NUREG/CR-2015, Vol. 1
UCRL-53021, Vol. 1

Seismic Safety Margins Research Program
Phase I Final Report—Overview

P. D. Smith, R. G. Dong, D. L. Bernreuter, M. P. Bohn, T. Y. Chuang, G. E. Cummings, J. J. Johnson, R. W. Mensing, J. E. Wells

Prepared for
U.S. Nuclear Regulatory Commission

LAWRENCE
LIVERMORE
LABORATORY

Available from

GPO Sales Program
Division of Technical Information and Document Control
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Printed copy price: $3.75

and

National Technical Information Service
Springfield, Virginia 22161

NUREG/CR-2015, Vol. 1
UCRL-53021, Vol. 1
RD, RM

Seismic Safety Margins Research Program

Phase I Final Report—Overview

Manuscript Completed: March 1981
Date Published: April 1981

Prepared by
P. D. Smith, R. G. Dong, D. L. Bernreuter, M. P. Bohn, T. Y. Chuang,
G. E. Cummings, J. J. Johnson,* R. W. Mensing, J. E. Wells

Lawrence Livermore Laboratory
7000 East Avenue
Livermore, CA 94550

Prepared for
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN No. A0126, A0130, A0138, A0139, A0142

*Structural Mechanics Associates

ABSTRACT

The Seismic Safety Margins Research Program (SSMRP) is a multiyear, multiphase program whose overall objective is to develop improved methods for seismic safety assessments of nuclear power plants, using a probabilistic computational procedure. The program is being carried out at the Lawrence Livermore National Laboratory and is sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. Phase I of the SSMRP was successfully completed in January 1981: A probabilistic computational procedure for the seismic risk assessment of nuclear power plants has been developed and demonstrated. The methodology is implemented by three computer programs: HAZARD, which assesses the seismic hazard at a given site, SMACS, which computes in-structure and subsystem seismic responses, and SEISIM, which calculates system failure probabilities and radioactive release probabilities, given (1) the response results of SMACS, (2) a set of event trees, (3) a family of fault trees, (4) a set of structural and component fragility descriptions, and (5) a curve describing the local seismic hazard. The practicality of this methodology was demonstrated by computing preliminary release probabilities for Unit 1 of the Zion Nuclear Power Plant north of Chicago, Illinois. Studies have begun aimed at quantifying the sources of uncertainty in these computations. Numerous side studies were undertaken to examine modeling alternatives, sources of error, and available analysis techniques. Extensive sets of data were amassed and evaluated as part of projects to establish seismic input parameters and to produce the fragility curves.

CONTENTS

LIST OF FIGURES

LIST OF TABLES

FOREWORD

The Seismic Safety Margins Research Program (SSMRP) is a multiyear, multiphase program aimed at developing improved methods for seismic safety assessments of nuclear power plants. The program is being carried out at the Lawrence Livermore National Laboratory (LLNL) and is sponsored by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research. The SSMRP is directed by P. D. Smith, program manager, and R. G. Dong, deputy program manager. Without the encouragement of their counterparts at the NRC, our successful work would have been impossible. We therefore extend special thanks to J. E. Richardson, NRC program manager and chief of the Mechanical Engineering Research Branch, and to C. W. Burger of the Structural Engineering Research Branch, NRC deputy program manager.

Planning for the SSMRP began in early 1978, and major technical work began in July of that year. This report is an overview of Phase I of the program, which was successfully completed in January 1981. Eight additional topical reports will document the detailed results of the projects that constituted Phase I. These projects, together with the names of the project managers, are as follows: Project I, Plant/Site Selection and Data Collection, T. Y. Chuang (LLNL) and G. Bagchi (NRC); Project II, Seismic Input, D. L. Bernreuter (LLNL) and R. J. Brazee (NRC, now with Teledyn-Geotech); Project III, Soil-Structure Interaction, J. J. Johnson (LLNL, now affiliated with Structural Mechanics Associates) and J. F. Costello (NRC); Project IV, Major Structure Response, J. J. Johnson and C. W. Burger; Project V, Subsystem Response, T. Y. Chuang and J. J. Burns (NRC); Project VI, Fragilities, M. P. Bohn (LLNL) and J. J. Burns; Project VII, Systems Analysis, G. E. Cummings (LLNL), J. E. Wells (LLNL), and J. J. Burns; and Project VIII, SMACS and BE-EM, J. J. Johnson and C. W. Burger. In a ninth topical report, the role of expert opinion in the SSMRP will be discussed by R. W. Mensing of LLNL, our in-house consultant on probabilistic methods and statistics.

The SSMRP staff appreciates the support of LLNL's Nuclear Systems Safety Program Office, Mechanical Engineering Department, Nuclear Test Engineering Division, Engineering Mechanics Section, and Methods Development Group, and we thank the Commonwealth Edison Company and Sargent & Lundy Engineers for being most cooperative. Over the course of our work, the dependable efforts of our secretaries, Sue Aubuchon and Virge Jaramillo, were indispensable and are deeply appreciated. Finally, we extend thanks to Douglas Vaughan of EG&G, Inc., the coordinating editor for this series of final reports; and to Jane Staehle and the other members of LLNL's Technical Information Department, who turned manuscript, computer output, and rough drawings into a final report.

EXECUTIVE SUMMARY

The Seismic Safety Margins Research Program (SSMRP) is a multiyear, multiphase program whose overall objective is to develop improved methods for seismic safety assessments of nuclear power plants, using a probabilistic computational procedure. The program is being carried out at the Lawrence Livermore National Laboratory (LLNL) and is sponsored by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research. Phase I of the SSMRP was successfully completed in January 1981: We have developed and demonstrated a probabilistic computational procedure for the seismic risk assessment of nuclear power plants. (For the SSMRP, we adopted a nonstandard definition of risk, namely, the probability of radioactive release.) The methodology includes the two essential features of a fully adequate safety evaluation procedure—an explicitly probabilistic approach and an emphasis on systems analysis. The methodology is implemented by three computer programs: HAZARD, which assesses the seismic hazard at a given site, SMACS, which computes in-structure and subsystem seismic responses, and SEISIM, which calculates structural, component, and system failure probabilities and radioactive release probabilities. The practicality of this methodology was demonstrated by computing preliminary release probabilities for Unit 1 of the Zion Nuclear Power Plant north of Chicago, Illinois.

This probabilistic methodology will ultimately be used as a tool for assessing the effect of seismic events on nuclear power plant safety. It can lend stability to the seismic safety licensing process by serving as a means for producing probabilistic estimates of seismic risk and by pointing to the most important areas of possible improvement aimed at decreasing this risk. To function in either of these roles, a methodology must explicitly acknowledge and evaluate sources of uncertainty. The SSMRP methodological chain thus considers uncertainties in each of its links, couples them, and propagates those uncertainties throughout the calculation.

The Phase I effort was organized into eight projects, the results and technical products of which are outlined in the following paragraphs.

Plant/Site Selection and Data Collection. Unit 1 of the Zion Nuclear Power Plant was chosen as an appropriate "typical" plant. An independent study, based on a comparison with other operating power plants in terms of important design features, concurred in our choice.

Seismic Input. We developed the tools and models necessary to describe probabilistically the seismic hazard at the Zion site and to generate appropriate acceleration time histories. The models include (1) a delineation of zones of roughly uniform seismic activity in the central United States, (2) an occurrence model that describes the seismicity for each zone, and (3) a ground motion model that accounts for earthquake source effects and regional attenuation of ground motion. The computer program HAZARD was developed to produce the necessary seismic hazard curve, based on these models. The hazard curve is divided into six acceleration ranges, and 30 time histories were generated for each range.

Soil-Structure Interaction. Analysis of the coupled soil-structure system by the substructure approach is the first step in the SMACS calculational procedure. We provided as input the necessary characterizations of the soil, foundations, and structures at the Zion site. In a separate study, foundation embedment, accounted for in our calculations, was found to have a significant effect on computed structure response. The angle of incidence of seismic waves, on the other hand, was found to affect only torsional response. In a comparison of two computer programs (FLUSH and CLASSI) that implement alternative approaches to the analysis of soil-structure interaction, we found varying agreement.

Major Structure Response. Major structure response was obtained as part of the computation of soil-structure interaction. Input included detailed finite element models of the containment building (the cylindrical containment shell and the internal structures were modeled separately) and the auxiliary-fuel-turbine (AFT) complex. To assess the uncertainty due to modeling assumptions, we analyzed four mathematical models constructed to represent the AFT complex. Disagreement among the results was marked in some cases.

Subsystem Response. The third segment of SMACS computed the responses of piping subsystems, given the structure response. We developed mathematical models of 13 piping systems as input and produced the software to perform the calculations. The software uses a pseudostatic-mode method with multisupport time-history input. Sensitivity studies have begun to evaluate the relative contributions of uncertainties in seismic input, soil-structure interaction, structure response calculations, and subsystem response calculations to the uncertainty in subsystem response.

SMACS and BE-EM. We developed the computer code SMACS to tie together the soil-structure interaction, structure response, and subsystem response calculations. Variations in input parameters (including ensembles of acceleration time histories for each acceleration level) reflected uncertainties about the Zion plant and site. Calculational results include peak and spectral accelerations at many points in the structures and subsystems, and peak moments in the piping subsystems. The input uncertainties are manifest in the range of responses computed for any node at each acceleration level. We also introduced the concept of comparing a best-estimate (BE) seismic analysis method, exemplified by the SSMRP methodology, with an evaluation method (EM), such as that embodied in the NRC's *Standard Review Plan.*

Fragilities. Fragility curves—normal or lognormal distributions describing the probability of failure as a function of a critical local response parameter—were necessary for all components and structures whose failure is accounted for in the SEISIM fault trees. Curves were thus developed for 37 generic categories of electrical and mechanical equipment and for 5 Zion structures. The curves were based on both available data and on carefully analyzed expert opinion.

Systems Analysis. To describe the Zion plant systematically, we developed (1) seven event trees that describe the possible event sequences that follow an earthquake and (2) fault trees that describe the possible failure modes for certain systems identified in the event trees as critical to safety. The computer program SEISIM accepts as input these event and fault trees, the responses computed by SMACS, the set of fragility curves (which, together with the calculated responses, establish the probabilities of the various fault tree failure modes), and a seismic hazard curve for the Zion site. SEISIM output includes structural, component, and system failure probabilities, and probabilities of radioactive release. Our first results are tentative, but reasonable.

SECTION 1: INTRODUCTION

1.1 Background: The Risk of Earthquakes

Earthquakes are not merely a local California problem. Measurable temblors have occurred in every state, and as shown in Fig. 1, even destructive earthquakes have been widely distributed. The 1811–12 series of earthquakes near New Madrid, Missouri, is believed by some to have included the most severe earthquake ever to occur in the United States. This series, which culminated in the event of February 7, 1812, destroyed the town of New Madrid, changed the configurations of major rivers, and was felt as far away as Boston and Washington, D.C.[2] This last historical observation is especially important and is consistent with the observation that even small seismic events in the eastern United States (for example, the July 28, 1980, earthquake near Lexington, Kentucky) are typically felt over wide areas. A potential thus exists for widespread damage during a major earthquake in the eastern United States. This potential, even when weighed against the more widely appreciated fact that earthquakes are less common in the East than in the West, suggests that earthquakes are a hazard that must be considered in every state.

Consideration of this pervasive earthquake hazard is especially important in the design of nuclear power plants. For conventional structures, we might be quite willing to accept some structural damage from an earthquake, as long as it does not cause loss of life. In a nuclear power plant, on the other hand, an earthquake could cause virtually no damage to the structures, yet could damage critical safety systems, thereby causing the release of radioactivity. And even conventional designs have not yet been shown to be completely adequate during earthquakes. Damage during both the 1971 San Fernando earthquake and the 1979 Imperial Valley earthquake demonstrates that even with considerable experience we still have much to learn about earthquake-resistant design.[3,4] The design requirements for nuclear facilities are more stringent than those for conventional structures, and we are

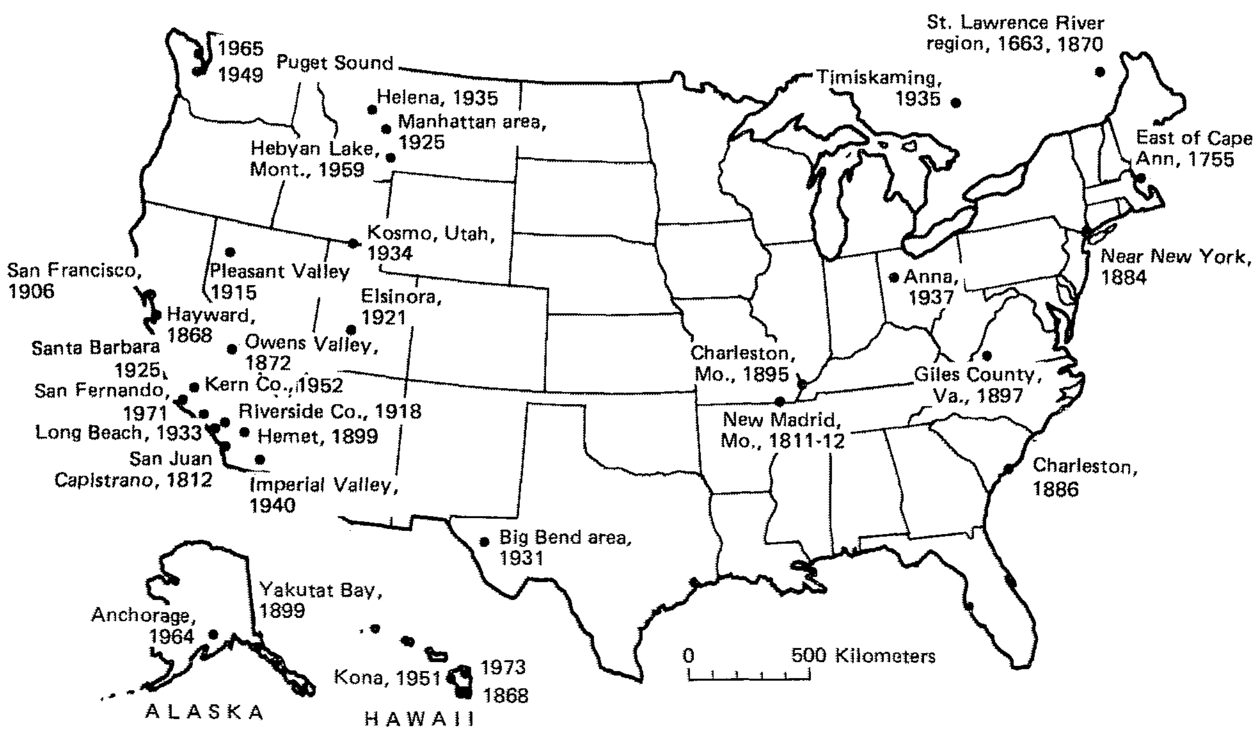


Fig. 1. Locations of past destructive earthquakes in the United States (from Ref. 1).

quite certain that nuclear structures are more reliable than conventional structures. However, we do not know that the increased reliability adequately reduces risk.

An especially important aspect of the earthquake hazard is the potential of seismic excitation for compromising redundancy. If an earthquake causes a diesel generator to fail, it is quite likely that a second, redundant diesel generator—which usually serves to back up the first—will also fail. But perhaps the most important reason for concern about earthquakes is the increasing reliability of nuclear plants in the absence of earthquakes: Nonseismic threats have diminished to the point that some studies now estimate earthquakes to be a dominant contributor to risk. The *Reactor Safety Study*[5] of 1975—the so-called Rasmussen report—concluded that earthquakes were not a major contributor to the risk of radioactive release from nuclear power plants, but a later Canadian study of seismic risk and nuclear power plants[6] showed that the risk may have been underestimated. A further study by Hsieh and Okrent[7] indicated that possible design errors, previously overlooked, could contribute to seismic risk.

1.2 Risk Assessment and the Role of the SSMRP

1.2.1 Features of an Adequate Risk Assessment Methodology

Currently, designing a nuclear power plant to resist seismic stresses reflects a deterministic concept of design. First, a design earthquake—the safe shutdown earthquake (SSE)—is selected as the "maximum credible earthquake" at the site. Then, the entire plant (that is, all critical structural, mechanical, and electrical systems) is designed according to procedures whose goal is absolute assurance that no system will fail in the event of an SSE.

This current design procedure is a natural extension of that used for conventional structures; however, such an extension to nuclear plants is clouded by one central difficulty.[8] Whereas the empirical basis for conventional design codes and procedures is considerable, our experience with nuclear power plants is far more limited. The absence of seismically induced failures in nuclear design cannot be interpreted as success, considering the rarity of the extreme events power plants must survive and the paucity of our experience of plant behavior during such events. Confidence in nuclear design depends instead on accepting the probabilistic nature of seismic threats and on evaluating the consequent risk probabilistically.

The two design steps described above, while explicitly deterministic, implicitly acknowledge their probabilistic base. The goal of absolute safety is, of course, impossible to achieve. We must therefore revise our design objective to something like "reduce the possibility of failure to some acceptable level." Furthermore, any economically reasonable choice of an SSE does not, and need not, establish an absolute maximum for seismic excitation in terms of peak ground acceleration at the power plant site. Indeed, any absolute pronouncement with regard to peak acceleration is suspect. The Imperial Valley earthquake of October 15, 1979, produced one record with a vertical acceleration of 1.74g, which is more than twice the design value for any plant and about 0.5g larger than any peak acceleration previously measured in the United States. Furthermore, plants are not designed for the most severe possible earthquake with a given peak (SSE) acceleration. Typical design basis response spectra are roughly equivalent to the mean plus one standard deviation (MSD) of spectra from a normalized set of recorded earthquakes. An MSD response value is less than the maximum response from this set, and this maximum is less than the maximum from the normalized set of all recorded earthquakes.

Even the empirical basis for conventional designs can do no more than raise confidence in conventionally engineered structures, for it cannot eliminate risk. The failures caused by the aforementioned San Fernando and Imperial Valley earthquakes underscore this assertion. As with any set of empirical data, failures are inherently probabilistic.

These reflections do not necessarily point to inadequacies in current designs, but rather to the need for an explicitly probabilistic procedure to evaluate them. Admitting the probabilistic nature of the earthquake hazard, of soil properties, material properties, and mathematical modeling methods should not lead to a lack of confidence in nuclear design. Rather, confidence must be redefined as assurance of low risk rather than no risk.

The first important feature of a fully adequate risk assessment methodology must therefore be its probabilistic basis. Equally important, however, it must take a systems approach to the question of risk. It must consider the interaction among the major elements in the problem, beginning with a definition of the seismic hazard and ending with a definition of the risks to the public. Figure 2 illustrates some of the elements of a representative systems analysis.

The first of these elements, the seismic hazard, depends primarily on seismological issues that are virtually independent of the plant, except for its location. The description of the seismic hazard should encompass all possible event scenarios, including the timing and intensities of aftershocks, surface faulting, and vibratory motion. The ground motions are propagated from the earthquake source through the earth and foundation to the structures, whose response is important in three ways. First, they may collapse on critical equipment, thus either directly initiating an accident that could lead to radioactive release or preventing safety systems from mitigating such an accident. Second, they act as transmitters of motion to safety-related components. Finally, the containment shell of the containment building acts as a secondary pressure vessel and the final barrier to release.

Accident sequences induced by the seismic environment thus reflect the behavior of a complex system comprising soil, foundation, and the entire nuclear power plant—its various components, subsystems, and structures. Other aspects of the system that must ultimately be considered include the reliability of offsite power (which may also be influenced by the seismic event), wind conditions, the temperature inside and outside the plant, and groundwater levels. The result of interactions among all these system variables may be one of a set of radioactive release patterns from the plant. Meteorological, demographic, and behavioral factors then determine the consequences to the public of any particular release.

Failure to take such a systems approach will inevitably yield an incomplete picture of plant safety. Without considering the entire nuclear power plant system, we cannot judge the importance of seismic adequacy in its constituent parts: Is it essential, merely reassuring, or irrelevant that a certain piping system is almost certain to survive an earthquake? How is the probability of the failure of a single pump related to the probability of failure for its parent system or for the plant as a whole? Moreover, a view that fails to take in the whole system can lead to a misleading picture, not merely an incomplete one. Design modifications aimed solely at increasing structural resistance to seismic loads might, for example, compromise the capacity of attached safety systems to resist seismic loads. Most important, only a systems analysis can go beyond an assessment of the probability of "failure," however defined, to an evaluation of public risk.

Only recently have these two essential features—an acknowledgment of the probabilistic nature of risk and the use of a systems approach—been combined in the analysis of a specific nuclear power plant (Diablo Cañyon).[10] This recent study, with its major emphasis on determining the public risk from a seismic event, indicates the potential of a seismic risk assessment methodology.

1.2.2 Toward an Improved Methodology: Goals of the SSMRP

The objectives of the SSMRP are (1) to estimate the degree of conservatism of the NRC's present *Standard Review Plan* seismic safety requirements[11] and (2) to develop improved requirements and methods for safety assessment. With these broad objectives in mind, we organized the program into three phases. In the initial phase, successfully completed in January 1981, we have developed and demonstrated a probabilistic computational procedure that, we believe, estimates the behavior of a nuclear power plant subjected to an earthquake more realistically than previous methodologies. The new procedure integrates state-of-the-art seismic analysis methods with a systems representation of critical plant components whose failure could lead to radioactive release. Ultimately, the calculated probability of release and its associated uncertainties will be the yardsticks for making judgments about current seismic design methods. Uncertainties have been propagated throughout our calculations, and we have begun sensitivity studies aimed at identifying the principal contributors to uncertainty, thus providing optimal direction for future research.

Compared with the current design procedure, sketched briefly above, the SSMRP computational procedure has several important features:

- It uses probability of release, rather than safety factors, as a figure of merit.

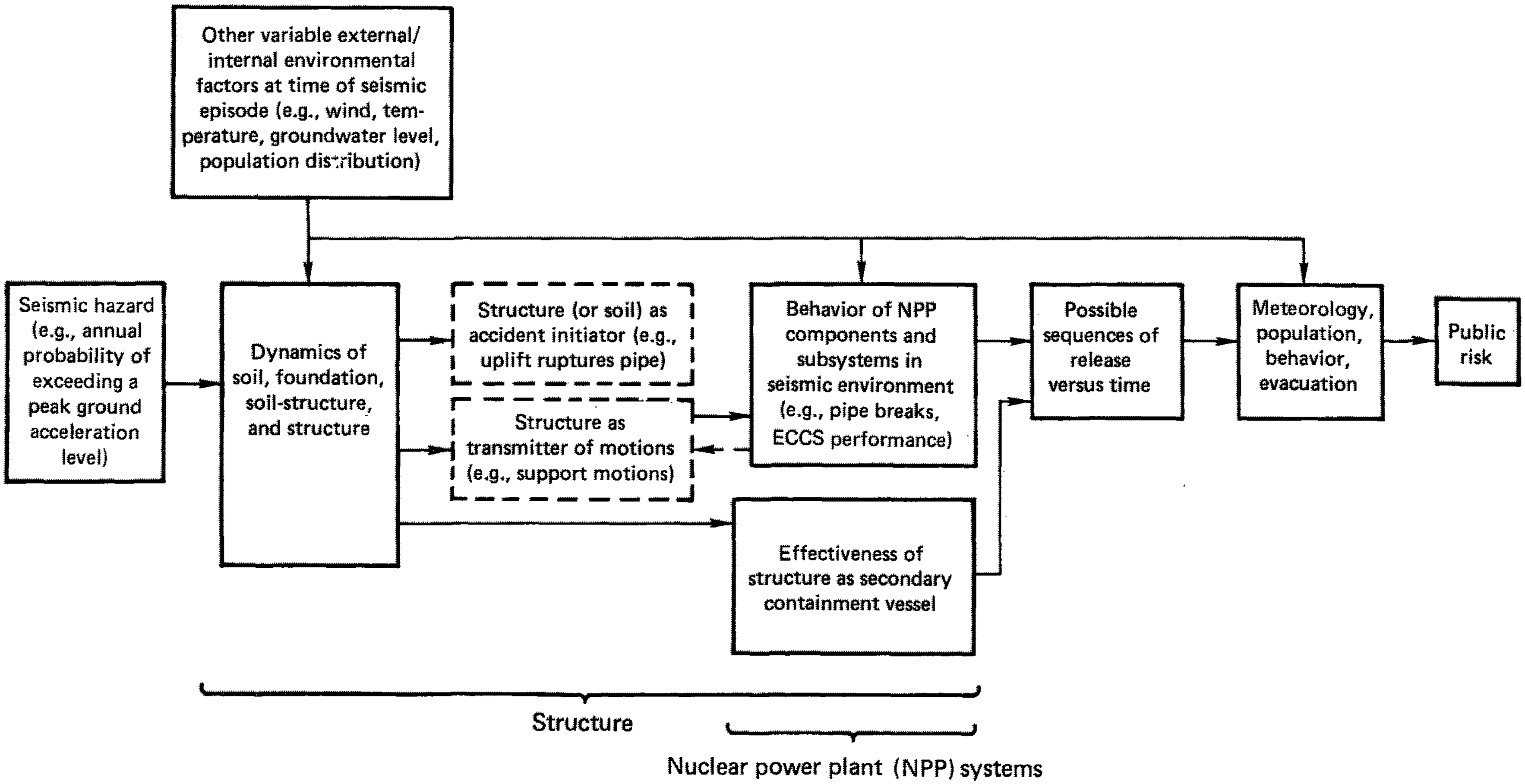


Fig. 2. Components of a generalized seismic safety analysis (adapted from Ref. 9).

● The SSMRP methodology couples the elements of the seismic safety methodology rather than treating each element separately. The current design procedure introduces conservatism at each step in the methodology to account for uncertainties. Many in the nuclear industry believe that this process has led to overly conservative designs.

● It can be applied to the safety evaluation of operating plants; design methodologies are not intended for this purpose.

● It specifically accounts for the "common-cause" threat of seismic events, that is, the simultaneity of the threat to all power plant components.

● It emphasizes state-of-the-art methods and experimental data.

● By taking a systems approach, the methodology can be used to evaluate the overall safety impact of proposed changes in design procedures or in individual safety systems.

● It can determine the overall effect of individual sources of uncertainty and can therefore be used to focus on the most important issues.

During the second phase of the SSMRP, we will perform further sensitivity studies, refine the procedure as necessary, validate the methodology using results from other, proven procedures and using experimental data wherever possible, and help the NRC implement for licensing use the tools developed by the SSMRP. The third phase will produce improved seismic safety requirements and will recommend changes in the *Standard Review Plan*.

1.3 An Overview of the SSMRP Methodology

The steps of the SSMRP methodology can be broadly outlined as follows:[12]

● Definition of the earthquake hazard.

● Calculation of plant response, which entails calculation of soil-structure interaction, the responses of major structures, and the responses of subsystems (for example, piping).

● Evaluation of failure, which requires definitions of the fragilities of structures, components, and systems, and a description of the operation and interaction of the systems within the plant.

These three segments of the analysis, each explicitly acknowledging its probabilistic basis, culminate in an estimate of the probability of radioactive release, which reflects the failures of primary importance. Since the ***Reactor Safety Study***[5] showed that the most significant release of radioactivity accompanies meltdown of the core, we have assumed that the objective of seismic analysis and design is to protect the public by preventing a core melt.

Key elements of the SSMRP methodology are illustrated in Fig. 3. The computation is divided into two parts. The first is performed by the computer program SMACS (Seismic Methodology Analysis Chain with Statistics), which calculates the seismic response of structures, systems, and components. The second part of the computation is carried out using the program SEISIM (Seismic Evaluation of Important Safety Improvement Measures), which calculates the probabilities of structural, component, and system failure and of radioactive release. These two computer codes are the major computational elements of the SSMRP methodology; the other elements illustrated in Fig. 3 are, of course, no less significant in their roles in the SSMRP.

The program SMACS was developed to link the seismic input with the soil-structure interaction, major structure response, and subsystem response calculations. The seismic input is defined by ensembles of acceleration time histories for three orthogonal directions (two horizontal and one vertical) on the surface of the soil. Soil-structure interaction and detailed structural response are determined simultaneously in a calculation that relies on the substructure approach to soil-structure interaction. This approach analyzes the coupled soil-structure system in a series of steps (determination of the foundation input motion, calculation of the foundation impedances, and analysis of the coupled system).[13] The result of this segment of the program is a detailed structural response in the form of time histories of accelerations, displacements, and forces. Using these results, SMACS then calculates the time-history responses of piping subsystems, using an advanced multisupport approach.[14]

Throughout these computations, uncertainties are accounted for probabilistically. The largest source of variability in seismic input is acknowledged by using ensembles of time histories; in the soil-structure interaction phase, the shear modulus

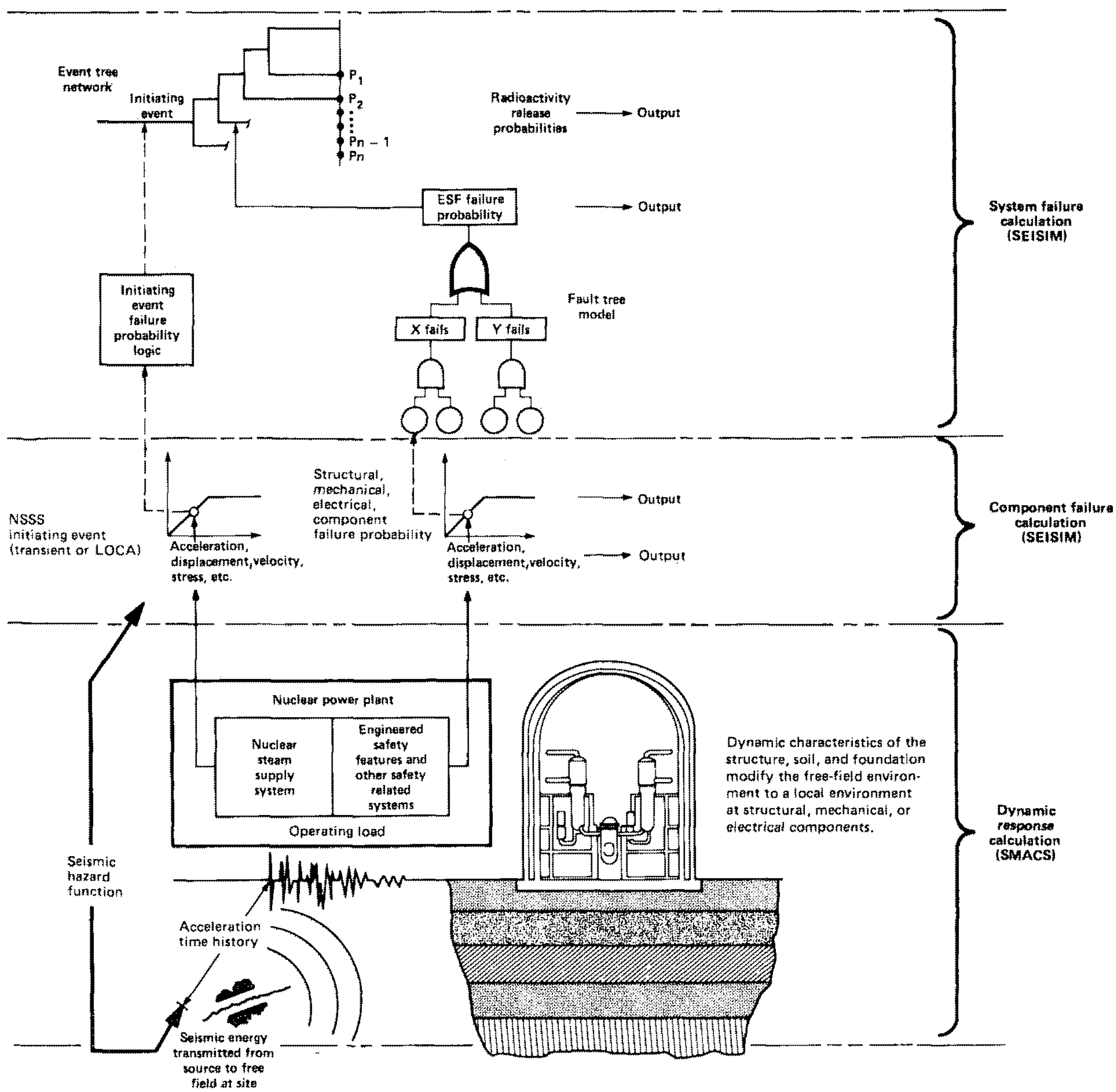


Fig. 3. A schematic representation of the computational approach taken in the SSMRP. Computed responses (of structures, systems, and components), together with fragility descriptions, establish the probabilities of initiating events and of structural and component failures. Fault tree models then allow system failure probabilities to be calculated, which in turn establish the probabilities of the accident sequences modeled in event trees.

and damping in the soil are varied; in the computation of structure and subsystem responses, variations in the eigensystems and modal damping properties account for the uncertainties.

The program SEISIM accepts as input (1) the responses calculated by SMACS for a range of earthquake levels, plus the probability of occurrence for each level; (2) fragility descriptions of structures and components; (3) event trees that characterize the possible accident sequences, given an initiating event; and (4) fault trees that describe the failure modes of systems designed to mitigate the consequences of an accident sequence.

Fragility curves express failure probability as a function of the local parameter that is critical for the given structure or component (spectral acceleration for many components, resultant moments for piping, etc.), where failure is defined as the inability of the item to serve its intended safety-related function. A typical curve is shown in Fig. 4. Fragility curves are based on existing test results, analytical information, or subjective opinion. For the Zion plant that was the subject of our Phase I analysis, such curves were produced for the failure calculations of 2300 components.

Event trees identify and model the important accident sequences. A simple example is shown in Fig. 5, where each node represents a point at which a system might operate as designed or fail. The input to SEISIM included event trees representing seven initiating events at the Zion plant (four sizes of loss-of-coolant accidents; two types of reactor transients, one in which the power conversion system is operable after transient initiation, a second in which it is not; and rupture of the reactor pressure vessel). The outcome of each initiating event then depends on whether mitigating safety features operate or fail. In the simple case of Fig. 5, for example, the consequences of the initiating event depend on the operation or failure of three systems, A, B, and C.

To evaluate the relative probabilities of the several possible accident sequences represented in an event tree, we must define system failure for the various systems involved; that is, we must describe

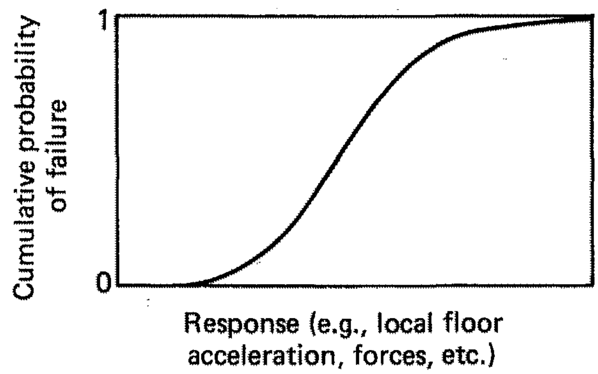


Fig. 4. A typical fragility curve, plotted as a cumulative probability distribution.

the various ways by which the systems can fail. It is the role of fault trees to provide these descriptions. The simple fault tree shown in Fig. 6 shows how the probability of failure of the hypothetical safety system A depends logically on the failure probabilities of its components (which depend in turn on fragility descriptions and computed responses). Over 3000 basic events (those at the lowest level in a fault tree) were considered during the construction of fault trees in Phase I of the SSMRP.

The major outputs from SEISIM, then, are failure and release probabilities. The responses

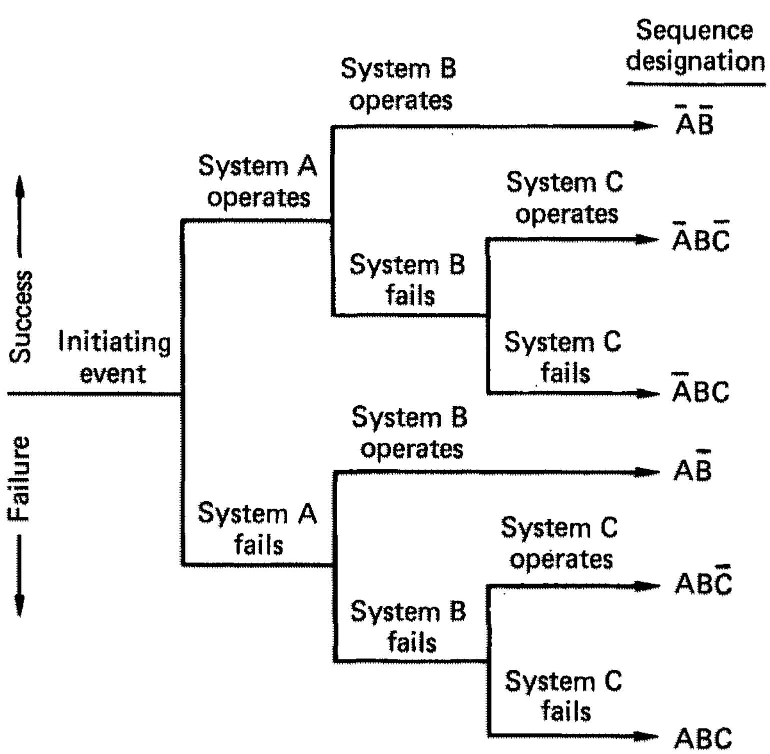


Fig. 5. A simple event tree, showing six potential sequences of events. The relative probabilities of the sequences are determined by the conditional probabilities of success or failure at each fork in the tree. In this example, which assumes that systems A, B, and C are independent, the success or failure of system C is irrelevant if system B operates.

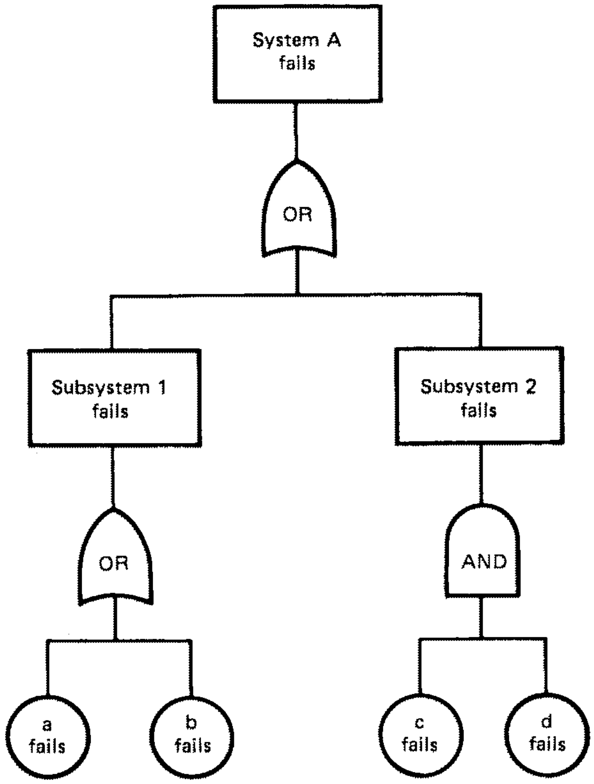


Fig. 6. A simple fault tree, showing the possible mechanisms by which system A might fail. If component a, or component b, or both component c and component d fail, the system fails.

calculated by SMACS for a specified level of seismic input are compared with the appropriate fragility curves to calculate the probabilities of structural and component failures. Logical descriptions of initiating events, event trees, and fault trees, each incorporating these failure probabilities, are then used to calculate the probability of each initiating event, the probabilities of system failures, and the probability of release. These calculated probabilities are conditioned on the specified level of ground motion, and they must finally be convolved with the seismic hazard curve to obtain a net value for the probability of release. In this calculational process, the uncertainties generated by the seismic response analyses and those associated with the fragility descriptions are considered, and their effects are propagated throughout the calculation.

1.4 Phase I: The First Step

The primary aim of Phase I was to develop the SSMRP risk assessment methodology and to demonstrate its workability. Developing the methodology meant

- Designing a probabilistic calculational procedure that is technically sound, that addresses the licensing issues facing the NRC, and that is reasonably economical.
- Creating the necessary computational software.
- Developing analytical models for seismic input and for soil, structures, and subsystems.
- Developing system models (event trees and fault trees) for the Zion plant.
- Establishing fragility curves for critical structures and components.

Demonstrating the workability of the methodology, of course, entailed running the program with our models and inputs, getting tentative but plausible results, and doing it at reasonable cost. The second section of this report provides brief technical descriptions of these accomplishments.

Beyond this primary goal, we began a series of sensitivity studies with two main purposes in mind. First, it is important to identify the principal contributors to the uncertainties in our final results: How does our best estimate of the probability of release change when we change the value of the soil shear modulus, the damping ratios, the frequencies of a subsystem, and so forth? Second, to improve our methodology (and to simplify it where possible), we must assess the effects of changes in our modeling methods and in our analytical approaches to complex problems such as soil-structure interaction. Again, the following pages will illustrate the progress we have made.

1.4.1 Features of the SSMRP Approach

Two important features of our approach to probabilistic risk assessment have not yet been discussed: the relationship of "side studies" to the on-line effort, and the role of peer review and expert opinion. Our on-line computations are those that point directly and logically toward our final result, namely, a value for the probability of release. Side studies, which were often sensitivity studies that focused on discrete portions of the computational procedure, furnished guidance and insight as we developed our calculational methods. The importance of the side studies, in addition to the assistance they provided, was their suitability as independent research topics with potential licensing applications. Some of the most important products of Phase I came from such side studies.

Expert opinion played an important part in several of the SSMRP projects, for example, in establishing the fragilities of electrical and mechanical components and in estimating the earthquake hazard in the region of the Zion plant. More will be said in the next section about some of the specific applications of subjective opinion. What should be emphasized here is the broad, general role of peer review. In broadening the base of support for our efforts in the SSMRP and in critically reviewing our ongoing work, the contributions of seven groups have been essential:

- Senior Research Review Group. Comprising NRC staff members and members from the research community, this peer review group monitored the progress of all facets of the SSMRP.
- Advisory Committee on Reactor Safeguards (Subcommittee on Extreme External Phenomena). The subcommittee periodically assessed the SSMRP as part of its mandate to review the research efforts of the NRC.
- Steering Committee on Subjective Inputs. The guidance provided by this committee helped define the proper role for subjective opinion in the SSMRP.
- Eastern United States Seismicity Panel. This panel helped define regions of roughly uniform earthquake hazard in the eastern United States and estimated the hazard for each region.
- Eastern United States Ground Motion Model Panel. Describing the regional attenuation of earthquake ground motion in the eastern United States is largely a matter of conjecture. This panel reviewed our subjective model.
- Seismic Input Peer Review Panel. This panel reviewed much of our work as we sought to specify the seismic hazard at the Zion site.
- Fragilities Panel. Fragility descriptions were based in part on solicited expert opinion. This panel was instrumental in shaping our opinion survey, and it provided guidance in the development of the fragility curves.

The members of these seven important panels are listed in Table 1. In addition, we sought the comments of numerous consultants from the nuclear industry, who reviewed the program from the industry's vantage point, primarily in two large technical coordination meetings. Table 2 lists those who attended these meetings.

1.4.2 Organization: Implementing the Computational Procedure

The SSMRP comprises eight projects, each directed toward a segment of the analysis strategy, or computational procedure, outlined above. The role of each can be seen in Fig. 7, which recasts the methodology of Fig. 3 in a different form. Project I was aimed at selecting the site and plant for detailed study and for assembling pertinent data. Projects II through VI provided input for the computational programs SMACS and SEISIM, which were developed and run as parts of Projects VIII and VII, respectively.

Project I: Plant/Site Selection and Data Collection. The objective of Project I was to select a nuclear power plant and site for detailed study and to collect design and construction data for use in the other projects. An independent assessment confirmed our choice of the Zion plant as a "typical" nuclear power plant.

Project II: Seismic Input. As Fig. 7 illustrates, Project II was responsible for input to both SMACS and SEISIM: time histories for the calculation of corresponding structure and component responses, and a hazard curve for the computation of unconditional risk estimates from the conditional release probabilities. The computer program HAZARD was developed to carry out our seismic hazard analysis.

Project III: Soil-Structure Interaction. The seismic response of a massive nuclear power plant structure depends strongly on the characteristics of the supporting soil. The first computational step in the SSMRP methodology is to account for the effects of this coupled soil-structure system. As part of Project III, we developed the software necessary for the computer code SMACS to compute basemat motions and in-structure response from the seismic input specified by Project II, and we supplied the SMACS input parameters that characterized the behavior of the soil, foundation, and structures at the Zion site. To carry out the necessary computations, we selected an analytical method—the substructure approach—that not only calculated the foundation impedances and scattering matrices, but also computed in-structure responses. Before this method was chosen and implemented, however, it was necessary to review and evaluate current analysis methodologies and to investigate appropriate computer software. These studies thus

Table 1. Members of seven important review panels.

Senior Research Review Group consultants

- S. H. Bush, Battelle Pacific Northwest Laboratories
- C. A. Cornell, Massachusetts Institute of Technology
- R. M. Hamilton, U.S. Geological Survey (former member)
- W. W. Hays, U.S. Geological Survey
- N. M. Newmark, University of Illinois (deceased January 25, 1981)

Advisory Committee on Reactor Safeguards (Subcommittee on Extreme External Phenomena)

- D. Okrent, UCLA, chairman
- M. Bender, self-employed
- H. Etherington, retired
- J. C. Mark, retired
- C. P. Seiss, University of Illinois
- P. G. Shewmon, The Ohio State University
- D. A. Ward, E. I. du Pont de Nemours and Co.
- R. Savio, NRC/ACRC
- J. C. Maxwell, University of Texas, consultant
- S. C. Saunders, Washington State University, consultant
- G. A. Thompson, Stanford University, consultant
- M. D. Trifunac, University of Southern California, consultant
- M. P. White, University of Massachusetts, consultant
- Z. Zudans, Franklin Institute Research Laboratory, consultant

Steering Committee on Subjective Inputs

- R. W. Mensing, LLNL, chairman
- D. H. Chung, LLNL, secretary
- L. R. Abramson, NRC
- R. J. Brazee, Teledyn-Geotech (former member)
- J. J. Burns, NRC
- P. D. Smith, LLNL
- W. E. Vesely, NRC

Eastern United States Seismicity Panel

- G. A. Bollinger, Virginia Polytechnic Institute and State University
- E. S. Chiburis, Weston Observatory
- M. A. Chinnery, Massachusetts Institute of Technology
- R. B. Herrmann, St. Louis University
- R. J. Holt, Weston Geophysical Research, Inc.
- O. W. Nuttli, St. Louis University
- P. W. Pomeroy, Columbia University
- M. L. Sbar, University of Arizona
- R. L. Street, University of Kentucky
- M. N. Toksoz, Massachusetts Institute of Technology

Seismic Input Peer Review Panel

- A. H.-S. Ang, University of Illinois
- O. W. Nuttli, St. Louis University
- L. R. Sykes, Columbia University
- D. Veneziano, Massachusetts Institute of Technology

Eastern United States Ground Motion Model Panel

- N. C. Donovan, Dames and Moore, Inc.
- R. K. McGuire, Dames and Moore, Inc.
- O. W. Nuttli, St. Louis University
- M. D. Trifunac, University of Southern California

Table 1. (Continued.)

Fragilities Panel

S. H. Bush, Battelle Pacific Northwest Laboratories
R. P. Kennedy, Structural Mechanics Associates
G. D. Shipway, Wyle Laboratories
J. D. Stevenson, Structural Mechanics Associates
J. M. Thomas, Failure Analysis Associates
P. P. Zemanick, Westinghouse Corporation

Table 2. Attendees at two technical coordination meetings. The affiliations are those given at the time of the meetings.

Meeting of December 11–12, 1978

A. H.-S. Ang, Univ. of Illinois	M. K. Kaul, Nuclear Services Corp.
R. D. Bailey, LLNL	R. P. Kennedy, EDAC
R. Bea, Woodward-Clyde	H. Lambert, TERA
D. L. Bernreuter, LLNL	R. T. Langland, LLNL
J. J. Burns, NRC	A. Lemoine, Systems Control
K. W. Campbell, TERA	L. Lewis, LLNL
C. K. Chou, LLNL	R. W. Mensing, LLNL
J. Chrostowski, J. H. Wiggins Co.	R. C. Murray, LLNL
D. H. Chung, LLNL	K. S. Pister, U.C., Berkeley
J. Collins, J. H. Wiggins Co.	P. D. Smith, LLNL
C. A. Cornell, MIT	J. Stevenson, JDS-McKee
G. E. Cummings, LLNL	S. W. Tagart, Jr., Nuclear Services Corp.
R. G. Dong, LLNL	F. J. Tokarz, LLNL
J. Goudreau, LLNL	V. Vagliente, LLNL
E. E. Hill, LLNL	I. Wall, NSC
J. Hudson, J. H. Wiggins Co.	J. Wells, LLNL
J. J. Johnson, LLNL	P. P. Zemanick, Westinghouse

Meeting of July 16–17, 1979

P. Albrecht, NRC	L. L. George, LLNL
A. H.-S. Ang, Univ. of Illinois	A. C. Heidebrecht, McMaster Univ.
G. T. K. Asmis, Atomic Energy Control Board (British Columbia)	J. M. Hudson, J. H. Wiggins Co.
	J. J. Johnson, LLNL
	J. E. Kelly, SAI-Palo Alto
G. Bagchi, NRC	R. P. Kennedy, EDAC
R. D. Bailey, LLNL	T.-Y. Lo, LLNL
D. L. Bernreuter, LLNL	R. W. Mensing, LLNL
R. J. Brazee, NRC	J. O'Brien, NRC
C. W. Burger, NRC	K. S. Pister, U.C., Berkeley
K. W. Campbell, TERA	J. E. Richardson, NRC
R. D. Campbell, EDAC	M. P. Singh, LLNL
C. K. Chou, LLNL	P. D. Smith, LLNL
T. Y. Chuang, LLNL	J. C. Stepp, FUGRO
D. H. Chung, LLNL	J. D. Stevenson, Woodward-Clyde
J. D. Collins, J. H. Wiggins Co.	M. Taylor, NRC
J. F. Costello, NRC	F. J. Tokarz, LLNL
G. E. Cummings, LLNL	N. C. Tsai, NCT Engineering
R. G. Dong, LLNL	R. W. K. Tso, McMaster Univ.
A. A. Garcia, SAI-Bethesda	K. Vepa, LLNL
B. J. Garrick, PLG	P. P. Zemanick, Westinghouse

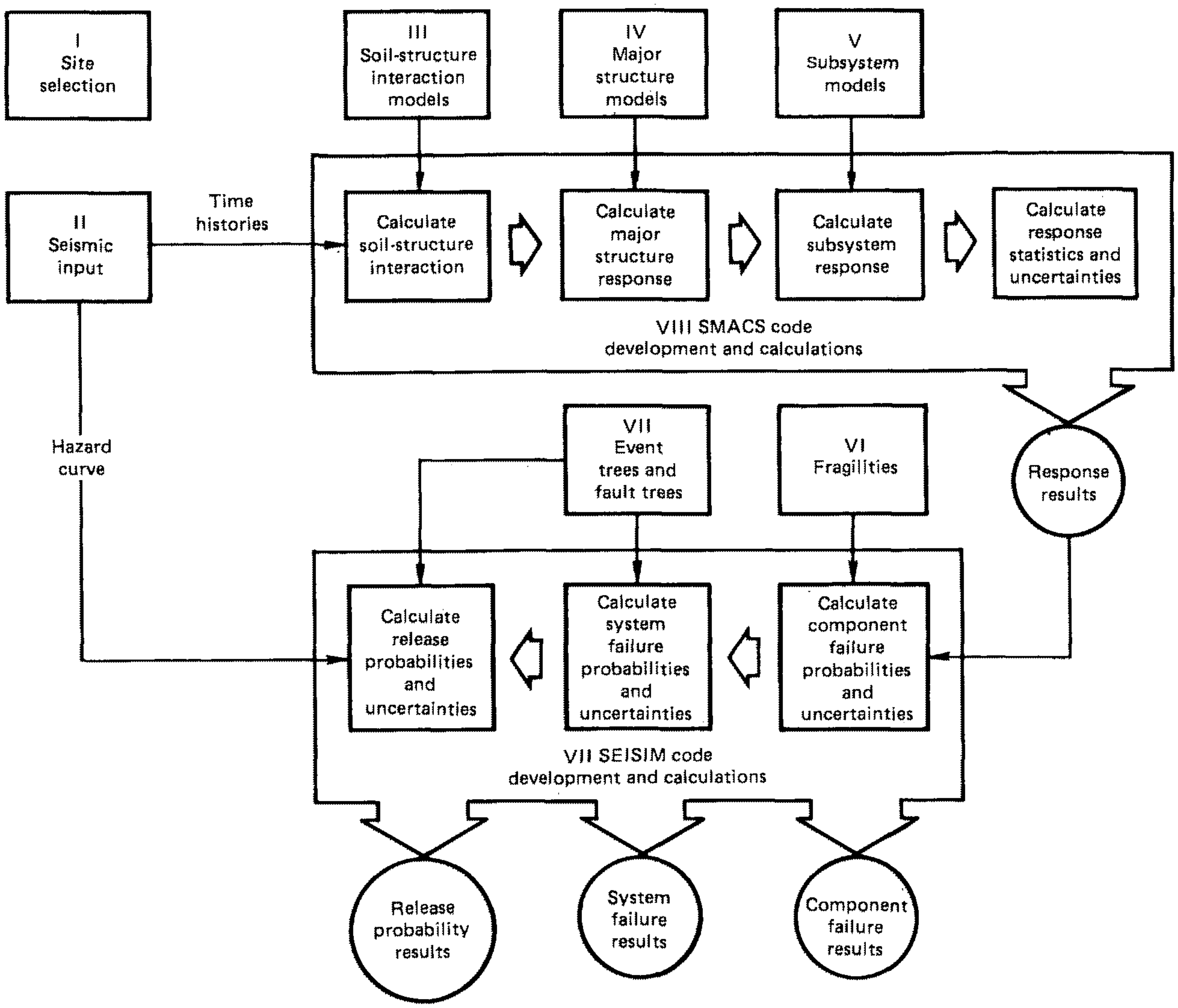


Fig. 7. A flow chart of the computational procedure, showing the roles of the several Phase I projects (indicated by Roman numerals).

became secondary goals of the soil-structure interaction project. Finally, we studied the effects on the results for the Zion plant of different modeling assumptions and different analysis techniques.

Project IV: Major Structure Response. Project IV provided the models needed for the second segment of SMACS: calculation of structural response (carried out in tandem with the soil-structure interaction calculations by the same software package). The choices of appropriate modeling parameters rested on thorough studies of structural modeling, methods of dynamic analysis, damping, nonlinear behavior of material and structures, and the uncertainties in each.

Project V: Subsystem Response. Given the input motions calculated in its first two segments (the soil-structure interaction software), SMACS then computes the response parameters for all subsystems identified as part of any accident scenario. As part of Project V, we evaluated the ability of analytical techniques to predict dynamic behavior, assessed the uncertainties, and developed the software necessary to implement this third segment of SMACS. Then, data were collected and models established for the pertinent piping subsystems. A final part of Project V directly involved the concerns of Projects II through IV as well: A sensitivity study was undertaken to determine the contributions of the several links in the seismic methodology chain—seismic input, soil-structure interaction, structure response, and subsystem response—to the uncertainty in subsystem response.

Project VI: Fragilities. Together with the response results from SMACS, the fragility curves provided by Project VI constitute the initial input to SEISIM for the calculation of structure and component failure probabilities and uncertainties. These curves provide all of the fragility descriptions necessary for Phase I for safety-related equipment, piping systems, and structures at the Zion plant. We based these descriptions on available data, expert opinion (solicited and appraised as part of Phase I), and engineering judgment. We also established a panel of consultants (see §1.4.1) to review the progress of Project VI, as well as to evaluate the fragility descriptions developed.

Project VII: Systems Analysis. The objectives of Project VII were to develop the analytical procedure (SEISIM) for calculating the probabilities of structural failure, component failure, system failure, and radioactive release, together with their associated uncertainties; and to develop the event trees and fault trees for the Zion plant needed for these calculations. As a side study, we commissioned a review of systematic errors at the Zion plant. In developing the computational code itself, we undertook an extensive study of statistical methods.

Project VIII: SMACS and BE-EM. The principal aim of Project VIII was to develop and execute SMACS, the computer code that comprises the calculational procedures developed by Projects III (soil-structure interaction), IV (major structure response), and V (subsystem response). The output included the responses and uncertainties for the Zion plant. This output was, in turn, input for SEISIM. A second objective was to introduce the concept of comparing a probabilistic "best-estimate (BE) method" with the kinds of evaluation method (EM) exemplified by the NRC's *Standard Review Plan*.

1.5 Beyond Phase I

Since the main thrust of Phase I was simply to develop a workable risk assessment methodology, much remains to be done in subsequent phases. We shall discuss our projections in more detail in Sec. 3, but we can look ahead toward general goals here. Our early attention in Phase II will be directed toward completing the sensitivity studies started in Phase I, improving our calculations of initiating-event probabilities, and refining our estimates of release probabilities. Along these same lines, we shall develop models of additional subsystems, implement additional fault trees, and improve our fragility characterizations. We shall pursue side studies as necessary for insight and technical guidance.

The results of the sensitivity studies will be used to focus our efforts on the links in the methodology chain where further work will be most profitable. Likely improvements include:

- New methods for generating simulated time histories.
- A means of accounting for local soil-column effects in the descriptions of seismic input.
- Means of accounting for complex foundation shapes, flexible foundations, and nonlinear structural responses.
- Software simplifications (aimed at making the programs more user-oriented) that do not sacrifice accuracy.

Finally, a significant effort will be devoted in the coming months to applying the SSMRP methodology to a study of the auxiliary feedwater system on Unit 1 of the San Onofre Nuclear Generating Station, located in southern California. This system was selected by the NRC for immediate scrutiny and will thus become the first application of the computational procedure we developed in Phase I to address a licensing concern.

SECTION 2: MAJOR ACHIEVEMENTS OF PHASE I

2.1 Technical Products and Results

The most important accomplishment of Phase I was to develop and demonstrate a generally applicable probabilistic computational procedure, using an interdisciplinary approach and state-of-the-art modeling techniques. Our first computations for the Zion plant produced higher estimates of failure and release probabilities than we expected, but these results are tentative, and were obtained more as a first exercise for our computational procedure than as defensible estimates of risk. More important than these preliminary results are the many other products of the efforts that led to them. These products may be broadly categorized as models (including fragility descriptions), computer programs, data bases, and results from various side studies.

The models include many that are specific to the Zion plant: structural finite element models, subsystem models, and fault trees. Some of the fragility curves are likewise plant specific. More generally useful were the models used to generate the hazard curve and the time histories for the Zion site. These models embody generally applicable seismological relationships; thus, they can be useful for other sites as well. Indeed, a mathematical relation established between magnitude scales used in the eastern and western United States is already being used by the NRC. Likewise, the usefulness of the event trees (perhaps with some modification) and many of the fragility curves is not restricted to the Zion plant. The generic fragility curves, especially, represent a significant achievement, since some skepticism had been expressed in the outside community about the feasibility of generating such curves.

Three useful and reasonably economical computer programs were developed as part of Phase I: SMACS, SEISIM, and HAZARD. The first and second of these, which, in tandem, carry out the probabilistic computations of the SSMRP, are among our principal products. We expect in the near future that SMACS will become generally useful to the NRC, either as part of a risk assessment program or as a tool for calculating response alone. SEISIM will not play a practical role as quickly as SMACS, but it currently appears to be the only event tree–fault tree risk assessment code that properly accounts for the common-cause nature of earthquakes. Both of these codes were developed in a research environment, but our efforts in Phase II will include steps to make them more user-oriented. The third code, HAZARD, assesses the seismic hazard at a given site and is currently used by the NRC in its Systematic Evaluation Program.[8]

In establishing the earthquake hazard in the vicinity of the Zion plant (and in the entire central United States) and in developing fragility curves, we sought to collect and evaluate all available pertinent data. The collected data are now stored at LLNL and can be used in studies of other plants. Finally, several of the side studies undertaken as part of the SSMRP (notably the reviews of soil-structure interaction, structure response, and subsystem response) provided assessments of modeling uncertainties and comparisons of analytical techniques.

The following sections expand considerably on the foregoing paragraphs. The achievements of the eight Phase I projects are summarized, and our progress is illustrated with representative results. The projects are discussed in numerical order, except that Project VIII (SMACS and BE-EM), which ties together Projects III, IV, and V, immediately follows the last of those three projects.

2.1.1 Project I: Plant/Site Selection and Data Collection

As its title implies, Project I had two parts: the selection of a representative nuclear power plant and site, and the collection of data needed by the other SSMRP projects. We selected Unit 1 of the Zion Nuclear Power Plant north of Chicago, Illinois, (Fig. 8) to fulfill the first of these goals. The operating utility, Commonwealth Edison Company (CECo), agreed to provide available data and records.

To confirm the choice of Zion 1, Engineering Decision Analysis Company, Inc., (EDAC) compared the plant and site with other plants.[15] They concluded that Zion is representative with regard to the design of its nuclear steam supply system (NSSS; the plant is a Westinghouse PWR); the type of containment structure (prestressed concrete); its

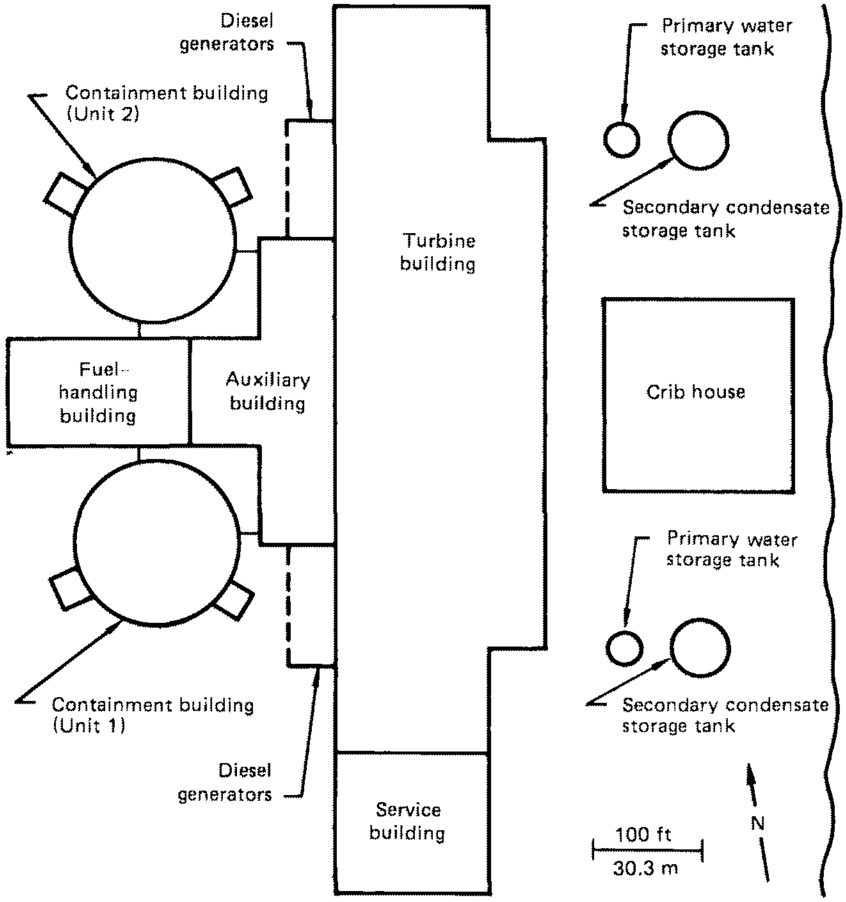


Fig. 8. Site plan of the Zion Nuclear Power Plant.

electrical capacity (1100 MWe); its location (the Midwest); the peak seismic acceleration used for design (0.17g); and the properties of the underlying soil (the low-strain shear-wave velocity is 1650 ft/s in a 50- to 100-ft-thick layer of soil overlying sedimentary bedrock). However, the proximity of bedrock, with its much higher shear-wave velocity, does introduce some uncertainty in our specification of input motion (see §3.1).

Data on the Zion plant were gathered from several sources. The Southwest Research Institute provided system and component data from the Nuclear Plant Reliability Data System.[16] They collected 22 pieces of engineering data on each of 26 systems and 5384 components. Sargent & Lundy Engineers (S&L; the original architect-engineers) and CECo were very cooperative in supplying design information, plant descriptions, design information, plant records, and test results. Westinghouse Electric Corporation supplied data on the NSSS to S&L, who were responsible for developing a mathematical model of the system.

2.1.2 Project II: Seismic Input

Project II was charged with developing "a probabilistic statement of the seismic hazard," which was to comprise the time histories upon which the SMACS computation was based and the hazard curve necessary for the calculation of unconditional release probabilities by SEISIM. Our approach can be broken down into four parts:[17,18]

- Specification of seismic zones of roughly uniform seismicity.
- Development of an occurrence model, which specifies for each zone the probability distribution of earthquake magnitudes, the largest possible earthquake, and the time dependence of seismic events.
- Development of a ground motion model, which accounts for earthquake source effects,

regional attenuation of ground motion, and local site effects.

- Development of the computer program HAZARD to assess the seismic hazard and to develop and correlate the information necessary to generate time histories.

The short period for which we have seismic data, the scarcity of strong-motion data for the central United States, and the considerable uncertainty about earthquake mechanisms and causes east of the Rockies led to a heavy reliance on expert opinion. We therefore assembled three panels of experts as sources of professional judgment (see Table 1). Seismic zonation, for example, is a matter of conjecture, and we based our zonation wholly on the opinion of the Eastern United States Seismicity Panel. Nonetheless, it was the variations in earthquake occurrence models and ground motion models, together with local site effects such as the shallow bedrock, that led to the largest uncertainties in the estimated peak ground accelerations at the Zion site.

The earthquake occurrence model depended in part on the historical record. We therefore assembled available historical catalogs of earthquakes and sought to resolve the inevitable differences among them. This corrected historical record was used along with expert opinion to develop the occurrence models for each zone.

As part of the ground motion model, TERA/DELTA developed a source model for a point dislocation that allowed insight into the effect on ground motion of such parameters as stress drop, focal depth, and the seismic quality factor Q (which is inversely proportional to the damping).[19] The treatment was then modified to account for extended ruptures. The results of the point-source model were especially important for Project III (soil-structure interaction), which required predicted angles of incidence for different wave types at the Zion site.

To develop our ground motion model, we assumed that in the near field the ground motion due to the "same" earthquake would be identical in the eastern and western United States. In the intermediate and far fields, the differences in ground motion were taken to arise solely from differences in regional attenuation. We confirmed the regionalization of attenuation properties and drew inferences about the effect of Q on the frequency content of earthquakes in the East and West.[20] Various earthquake magnitude scales were also compared: An important result showed that $M_L = 0.57 + 0.92m_b$, where M_L is the local magnitude for a western U.S. earthquake and m_b is the body-wave magnitude recorded for an earthquake in the East.[21]

Based on these assumptions and results, and on discussions with our expert consultants, we chose a ground motion model that relies on "magnitude weighting." The local ground motion is established by combining (1) a relation giving site intensity (as observed in the central United States) as a function of earthquake magnitude and epicentral distance, and (2) a relation (based on western U.S. data sets)[22] between observed ground motion, site intensity, and earthquake magnitude. Figure 9 compares the peak ground acceleration, as a function of distance, predicted by the model we chose with that predicted by several other plausible ground motion models. The resulting hazard curve is shown in

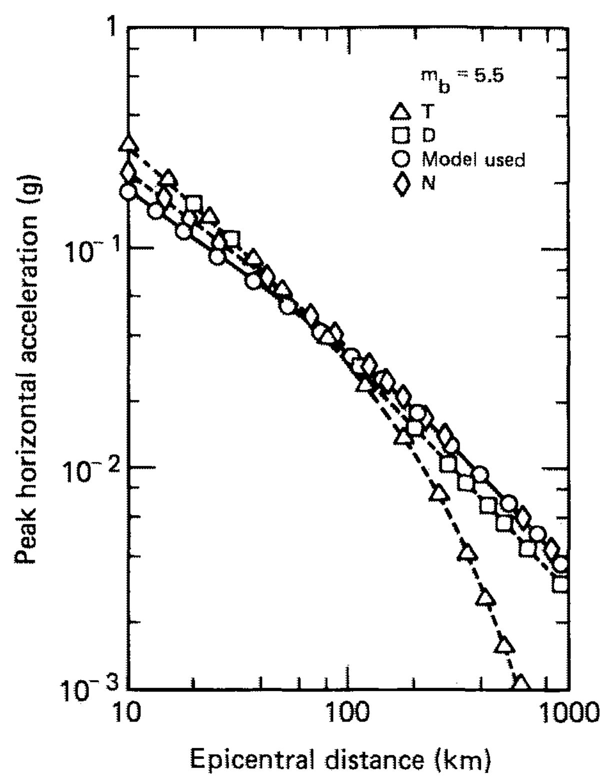


Fig. 9. Ground motion attenuation for an earthquake of $m_b = 5.5$, predicted by several ground motion models for the central United States: T, theoretical model proposed by Nuttli;[23] D, distance-weighted model;[24] N, unweighted model.

Figs. 10 and 11. In Fig. 10, the hazard curve is compared with curves drawn from the recent historical record and curves that represent the extreme views of our panel of experts. All of the curves in Fig. 10 assume our magnitude-weighted ground motion model, so the extremes of expert opinion result solely from variations among occurrence models and differences in zonation. (Uncertainty in the hazard curve was not explicitly accounted for in Phase I.)

To generate simulated time histories consistent with the hazard curve, we used the modified Monte Carlo approach illustrated in Fig. 12. On the basis of the characteristics of the various seismic zones and the information developed for the occurrence model, each ring Z_i at a distance R_i from the site was described by an earthquake occurrence frequency λ_{ij} for each magnitude range $M_j \pm \Delta M_j/2$. For each magnitude-distance pair, we next introduced a Monte Carlo simulation of N_{ij} trials, where N_{ij} is proportional to λ_{ij}. Each trial produced a peak spectral acceleration and spectral shape, selected from lognormal distributions corresponding to R_i and M_j. The distributions were based on our ground motion model. From the many such spectra generated (Fig. 13a is an example), we randomly selected 30 in each of six ranges of peak ground acceleration (0.15–0.30g, 0.30–0.45g, 0.45–0.60g, 0.60–0.75g, 0.75–0.98g, and >0.98g). Finally, time histories (such as the one illustrated in Fig. 13b) were generated for each response spectrum, using the program SIMQ.[25]

2.1.3 Project III: Soil-Structure Interaction

When subjected to seismic excitation, the response of a massive structure supported on soft soil differs substantially from that of an identical structure supported on a very stiff soil or rock. Acknowledging and computing the effects of this

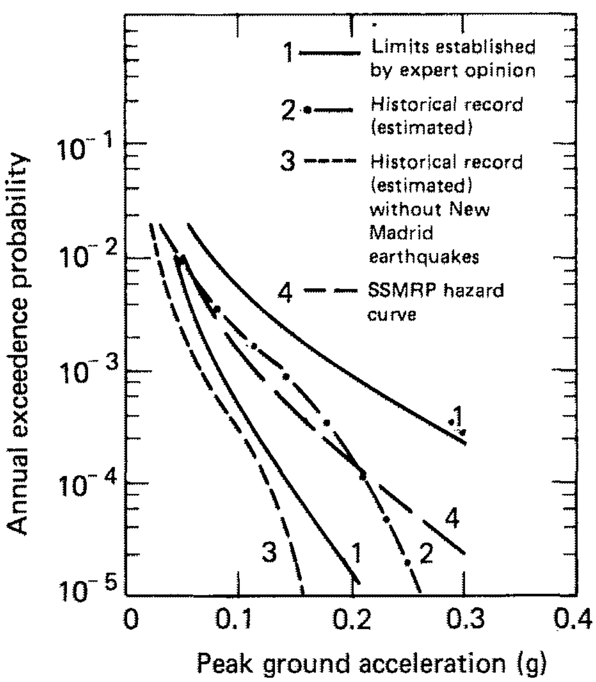


Fig. 10. A portion of the seismic hazard curve for the Zion site, compared with the historical record and with curves based on the extremes of expert opinion. All curves are based on the same ground motion model; thus, the solid curves represent the uncertainties in seismic zonation and the occurrence model.

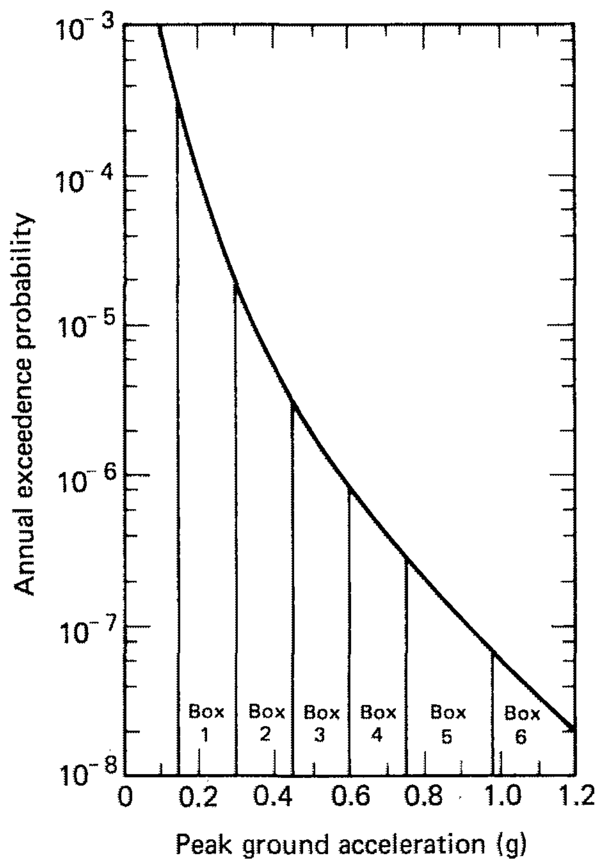


Fig. 11. Seismic hazard curve for the Zion site. The six "boxes" define the acceleration ranges for which independent calculations, based on the median of each range, were carried through the SSMRP computational procedure.

coupled soil-structure system was the object of Project III. As part of this project, we (1) developed the software necessary for the computer code SMACS to compute basemat motions and in-structure response from the seismic input specified by Project II; (2) supplied the SMACS input parameters that characterized the behavior of the soil, foundation, and structures at the Zion site; and (3) studied the effects of variations in a number of critical parameters.

With the help of consultants and subcontractors, we first reviewed the status of current analysis techniques.[13,26-30] Two classes of such techniques are available for analyzing soil-structure interaction effects: the direct method, which analyzes the idealized soil-structure system in a single step, and the substructure approach, which treats the problem in a series of steps (determination of the foundation input motion, determination of the foundation impedances, and analysis of the coupled

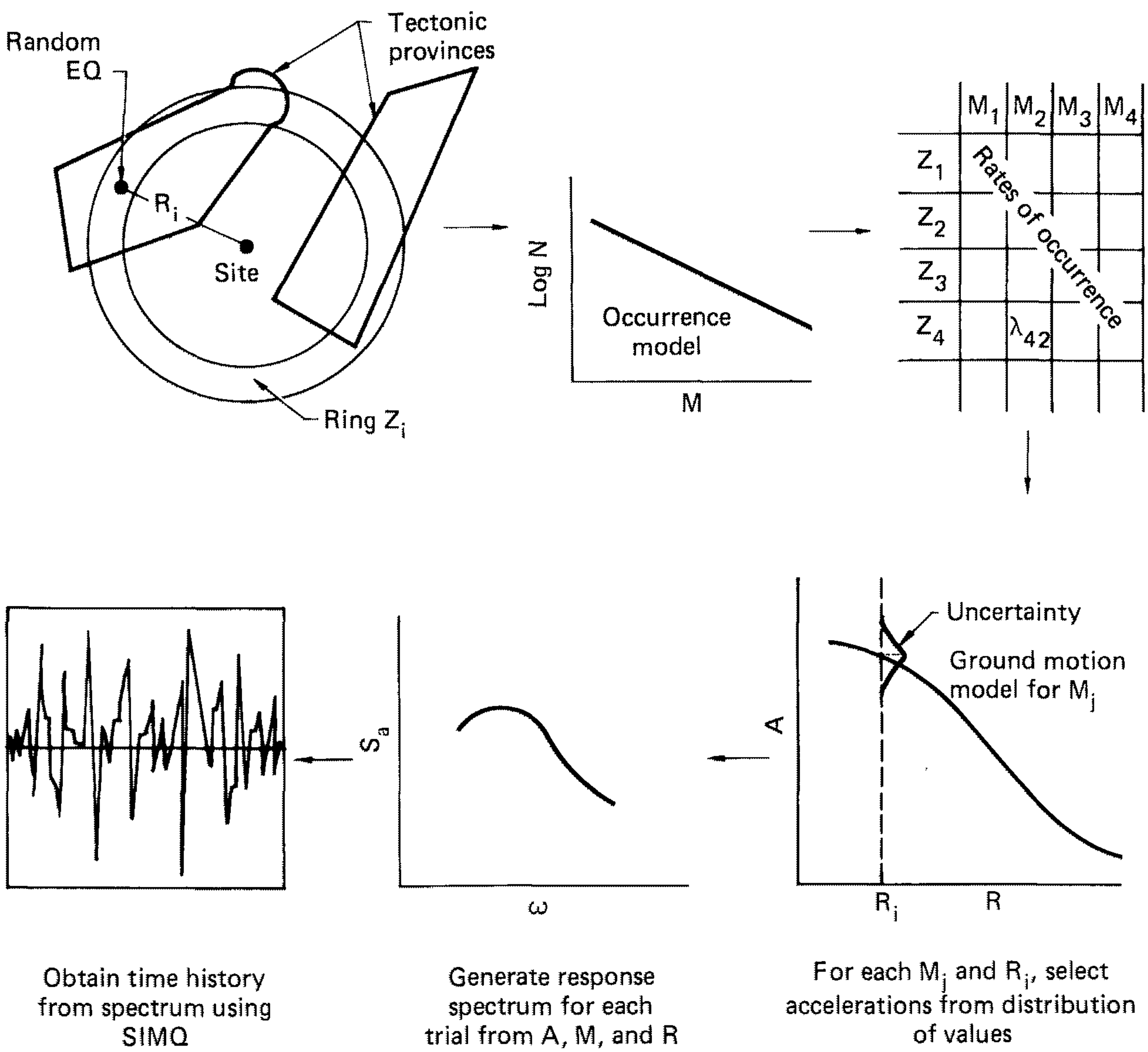
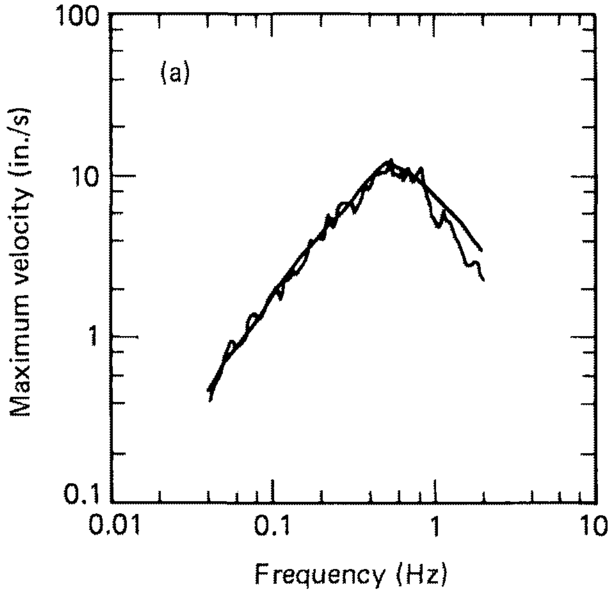


Fig. 12. Multistep procedure for deriving simulated time histories. The procedure demands an estimate of seismic zonation, an occurrence model, and a ground motion model. The procedure was implemented using the program HAZARD.



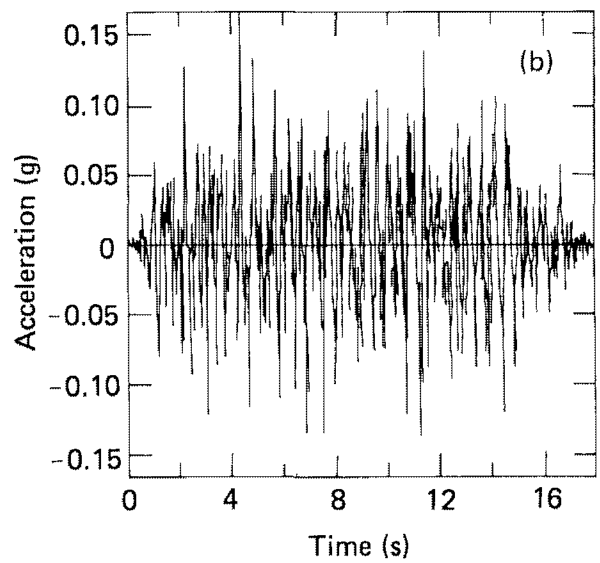


Fig. 13. A typical response spectrum (a) and a typical time history (b) for the Zion site.

system). In the review reports, attention was focused on the accuracy of the two methods, sources of uncertainty, the means of accounting for the three-dimensionality of the soil-structure interaction problem, the consequences of nonlinear soil properties, the effect of embedment, and the effect of interactions among neighboring structures. Soil-structure interaction was acknowledged to be a complex phenomenon for which no exact solution is currently possible. The greatest uncertainties persist in specification of the free-field ground motion (see §2.1.2), followed by uncertainties in the material behavior of soil and differences in soil-structure interaction modeling procedures. Despite these uncertainties, adequate solutions are made possible by judiciously applying engineering judgment and the results of parametric studies.

In part based on these reviews, a decision was made to adopt the substructure approach for our calculations. To implement this approach, illustrated schematically in Fig. 14, we used the CLASSI family of programs, developed by Luco and Wong.[31] CLASSI is organized according to the steps of the substructure approach. The set of CLA codes solves the first two steps—determination of the foundation input motion and the foundation impedances. The coupled soil-structure system is then analyzed by SSIN, the program that formed the core of our soil-structure interaction and major structure response calculations in SMACS. We also used the highest-order member of the set of CLA codes, which analyzes deeply embedded foundations and axisymmetric geometries.

Before carrying out our Phase I response calculations, we investigated several phenomena to assess their impact on the dynamic response of the Zion plant. Two such investigations were aimed at understanding the effects of embedment and nonvertically incident seismic waves on structural response.[32] The effect of embedment on foundation impedances appeared not to be large, but the effect on in-structure response was significant, as shown in Fig. 15. The effect of nonvertically incident P/SV waves and SH waves was examined in conjunction with Project II. For the Zion site and structures, such waves significantly affect only the expected torsional motions; Table 3 displays representative results.

Among our side studies were comparisons of the direct and substructure approaches. First, a finite element direct approach using the program FLUSH was compared to the substructure approach using the program CLASSI.[33] Two cases were considered: the Zion containment building as an isolated structure and the entire complex of structures at the Zion plant. Being a well-defined problem, the first case served as a benchmark for the two analytical procedures. Figure 16 illustrates the generally good agreement between the two techniques, though either can produce the higher

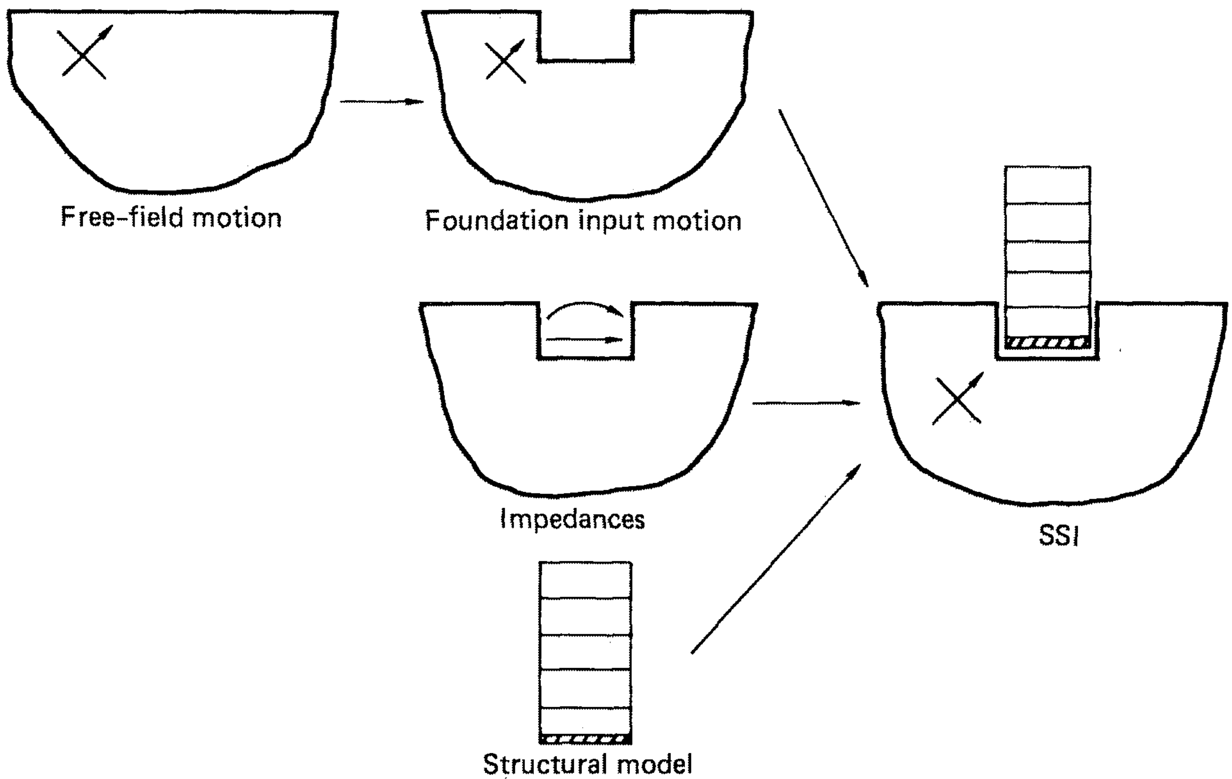


Fig. 14. Schematic representation of the elements of the substructure approach to soil-structure interaction analysis. This approach calculates foundation input motion, foundation impedances, and in-structure responses in three separate steps.

response. This case reinforced the conclusion of the review reports that careful specification of the problem leads to consistent results for the two analysis methods, provided the basic assumptions of the methods apply.

Analysis of the entire Zion facility required extensive simplifications and judgment in the use of both the direct method and the substructure approach. The FLUSH analysis is essentially two-dimensional, which dictates consideration of a number of slices through the facility. Each slice must be assumed to behave independently, and the results must be superposed. CLASSI permits the analysis of embedded foundations only if they have axisymmetric geometry. We therefore developed equivalent cylindrical properties to represent the complicated foundation of the auxiliary-fuel-turbine (AFT) complex. Results for this second comparison

Table 3. Response, in units of gravity, at the top of the Zion containment shell for vertically and nonvertically incident seismic waves.

Response component	For vertically incident waves	For intermediate wave velocities[a]	For slow wave velocities[b]
Horizontal (x)	0.6303	0.6396	0.6480
Horizontal (y)	0.4519	0.4513	0.4485
Vertical	0.2555	0.2514	0.2447
Rocking (x)	0.2018	0.2003	0.1967
Rocking (y)	0.2718	0.2771	0.2810
Torsional	0.0119	0.0437	0.0783

[a]Apparent wave velocities: P/SV = 6 km/s, SH = 4 km/s. Vertical waves have an infinite apparent velocity.
[b]Apparent wave velocities: P/SV = 3 km/s, SH = 2 km/s.

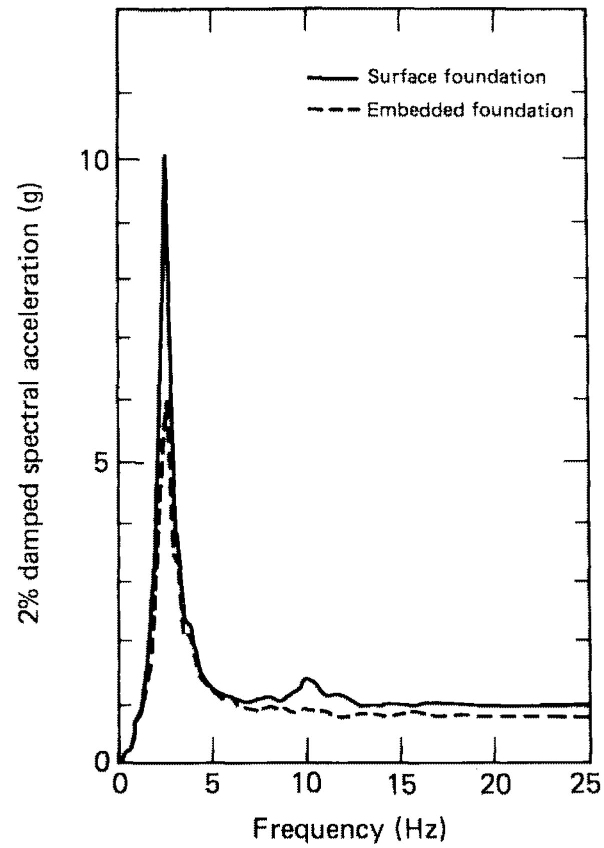


Fig. 15. Response spectra at the top of the containment shell, for embedded and surface foundations. The input was a single recorded accelerogram. The computation used CLASSI.

indicate the variability possible between results based on different soil-structure interaction analysis procedures.

We also began a preliminary study of the significance of nonlinear soil material properties on soil-structure interaction. The emphasis in this continuing study is on identifying those aspects of nonlinear behavior that can be adequately modeled by simpler, equivalent linear techniques. We are looking at two situations: soil amplification and three-dimensional soil-structure interaction. Soil material models include linear viscoelastic and multisurface plasticity representations. Results will serve to quantify the variability in soil and structure response due to modeling.

2.1.4 Project IV: Major Structure Response

Major structure response was obtained as one of the results of the SMACS computation. Con-

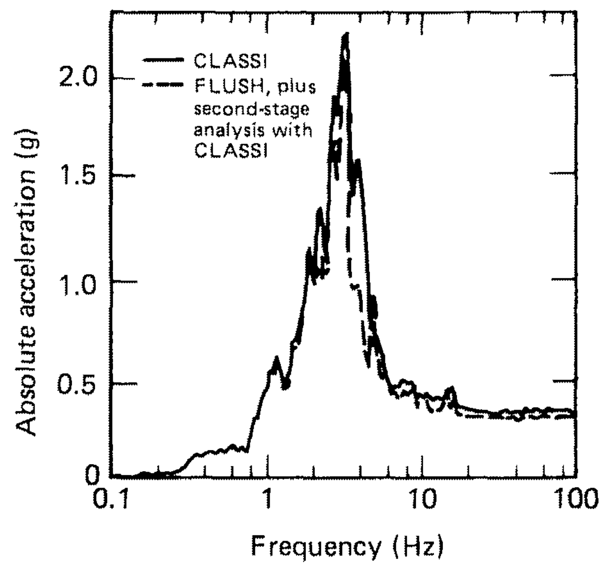


Fig. 16. Response spectra computed for the top of the containment shell, using the direct method (FLUSH) and the substructure approach (CLASSI).

struction of the mathematical models used to represent the containment building (Fig. 17) and the AFT complex was the responsibility of this project. We developed detailed finite element models of each. The containment building is composed of the containment shell and an internal structure, connected only through the basemat; hence, we modeled each structure separately. The containment shell was modeled with a series of beam elements with shear and bending characteristics appropriate for a circular cylindrical shell. Masses and rotary inertias were lumped at nodal points. Inertias affecting bending and torsional response of the shell were included. Thirteen modes were included in the analysis. The internal structure, including a simplified model of the NSSS (reactor pressure vessel, steam generators, reactor coolant pumps, and piping), was modeled with three-dimensional beam and plate finite elements. Masses were again lumped at selected nodes. Sixty modes were included in the analysis.

The AFT complex consists of the T-shaped auxiliary building, the turbine building, the fuel-handling building, and the diesel generator buildings. These structures are founded on a common base slab of varying elevation. Common floor slabs in the superstructure provide additional structural connections. Constructed of reinforced

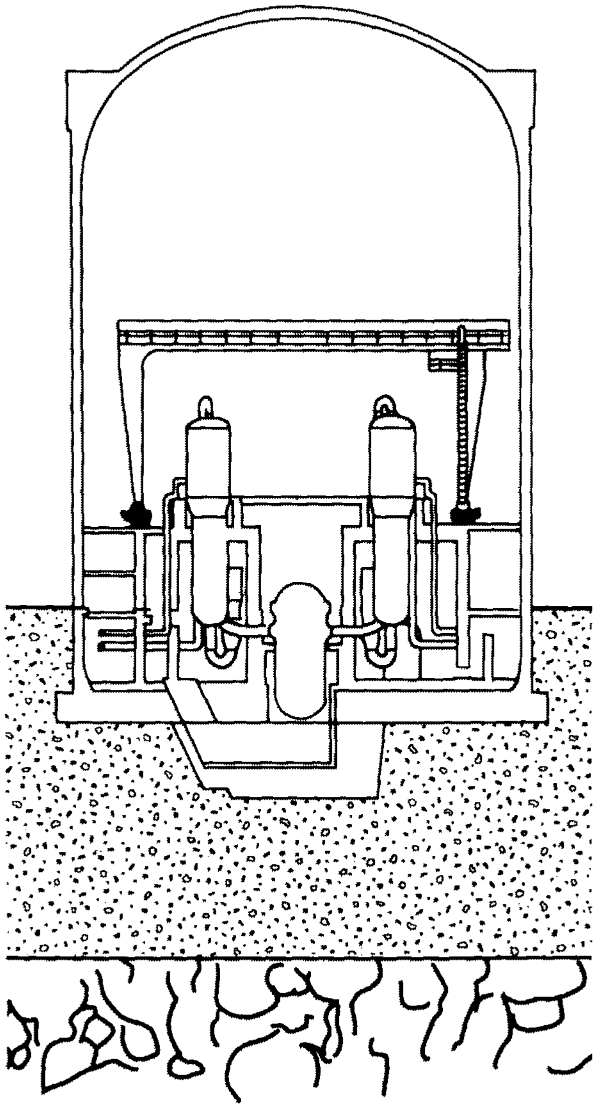


Fig. 17. Schematic cross section of the containment building at Zion, showing the approximate extent of embedment.

concrete and braced steel frames, the complex is essentially symmetric with respect to an east-west axis that divides the two generating units of the plant.

The finite element model of the AFT complex employed thin plate and shell elements to represent the concrete shear walls, and beam and truss elements to model the braced frames. Because of its symmetry, only half of the structure was modeled (Fig. 18). Nonetheless, the size and complexity of the model required a reduction in the number of dynamic degrees of freedom. We therefore lumped masses at selected node points, thereby eliminating all massless degrees of freedom while retaining the stiffness definition of the detailed model. The location and number of lumped mass points were chosen to minimize the effect of the simplification procedure on the response in the auxiliary building area and to suppress "local modes" in the turbine building. One hundred and thirteen modes were extracted from the reduced model to define the dynamic characteristics of the AFT complex for the SMACS computations.

As part of Project IV, we also undertook several side studies to quantify major sources of uncertainty in the computation of structure response. In one study, we sought to quantify effects of different modeling assumptions.[34] Four different mathematical models were created to describe a portion of the AFT complex that excluded the turbine building and was truncated at grade. The models were a detailed finite element model, a detailed finite element model with masses lumped at selected nodes, a detailed finite element model with the constraint of rigid floors, and a shear-beam model. Dynamic characteristics (frequencies and mode shapes) and response quantities (peak nodal accelerations and in-structure response spectra) were determined for the models and compared.

We found that all modeling approaches preserved total mass and rotational moments of inertia. The coefficients of variation for these quantities were small. A greater variation was seen in the frequencies of comparable modes: Coefficients of variation varied from 0.09 to 0.31. The variation among the four models in nodal accelerations at floor slab centers of gravity was comparable to the variation in the determined frequencies. No model consistently gave either the maximum or the minimum response.

The effect of the different modeling assumptions was most evident in the large variations seen (for example, in Fig. 19) in the response of locally flexible areas of the structure. The method used in each model to define the relative distribution of stiffness and mass in these areas dictated whether the local response was accurately determined. In particular, the location of lumped-mass points profoundly affected dynamic behavior. Lumping mass at nodes allows the analyst to selectively include or exclude vibratory modes and to subsequently bias response values.

A second side study looked at the effects of random variability in properties such as concrete density and member dimensions, and at the effects

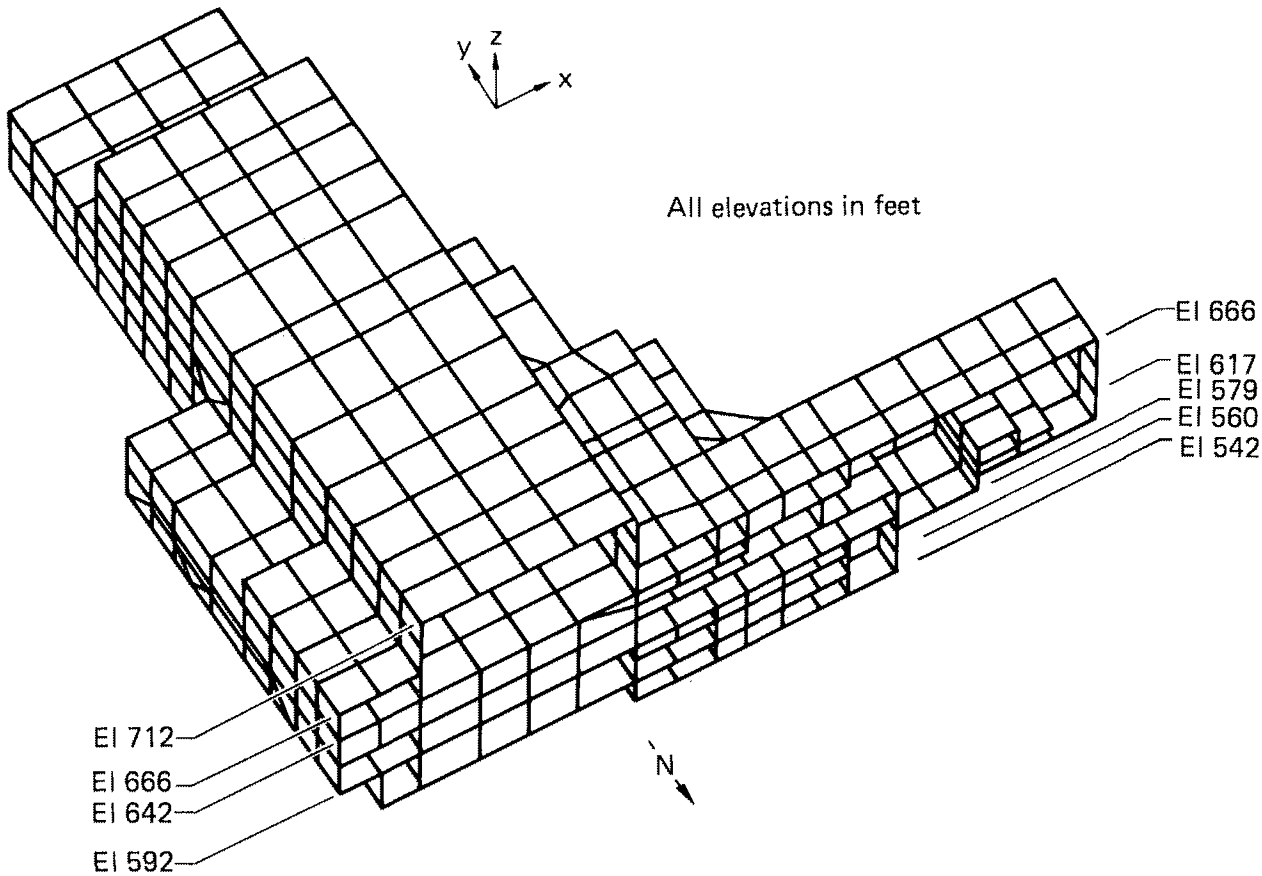


Fig. 18. Detailed finite element model of the AFT complex.

of the consequent variability in structural damping.[35] Another study of structural uncertainties produced a methodology for explicitly modeling these uncertainties as three normalized random variables that relate to modal frequency, amplitude, and damping.[36] The methodology relies on experimental data, either from the structure being modeled or from similar structures.

Among the preliminary studies that led to our decisions about modeling techniques and calculational procedures were two extensive reviews of state-of-the-art structural analysis methods.[37,38] The review reports discussed modeling options, dynamic analysis techniques, damping, nonlinearity of materials and structures, and uncertainties.

2.1.5 Project V: Subsystem Response

The primary aim of the subsystem response project was to determine the appropriate response parameters for components and systems appearing in the fault trees developed by Project VII, and to provide SMACS with the input and software it needed to calculate responses for critical subsystems, given the input motion computed for the subsystem supports. (For our purposes, a subsystem is any component or system whose behavior during a seismic event can be decoupled from the major structure response.) Before undertaking development of the necessary models and software, we solicited two extensive reviews of subsystem response methodologies.[39,40] These reviews, provided by the Nuclear Services Corporation/Quadrex and EDAC, achieved several aims. They

- Reported on existing methods and methods under development for the seismic qualification of subsystems.
- Assessed the accuracy of available analysis techniques.
- Identified and quantified sources of random and modeling uncertainty.
- Recommended appropriate analysis techniques for the SSMRP.

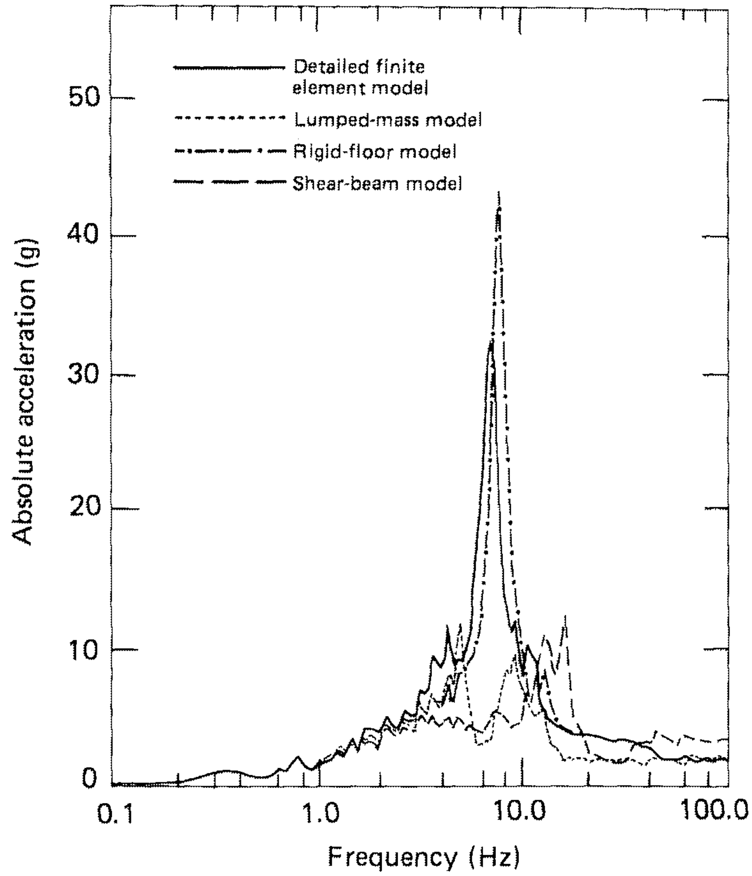


Fig. 19. Comparison of responses at the fuel-handling building wall in four different models of the modified AFT complex.

Before embarking on the construction of our own models, we reviewed and summarized information on available models of piping systems, including the main steam and main feedwater system, auxiliary feedwater system, residual heat removal system, service water system, safety injection system, component cooling water system, containment spray system, and reactor coolant system. Then, with this information and using current modeling techniques, we constructed the mathematical models needed for response computations. The new piping models were for the auxiliary feedwater system (2 models), service water system (3), residual heat removal system and safety injection system (6), and reactor coolant loop system (2).

The reactor coolant loop model[41] contains all four main reactor coolant loop piping systems, six branch lines (the pressurizer surge line, the residual heat removal line, the bypass line, the safety injection line, and two pressurizer spray lines), supports, and all major equipment (including the reactor pressure vessel, four steam generators, four reactor coolant pumps, and a pressurizer). Each coolant loop system includes three sections of pipe—a hot leg from the reactor pressure vessel to the steam generator, a crossover leg to the reactor coolant pump, and a cold leg back to the reactor pressure vessel. The model, which is represented by 760 nodes and 2941 equations, was prepared for a modified version of SAP4, which is able to deal with grouped supports. Pipe elements were used to describe straight and curved pipes; beam, truss, and boundary elements were used to simulate supports. Finally, stiffness matrix elements were used to represent the stiffness effects of the main steam lines and the feedwater lines, but constant and variable spring hangers, which have small stiffnesses compared to the stiffnesses of other supports, were not included in the model.

Our software development efforts centered on developing a multisupport time-history analysis capability, which involved the modification of SAP4, plus a new program named SAPPAC.[14] Our final response calculations were based on a linear elastic analysis, using the pseudostatic-mode method of multisupport time-history analysis. We accounted for flexibility by incorporating the ASME flexibility factor for piping elbows,[42] and we assumed all piping supports to be rigid, except for part of the auxiliary feedwater line and the reactor coolant loop (see above).

Economy in the subsystem response calculations was achieved by taking two important steps. First, responses were calculated only for components and subsystems identified as potential contributors to the failure of a system. Second, the responses were computed only in terms of the parameters used in the fragility descriptions (for example, accelerations for many structures and components, peak resultant moments for piping).

Figure 20 illustrates a result typical of our subsystem response calculations. Each pair of points (one circle and the triangle above it) represents the distribution of stresses calculated for 30 earthquake simulations, whose median peak ground acceleration is indicated on the abscissa. (The two plots of Fig. 21, discussed next, explicitly show two such distributions for 90 earthquake simulations.) The variation in response is due in part to the differences among the input time histories and in part to variations in the other input parameters (soil shear modulus, soil damping, structure frequency, structural damping, subsystem frequency, and subsystem damping). Figure 20 illustrates that the SSE design stress (square symbol) is well above our calculated ninetieth-percentile value for the same peak ground acceleration.

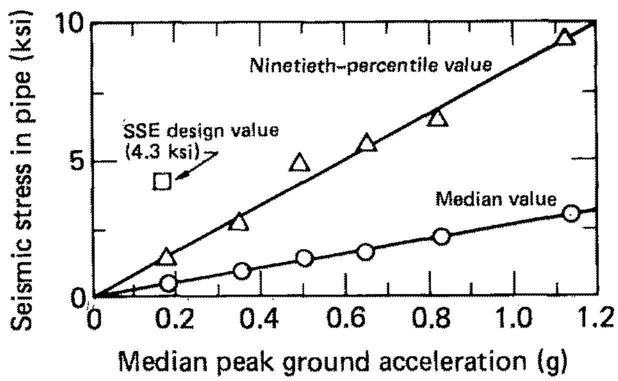


Fig. 20. Stress calculated at a reactor pressure vessel nozzle for six median peak ground accelerations. Thirty time histories were used in the computation of each pair of points (circle and triangle), which reflect the median and ninetieth-percentile values of stress.

As a final part of Project V, we began a study that is looking at the contribution of each link in the

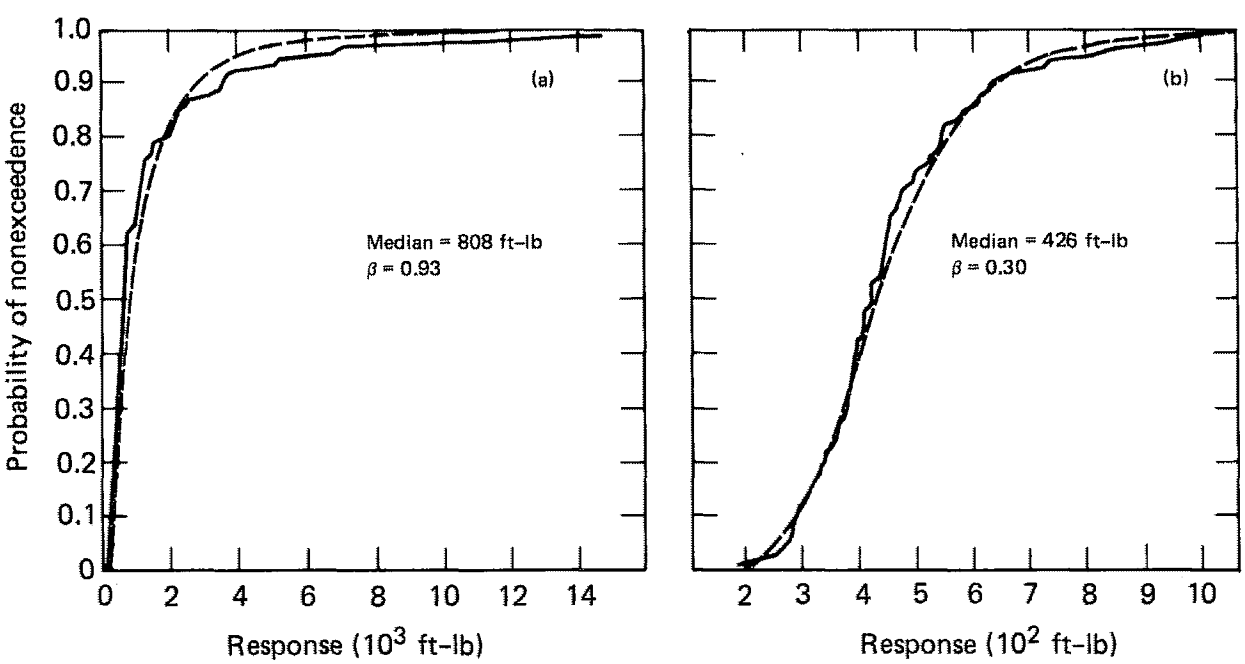


Fig. 21. Distribution of computed peak moments at an auxiliary feedwater nozzle: (a) all input parameters varied to reflect uncertainty; (b) only input ground motion varied. Each solid line represents 90 data points at equally spaced ordinate values. The dashed lines are lognormal fits to the data; β is the standard deviation of the logarithm of response.

seismic methodology chain (seismic input, soil-structure interaction, major structure response, and subsystem response) to the uncertainty in the subsystem responses calculated for a portion of the auxiliary feedwater line and the reactor coolant loop. In each of eight experiments, 90 runs (each including a different earthquake time history with a peak acceleration between 0.15g and 0.30g) are being made to evaluate the effects of several input parameters, which will be varied randomly to reflect the uncertainties in a broad range of variables. The input parameters are soil shear modulus and soil material damping (soil-structure interaction); structure frequencies and structural damping (major structure response); and subsystem frequencies and damping (subsystem response). In each of the eight experiments, a different set of input parameters will be varied. The results of the first two experiments are shown in Fig. 21. In Fig. 21a, the variation in the predicted response is due to random variation of all input parameters. In Fig. 21b, the variation is caused only by the variation in the 90 earthquake time histories; all other input parameters were held at nominal values for the 90 runs.

2.1.6 Project VIII: SMACS and BE-EM

Though designated Project VIII, the development of SMACS is logically discussed here. SMACS (Seismic Methodology Analysis Chain with Statistics) is the computer code that links the seismic input with the soil-structure interaction, major structure response, and subsystem response calculations. Its two main components are the SSIN portion of CLASSI (see §2.1.3) and the multisupport time-history analysis program developed in part by Project V (subsystem response).

Development of the multisupport analysis capability was completed as part of the implementation of SMACS. As part of this effort, we developed a modified version of SAP4 and wrote a new program, SAPPAC.[14] Three methods of performing multisupport time-history analyses were implemented in SAPPAC: the pseudostatic-mode method, the absolute displacement formulation, and the datum acceleration/relative displacement formulation. The pseudostatic-mode method was selected as the most efficient for our purposes. SAPPAC is a stand-alone modular program from which the time-history analysis module was extracted and incorporated into SMACS.

Our initial calculations generated peak accelerations and spectral accelerations at many locations in the structures and subsystems, and peak resultant moments in subsystem piping. The points for which responses were computed were selected according to the requirements of the fault trees developed by Project VII and the fragility descriptions. The fragilities of the structures and floor-mounted components are expressed in terms of structural accelerations, the fragilities of valves mounted on piping systems are tied to subsystem accelerations, and the fragilities of the pipes themselves are characterized by peak resultant moments over the duration of the earthquake.

Figure 22 illustrates results typical of SMACS. Each solid line represents 30 data points (spaced at equal distances along the ordinate), each of which represents the response for one earthquake simulation. Each simulation assumed different values of the input parameters (soil shear modulus and damping, and structure and subsystem frequencies and damping), selected according to a Latin hypercube experimental design.[43] Hence, the variability in the responses is a result of variations in all elements of the seismic methodology chain. These results are for the range of accelerations between 0.15g and 0.30g. Analyses for other acceleration ranges led to similar variations in response.

Our initial selection of input parameter distributions was designed to encompass all aspects of uncertainty and assumed minimal knowledge of the Zion facility. For example, variations in soil shear modulus and damping represent not only our uncertainty in the definition of the viscoelastic material constants at a point, but also phenomena not modeled: the irregular geometry and stiffness of the foundation, nonuniform embedment, nonlinear soil material behavior, separation of soil and structure during the earthquake, and others. Similarly, for structures and subsystems, variability in eigensystem and modal damping properties represent the effects of a wide range of phenomena. Table 4 shows the ranges of the original input parameters.

A second case was considered, where the variability in the input parameters was reduced to reflect a significantly improved state of knowledge; Table 4 also summarizes this case. Figure 23 shows the response results for this second case and should be compared d ently with Fig. 22. The reduction in the variability of response is apparent. The sensitivity study described in §2.1.5 is aimed at learning

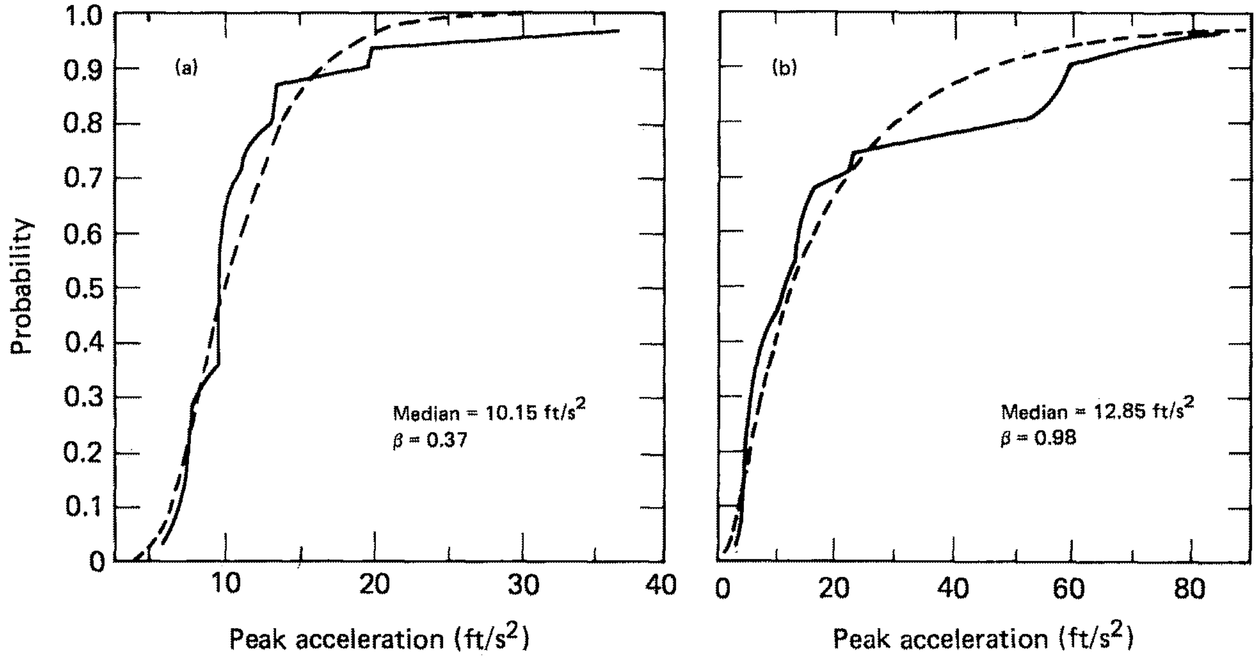


Fig. 22. Distributions of peak accelerations computed at the top of the containment shell (a) and at a check valve in the auxiliary feedwater system (b), using input parameters that reflect a state of minimum knowledge about the Zion plant. Each solid line represents 30 simulations (peak free-field ground accelerations between 0.15g and 0.30g), and the dashed lines are two-parameter lognormal fits. β is the standard deviation of the logarithm of response.

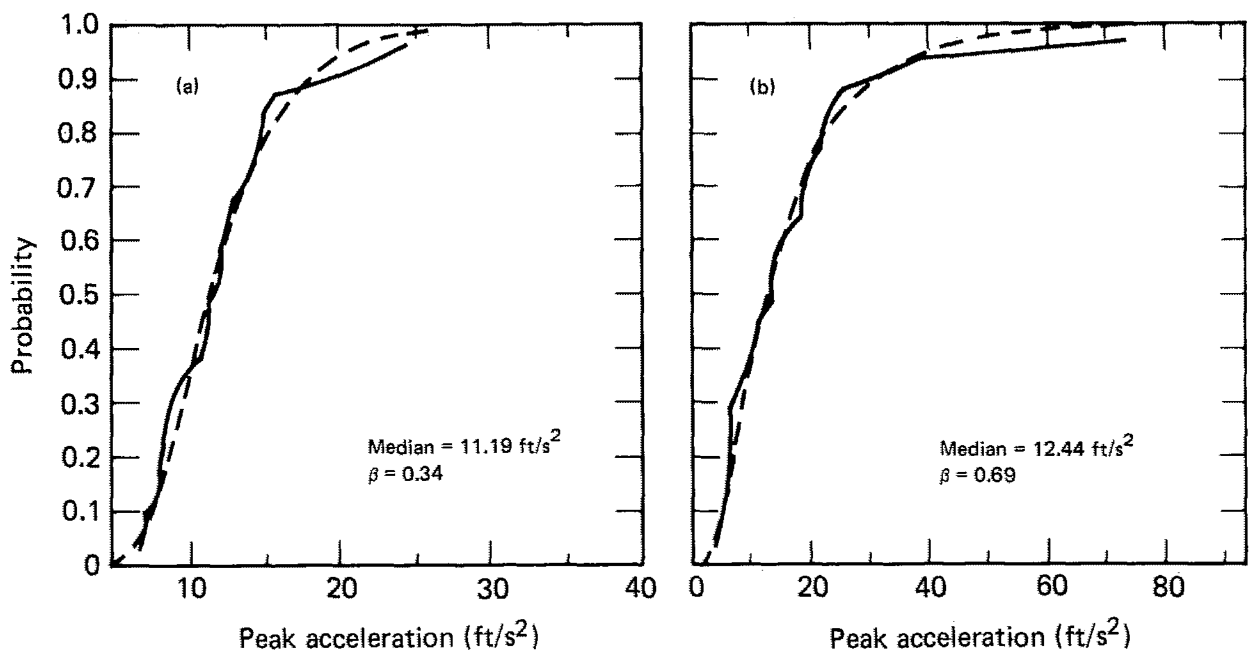


Fig. 23. Distributions of responses computed at the top of the containment shell (a) and at a check valve in the auxiliary feedwater system (b), using input parameters with reduced variability.

Table 4. Range of normalized input parameters used to compute the results of Figs. 22 and 23. Values are given for the median, the coefficient of variation (COV), and the tenth and ninetieth percentiles of lognormal distributions. Case 1 reflects a state of maximum uncertainty about the Zion plant; case 2 represents a state of significantly greater knowledge.

		Case 1			Case 2		
Parameter	Median	COV	10%	90%	COV	10%	90%
Soil shear modulus	1.0	0.7	0.45	2.24	0.35	0.65	1.55
Soil damping	1.0	1.00	0.34	2.91	0.5	0.55	1.90
Structure frequency	1.0	0.5	0.55	1.90	0.25	0.73	1.37
Structure damping	1.0	0.7	0.45	2.24	0.35	0.65	1.55
Subsystem frequency	1.0	0.5	0.55	1.90	0.25	0.73	1.37
Subsystem damping	1.0	0.7	0.45	2.24	0.35	0.65	1.55

more about the origin of the wide gaps between the median responses and the extremes (seen, for example, in Fig. 22). Our studies so far indicate that variations in no single parameter dominate the variations in the calculated response. The relationship among input parameters and input ground motion is complex.

We studied the adequacy of using 30 earthquakes to define the range of input ground motions by analyzing selected subsystems for 60 and 90 earthquakes. In general, the variability of the responses changed by only about 10%, which indicated that 30 earthquakes were adequate.

Finally, as part of Project VIII, we also introduced the concept of comparing a best-estimate (BE) seismic analysis method, exemplified by the SSMRP methodology, with an "evaluation method" (EM) or design methodology, such as that embodied in the NRC's *Standard Review Plan.*[11,44] In-structure response spectra derived using the BE method (relying on real three-dimensional time histories and statistical samples of stiffness and damping values) were compared with results from an EM computation that used synthetic time histories targeted at R.G. 1.60 design spectra[45] and broadened in-structure spectra. As shown in Fig. 24, the best-estimate method produced dramatically lower accelerations at all frequencies.

2.1.7 Project VI: Fragilities

In the SSMRP, a component or structure fails when it cannot perform its safety-related function. To predict failure, it was necessary to develop fragility descriptions (fragility curves) for all critical components and structures identified in the fault trees. In Phase I, all curves were developed in both normal and lognormal distributional forms, but examples are given here only for the lognormal forms. In Phase I, random and modeling uncertainties were not separated in the fragility curves used in SEISIM; however, attention was given to establishing this separation, and the results will be used in Phase II.

Fragility descriptions were based in part on available data and in part on carefully analyzed expert opinion. One of our important achievements was the beginning of a consistent data base for the characterization of fragilities. The data were drawn from sources such as U.S. Army Corps of Engineers test reports[46] and Zion design and qualification analysis results. Sargent & Lundy provided design specifications; design reports for the AFT complex, containment building, and crib house; and hanger and snubber catalogs. We subcontracted to EDAC two preliminary studies of potential structural and component failure modes at the Zion plant.[47,48] Finally, to establish fragilities for components when no data were found, we sent out 250 questionnaires to recognized experts in nuclear design and testing. The results of the survey were reviewed by a panel of consultants (see Table 1).[49]

Structural fragility curves were established by Structural Mechanics Associates (SMA) for the containment building, the AFT complex, the crib house, condensate storage tanks, and a section of buried pipe.[50] Building fragilities were based on the acceleration at which inelastic structural deformation would interfere with the operation of safety-related equipment. This failure acceleration was computed as the design acceleration times a factor of safety that accounts for original, conservative estimates of material strength and conservative design

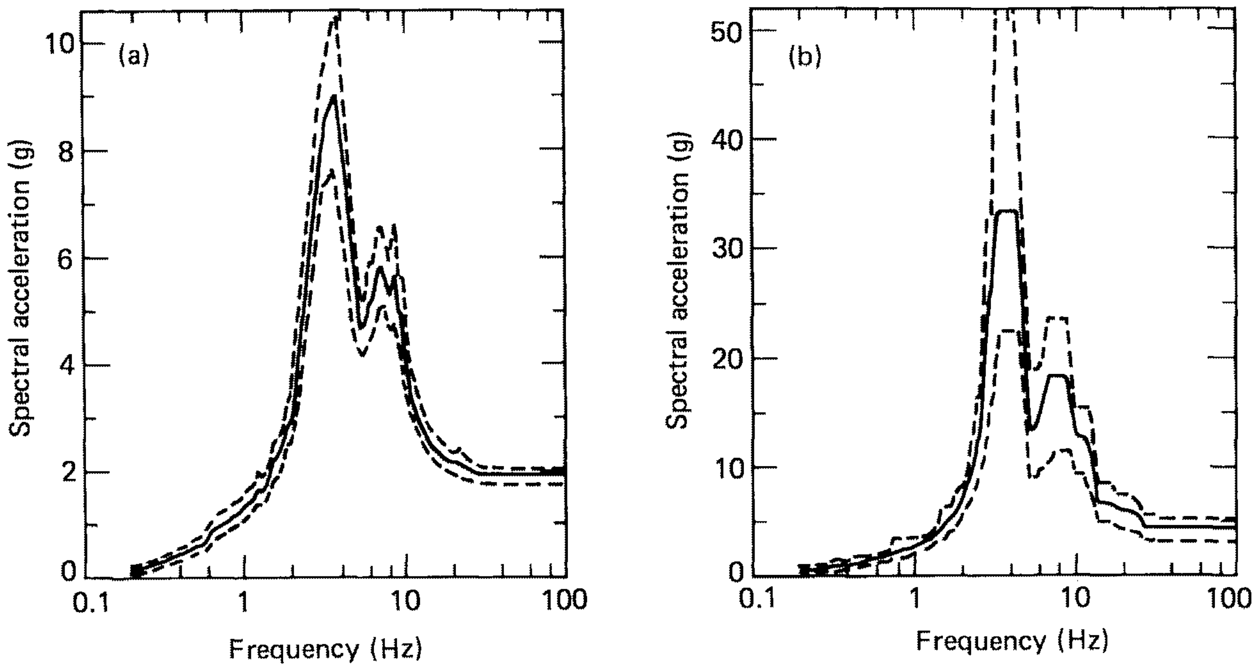


Fig. 24. Comparison of response spectra computed using the best-estimate method (a) and the evaluation method (b). A simple shear-beam model was used for the computations. The results for the best-estimate method show the mean and the 95% confidence limits; the evaluation method results show the maximum, mean, and minimum spectra.

analyses.[51] This factor of safety also includes a correlation for ductility that allows nonlinear failure criteria to be related to the linear responses calculated. Two typical fragility curves for structures are shown in Fig. 25.

Fragilities were derived for all electrical and mechanical components by grouping the equipment into the 37 generic categories listed in Table 5.[49,52] The data from the Corps of Engineers were especially useful here. As part of a missile site hardening program, the Army shock-tested about 300 items (representative of over 18,000 components) to failure. This information was combined with design analysis data and the results of our opinion survey, using a weighted least-squares procedure. We assigned weight factors to expert opinion according to the experience of the respondents, and we weighted data and analysis results more heavily than opinion. Finally, curves for different failure modes of the same component were combined to

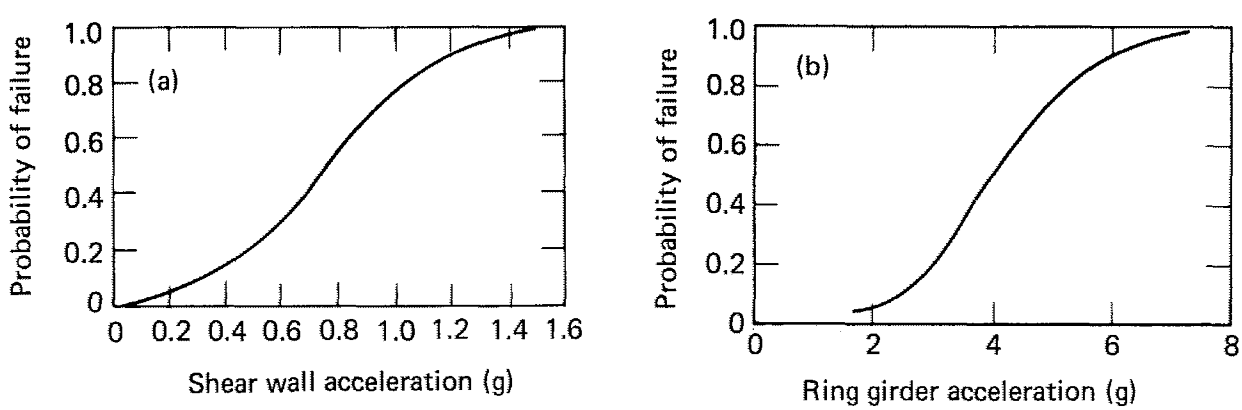


Fig. 25. Structure fragility curves for the auxiliary building shear walls (a) and shear failure of the containment building wall (b). The standard deviations of the logarithms of response are 0.345 and 0.34, respectively.

Table 5. Generic categories of mechanical, electrical, and miscellaneous components. One or more fragility curves were established for each.

Mechanical components	
Reactor core assembly	Small to medium vessels and heat exchangers
Reactor pressure vessel	Reactor coolant pump
Pressurizer	Large vertical centrifugal pumps with motor drive
Steam generator	Large vertical pumps
Piping	Motor-driven compressors
Large vertical storage vessels with formed heads	Large motor-operated valves
Large vertical storage tank, flat bottom	Large relief and check valves
Large horizontal vessel	Small valves
Electrical components	
Horizontal motors	Motor control centers
Generators	Light fixtures
Battery racks	Communications equipment
Switchgear	Invertors
Dry transformers	Cable trays
Control panels and racks	Circuit breakers
Auxiliary relay cabinets	Relays
Local instruments	Ceramic insulators
Instrument panels	
Miscellaneous components	
Air-handling units	Instrument racks
Duct work	Hydraulic snubbers

give a single effective fragility curve. Figure 26a shows the fragility curve for the reactor pressure vessel, and Fig. 26b includes curves for relay chatter and breaker trip. Relay chatter was a source of uncertainty in our system failure probabilities. A relay that chatters during an earthquake was assumed to regain its function afterwards (see §3.1.1).

Piping fragility curves were developed by SMA on the basis of analysis and static-collapse data.[49,52] Curves were developed for such elements as elbows, tees, reducers, and butt welds, and all were related by use of numerical conversion factors to a single master curve for a butt weld in 6-in. piping. This master curve, together with two other piping fragility curves, are shown in Fig. 27. Conversion factors also accounted for different temperatures and materials.

To relate these structure, component, and piping fragility curves to the 3000 basic seismic failure events that were considered during the construction

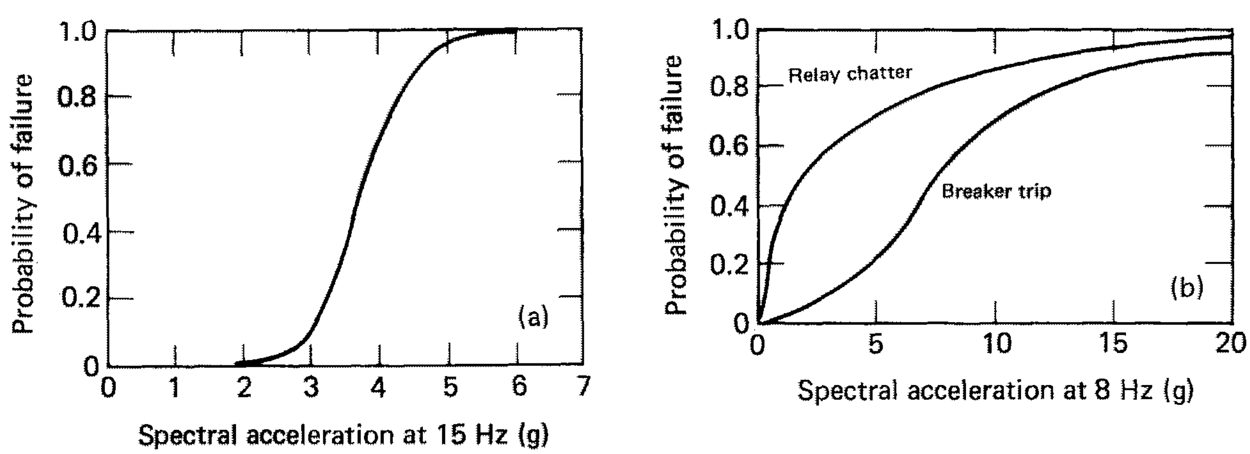


Fig. 26. Fragility curves for the reactor pressure vessel (a) and for relay chatter and breaker trip (b).

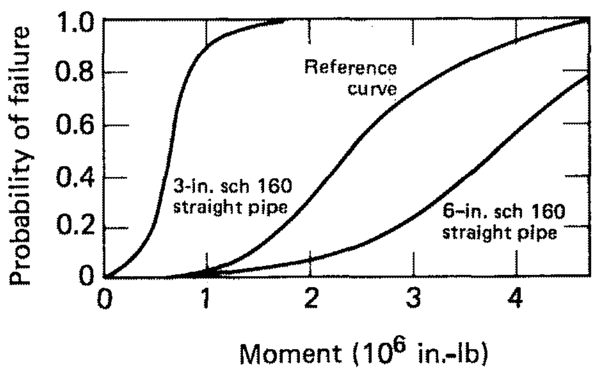


Fig. 27. Piping fragility curves, showing their relation to the master curve for a butt weld in 6-in. piping.

of fault trees for SEISIM, we developed a preprocessor code that correlates each basic event with the proper fragility curve, response parameter, and response location.

2.1.8 Project VII: Systems Analysis

The computer program SEISIM (Seismic Evaluation of Important Safety Improvement Measures) is one of the major computational tools of the SSMRP. It computes the probability of structural failure, component failure, system failure, and radioactive release by linking a local seismic hazard curve (generated by Project II), a set of fragility curves (Project VI), and the responses calculated by SMACS (Project VIII). The linkage, as shown schematically in Fig. 3, is achieved through eight event trees, which describe the scenarios that might follow an earthquake, and a set of fault trees, which define the mechanisms by which each system might fail. The essential features of SEISIM, therefore, are these event trees and fault trees, which must, in principle, model all possible seismic failure modes of a nuclear power plant.

Preliminary Studies. The development of the SEISIM code followed several preliminary studies. Reviews were undertaken of pertinent statistical techniques,[53] plausible sensitivity studies,[54] and systematic errors at the Zion plant.[55] We also solicited summaries of a computational approach from two independent subcontractors.[56,57] The final software design was a result of collaboration between LLNL and J. H. Wiggins Company, and its evolution has been amply documented.[58-62]

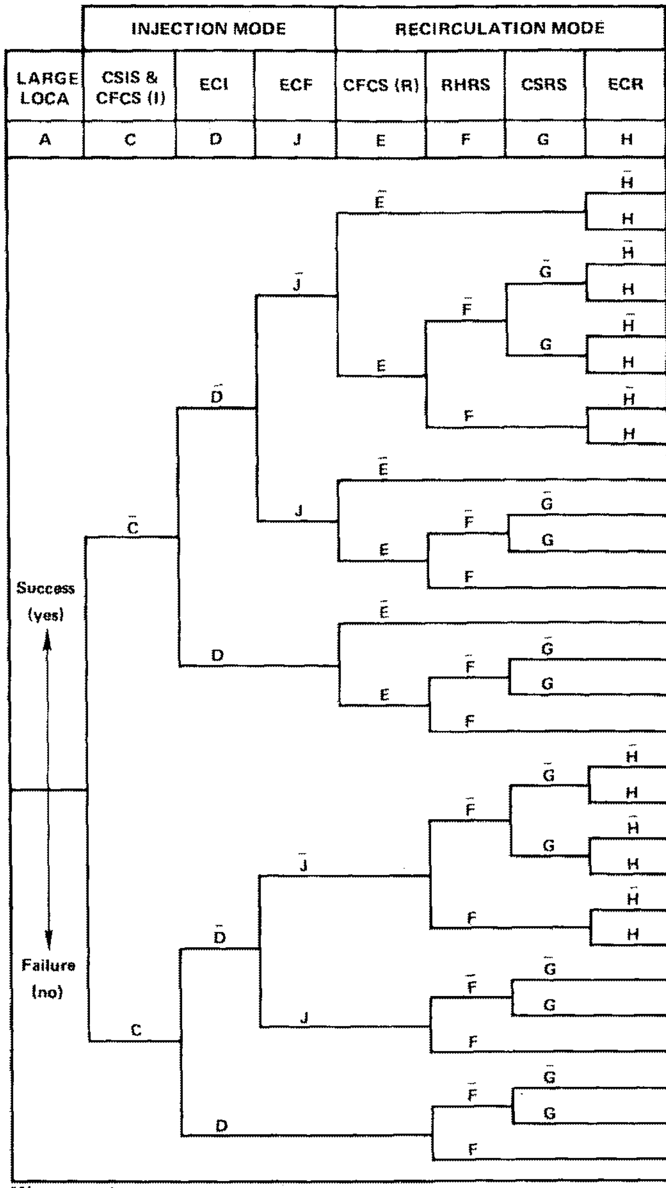
Event Trees. In Phase I, we identified seven initiating events that might be caused by an earthquake and that might lead to radioactive release. Each is associated with an event tree:[63]

- Reactor vessel rupture.
- Large LOCA (rupture of a pipe larger than 6 in. in diameter, or the equivalent).
- Medium LOCA (rupture of a pipe 3 to 6 in. in diameter, or the equivalent).
- Small LOCA (rupture of a pipe 1.5 to 3 in. in diameter, or the equivalent).
- "Small-small" LOCA (rupture of a pipe 0.5 to 1.5 in. in diameter, or the equivalent).
- Transient with power conversion system (PCS) operable.
- Transient with PCS inoperable.

We assumed that every earthquake (in the range of peak accelerations we considered) would initiate one of these event trees. This is equivalent to assuming that every earthquake in our range of interest would cause a reactor shutdown, either through a planned sequence of events or through an accident sequence. From these event trees, we identified 148 core melt accident sequences, each of which then feeds into an eighth event tree (containment failure) that establishes the mode by which radioactivity will be released. A core melt sequence, coupled with a containment failure mode, thus produces a radioactive release (a release sequence), which we categorized according to severity along the lines established by WASH-1400.[5] As an example, our event tree for a large LOCA, containing 23 core melt sequences, is shown in Fig. 28. Each of these sequences can lead to a release by way of two or more of the five potential containment failure modes, identified by the letters α, β, γ, δ, and ϵ.

Fault Trees. To evaluate the probability of system failure at each branch of an event tree, a fault tree is needed (see Fig. 6 for a simple example that includes only four basic events). For the event tree in Fig. 28, for example, fault trees are needed for the containment spray injection system and containment fan cooler system (CSIS & CFCS), the emergency coolant injection system (ECI), etc. In Phase I, we established fault trees for the Zion systems we estimated as most important:

- Auxiliary feedwater system (comprising 778 random and fragility-related basic events).
- Part of the service water system (172).
- Emergency core cooling system, including individual fault trees for the safety injection system (248), the residual heat removal system (298), the charging system (384), and the accumulators (58).



SEQUENCE	SEQ	CONTAINMENT FAILURE MODE	WASH-1400 EQUIVALENT
$A\ \bar{C}\ \bar{D}\ \bar{J}\ \bar{E}\ \bar{H}$*	1		
$A\ \bar{C}\ \bar{D}\ \bar{J}\ \bar{E}\ H$	2	$\alpha\beta\epsilon$	AH
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ \bar{F}\ \bar{G}\ \bar{H}$*	3		
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ \bar{F}\ \bar{G}\ H$	4	$\alpha\beta\epsilon$	AH
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ \bar{F}\ G\ \bar{H}$*	5		
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ \bar{F}\ G\ H$	6	$\alpha\beta\gamma\delta\epsilon$	AHF
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ F\ \bar{H}$	7	$\alpha\delta$	AG
$A\ \bar{C}\ \bar{D}\ \bar{J}\ E\ F\ H$	8	$\alpha\beta\delta\epsilon$	AHG
$A\ \bar{C}\ \bar{D}\ J\ \bar{E}$	9	$\alpha\beta\epsilon$	AE
$A\ \bar{C}\ \bar{D}\ J\ E\ \bar{F}\ \bar{G}$	10	$\alpha\beta\epsilon$	AE
$A\ \bar{C}\ \bar{D}\ J\ E\ \bar{F}\ G$	11	$\alpha\beta\delta\epsilon$	AEF
$A\ \bar{C}\ \bar{D}\ J\ E\ F$	12	$\alpha\beta\delta\epsilon$	AEG
$A\ \bar{C}\ D\ \bar{E}$	13	$\alpha\beta\epsilon$	AD
$A\ \bar{C}\ D\ E\ \bar{F}\ \bar{G}$	14	$\alpha\beta\epsilon$	AD
$A\ \bar{C}\ D\ E\ \bar{F}\ G$	15	$\alpha\beta\epsilon$	AF
$A\ \bar{C}\ D\ E\ F$	16	$\alpha\beta\delta\epsilon$	ADG
$A\ C\ \bar{D}\ \bar{J}\ \bar{F}\ \bar{G}\ \bar{H}$*	17		
$A\ C\ \bar{D}\ \bar{J}\ \bar{F}\ \bar{G}\ H$	18	$\alpha\beta\epsilon$	ACH
$A\ C\ \bar{D}\ \bar{J}\ \bar{F}\ G\ \bar{H}$*	19		
$A\ C\ \bar{D}\ \bar{J}\ \bar{F}\ G\ H$	20	$\alpha\beta\epsilon$	ACH
$A\ C\ \bar{D}\ \bar{J}\ F\ \bar{H}$	21	$\alpha\delta$	ACG
$A\ C\ \bar{D}\ \bar{J}\ F\ H$	22	$\alpha\beta\delta\epsilon$	ACHG
$A\ C\ \bar{D}\ J\ \bar{F}\ \bar{G}$	23	$\alpha\beta\epsilon$	ACE
$A\ C\ \bar{D}\ J\ \bar{F}\ G$	24	$\alpha\beta\epsilon$	ACE
$A\ C\ \bar{D}\ J\ F$	25	$\alpha\beta\delta\epsilon$	ACEG
$A\ C\ D\ \bar{F}\ \bar{G}$	26	$\alpha\beta\epsilon$	ACD
$A\ C\ D\ \bar{F}\ G$	27	$\alpha\beta\epsilon$	ACD
$A\ C\ D\ F$	28	$\alpha\beta\epsilon\delta$	ACDG

*No core melt

Fig. 28. Event tree for a large LOCA. Each subheading under "Injection mode" and "Recirculation mode" indicates a system, the successful operation or failure of which determines the path of an event sequence. The sequences that lead to core melt then produce radioactive releases by one of the indicated containment failure modes (α, β, γ, δ, or ϵ).

- Electric power system (158).

The failure probabilities of the remaining systems were based on other seismic risk studies (for example, the Diablo Canyon study)[10] and expert judgment on the similarities and differences between, say, Diablo Canyon and Zion.

Each fault tree was represented in SEISIM by a union of cut sets, each of which is a minimum set of basic events necessary to cause system failure. In the simple example of Fig. 6, for example, the fault tree for system A can be represented by the cut sets *a, b,* and *cd.* That is, system A fails if component a, or component b, or components c and d fail.

The probabilities of the basic events in a fault tree, hence the probability of system failure, depend on component failure probabilities, each of which we characterized as random or fragility related.

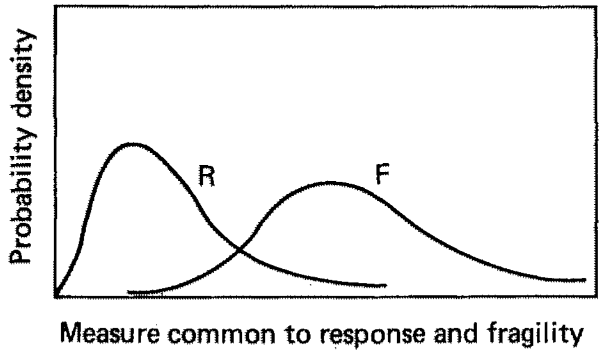


Fig. 29. Schematic representation of the method used for calculating fragility-related component failure probabilities. Failure is assumed to occur when a random value of the response R exceeds a random value of the fragility F.

Random failure probabilities were taken as independent of seismic activity, whereas fragility-related failure probabilities were computed as shown in Fig. 29. The probability density function for R is a result of the SMACS computation; the function for F is the derivative of a fragility curve. Failure occurs when a random value of R exceeds a random value of F.

Accounting for the common-cause nature of earthquakes is one of the features of the SEISIM methodology. In the computation of system failure probabilities and sequence probabilities, the failure probabilities of the fragility-related basic events represented in the fault trees were correlated in recognition of the fact that an earthquake affects all components simultaneously: Failure events are, in general, not independent.

With all the foregoing information, SEISIM computes the unconditional probability of a single release sequence (for a given earthquake acceleration range) as the product of four probabilities: (1) the probability of an earthquake producing the given input ground motion, taken from the seismic hazard curve, (2) the probability of the necessary initiating event, given the input motion, (3) the probability of the accident sequence, given the initiating event, and (4) the probability of the given containment failure mode, given the accident sequence. The probability of release in one of the seven WASH-1400 release categories, still for the given earthquake acceleration range, is simply the sum of the probabilities for all release sequences applicable to that category. Finally, the unconditional probability of release for a given release category, for the entire range of earthquake accelerations, is obtained by integrating over all accelerations.

Results. The principal aim of Phase I was to demonstrate the methodology outlined above. Our numerical results are tentative and are based on a number of simplifying assumptions:

- Some initiating event probabilities were only estimated.
- Not all accident sequences were explicitly included in the final computation of release probability (about two-thirds of the 148 were included).
- Not all fault trees were constructed; some system failure probabilities were estimated.
- Human error probabilities and release categories were, in general, based on WASH-1400.[5]
- Effect of design errors was not included.

Nonetheless, we succeeded in establishing the workability of the SSMRP computational procedure.

Throughout the seismic methodology chain of the SSMRP, computations were carried out in one of six "boxes," each corresponding to a range of peak accelerations (see Fig. 11). All calculated responses and probabilities are therefore associated with one of these boxes. Table 6 shows the acceleration range for each of the boxes, together with the annual probabilities of an earthquake producing such accelerations at the Zion site. Table 7 then gives the probabilities for each of the seven initiating events, given the specified input motions.

The various system failure probabilities calculated by SEISIM, again assuming the specified range of input motion, are shown in Table 8. Finally, Table 9 shows the calculated unconditional probabilities of release for the seven release categories. In each category, the unconditional probabilities for the six boxes have been summed.

Table 6. Earthquake occurrence probabilities at the Zion site, based on the hazard curve established by Project II (Fig. 11).

Box designation	Acceleration range (g)	Annual probability
a_1	0.15–0.30	2.52×10^{-4}
a_2	0.30–0.45	4.55×10^{-5}
a_3	0.45–0.60	6.57×10^{-7}
a_4	0.60–0.75	1.61×10^{-7}
a_5	0.75–0.98	5.31×10^{-8}
a_6	above 0.98	4.10×10^{-8}

Table 7. Conditional annual probabilities for the seven initiating events[a] identified for Zion.

Box designation	RPV	Lg LOCA	Md LOCA	Sm LOCA	SS LOCA	T_1	T_2
a_1	0	6.25E-4	6.25E-4	6.26E-4	6.26E-4	4.04E-1	5.93E-1
a_2	0	1.26E-2	1.28E-2	1.31E-2	1.36E-2	1.89E-2	9.29E-1
a_3	0	5.18E-2	5.46E-2	6.09E-2	7.23E-2	0	7.60E-1
a_4	1.42E-14	6.97E-2	7.49E-2	8.70E-2	1.11E-1	0	6.57E-1
a_5	2.37E-11	1.16E-1	1.31E-1	1.71E-1	2.69E-1	0	2.92E-1
a_6	7.43E-6	2.48E-1	3.30E-1	4.22E-1	0	0	0

[a]The seven events are: rupture of the reactor pressure vessel (RPV), large LOCA, medium LOCA, small LOCA, small-small LOCA, transient with PCS operable (T_1), and transient with PCS inoperable (T_2). Probabilities for all but the RPV rupture, the large LOCA, and T_2 were estimated.

Table 8. Conditional annual system failure probabilities computed by SEISIM; other system failure probabilities were estimated on the basis of other studies and engineering judgment.[a]

Box designation	AFWS	SWS	SIS[b]	CHG[b]	RHR[b]	ACC
a_1	8.4E-3	6.7E-5	4.7E-2	1.4E-1	4.8E-1	3.8E-4
a_2	1.4E-1	1.8E-3	1.2E-1	5.8E-1	9.0E-1	4.2E-3
a_3	3.0E-1	1.4E-2	2.6E-1	8.2E-1	9.6E-1	2.1E-2
a_4	5.9E-1	1.2E-1	5.0E-1	9.8E-1	9.8E-1	1.3E-2
a_5	8.6E-1	4.9E-1	7.5E-1	~1.0	~1.0	2.6E-1
a_6	~1.0	9.3E-1	9.9E-1	~1.0	~1.0	9.9E-1

[a]Abbreviations: AFWS, auxiliary feedwater system; SWS, service water system; SIS, safety injection system; CHG, charging system; RHR, residual heat removal system; ACC, accumulators.
[b]Injection phase; loss of one of two pumps causes failure.

Table 9. Calculated annual release probabilities for the seven release categories established by WASH-1400.[5]

Release category	Probability of release[a]
1	2×10^{-8}
2	1×10^{-8}
3	2×10^{-7}
4	1×10^{-10}
5	2×10^{-6}
6	2×10^{-8}
7	1×10^{-5}

[a]Based on the most probable accident sequences. Approximately two-thirds of the 148 sequences were accounted for.

2.2 Administrative Accomplishments

The principal accomplishments of the SSMRP have been technological. However, smooth administration has been an essential means to the technical end. Creation of a large, diverse research program in line with the needs of the NRC's Office of Nuclear Regulatory Research and Office of Nuclear Reactor Regulation necessarily demanded much thought about administrative as well as technical details. In addition, we at first encountered appreciable skepticism in the general nuclear power community about the feasibility of such a complex program. Overcoming this skepticism was again an administrative achievement as much as a technical one.

Our administrative efforts can be divided for convenience into three categories: planning and budgeting, coordination and balancing of technical disciplines, and communication with the outside technical community. Planning and budgeting was a matter of establishing at the outset realizable goals and a realistic approach. We established a level of effort that has produced a reasonable rate of progress, yet one that has been consistent with the available resources. Periodic review and adjustment was a necessary part of such a large, multidisciplinary program, but when it became necessary to realign our emphasis, we did so in light of the most important goal of Phase I: to produce and demonstrate a seismic risk assessment methodology that is workable and technically sound. As part of our planning efforts during Phase I, a speculative five-year plan for the SSMRP was written, consisting of research topic descriptions and proposed costs and schedules.[64]

Putting together a cohesive team from a broad range of professional disciplines, including civil, mechanical, and nuclear engineering, seismology, and statistics, was perhaps the central administrative achievement of the SSMRP. Because of the scope of the program, the team comprised not only LLNL staff members from these diverse disciplines, but also the many subcontractors whose efforts were discussed in the foregoing paragraphs. A balanced perspective on risk demanded a balance among the several disciplines represented, both on the in-house staff and among the contractors. The coordination of technical disciplines extended to the coordination among the eight projects that constituted Phase I of the SSMRP. In the early stages, a significant challenge was simply establishing a common technical language and pulling together the ways the program was seen by the different disciplines.

A daughter program was generated from the SSMRP in March 1979. The Load Combination Program[65] (LCP) is being carried out at LLNL under the management of C. K. Chou and is sponsored by the NRC, Office of Nuclear Regulatory Research. A diverse team was again formed, and the program plan and approach have been established. The objectives of the LCP are

- To develop a methodology for appropriately combining dynamic loads for nuclear power plants under normal plant operation, transients, accidents, and natural hazards. The methodology is to be based on the probabilistic assessment of the reliability of components, systems, and structures.
- To establish design criteria, load factors, and component service levels for appropriate combinations of dynamic loads or responses to be used in nuclear power plant design.
- To determine the reliability of typical piping systems, both inside and outside the containment structure, and to provide the NRC with a sound technical basis for defining the criteria for postulating pipe breaks. (We expect that NRC Regulatory Guide 1.46, *Protection Against Pipe Whip Inside Containment,* will be revised in accordance with the findings.)
- To determine probabilities of a large LOCA induced directly and indirectly by a range of earthquakes.

Communication with the technical community largely took the form of consultation with (and review by) the expert panels and review groups mentioned in §1.4.1. These groups, which included members from the NRC, universities, and the nuclear industry, helped us tackle many specific problems (especially those involving subjective issues, such as establishing fragility descriptions and a realistic ground motion model) and strengthened our confidence in our models and computational results. More important, these consultants and reviewers served as overseers of the whole program and provided a forum by which confidence in the SSMRP could be spread beyond its participants. Our efforts at communication also took the form of a broad overview report on risk analysis as applied to nuclear power plants,[12] written from the vantage point provided by our experience with the SSMRP. We were also a participant in an effort with the NRC to exchange technical information pertinent to reactor safety with Japan, France, West Germany, and the United Kingdom.

SECTION 3: CONCLUDING REMARKS

3.1 A Summary of Phase I Uncertainties

One of the features of the SSMRP methodology is its explicit accounting of uncertainties. And one of the aims of the program is to uncover the greatest contributors to the total uncertainty in the final computed probability of release. It is therefore appropriate that we look carefully at the sources of uncertainty in our Phase I calculations. Most were mentioned as part of the discussions of technical products and results in §2.1, but they are summarized and discussed here. This discussion will provide (1) an overview of the sources of uncertainty in Phase I, (2) a basis for planning the sensitivity studies to be carried out in Phase II, and (3) some basis on which the NRC might estimate the uncertainties in their licensing procedures.

The SSMRP probabilistic computational procedure, as shown in Fig. 3, consists of two main parts: response computations and computations of the probabilities of failure and release. This division was established to provide the necessary flexibility for later phases of the program and to produce the greatest number of practical spinoffs and tools for NRC use. Accordingly, our discussion of uncertainties will follow this division. We begin by looking at the uncertainties in the inputs to SEISIM, one of which is the response results of SMACS. We then turn to the several sources of uncertainty in the response results. When possible, we tentatively rank the uncertainties, basing our ranking on a combination of available data and our engineering judgment.

3.1.1 Uncertainties in SEISIM Inputs

The computation of release probabilities by SEISIM requires five inputs: the responses calculated by SMACS, a set of fragility curves, a set of event trees, a family of fault trees, and seismic occurrence data. Uncertainties in each of these inputs contribute to the uncertainty in our final results, but a definitive ranking according to importance is not yet possible. As mentioned in §2.1.2, the bounds on the uncertainty in the seismic hazard curve were not established in Phase I. However, our current state of development reveals that the uncertainty is considerable.

As discussed in §2.1.8, fault trees were constructed for seven Zion systems. Failure probabilities for the rest were based on other seismic risk studies (including the Diable Canyon study)[10] and our engineering judgment. These estimated failure probabilities contribute to the uncertainty in our final results. Fault trees that may require development in Phase II include the containment spray system, the containment fan cooler system, the component cooling system, and the instrumentation and control system.

Uncertainties associated with event trees arise from two sources. First, the initiating-event probabilities were calculated for only three of the seven initiating events. Probabilities for the other four were estimated (see Table 7). Second, in the computation of release probabilities, not all of the 148 accident sequences were included. About two-thirds were accounted for, and Boolean expressions for the rest were not included in Phase I. Our best estimates of the probabilities of release (Table 9) would inevitably increase if all accident sequences were included.

Fragility descriptions were based on test results and analysis where available, but for most components we relied partly or solely on the results of our expert opinion survey. Therefore, uncertainty in the fragility curves, reflected in the size of β (the standard deviation of the logarithm of response for lognormal distributions), varied considerably. Significant sources of fragility-related uncertainty lay in evaluating the fragilities of the following:

- Electrical components: relays and breakers.
- Local instrumentation, including sensors and associated electronics.
- Diesel generator accessories: fuel system, lubrication system, etc.
- Piping.
- Valves, generally. Much of the uncertainty arises from the decision to assign all valves to one of only three categories.
- Spring-operated safety relief valve.

A unique problem arose in predicting the behavior of relays. Seismic excitation may be accompanied by relay chatter (see Fig. 26b) that only *temporarily interrupts function; these components* may regain their function after the earthquake. It is not clear how best to treat this phenomenon in computing system failure probabilities. For Phase I computations, we assumed that the relays functioned after the earthquake.

3.1.2 Uncertainties in Response and in SMACS Input

The computer program SMACS calculated in-structure response in terms of acceleration, and piping response in terms of both acceleration (used to evaluate failure probabilities of components mounted on piping systems) and moment (used to evaluate the failure probabilities of the piping systems themselves). The greatest uncertainty was found in the piping moments, followed by piping accelerations and structure accelerations. For each of these responses, some uncertainty is introduced by each of the first three links in the seismic methodology chain: specifying seismic input, accounting for soil-structure interaction, and computing structure response. Calculating subsystem (piping) response introduces further uncertainty. The sensitivity study described in §2.1.5 is currently being carried out to determine the relative contributions of these four sources of uncertainty.

Seismic Input. Within each range of peak ground accelerations (Table 6), our SMACS computations were based on 30 randomly selected time histories. The use of 60 or 90 time histories did not significantly change our results (see §2.1.6); hence, we concluded that 30 time histories adequately reflect the distribution of simulated ground motions within a given range of peak free-field accelerations. Thus, our uncertainty regarding the origin of a specified acceleration (nearby, small earthquake or distant, large earthquake) is explicitly accounted for in the SMACS output. However, the set of simulated time histories from which the 30 are selected were generated on the basis of an assumed ground motion model and an assumed occurrence model (which, however, included variable parameters). Uncertainties in these assumptions lead to uncertainties in the distribution of time histories for a given range of peak ground accelerations, as well as uncertainties in the hazard curve. These uncertainties have not been explicitly accounted for.

Soil-Structure Interaction. Soil shear modulus and soil material damping were varied randomly in the SMACS calculations to reflect our uncertainty both in these two soil properties and in a number of other factors associated with soil-structure interaction, including approximations in foundation modeling and choice of analysis technique. Our current judgment is that uncertainties in soil properties are more important than modeling and analysis approximations, and that all are overwhelmed by uncertainties in specifying the free-field ground motion.

Structure Response. In the SMACS computations of major structure response, random variations in structure damping and frequency accounted for uncertainties in modeling techniques as well. The variation possible among results based on different modeling procedures was illustrated in Fig. 19. We now believe that the choice of damping values is less important than modeling variability. The contribution of uncertainty in the structure response computation to the overall uncertainty, as measured by the increase in β from the distribution of foundation response to the distribution of in-structure response, can vary significantly, depending on the location for which in-structure response is being computed.

Subsystem Response. Again, we chose variations in modal damping and frequency to reflect not only uncertainties in these two parameters but also uncertainties introduced by modeling. The most important modeling assumption, and the greatest contributor to the uncertainty introduced during the computation of piping response, concerns the stiffness and failure of piping supports.

3.2 Applications to Licensing

The possible applications of SSMRP results to licensing are broad. Ultimately, we plan to provide results and practical tools that help focus safety assessments on the most important issues, that direct research and development efforts for the biggest payoffs, that aid in the evaluation of hardening options for a given plant, and that rank the relative hardnesses of different plants (with a goal of establishing for all plants an acceptable level of risk). As LLNL continues to coordinate its efforts with NRC licensing needs, other applications of these tools can be identified. Here, we focus on possible applications of the products already generated in Phase I. These products include

- Computer codes: SMACS, SEISIM, and HAZARD.
- Analytic models and modeling techniques for seismic input, response computations, and probabilistic failure and release computations.
- Characterizations of uncertainties in the technical areas of the SSMRP.

- Characterizations of parameters, fragilities, damping, frequencies, and seismic hazard curve.
- Collections of available data and expert opinion for seismic input and fragilities.
- State-of-the-art reviews for soil-structure interaction, major structure response, and subsystem response.
- Side studies.

Among the most important of these products are the computer programs SMACS, SEISIM, and HAZARD. SMACS not only reflects the state of the art in accounting for soil-structure interaction and in computing in-structure and subsystem responses, but it is also reasonably economical and adaptable to other computer systems. Its role in the licensing process could be to compute responses of structures or systems already in place as part of reanalysis programs, or to aid in the evaluation of proposed designs and alternative "retrofits." The current research-oriented version of SMACS can be available to the NRC this year. However, we recommend waiting for a forthcoming, more user-oriented version. We also expect SEISIM to become a licensing tool, but transfer to the NRC must await further refinements to the code, as well as the additional efforts necessary to modify this research tool for practical use. HAZARD is currently being used by the NRC, Office of Nuclear Reactor Regulation, in their Systematic Evaluation Program to determine the seismic hazard at specific sites.[8]

Our first opportunity to apply the SSMRP computational procedure (using SMACS and a portion of SEISIM) to a licensing issue will be an analysis of the auxiliary feedwater system on Unit 1 of the San Onofre Nuclear Generating Station (SONGS), located in southern California. The NRC's Office of Nuclear Reactor Regulation is currently evaluating non–seismically qualified auxiliary feedwater systems of several operating PWRs. Our analysis of the SONGS system, to be carried out as part of our Phase II, will be a part of this evaluation.[66]

Among the analytic models developed in Phase I, the ground motion and occurrence models that were used to characterize seismic input are already finding practical use in NRC licensing and review programs. These models are being used, for example, in the NRC's Systematic Evaluation Program, whose purpose is the reevaluation of older nuclear power plants in light of current criteria.

Aside from these practical tools, the most significant product of the SSMRP is likely to be the insights provided into the magnitudes and sources of uncertainty. These insights, conveyed in §3.1.1, can be applied immediately as input to the NRC to assess important safety issues.

Finally, many of the generic models, data compilations, and technical reviews can be used immediately by the NRC. The most important of these products include generic fragility curves, the seismic input data compilation, the soil-structure interaction review,[13] the major structure response reviews,[37,38] and the event trees.

Looking beyond Phase I, we see a steady growth in the applicability of SSMRP results to licensing. Most importantly, our objective will be to identify, among the systems and components of a nuclear power plant, the most significant contributors to risk (radioactive release). From these results, an assessment of the current *Standard Review Plan*[11] will be possible.

3.3 Future Efforts

A major emphasis in the early part of Phase II will be the completion of sensitivity studies started in Phase I. We have identified four types of such studies:

I. Evaluation of effects on the results due to modeling changes.

II. Estimation of the probability of release and the associated uncertainties.

III. Ranking of contributors to the uncertainty associated with the probability of release.

IV. Ranking of contributors to the probability of release.

Results from these sensitivity studies will identify

- Most promising research areas.
- Areas in the computational procedure where simplifications can be made without significant loss of accuracy.
- Areas where refinements are warranted to improve the quality of the results.
- Areas where nonlinear techniques can be avoided.
- Possible courses of action for the NRC to improve its licensing procedures and requirements.

Validation will also receive major emphasis in Phase II. SSMRP results will be compared with results produced by existing, proven methods and with available experimental data. Depending on the useful information available, comparisons may focus on specific portions of the computational procedure, such as major structure response, or on the results of the overall procedure, such as probabilities of release. Since experimental data are generally in short supply, we may wish to begin a very modest experimental investigation, say, on a component found by our sensitivity studies to be highly important but for which no other source of data is available. In general, however, we intend to use available experimental data from domestic and foreign sources.

Finally, based on our Phase I experience, we have identified several other tasks needing further attention in the seven technical areas of the SSMRP. These are summarized in Table 10. Whether these tasks are pursued, and the priorities assigned to them, depend on our judgments of their relative importance to the overall program, as indicated by sensitivity studies; their appropriateness to the state of development of the program; and the magnitude and depth of the required effort.

Table 10. Areas needing further attention, as suggested by our Phase I experience.

Seismic Input

- Incorporation of results from earthquake modeling studies.
- Inclusion of additional data into data base for the ground motion model.
- Study of alternative methods for generating free-field response spectra and time histories, including ARMA (auto-regression/moving average) models for simulating time histories.
- Assessment of uncertainties introduced by occurrence and ground motion models.

Soil-Structure Interaction

- Sensitivity studies aimed at understanding the effects of (1) soil property variations, (2) basemat flexibility in the AFT complex, (3) structure-to-structure interaction, and (4) local nonlinear behavior.
- Study of the effects of local site conditions on seismic input.
- Identification, investigation, and comparison of characteristics of soil-structure interaction analysis techniques; definition of their applicability and limitations.

Major Structure Response

- Continued study of uncertainty in structural dynamic behavior: additional consideration of modeling sensitivity (in theory, as applied by engineers, etc.) and eigensystem sensitivity.
- Study of nonlinear behavior of structures and equivalent linear simplifications.
- Review and assessment of structural damping information currently available.

Subsystem Response

- Further Zion subsystem modeling, based on sensitivity study results.
- Investigation of the effects of the assumption of piping support rigidity.

SMACS and BE-EM

- Implementation of machine-independent version of SMACS; emphasis on practical, user-oriented aspects.
- Assessment of the appropriateness of input parameters to incorporate uncertainty in response calculations.

Fragilities

- Implementation of results for establishing upper and lower bounds for the fragility curves.
- Expansion of the data base and benchmarking of the fragility curves.

Systems Analysis

- Implementation of additional fault trees.
- Improvement of the definition of initiating events and computation of all initiating-event probabilities.
- Revision of event and fault trees to reflect current conditions at Zion.
- Culling of fault trees and accident sequences on the basis of cut-set probabilities.
- Improvements to SEISIM code.

REFERENCES

1. W. W. Hays, *Procedures for Estimating Earthquake Ground Motions,* U.S. Geol. Survey Prof. Paper 1114, 1980.
2. O. W. Nuttli, "The Mississippi Valley Earthquakes of 1811 and 1812: Intensities, Ground Motion, and Magnitudes," *Bull. Seismol. Soc. Am.,* **63:** 227-248, 1973.
3. L. M. Murphy, *San Fernando, California, Earthquake of February 9, 1971,* National Oceanic and Atmospheric Administration, U.S. Dept. of Commerce, Washington, D.C., 1973.
4. H. A. Levin, J. A. Martore, L. Reiter, D. Jeng, and L. W. Heller, *Reconnaissance Reports—Imperial Valley Earthquake—October 15, 1979,* U.S. Nuclear Regulatory Commission, Washington, D.C., memorandum for D. G. Eisenhut, November 2, 1979.
5. U.S. Nuclear Regulatory Commission, *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,* Washington, D.C., WASH-1400 (NUREG 75/014), October 1975.*
6. D. L. Anderson, R. G. Charlwood, and C. B. Chapman, *Seismic Risk Analysis Procedures for Canadian Nuclear Power Plants,* Acres Consulting Services Limited, 1975.
7. T.-M. Hsieh and D. Okrent, *Some Probabilistic Aspects of the Seismic Risk of Nuclear Reactors,* School of Engineering and Applied Science, University of California, Los Angeles, Calif., UCLA-ENG-76113, 1976.
8. D. W. Coats, *Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria,* U.S. Nuclear Regulatory Commission, Washington, D.C., NUREG/CR-1161, May 1980.**
9. C. A. Cornell and N. M. Newmark, "On the Seismic Reliability of Nuclear Power Plants," paper delivered at the topical meeting "Probabilistic Reactor Safety" of the American Nuclear Society, Newport Beach, Calif., May 8–10, 1978.
10. Pacific Gas and Electric Company, *Seismic Evaluation for Postulated 7.5M Hosgri Earthquake, Units 1 and 2, Diablo Canyon Site, Vol. V: Analysis of the Risk to the Public from Possible Damage to the Diablo Canyon Nuclear Power Station for Seismic Events,* U.S. Nuclear Regulatory Commission, Washington, D.C., docket Nos. 50-275 and 50-323, 1977.
11. U.S. Nuclear Regulatory Commission, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,* Office of Nuclear Reactor Regulation, Washington, D.C., NUREG-75-087, 1975. ***
12. P. D. Smith, D. L. Bernreuter, M. P. Bohn, T. Y. Chuang, G. E. Cummings, R. G. Dong, J. J. Johnson, R. W. Mensing, and J. E. Wells, *An Overview of Seismic Risk Analysis for Nuclear Power Plants,* Lawrence Livermore National Laboratory, Livermore, Calif., UCID-18680, September 1980.
13. J. J. Johnson, *Soil-Structure Interaction: The Status of Current Analysis Methods and Research,* Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-53011, October 1980 (published as U.S. Nuclear Regulatory Commission NUREG/CR-1780). **
14. G. L. Goudreau, *Preliminary Users' Manual for SAPPAC,* Methods Development Group, Mechanical Engineering Department, Lawrence Livermore National Laboratory, Livermore, Calif., draft report, August 1980.
15. R. L. Sharpe, J. W. Reed, and C. B. Waugh, *Plant/Site Selection Assessment Report,* Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15110, July 1979 (prepared by Engineering Decision Analysis Company, Inc.; to be published as U.S. Nuclear Regulatory Commission NUREG/CR-1705).**
16. Southwest Research Institute, *Nuclear Plant Reliability Data System: Special Report of Zion 1 Engineering and Failure Data,* San Antonio, Texas, October 1976.
17. TERA Corporation, *Seismic Hazard Analysis: A Methodology for the Eastern United States,* U.S. Nuclear Regulatory Commission, Washington, D.C., NUREG/CR-1582, Vol. 2, 1980.**
18. TERA Corporation, *Seismic Hazard Analysis: Solicitation of Expert Opinion,* U.S. Nuclear Regulatory Commission, Washington, D.C., NUREG/CR-1582, Vol. 3, 1980.**
19. R. J. Apsel, G. A. Frazier, A. Jurkevics, and J. C. Fried, *Application of Earthquake Source Modeling to Assess the Relative Differences between Seismic Ground Motion in the Eastern and Western Regions of the*

United States and to Characterize the Type and Direction of Incoming Seismic Waves: Final Report, Del Mar Technical Associates, Del Mar, Calif., DELTA-R-80-043, September 1980.

20. D. H. Chung and D. L. Bernreuter, *The Effect of Regional Variation of Seismic Wave Attenuation on the Strong Ground Motion from Earthquakes*, to be published as Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-53004 (also to be published as U.S. Nuclear Regulatory Commission NUREG/CR-1655).
21. D. H. Chung and D. L. Bernreuter, *Regional Relationships among Earthquake Magnitude Scales*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-52745, May 1980 (also published as U.S. Nuclear Regulatory Commission NUREG/CR-1457). **
22. C. B. Crouse, J. A. Hileman, B. E. Turner, and G. R. Martin, *Compilation, Assessment, and Expansion of the Strong Earthquake Ground Motion Data Base*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15227, April 1980 (prepared by FUGRO, Inc.; also published as U.S. Nuclear Regulatory Commission NUREG/CR-1660).**
23. O. W. Nuttli, *The Relation of Sustained Maximum Ground Acceleration and Velocity to Earthquake Intensity and Magnitude*, Report 16 of *State-of-the-Art for Assessing Earthquake Hazards in the United States*, U.S. Army Engin. Waterways Exp. Sta., Vicksburg, Miss., Misc. Paper S-73-1, November 1979.
24. O. W. Nuttli and R. B. Herrmann, *Credible Earthquakes for the Central United States*, Report 12 of *State-of-the-Art for Assessing Earthquake Hazards in the United States*, U.S. Army Engin. Waterways Exp. Sta., Vicksburg, Miss., Misc. Paper S-73-1, December 1978.
25. D. Gasparini and E. H. Vanmarke, *Simulated Earthquake Motions Compatible with Prescribed Response Spectra*, Report 2 of *Evaluation of Seismic Safety of Buildings*, Department of Civil Engineering, Massachusetts Institute of Technology, Cambridge, Mass., Publ. No. R76-4, January 1976.
26. J. E. Luco, *Linear Soil-Structure Interaction*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15272, July 1980 (included in Ref. 13).
27. J. M. Roesset, *A Review of Soil-Structure Interaction*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15262, June 1980 (included in Ref. 13).
28. H. B. Seed and J. Lysmer, *The Seismic Soil-Structure Interaction Problem for Nuclear Facilities*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15254, April 1980 (included in Ref. 13).
29. J. T. Christian, J. R. Hall, and E. Kausel, *Soil-Structure Interaction*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15230, February 1980 (prepared by Stone and Webster Engineering Corporation; included in Ref. 13).
30. J. Isenberg, D. K. Vaughan, G. L. Wojcik, F. S. Wong, and J. P. Wright, *A Review of Structure-Medium Interaction Analysis Methods with Applications to SSMRP*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15255, May 1980 (prepared by Weidlinger Associates; included in Ref. 13).
31. H. L. Wong and J. E. Luco, *Soil Structure Interaction: A Linear Continuum Mechanics Approach (CLASSI)*, Report CE, Dept. of Civil Engineering, University of Southern California, Los Angeles, Calif., 1980.
32. H. L. Wong and J. E. Luco, *Identification of Sensitive Parameters for Statistical Simulation of Soil-Structure Interaction*, to be published by Lawrence Livermore National Laboratory, Livermore, Calif.
33. O. R. Maslenikov, J. C. Chen, and J. J. Johnson, *Uncertainty in Soil-Structure Interaction Analysis of a Nuclear Power Plant—A Comparison of Two Analysis Techniques*, to be published at Lawrence Livermore National Laboratory, Livermore, Calif.
34. B. J. Benda and J. J. Johnson, *Variability in Dynamic Characteristics and Seismic Response Due to the Mathematical Modeling of Nuclear Power Plant Structures*, to be published as U.S. Nuclear Regulatory Commission NUREG/CR-1897.
35. D. A. Wesley, P. S. Hashimoto, and R. B. Narver, *Variability of Dynamic Characteristics of Nuclear Power Plant Structures*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15267, July 1980 (prepared by Structural Mechanics Associates; also published as U.S. Nuclear Regulatory Commission NUREG/CR-1661).**

36. T. K. Hasselman and S. S. Simonian, *Structural Uncertainty in Seismic Risk Analysis*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15218, March 1980 (prepared by J. H. Wiggins Company; also published as U.S. Nuclear Regulatory Commission NUREG/CR-1560).**
37. J. J. Healey, S. T. Wu, and M. Murga, *Structural Building Response Review*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15185, February 1980 (prepared by Ebasco Services Incorporated; also published as U.S. Nuclear Regulatory Commission NUREG/CR-1423, Vol. 1).**
38. A. K. Singh, T. I. Hsu, and T. P. Khatua, *Structural Building Response Review*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15183, January 1980 (prepared by Sargent & Lundy Engineers; also published as U.S. Nuclear Regulatory Commission NUREG/CR-1423, Vol. 2).**
39. J. Fogelquist, M. K. Kaul, R. Koppe, S. W. Taggart, Jr., H. Thailer, and R. Uffer, *Subsystem Response Review*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15215, March 1980 (prepared by Nuclear Services Corporation; also published as U.S. Nuclear Regulatory Commission NUREG/CR-1700).**
40. R. P. Kennedy, R. D. Campbell, D. A. Wesley, H. Kamil, A. Gantayat, and R. Vasudevan, *Subsystem Response Review*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15216, March 1980 (prepared by Engineering Decision Analysis Company, Inc.; to be published as U.S. Nuclear Regulatory Commission NUREG/CR-1706).
41. Sargent & Lundy Engineers, *Nuclear Safety Related Reactor Coolant Loop Model: Zion Station*, Chicago, Ill., Report SL-3806, September 1980.
42. American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code. Section III. Rules for Construction of Nuclear Power Plant Components*, New York, N.Y., July 1980.
43. R. L. Iman, W. J. Conover, and J. E. Campbell, *Risk Methodology for Geologic Disposal of Radioactive Waste: Small-Sample Sensitivity Analysis Techniques for Computer Models, with an Application to Risk Assessment*, Sandia Laboratories, Albuquerque, N.M., SAND80-0020, March 1980 (also published as U.S. Nuclear Regulatory Commission NUREG/CR-1397).**
44. S. E. Bumpus, J. J. Johnson, and P. D. Smith, *Best Estimate Method vs Evaluation Method: A Comparison of Two Techniques in Evaluating Seismic Analysis and Design*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-52746, May 1980 (also published as U.S. Nuclear Regulatory Commission NUREG/CR-1489).**
45. U.S. Atomic Energy Commission, *Design Response Spectra for Seismic Design of Nuclear Power Plants*, Washington, D.C., Regulatory Guide 1.60, 1973.
46. U.S. Army Corps of Engineers, *Subsystem Hardness Assurance Analysis*, Huntsville, Ala., HNDDSP 73-161-ED-R, Vols. 1 and 2, June 1975.
47. R. D. Campbell and D. A. Wesley, *Preliminary Failure Mode Predictions for the SSMRP Reference Plant (Zion 1)*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15042, October 1979 (prepared by Engineering Decision Analysis Company, Inc.; also published as U.S. Nuclear Regulatory Commission NUREG/CR-1703).**
48. R. D. Campbell and D. A. Wesley, *Potential Seismic Structural Failure Modes Associated with the Zion Nuclear Plant*, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15140, October 1979 (prepared by Engineering Decision Analysis Company, Inc.; also published as U.S. Nuclear Regulatory Commission NUREG/CR-1704). **
49. M. P. Bohn, manuscript in preparation, Lawrence Livermore National Laboratory, Livermore, Calif.
50. D. A. Wesley and P. S. Hashimoto, *Seismic Structural Fragility Investigation for the Zion Nuclear Power Plant*, Structural Mechanics Associates, Inc., Newport Beach, Calif., 12205.05, September 1980.
51. R. P. Kennedy, C. A. Cornell, H. F. Perle, R. D. Campbell, and S. Kaplan, "Probabilistic Seismic Safety Study of an Existing Nuclear Power Plant," paper presented at 5th SMiRT Conference, Berlin, August 1979.
52. R. D. Campbell, R. P. Kennedy, G. Hardy, and H. Banon, *SSMRP Subsystem Fragility*, Structural Mechanics Associates, Inc., Newport Beach, Calif., 12205.06.01, January 1981.
53. L. L. George, *Probability Computational Methods in SSMRP*, to be published as Lawrence Livermore National Laboratory, Livermore, Calif., UCID-18686.

54. L. L. George, *Statistical Sensitivity Analysis Methods in SSMRP,* to be published at Lawrence Livermore National Laboratory, Livermore, Calif.
55. P. Moieni, G. Apostolakis, and G. E. Cummings, *Interim Report on Systematic Errors in Nuclear Power Plants,* Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-15274, September 1980 (also published as U.S. Nuclear Regulatory Commission NUREG/CR-1722). **
56. J. D. Collins, J. M. Hudson, and J. D. Chrostowski, *Specification of Computational Approach,* Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-13986, February 1979 (prepared by J. H. Wiggins Company; also published as U.S. Nuclear Regulatory Commission NUREG/CR-1701). **
57. I. B. Wall, M. K. Kaul, R. I. Post, S. W. Tagart, Jr., and T. J. Vinson, *Specification of Computational Approach,* Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-13985, February 1979 (prepared by Nuclear Services Corporation; also published as U.S. Nuclear Regulatory Commission NUREG/CR-1702).**
58. J. H. Hudson and J. D. Collins, *SEISIM Design Concept,* J. H. Wiggins Company, Redondo Beach, Calif., Technical Report 78-1645-2, June 1979.
59. J. H. Hudson and J. D. Collins, *SEISIM Code Development Specification,* J. H. Wiggins Company, Redondo Beach, Calif., Technical Report 79-1366, October 1979.
60. J. H. Hudson, J. D. Gasca, and J. D. Collins, *SEISIM Option 1 Design Specification,* J. H. Wiggins Company, Redondo Beach, Calif., Technical Report 80-1366-1, April 1980.
61. J. H. Hudson and J. D. Gasca, *SEISIM Option 1 User's Manual,* J. H. Wiggins Company, Redondo Beach, Calif., Technical Report 80-1366-2, January 1981.
62. L. L. George and J. E. Wells, "The Reliability of Systems of Dependent Components," paper to be delivered at ASQC Quality Congress, San Francisco, Calif., May 1981 (also published as Lawrence Livermore National Laboratory UCRL-84154).
63. A. A. Garcia and J. E. Kelly, *Event Tree Development and Construction,* Science Applications, Inc., Palo Alto, Calif., SAI-003-79-BE, August 1979.
64. R. G. Dong, *Seismic Safety Margins Research Program (Phase I) Long-Range Plan,* Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-53002, April 1980.
65. S. Lu, R. D. Streit, and C. K. Chou, *Large LOCA-Earthquake Combination Probability Assessment—Load Combination Program Project 1 Summary Report,* Lawrence Livermore National Laboratory, Livermore, Calif., UCID-18694, January 1981 (also published as U.S. Nuclear Regulatory Commission NUREG/CR-1889). **
66. T. Y. Chuang, P. D. Smith, R. G. Dong, D. L. Bernreuter, M. P. Bohn, G. E. Cummings, and J. E. Wells, *Seismic Safety Margins Research Program Project I: SONGS 1 AFWS Project,* Lawrence Livermore National Laboratory, Livermore, Calif., UCID-18959, February 1981.

*Available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

**Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and/or the National Technical Information Service, Springfield, VA 22161.

***Available for purchase from the National Technical Information Service.

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG/CR-2015, Vol. 1
UCRL-53021, Vol. 1

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)

Seismic Safety Margins Research Program
Phase I Final Report - Overview

5. DATE REPORT COMPLETED

MONTH	YEAR
March	1981

7. AUTHOR(S) P. D. Smith, R. G. Dong, D. L. Bernreuter, M. P. Bohn, T. Y. Chuang, G. E. Cummings, J. J. Johnson, R. W. Mensing, J. E. Wells

DATE REPORT ISSUED

MONTH	YEAR
April	1981

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Lawrence Livermore National Laboratoy
P.O. Box 808
Livermore, California 94550

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

U.S. Nuclear Regulatory Commission
Division of Reactor Safety Research
Office of Nuclear Regulatory Research
Washington, DC 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.
FINs A0126, A0130, A0138, A0139, A0142

13. TYPE OF REPORT

Technical

PERIOD COVERED (Inclusive dates)

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

The Seismic Safety Margins Research Program (SSMRP) is a multiyear, multiphase program whose overall objective is to develop improved methods for seismic safety assessments of nuclear power plants, using a probabilistic computational procedure. The program is being carried out at the Lawrence Livermore National Laboratory and is sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. Phase I of the SSMRP was successfully completed in January 1981: A probabilistic computational procedure for the seismic risk assessment of nuclear power plants has been developed and demonstrated. The methodology is implemented by three computer programs: HAZARD, which assesses the seismic hazard at a given site, SMACS, which computes in-structure and subsystem seismic responses, and SEISIM, which calculates system failure probabilities and radioactive release probabilities, given (1) the response results of SMACS, (2) a set of event trees, (3) a family of fault trees, (4) a set of structural and component fragility descriptions, and (5) a curve describing the local seismic hazard. The practicality of this methodology was demonstrated by computing preliminary release probabilities for Unit 1 of the Zion Nuclear Power Plant north of Chicago, Ill. Studies have begun aimed at quantifying the sources of uncertainty in these computations. Numerous side studies were undertaken to examine modeling alternatives, sources of error, and available analysis techniques. Extensive sets of data were amassed and evaluated as part of projects to establish seismic input parameters and to produce the fragility curves.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT

UNLIMITED

19. SECURITY CLASS (This report)
UNCLASSIFIED

20. SECURITY CLASS (This page)
UNCLASSIFIED

21. NO. OF PAGES

22. PRICE
S

NRC FORM 335 (7-77)

NUREG/CR-2015, Vol. 1

SEISMIC SAFETY MARGINS RESEARCH PROGRAM
PHASE I FINAL REPORT—OVERVIEW

APRIL 1981