Application of the SSMRP Methodology to the Seismic Probabilistic Risk Analysis at the Zion Nuclear Power Plant

M.P. Bohn, L.C. Shieh, J.E. Wells, L.L. George, R.W. Mensing, L.E. Cover, D.L. Bernreuter

University of California, Lawrence Livermore National Laboratory, P.O. Box 808, Livermore, California 94550, U.S.A.

J.J. Johnson

Structural Mechanics Associates, 2400 Old Crow Canyon Road, San Ramon, California 94583, U.S.A.

SUMMARY

The Seismic Safety Margins Research Program (SSMRP) is an U.S. NRC-funded multiyear program conducted by Lawrence Livermore National Laboratory (LLNL). Its goal is to develop a complete fully coupled analysis procedure (including methods and computer codes) for estimating the risk of an earthquake-induced radioactive release from a commercial nuclear power plant. The analysis procedure is based upon a state-of-the-art evaluation of the current seismic analysis and design process and explicitly accounts for uncertainties inherent in such a process.

The SSMRP is the first effort to trace seismically induced failure modes in a reactor system down to the individual component level, and to take into account common cause earthquake-induced failures at the component level. This paper presents the results of our seismic risk analysis of the Zion nuclear power plant using the SSMRP methodology.

The risk analysis included a detailed seismological evaluation of the region around Zion, Illinois which provided the earthquake hazard function and an appropriately randomized set of 180 time histories (having pga values up to 1.8 g). These time histories were used as input to dynamic structural response calculations for four separate Zion buildings. Detailed finite element models of the buildings were used. Calculated time histories at piping support points were then used to determine moments throughout critical piping systems. Twenty-one separate piping systems were analyzed. Finally, the responses of piping and safety system components within the buildings were combined with probabilistic failure criteria and event tree/fault tree models of the plant safety systems to produce an estimate of the probability of core melt and radioactive release due to the occurrence of earthquakes.

A number of important conclusions have been drawn concerning the relative importance of each element of the seismic methodology chain on seismic risk. For example, the random variations in earthquake time histories did not totally dominate the final uncertainty in response calculations. Uncertainty in underlying soil conditions was found to be roughly equal in importance to uncertainty in structure parameters. In addition, the importance of considering dependent (common-cause) earthquake-induced failures is demonstrated. Finally, a number of structural areas are identified which, because of their importance in the final risk numbers, require additional research.

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

1. Introduction

To assist the NRC in its licensing and evaluation role, the NRC funded the Seismic Safety Margins Research Program (SSMRP) at LLNL with the goal of developing tools and data bases to evaluate the risk of earthquake-caused radioactive release from a commercial nuclear power plant. This program began late in 1978, and the methodology was finalized in 1982. A complete seismic risk assessment for the Zion plant was finished in October 1982. This paper describes the SSMRP risk assessment methodology and the results of the Zion Analysis. For a complete description, see the 10 volume SSMRP Phase I report, Ref. 1.

1.1 Scope of the SSMRP

A nuclear power plant is designed to ensure the survival of all buildings and emergency safety systems in a worst-case ("safe shutdown") earthquake. The assumptions underlying this design process are deterministic. In practice, however, these assumptions are clouded by considerable uncertainty. It is not possible, for example, to accurately predict the worst earthquake that will occur at a given site. Soil properties, mechanical properties of buildings, and damping in buildings and internal structures also vary significantly among plants. To model and analyze the coupled phenomena that contribute to the total risk of radioactive release, it is therefore necessary to consider all significant sources of uncertainty as well as all significant interactions. Total risk is then obtained by considering the entire spectrum of possible earthquakes and integrating their calculated consequences.

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

1. Determine the local earthquake hazard.
2. Identify potential accident scenarios for the plant which lead to radioactive release.
3. Determine failure modes for the plant emergency safety systems.
4. Compute failure probabilities of the critical components in the emergency safety systems.
5. Compute probability of radioactive release using information from Steps 1 through 4.

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

Step 1 - Determine the Earthquake Hazard

The earthquake hazard at a given power plant site is characterized by a frequency plot which gives the probability of occurrence (per year) of earthquakes causing different peak ground accelerations. Figure 1 shows this so-called Hazard Curve for the Zion Nuclear Power Plant, located at Zion, Illinois, approximately 40 miles north of Chicago. This curve is derived from a combination of recorded earthquake data, estimated earthquake magnitudes of known events for which no data are available, review of local geological investigations, and use of expert opinion based on a survey of seismologists and geologists familiar with the region in question.

In addition to computing the seismic hazard curve, a number (usually 30) of random synthetic earthquakes are generated by using the data just discussed and a Monte Carlo procedure incorporated in our HAZARD code. These earthquake time histories provide the random ground motion uncertainty inherent in real earthquakes, and are used as input to the building response calculations described below. Each synthetic earthquake is described by three time histories in three orthogonal directions. A full discussion of this process is given in Refs. 1 and 2.

Step 2 - Identify Accident Sequences

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

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

For each of these initiating events, an event tree is constructed. Each branch of an event tree is an accident sequence. As an example, Fig. 2 shows the event tree for a transient (T2) in which the power conversion system is assumed to be initially inoperative (due to emergency shutdown or failures in the main steam turbines, for example). Fifteen accident sequences (branches) are shown on this tree. Accident Sequence 4 is highlighted. It consists of the initiating event (T2), successful operation of the RPS (denoted $\bar{K}$), failure of the AFWS (denoted L), successful operation of the pressurizer relief valves (denoted $\bar{P}$) and successful closing of the relief valves (denoted Q). Thus, this accident sequence is the set of five events (T2 $\bar{K}$ L $\bar{P}$ $\bar{Q}$). In computing the probability of core melt, we compute the probability of these five events occurring together. Similarly, the probability of all the other accident sequences is computed.

Step 3 - Determine Failure Modes of Safety Systems

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

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

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

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

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

(a) Response Calculations

The buildings, foundations, major components, and pioing systems are all modeled by the finite element method. SSI and structure response were calculated by the substructure approach. Pioing analysis was performed by multi-support time history analysis. The model generated for the Zion turbine/auxiliary building is shown in Fig. 3. Such models were developed for four buildings and five different piping systems in the Zion power plant analysis. For Zion, responses at over 400 points in the buildings and over 1000 points in the pioing systems were comouted for each input time history.

To incorporate the uncertainties, multiple analyses of the entire power plant are made. In each of these repeated calculations, the magnitudes of the input parameters are varied in a random fashion, and each calculation is performed using a different set of three input time histories. Typically, 30 calculations are made (at each earthquake level) with the result that 30 values of response are computed for each building wall, slab, pipe segment, valve and component. From these 30 values, a statistical distribution of the response of each wall, component, etc., can be constructed. Figure 4 shows a typical result for the moment in a pipe segment, plotted as a cumulative probability distribution. Each circle is the moment resulting from a single, deterministic calculation, and the set of 30 circles corresponds to the 30 repeated calculations. Such distribution functions were determined for the responses of every wall, slab, pipe segment, and electro-mechanical component identified on the fault trees.

(b) Determination of Fragility Functions

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

A data base of the necessary fragility functions was developed (Ref. 1). As a first step, all components identified on the fault trees were grouped into 37 generic categories. Fragility functions for each generic category were developed based on a combination of design analysis reports, experimental data and an extensive expert opinion survey. Statistical methods were used to combine data from several sources. Typical fragility curves are shown in Fig. 5.

Step 5 - Compute Probability of Core Melt and Radioactive Release

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

(a) Calculation of Cut Set Probabilities

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

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

(b) Calculations of Probability of Radioactive Release

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

Finally, accident sequence probabilities are assigned to different release categories to reflect their severity with respect to radioactive release to the surrounding population. These release categories relate to the type and energy content of the radioactive fission product release, as well as the mode and timing of the release. They range from rupture of the top of the containment with a rapid, high energetic release (due to a fuel/water explosion or due to steam overpressure) down to slow melt-through of the containment concrete foundation, which is expected to have the least effect on the surrounding population. The containment failure modes and the release categories are those derived and used in the Reactor Safety Study (Ref. 3).

2. Results of Zion Risk Analysis and Confidence Bounds

This section presents the results of the calculations made for the seismic risk analysis of the Zion nuclear power plant, and describes the confidence bounds on these results. The base case is our best estimate of the configuration of the Zion plant and its emergency systems and procedures. A number of important assumptions were made.

1. The identified structural failure modes were assumed to have the most serious hypothesized consequences. Two structural failure modes play significant roles.
   (i) The failure of the roof of the service water pump enclosure room (at top of the crib house) is assumed to fail all six service water pumps beneath it. This results in loss of the emergency AC power diesel generators, due to lack of cooling water.
   (ii) The failure of the wall between the turbine building and the auxiliary building is assumed to cause loss of all electrical wiring and control air conduits, so both power and control to the reactor building are lost.
2. Soil failure under the toe of the containment was assumed to result in sufficiently large rocking motions so as to fail the SIS, CHG and RHR piping between the AFT building and the reactor building.

These assumptions play crucial roles in the base case results.

2.1 Probability of Radioactive Release

The median frequency of radioactive release (total) was computed to be 2.0 x 10^{-5} per year. This value reflects inherent randomness in all the input variables and the hazard curve, as well as systematic uncertainties in all the input variables due to lack of exact knowledge of their mean values. The 10-90% confidence band on the release frequency was found to be about 3 orders of magnitude. The median values and confidence bounds were obtained by making repeated calculations of the release frequencies, while varying the median values of all input variables according to an experimental design. Fourteen repeated calculations were performed, and new sets of structural responses and hazard curves were used for each. Median values and confidence bounds were inferred from these fourteen runs for each of the seven WASH-1400 (Ref. 3) release categories, as seen in Fig. 6. These core melt frequencies are due primarily to the failure of certain structural elements which result in common-cause failures of the safety systems.

The release frequencies at earthquake levels 2, 3, and 4 are dominant both for probability of release and dose. The probabilities of both release and dose drop off at earthquake levels 1, 5 and 6. This indicates that the bulk of the risk is at the intermediate earthquake levels, and that the range of peak ground accelerations considered is adequate.

At the three lower earthquake levels, the initiating events are dominated by the transients T1 and T2. At earthquake level 4, it is primarily the small LOCAS which are important. At earthquake level 5 the initiating event probabilities are fairly evenly spread over the initiating events and the large LOCA and RPV initiating event become significant. Finally, at level 6 the dominant initiating events are the RPV and large LOCA events. The initiating component failure for the two transient event trees is primarily the loss of offsite power due to failure of the ceramic insulators at the point where offsite power is brought into the switchyard. The component failures which cause the large LOCA and the RPV initiating events are failures of the supports of the steam generators and reactor coolant pumps which are assumed to result in the failure of the primary coolant piping . Thus, it can be seen that it is not failure of the piping which results in a large LOCA but rather the failure of the supports of the major components.

Radioactive release in category 7 at earthquake levels 2, 3, and 4 dominates (over 80%) the total release probability. Examination of the transient event trees shows that four accident sequences are the major contributors. The four accident sequences are those in which the auxiliary feedwater system has failed to function due to the failure of the piping between the reactor building and the auxiliary building. The second major contributor is the failure of the service water pump enclosure roof. Since the auxiliary feedwater system depends on component cooling, the loss of the service water pumps was presumed to result in loss of function of the auxiliary feedwater system. The remaining important accident sequence at level 2 is a small LOCA accident sequence which also results from the loss of the auxiliary feedwater system. The two contributing components failures are again uplift of the containment and failure of the pump enclosure roof in the crib house.

At earthquake levels 3 and 4 the same four transient accident sequences are important. At these levels, however, we also see a number of small LOCA sequences contributing significantly, due again to the pump enclosure roof and uplift of the containment basemat. One large LOCA accident sequence is important and this is due to failure of a pipe in the RHR system between the containment and the auxiliary building. At earthquake level 4, there is one important small LOCA accident sequence which is due to the failure of pairs of pipes in the RHR system. Here again, these are pairs of interconnecting pipes between the auxiliary building and the reactor building, failing because of moment induced by differential building motion.

In summary, in computing the probability of radioactive release, the dominant contributors are the uplift of the containment basemat and failure of the service water system due to failure of the pump enclosure roof of this service water system and, to a smaller extent, there are contributions from failure of interconnecting pipes due to differential motion between the reactor building and the auxiliary building. If uplift occurs, but the interconnecting pipes are not damaged, and if the pump enclosure roof fails but the service water system still functions, then the risk decreases by a factor of five. Thus the assumptions as to the effects of the structural failures play an important role in determining the risk at the plant, and these assumptions need careful examination on a plant-by-plant basis.

References

[1] P. D. Smith, et al., Seismic Safety Margins Research Program Phase I Final Report, Volumes 1-10, NUREG/CR-2015, 1981, 1982.

[2] D. L. Bernreuter, et al., Seismic Hazard Analysis, Volumes 1-4, NUREG/CR-1582, 1980-1981.

[3] U.S. Nuclear Regulatory Commission, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, NUREG-75/014, October 1975.

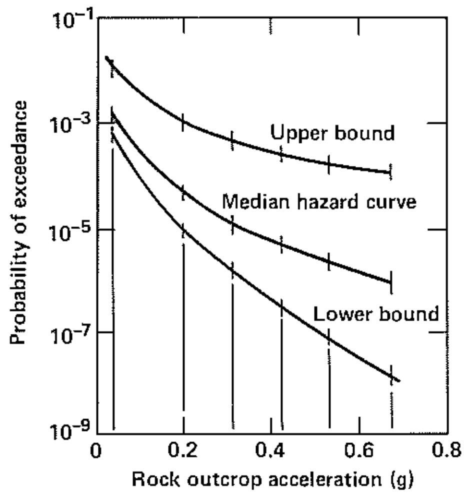

Fig. 1 Seismic hazard curve for the Zion site.

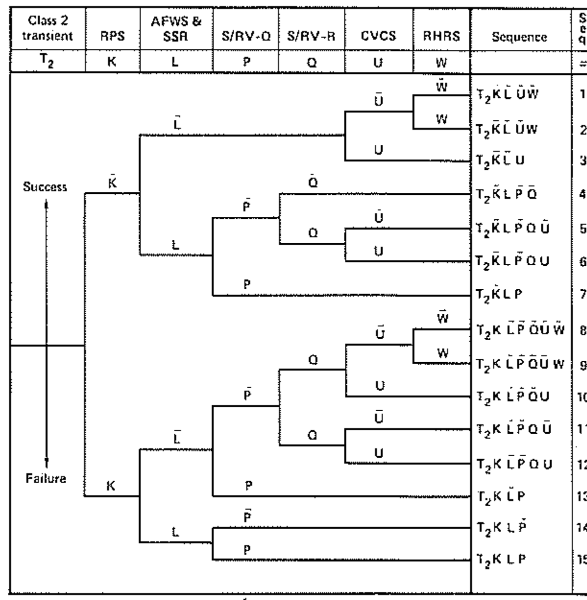

Fig. 2 Event tree for a transient in which the secondary coolant loop is assumed to be initially inoperative.

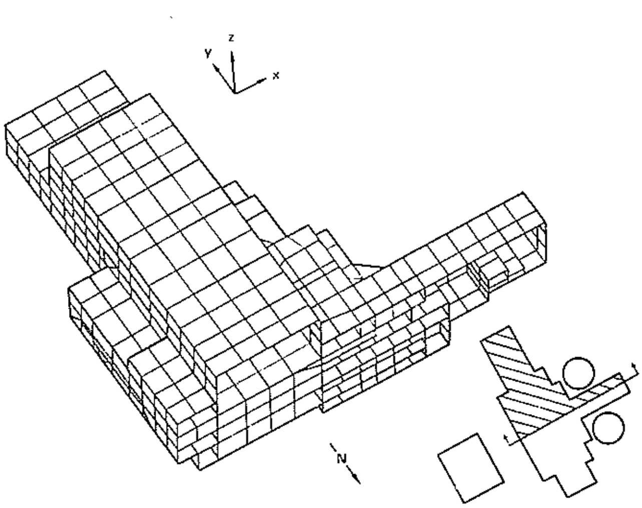

Fig. 3 Finite element model of half of the Zion turbine/auxiliary building.

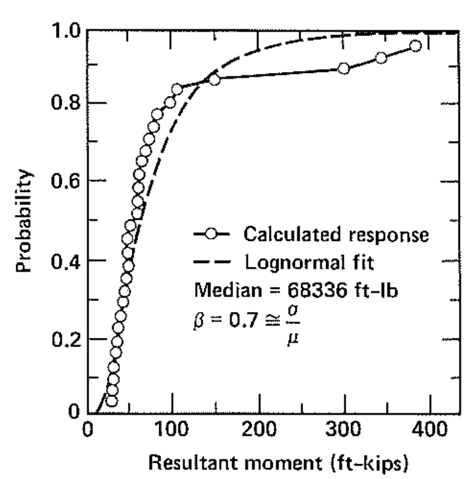

Fig. 4 Computed cumulative distribution function for the moment in a pipe segment.

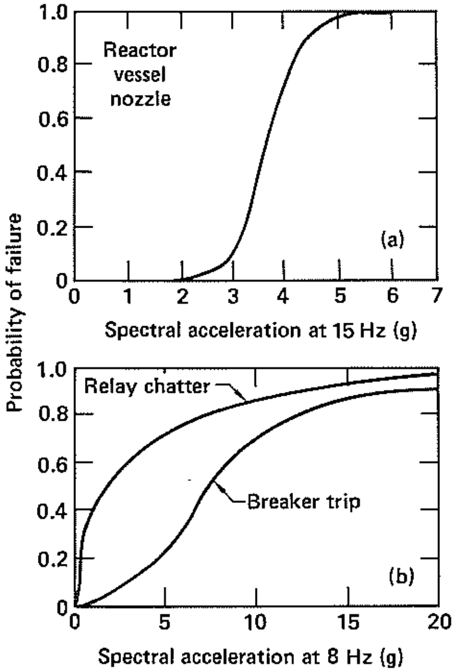

Fig. 5 Examples of fragility functions.

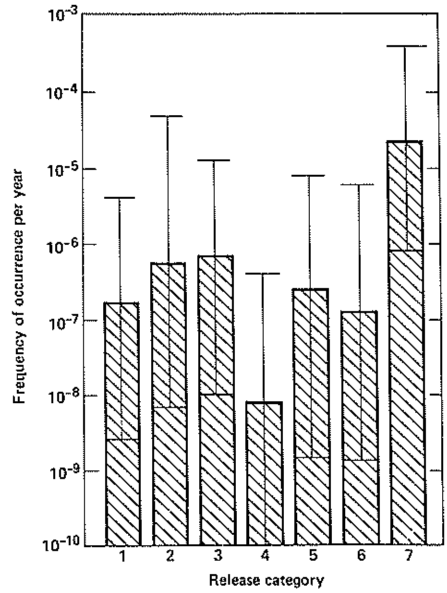

Fig. 6 Histogram of frequencies of release in the seven WASH-1400 release categories.