and outside the nuclear industry would be a start.

- SSMRP component and structure failure analysis relied on linear analysis with corrections, e.g. ductility factors, to account for non-linear behavior. Some phenomena might not be adequately accounted for by this process, such as the frequency shift observed in concrete structures that are subjected to vibratory motion. Structures have been found of key importance in SSMRP and other seismic risk assessments, yet there is great uncertainty as to how buildings respond or fail due to earthquakes. Work in this area is clearly warranted. The non-linear behavior of piping needs more study, particularly when subject to large displacements, because of its possible contribution to risk. The reaction of pipe mountings may be important in this case.

- With regard to systems analysis and modeling, further study is needed in several areas. The SSMRP uncertainty analysis relied on 14 data points and an experimental design scheme to calculate the median, mean, and confidence limits. Further calculation would be required to see whether an uncertainty analysis using 14 data points is sufficient to adequately estimate these quantities. Better modeling of seismic hazard and fragility uncertainties would also be useful. SSMRP initiating events provide a complete but not unique set. Their definition and calculation of probability of occurrence could use further study including better correlation between component failures leading to initiating events and to system failure. Also, fault trees were not constructed for all Zion systems. System failure probabilities were assumed for the reactor protection (RPS), containment spray (CSIS, CSRS) and containment fan cooler (CFCS) systems. Because several of these systems were found important, fault tree construction of these may impact the results.

Research concerning seismic risk continues within NRC and the industry. Also, validation studies are underway with an attempt to experimentally validate seismic risk methods where possible. As this research and validation continues, improvements to the methods will result.

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