Seismic Safety Margins Research Program

An Assessment of Potential Increases in Risk Due to Degradation of Steam Generator and Reactor Coolant Pump Supports

Michael P. Bohn, James E. Wells, Long C. Shieh,
Lester E. Cover, Ronald L. Streit

Prepared for
U.S. Nuclear Regulatory Commission

DO NOT MICROFILM
THIS PAGE

Lawrence
Livermore
National
Laboratory

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

MASTER

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

NUREG/CR-3345
UCID-19719
RD/RM

NUREG/CR--3345

DE83 016943

Seismic Safety Margins Research Program

An Assessment of Potential Increases in Risk Due to Degradation of Steam-Generator and Reactor- Coolant-Pump Supports

Manuscript Completed: July, 1983
Date Published: August, 1983

Prepared by
Michael P. Bohn, James E. Wells, Long C. Shieh
Lester E. Cover, Ronald L. Streit

Lawrence Livermore National Laboratory
7000 East Avenue
Livermore, CA 94550

Prepared for
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN No. A-0439

NOTICE
PORTIONS OF THIS REPORT ARE ILLEGIBLE.
It has been reproduced from the best
available copy to permit the broadest
possible availability.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

Abstract

During the NRC licensing review for the North Anna Units 1 and 2 pressurized-water reactors (PWRs), questions were raised regarding the potential for low-fracture toughness of steam-generator and reactor-coolant-pump supports. Because other PWRs may face similar problems, this issue was incorporated into the NRC Program for Resolution of Generic Issues. The work described in this report was performed to provide the NRC with a quantitative evaluation of the value/impact implications of the various options of resolving the fracture-toughness question.

This report presents an assessment of the probabilistic risk associated with nil-ductility failures of steam-generator and reactor-coolant-pump structural-support systems during seismic events, performed using the Seismic Safety Margins Research Program codes and data bases.

Two cases were analyzed. In Case 1 the support failure was assumed to occur at the safe-shutdown earthquake (SSE), while in Case 2 the systems were assumed to fail at twice the SSE. The public dose assuming reduced capacity supports was found to be about 32 man-REM/year (Case 1) and 12 man-REM/year (Case 2), compared to the base case (normal support capacity) of 6.6 man-REM/year.

Although this analysis was performed using the seismic hazard curve, and logic and structural models specific to Zion, the resulting risk numbers are in no way to be construed as applying to the Zion Plant, because the fracture toughness of the Zion supports has been found to satisfactorily meet all design requirements.

CONTENTS

	<u>Page</u>
Abstract	iii
List of Figures	v
List of Tables	vi
Executive Summary	1
Section 1: Introduction and Background	5
Section 2: Overview of the SSMRP Methodology	8
2.1 Scope of the SSMRP	9
Section 3: Overview of the Fragilities Development	36
3.1 Data Sources for Components	37
3.2 Fragilities Used for Base Case Risk Assessment of Zion	41
3.3 Base Case Fragilities for the Steam Generator and Reactor Coolant Pumps	46
3.3.1 Fragilities of SG and RCP Used in Base Case Analysis	54
3.4 Degraded Capacity Fragilities for the Steam Generators and Reactor Coolant Pumps	54
3.4.1 Supporting Rationale for Degraded Support Fragility Values	55
Section 4: Summary of Degraded Support Seismic Risk Calculations	60
4.1 Summary of Results	60
4.2 Earthquake Level Probabilities	62
4.3 Probability of Initiating Events	64
4.4 Probability of Accident Sequences and Containment Failure Modes	69
Section 5: Estimation of Median Values and Confidence Limits	77
5.1 Uncertainty Analysis	77
5.2 SSMRP Uncertainty Analysis: Design & Methodology	80
5.2.1 Response Analysis Uncertainties	81
5.2.2 Fragilities Uncertainties	82
5.2.3 Hazard Curve Uncertainties	82
5.3 Experimental Design	82
5.4 Estimates of Confidence Bounds for the USI-A12 Results	84
5.5 Estimated Confidence Limits	88
Section 6: References	90
Appendix A	A-1

List of Figures

	<u>Page</u>
1. Hazard curve at the Zion site.	11
2. Schematic of the development of the hazard curve	12
3. Reactor vessel-rupture event tree	15
4. Large LOCA event tree	16
5. Median LOCA event tree	17
6. Small LOCA event tree	18
7. Small-small LOCA event tree	19
8. Transient (T2) event tree	20
9. Transient (T2) event tree	21
10. Finite element half-structure model of Zion auxiliary - fuel-handling - turbine building.	25
11. Variation in moment at RPV nozzle (loop 1 hot leg) with random uncertainties only	27
12. Schematic flow diagram of input and output for the SMACS code	30
13. Examples of fragility curves	32
14. Steam generator upper lateral support	47
15. Steam generator lower lateral support	48
16. Steam generator vertical support.	49
17. Reactor coolant pump lateral support	50
18. Reactor coolant pump vertical support	51
19. Experimental values of fracture toughness for A-36 (grade 50) and A-572 (grade 50) from Reference 16	57
20. Family of hazard curves for Zion site from which median curve and bounds were computed	65
21. Probability of core melt frequency (per year) based on inclusion of systematic uncertainty.	87

List of Tables

		<u>Page</u>
1.	SSMRP initiating events	14
2.	Input parameters with associated uncertainty specified as a coefficient of variation (COV)	28
3.	Generic categories for components	31
4.	Summary of component fragilities	42
5.	Estimated public dose in WASH-1400 release categories (man-REM) from NUREG/CR-2800	61
6.	Earthquake hazard at Zion	63
7.	Computed median (m) and log-standard deviation (β) of base spectral accelerations (ft/sec^2) for steam generators and reactor coolant pumps	67
8.	Conoitional probabilities of failure of steam generator and reactor coolant pump at each earthquake level	67
9.	Initiating event probabilities (conditional) at each earthquake level	68
10.	Definition of events used on the containment- event tree	70
11.	Probability of release categories per year for each earthquake level - 3 cases	71
12.	Dominant accident sequences - Case 1	73
13.	Dominant accident sequences - Case 2	74
14.	Dominant accident sequences - Base Case	75
15.	Probablity of release categories per year for weak support case - with and without building failure effects	86

Executive Summary

During the NRC licensing review for the North Anna Units 1 and 2 pressurized water reactors (PWRs), questions were raised regarding the potential for low fracture toughness of steam generator and reactor coolant pump supports. Because other PWRs use similar materials and designs and may face similar problems, the review of this issue was incorporated into the NRC Program for Resolution of Generic Issues as Generic Technical Activity A-12. The NRC staff formulated a plan (Task Action Plan A-12) to resolve these safety issues. The issues involved and proposed resolution were discussed in the draft NUREG-0577 (issued for comment). The work described in this report was performed to provide the NRC with a quantitative evaluation of the value/impact implications of the various options of resolving the fracture toughness question.

The scope of this work was to perform an assessment of the probabilistic risk, in terms of man-rem/reactor year, associated with nil-ductility failures of steam generator (SG) and reactor coolant pump (RCP) structural support systems during seismic events. The assessment was performed using the Seismic Safety Margins Research Program codes and data bases.

The analysis performed was necessarily specific to the Zion Plant (Unit 1) which is a late 1960's vintage plant having pipe column vertical supports for the steam generators and reactor coolant pumps. And, as pointed out in the draft revision of NUREG-0577, the fracture toughness of the Zion column supports has been found to satisfactorily meet all design requirements. Thus, although this analysis was performed for Zion, the resulting risk numbers are in no way to be construed as applying to the Zion Plant.

Two sets of calculations were performed to assess the change in risk due to two levels of reduced capacity of the steam generator and reactor coolant pump supports. The base case corresponds to the supports having as-designed capacities, with a median failure level of approximately four times the

safe-shutdown earthquake (SSE) level (0.17 g). Case 1 is the weakest case, where-in the support capacity was reduced to a median failure level equal to the SSE, while Case 2 is an intermediate case with the support capacities reduced to twice the SSE level. This reduction was per Office of Nuclear Reactor Regulation request, and Case 1 represents a case of severely inadequate support capacity. However, our preliminary examination into the support capacity at temperatures below the nil ductility temperature of A-572 steel shows that reductions in capacity of 3 to 5 times could be expected. Thus, the reduction is not unreasonable.

The main result of interest is that the public dose assuming reduced capacity supports was found to be about 32 man-REM/year (Case 1) and 12 man-REM/year (Case 2), compared to the base case (normal support capacity) of 6.6 man-REM/year. These values are based on the seismic hazard specific to the Zion site, and include the effects of all sizes of earthquakes thought to be possible at the site, but taking into account the decreasing probability of occurrence of larger earthquakes. The estimated public dose (given a core melt) in the seven release categories for a PWR was taken from NUREG/CR-2800. It is a simple matter to modify these results for different seismic hazard curves or different exposure values.

The increase in dose due to reduced capacity of the steam generator and reactor coolant pumps is due to the following. The reduced capacity results in a high probability of failure of the steam generators and/or reactor coolant pumps at relatively low earthquake levels, as shown below for Case 1:

Earthquake Level	Conditional Steam Generator Failure Probability Per Year	Conditional Reactor Coolant Pump Failure Probability Per Year
1	0.002	0.013
2	0.076	0.139
3	0.631	0.681
4	0.945	0.925
5	0.957	0.946
6	0.995	0.985

These values are considerably higher than in the base case of normal support capacity. The failure of the supports of either a steam generator or a reactor coolant pump is assumed to result in shear failure and rupture of the 34 inch OD primary coolant piping. If any one steam generator or one pump fails, then a large LOCA results. If failures occur in two loops (out of four) then an initiating event denoted "reactor vessel rupture" occurs. This is defined as a large LOCA caused by a break which prevents any effective ECCS action, and thus always results in a core melt. Multiple primary coolant piping failures (in different loops) give rise to such an event. It was found that this initiating event totally dominated the analysis of Case 1 and was the major contributor in Case 2.

The computed public dose per year in each of the seven release categories (as defined in WASH-1400) is shown in the table below.

Dose per year in the WASH-1400 release categories (man-REM/year).

<u>Release Category</u>	<u>Case 1 Weak Supports</u>	<u>Case 2 Int. Supports</u>	<u>Base Case</u>
1	0.7	0.7	0.1
2	9.8	9.6	2.7
3	15.1	1.0	3.6
4	0	0	0
5	0	0	0
6	0	0	0
7	6.4	0.4	0.2
Total	<hr/> 32.0	<hr/> 11.8	<hr/> 6.6

It can be seen that the dose in Case 1 (the weakest support case) is dominated by release categories 2 and 3 (containment failure due to overpressure) and category 7 (melt-through of basemat).

It should be noted that brittle failure of the support columns could only occur during those times when the reactor supports were at low temperatures relative to the nil-ductility temperature (NDT) of the support material. Since temperatures inside containment are normally over 100°F, and the NDT of A-36 and A-572 steels is usually below 75°F, this could only occur during some small percentage of the time. This could reduce the risk of the reduced capacity support case significantly.

Section 1: Introduction and Background

During the NRC licensing review for the North Anna Units 1 and 2 pressurized water reactors (PWRs), questions were raised regarding the potential for low fracture toughness of steam generator and reactor coolant pump supports. Because other PWRs use similar materials and designs and may face similar problems, the review of this issue was incorporated into the NRC Program for Resolution of Generic Issues as Generic Technical Activity A-12. The NRC staff formulated a plan (Task Action Plan A-12) to resolve these safety issues. The issues involved and proposed resolution were discussed in the draft NUREG-0577 (issued for comment). The work described in this report was performed to provide the NRC with a quantitative evaluation of the value/impact implications of the various options of resolving the fracture toughness question.

The scope of this work was to perform an assessment of the probabilistic risk, in terms of man-rem/reactor year, associated with nil-ductility failures of steam generator (SG) and reactor coolant pump (RCP) structural support systems during seismic events. The assessment was performed using the Seismic Safety Margins Research Program codes and data bases, and was made for the Zion PWR nuclear power plant.

The Seismic Safety Margins Research Program (SSMRP) is an 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-caused 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 includes the uncertainties inherent in such a process.

The SSMRP was begun in 1978 when it became evident that an accurate seismic risk analysis must simultaneously consider all the interrelated factors that affect the final probability of radioactive release. These closely coupled factors are:

- The likelihood and magnitude of an earthquake.
- The transfer of earthquake energy from a fault source to a power plant, a phenomenon that varies greatly with the magnitude of an earthquake.
- Interaction between the soil underlying the power plant and the structural response, a phenomenon that depends on the soil composition under the plant and the location of the fault source relative to the plant.
- Coupled responses of a power plant's buildings and the massive reactor vessels, piping systems, and emergency safety systems within.
- Numerous accident scenarios, which vary according to the types of failures assumed and the success or failure of the engineered safety features intended to mitigate the consequences of an accident.

All these factors, and the uncertainties associated with them, were considered in the analyses reported herein.

This report presents the results of two sets of calculations, in which the capacity of the steam generator and reactor coolant pumps have been reduced to 18% and 36% of their designed capacity values. These results are compared with the results of a base case SSMRP analysis of the seismic risk at Zion. The changes in support capacities are the only differences from the SSMRP base case. Thus differences in the resulting risk numbers are due entirely to differences in support capacity, and a meaningful comparison of changes in risk level can be made.

This report is structured as follows. Section 2 presents an overview of the SSMRP and the calculational process used to generate the risk numbers. Section 3 describes the fragility relations used for the steam generator and reactor coolant pumps, and describes how the reduced capacities were obtained. Section 4 presents the results of the calculations. Section 5 describes the confidence bounds on these results, and the basis for these bounds.

The analysis we performed was necessarily specific to the Zion plant. Thus the results presented are based on the seismic hazard curve specific to the Zion site. The reader should be cautioned against using these results for a plant at a different site without investigating the applicability of the Zion hazard curve to that site.

The Zion Plant (Unit 1) is a late 1960's vintage plant having pipe column vertical supports for the steam generators and reactor coolant pumps. And, as pointed out in the draft revision of NUREG-0577, the fracture toughness of the Zion column supports has been found to satisfactorily meet all design requirements. Thus, although this analysis was performed using the structural and logic models developed for Zion, the resulting risk numbers are in no way to be construed as applying to the Zion plant.

Section 2: Overview of the SSMRP Methodology

To assist the NRC in its seismic licensing and evaluation role, the NRC funded the Seismic Safety Margins Research Program at LLNL with the goal of developing tools and data bases to evaluate the risk of earthquake-caused damage and subsequent radioactive release from a commercial nuclear power plant. This program began late in 1978, and the first phase of the program--to assemble the methodology and demonstrate it for a typical reactor plant--was completed early in 1981. The methodology was finalized in 1982 and a complete seismic risk assessment for the Zion plant was presented at the 10th Water Reactor Safety Research Information Meeting in October 1982. This section describes some of the unique aspects of earthquake design and the risk assessment methodology developed in the SSMRP. For a complete description see the 10 volume SSMRP Phase I report, (Ref. 1).

Earthquakes present a unique challenge in the design and evaluation of a nuclear power plant. A somewhat surprising aspect of the threat of earthquakes is the fact that strengthening one part of a nuclear power plant may actually increase the risk of radioactive release. For example, one could greatly strengthen the walls of the containment building until there was virtually no possibility of damaging it during any conceivable earthquake. By strengthening these walls, however, they are made much stiffer. The ground shaking under the plant is transmitted up through the walls with little attenuation or modification. Important safety equipment mounted on or tied to the walls on the inside of the containment would then see a higher level of excitation than if the walls of the containment were more flexible and allowed to undergo some inelastic deformation (at high earthquake levels) with attendant energy absorption. Thus, the overall risk of radioactive release could be increased by strengthening part of the system. This point underscores the crucial feature of the SSMRP: the nuclear power plant is examined in its entirety, as a system.

2.1 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.

For a given site, 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. The process of deriving this curve is illustrated in Fig. 2. The region around the site (say within 100 km) is divided into zones, each zone having a uniform mean rate of earthquake occurrence. This mean occurrence rate is determined from the historical record along with the distribution of earthquake magnitudes. Then, for the region under consideration, an attenuation law is determined, which relates the ground acceleration at the site to the ground acceleration at the earthquake source, as a function of the earthquake magnitude. The uncertainty in the attenuation law is specified by the standard deviation of the data (from which the law was derived) about the mean attenuation curve. These three pieces of information (zonation, mean occurrence rate and magnitude distribution for each zone, and attenuation law) are then combined statistically to compute the hazard curve as shown in Fig. 1. A full discussion of this process is given in Refs. 1 through 3.

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.

Fig. 1. Hazard curve at the Zion site.

Fig. 2. Schematic of the development of the hazard curve.

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. Should the earthquake damage essential components in the safety systems, it is possible that the effectiveness of the safety systems would be diminished. In an extreme case, it is possible that damage to the fuel rods or core melt could result. In this step of the risk analysis process, 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 the Zion power plant, 315 accident sequences were identified.

All the identified accident sequences result from one or more seismically induced initiating events (events requiring immediate shutdown of the plant). In the SSMRP analysis of the Zion plant, we identified 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. This event is really a LOCA, but it is assumed that the break occurs in the reactor vessel wall or in a nozzle, and hence the ECCS cannot effectively flood the core. Thus, in this case, the only mitigating safety systems are those in the containment (CSIS and CFCS). These seven initiating events are shown in Table 1.

For each of these seven initiating events, an event tree is constructed. These seven event trees are shown in Figs. 3 through 9. Each branch of an event tree is an accident sequence. As an example of an event tree, Fig. 9 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. Consider, for example, accident sequence 4. 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 $\bar{Q}$). Thus, this accident sequence is the set of five

Table 1. SSMRP initiating events.

Event	Symbol	Characteristics
1. Vessel Rupture	RPV	(Break with ECCS Ineffective)
2. Large LOCA	LLOCA	(ID > 6 Inches Diameter)
3. Medium LOCA	MLOCA	(3 < ID < 6 Inches Diameter)
4. Small LOCA	SLOCA	(1.5 < ID < 3 Inches Diameter)
5. Small-Small LOCA	SSLOCA	(0.5 < ID < 1.5 Inches Diameter)
6. Transient T1	T1	(PCS Initially Available)-
7. Transient T2	T2	(PCS Initially Inoperative)

Figure 3. Reactor vessel-rupture event tree.

Figure 4. Large LOCA event tree.

*No core melt

Figure 6. Small LOCA event tree.

Figure 7. Small-small LOCA event tree.

^aNo core melt

^bThese accident sequences include the following events: safety relief valve opens and fails to reclose, which initiates a LOCA. The LOCA event tree is appended to the footnoted sequences.

Figure 8. Class 1 transient event tree.

^a No core melt

^b These accident sequences include the following events: safety relief valve opens and fails to reclose, which initiates a LOCA. The LOCA event tree is appended to the footnoted sequences.

Figure 9. Transient (T2) event tree.

events ($T2 \bar{K} \bar{L} \bar{P} \bar{Q}$). In computing the probability of core melt, we computed the probability of these five events occurring together. Similarly, the probability of all the other accident sequences identified for the plant in question is computed. We use fault trees to determine the success or failure of each of the safety systems whose success or failure make up the accident sequences.

Step 3 - Determine Failure Modes of Safety Systems

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

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

<u>System</u>	<u>Number of Basic Events</u>	<u>Number of Logic Gates</u>
ACC	54	17
SIS	242	117
RHR	309	130
CHG	378	172
AFWS	1288	786

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

Human and maintenance errors	533
Other random failures	20
Seismically-induced failures	1923

In all, a total of 2476 basic events were considered.

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

To compute the failure of critical components and safety systems, it is necessary to have both a measure of the maximum response (load or acceleration) that the component experiences during an earthquake as well as a measure of the response level at which it fails. At this point it becomes necessary to utilize the concepts of probability and statistics, for neither the response nor the strength at failure of any component are known exactly.

For example, the acceleration experienced by a certain valve depends on the nature of the ground shaking and the properties and layering of the soil under the plant, the material properties of the steel and concrete of which the buildings and foundations are constructed, the design and layout of the buildings themselves, and the material and manner of supporting the piping systems. None of these are known exactly. Indeed, even the building design

and layout are not known precisely, for there are tolerances allowed in thickness of poured concrete walls and in the location of piping support. And, in fact, there can be errors in construction (such as using too few steel reinforcing rods in a wall) which result in differences between the building as designed and as actually constructed. Further, 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. All of these uncertainties give rise to uncertainties in calculating the response and onset of failure of every building and component in the power plant. The most important feature of the SSMRP is that these uncertainties are explicitly recognized and propagated through the calculational scheme, so that the result is not a single number, but rather, the statistical probability of the occurrence of core melt and radioactive release.

A. Response Calculations

The buildings, foundations, major components, and piping systems are all analyzed mathematically using the Finite Element Method, which models the actual structure by a large assemblage of point masses interconnected by springs. The model generated for the Zion turbine/auxiliary building is shown in Fig. 10. Such models were developed for four buildings and five different piping systems in the Zion power plant analysis. Using these models, the response of the buildings and piping systems can be computed for any given time history of the ground shaking under the foundation. For the Zion power plant, responses at over 4000 points in the buildings and over 1000 points in the piping systems were computed, for each input time history.

To incorporate the uncertainties in the input parameters discussed above, multiple calculations 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 input time history, (but having the same nominal peak ground acceleration). Typically, 30 calculations are made (at each earthquake

Fig. 10 Finite element half-structure model of Zion auxiliary - fuel-handling - turbine building.

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 11 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. The dashed line is the best-fitting (two-parameter) statistical lognormal distribution, which is clearly adequate to model these responses. Such distribution functions were determined for the responses of every wall, slab, pipe segment, and component which is identified on the fault trees of the safety systems.

As mentioned above, for each deterministic calculation of the buildings and components responses, a specific set of input variables is used. But a different set of input variables is used in each of the 30 response calculations. The different sets of input variables are chosen from statistical distributions of the input parameters. These input parameter uncertainties were determined from a combination of data, literature review, peer survey and calculation. The uncertainties in these input parameters are intended to include all sources of uncertainty, including random variation in material properties, tolerances in design and construction, friction and building connection uncertainties and all other effects. The input parameters incorporating uncertainty in the soil-structure interaction calculations are the soil shear modulus and the soil material damping. In both the building and piping calculations, superposition of natural mode shapes is used to compute responses, so here the input parameters varied are the natural frequencies (fixed-base) and modal damping. (All natural frequencies for a given building or piping system are assumed to have the same variation.) The input parameters and their coefficients of variation (the ratio of the standard deviation to the mean of the distribution) are shown in Table 2. A computer code, SMACS, was written to perform the repeated calculations, sampling from the input parameter distributions using a Latin Hypercube

Fig. 11. Variation in moment at RPV nozzle (loop 1 hot leg) with random uncertainties only.

Table 2. Input parameters with associated uncertainty specified as a coefficient of variation (COV).

	Key parameter	Random COV	Modeling* COV
Seismic Input	Time histories	N/A	N/A
Soil-Structure Interaction	Shear modulus	0.40	0.57
	Damping	0.50	0.87
Structure Response	Frequency	0.25	0.43
	Damping	0.35	0.61
Subsystem Response	Frequency	0.25	0.43
	Damping	0.35	0.61

*Modeling uncertainties are used in estimating confidence bounds as described in Section 5.

sampling scheme. The input and output of the SMACS code are shown schematically in Fig. 12.

B. Determination of Fragility Functions

Component failure is defined as either loss of pressure boundary integrity or loss of operability. Failure (fragility) is characterized by a cumulative distribution function which describes the probability that failure has occurred given a value of loading. In the context of the SSMRP, loading may be local spectral acceleration, local zero period acceleration or internal force resultant such as 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 acceleration predicted under the foundation.

The development of the necessary fragility functions is reported in Vol. 6 of Ref. 1. As a first step, all components identified in the reactor fault tree analyses were grouped into 37 generic categories. The generic categories are shown on Table 3. 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. 13, with the ordinate giving the probability that the acceleration at failure of the component is less than the abscissa. The development of the fragility functions is discussed in Section 3.

Step 5 - Compute Probability of Core Melt and Radioactive Release

Accident sequence probabilities must be calculated to determine radioactive release probabilities. As discussed in Step 2 above, each accident sequence consists of groups of events (successes or failures of safety systems) which must occur together. And from Step 3, the failure of each safety system can be represented in terms of cut sets, which are groups of component failures

Fig. 12. Schematic flow diagram of input and output for the SMACS code.

Table 3. Generic categories for components.

Mechanical

. Reactor core assembly	. Large horizontal vessel
. Reactor pressure vessel	. Small to medium vessels and heat exchangers
. Pressurizer	. Large vertical centrifugal pumps with motor drive
. Steam generator	. Motor-driven pumps and compressors
. Reactor coolant pump	. Large motor-operated valves
. Piping	. Small motor-operated valves
. Large vertical storage vessels with formed heads	. Large hydraulic- and air-actuated valves
. Large vertical flat bottom storage tank	. Large relief, manual, and check valves
	. Miscellaneous small valves

Electrical

. Horizontal motors	. Auxiliary relay cabinets	. Invertors
. Generators	. Local instruments	. Cable trays
. Battery racks	. Motor control centers	. Circuit breakers
. Switchgear	. Communications equipment	. Relays
. Dry transformers	. Light fixtures	. Ceramic insulators
. Control panels and racks		

Miscellaneous

. Air handling units	. Duct work
. Instrument racks and panels	. Hydraulic snubbers and pipe supports
-

Fig. 13. Examples of fragility curves.

which will cause the safety system to fail. These cut sets and the accident sequences are combined so that every accident sequence can be expressed in a Boolean expression of the form

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

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

A. Calculation of Component Failure Group Probability

It can thus be seen that to calculate the probability of the accident sequences (from which radioactive releases probability can be found) it is necessary to evaluate the probability of the component failure groups, $\text{Prob}(C_i C_j \dots C_k)$. That is, to evaluate the probability of all the component failures in a group $C_i \dots C_k$ occurring together. If these events were independent of one another, (so that the failure of one component had no correlation with the failure of any other component) then a well-known law in probability theory states that the probability of several independent events occurring simultaneously is equal to the product of their individual failure probabilities, i.e.

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

But when failure is caused by ground shaking due to an earthquake, all parts of the plant are shaken together, and the accelerations or loads the various buildings and components experience are highly correlated, and this correlation increases the probability of occurrence of the component failure group. In fact, two components sitting side by side would experience essentially the same floor accelerations, and thus their

responses would be said to be "perfectly correlated". Hence if one component failed, it is highly likely that both components would fail. The computer code SEISIM was written expressly to calculate such dependent 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 building response calculations at each earthquake level), SEISIM constructs a multi-variate normal 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. The accident sequence probability (which is a measure of the probability of core melt) is then 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, radioactive releases are sorted into several different release categories to reflect their severity with respect to the surrounding population. These release categories relate to the type, and energy content of the radioactive fission product release, as well as to the mode and timing of the release. They range from rupture of the top of the containment with a rapid, 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 were derived from those used in the Reactor Safety Study (Ref. 4).

In summary, the five step SSMRP calculational process leads to a single value (a point estimate) for the frequency of core melt per year and the frequency of occurrence of radioactive release (per year) in each of the seven release categories defined in the WASH-1400.

If confidence bounds on these point estimate values are desired, then one must perform an uncertainty analysis, which involves multiple replications of the entire five step process described above. The uncertainty analysis process is described in Section 5.

Section 3: Overview of the Fragilities Development

Fragility relations are required for both the buildings and the piping and components. Building fragilities are necessarily specific to the plant being analyzed. For the Zion Unit 1 reactor being studied in the SSMRP, fragilities were developed for the reactor containment, turbine, auxiliary, fuel handling and crib house buildings. Development of these fragilities is described in Ref. 5 and Vol. 6 of Ref. 1.

The components for which fragility curves must be developed are determined by the depth of detail in the event and fault tree analyses of the reactor system under consideration. For Zion, 7 event trees and 11 fault trees (Ref. 1) have been devised to cover all the hypothesized reactor transients and potential modes of release of radioactivity. Taken together, these event and fault trees require the determination of the probability of failure (due to seismic loading) of over 2300 basic events. (A basic event could be failure of a certain valve, for example). Since it was clearly not feasible to generate fragility curves for thousands of specific components, the first step in the development of the fragility data base was to group all the components identified on the event and fault trees into a set of generic categories. For example, all motor-operated valves with piping diameters between 2 1/2 and 8 inches were placed in a single generic category, and similarly all electrical motor control centers were placed in another generic category. Then a single fragility curve was derived for each generic category. A detailed review of the components showed that a set of 37 different categories would suffice to cover all the required fragilities. These categories, selected on the basis of equipment function, governing design criteria, method of seismic qualification, and response characteristics, are shown in Table 3. While the equipment in most of these categories is considered generic, certain components are specific to Zion. These include the reactor pressure vessel, the pressurizer, the steam generators and the reactor coolant pumps.

3.1 Data Sources for Components

Actual experimental data on failure of components as a function of local base acceleration are scarce. The type of data most commonly available results from qualification tests in which the component is experimentally shown to function as designed for a prescribed acceleration spectrum input. While such data do provide a lower limit to the fragility level, it is difficult to extrapolate from these data to higher response levels. One notable exception to the lack of actual fragility data was the data obtained in the U.S. Army Corps of Engineers SAFEGUARDS program (Ref. 6). This 11-year program, conducted as part of a missile-site hardening effort, included tests of both mechanical and electrical components. The items tested were off-the-shelf items and were typical of components used in commercial reactors in the late 1960s, and the results are thus directly applicable to the Zion power plant. Sixty-four test programs involving shaker table tests of approximately 300 items were conducted. Excitation consisted of sine beat pulse tests, selected to fit a prescribed acceleration spectrum. Equipment function was monitored during each test. Thus these were truly tests of fragility with respect to both functional and structural failure. Typically, components were tested to over 15-g peak acceleration. Out of the nearly 300 reports generated in the SAFEGUARDS program, 64 were found to be directly applicable to components needed in the SSMRP. In particular, these were the only data available for electrical components, and thus all our electrical component fragilities are derived from this source.

A second source of information was the generic design analyses performed by Westinghouse and various component manufacturers for components used in the Zion plant. In these analyses, the component was assumed to be excited by a base acceleration corresponding to a prescribed design spectrum. Then an analytical solution for the stresses or loads in the component was obtained. From these analytical solutions we obtained the acceleration at failure by extrapolating the stresses to our estimate of the ultimate stress capacity using a procedure due to Newmark (Ref. 7). In this procedure, the acceleration at failure is determined from the relation

$$A_F = A_D F_S F_\mu \quad (1)$$

where A_D = design acceleration,
 F_S = strength factor accounting for the limit load capacity,
 F_μ = ductility factor accounting for the inelastic energy absorption.

The factor accounting for the limit load capacity is computed from

$$F_S = \frac{\sigma_{lim} - \sigma_{dead}}{\sigma_{seismic}} \quad (2)$$

where σ_{lim} is the failure load or stress, σ_{dead} is the static load due to weight, pressure, thermal, etc., and $\sigma_{seismic}$ is the peak load induced by the seismic excitation. Thus F_S scales the design acceleration to the failure acceleration, assuming all loads (or stresses) are calculated by a linear elastic analysis, since the peak load (or stress) is proportional to peak acceleration.

Before failure occurs, however, a significant amount of inelastic deformation (and hence energy absorption) takes place. In this inelastic response range, the stress increases much more slowly than the peak acceleration. Hence, the actual acceleration at failure is much higher than that predicted by the product $A_D F_S$ alone. This additional acceleration capacity is taken into account by the ductility factor F_μ . This ductility factor was introduced by Newmark (Ref. 8) and is a function of both the ductility of the component and the component damping. The ductility μ is usually estimated on the basis of engineering judgment and a knowledge of component construction details, and then the ductility factor can be approximated by the relation

$$F_\mu = \sqrt{2\mu - 1} \quad (3)$$

The statistical distribution of the acceleration at failure (the fragility relation) is obtained by assuming that the factors F_S and F_μ are lognormally distributed random variables. This choice of distributional form has been found to be appropriate in several studies (Refs. 9 - 11) and also results in considerable computational convenience. If M_S and M_μ denote the median values of F_S and F_μ , and if β_S and β_μ denote the

standard deviations of the natural logarithms of the variables F_S and F_μ , then by the multiplicative property of log-normal random variables, the median and log-standard-deviation of the acceleration at failure are given by,

$$M_A = A_D M_S M_\mu \quad \text{and} \quad \beta_A = \sqrt{\beta_S^2 + \beta_\mu^2} \quad (4)$$

respectively.

These two parameters completely define the distribution of acceleration at failure. Values of the uncertainty in the factors F_S and F_μ are estimated from data, analysis, or engineering judgment, depending on the component. While this method of estimating fragility of components is not based directly on failure tests, it does allow an estimate of failure incorporating experimental determination of ultimate strength, weld and connector ductilities, etc., and the choice of the uncertainty factors β_S and β_μ may be made so as to reflect our confidence (or lack thereof) in the analysis. This measure of confidence can then be propagated through the entire SSMRP calculational scheme, and its effect on the final prediction of radioactive release probability can be determined.

The final source of information on fragility of components was an expert opinion survey performed in the spring of 1980. In this survey, a carefully worded questionnaire was mailed to several hundred well-known specialists in the nuclear industry. These individuals were selected from the Nuclear Steam Supply System (NSSS) vendors, architect/engineering firms, consultants to the nuclear industry, and from the ranks of colleges and universities. In each case, the individual was asked to respond only for those components for which he felt a high degree of expertise. For each component, the respondent was asked to provide:

- The three lowest (weakest) failure modes.
- The appropriate response quantity for each mode (eg., peak acceleration, spectral acceleration at some frequency and damping, or force resultant, etc.).

- the response values at 10, 50, and 90% probability of failure.
- The primary source of his information (i.e., experience, test data, etc.).

The expert opinion responses covered virtually every category of component needed for the SSMRP, with 147 detailed responses being returned. Comparison of responses from different experts for the same component showed, in general, surprisingly good agreement. Inasmuch as the expert opinion responses were provided at three probability levels, it was necessary to develop a method of statistically combining them.

The procedure adopted was based on a combined least squares and nested analysis of variance approach. The equations used are described in detail by George (Ref. 12). In this approach, each failure mode (for each component) is treated as independent, and a single fragility curve is developed for each mode based on the responses of all experts who identified that particular failure mode.

Based on our subjective evaluation of the expert opinion responses, we combined different expert's responses into a common group if we had reason to believe that these experts were all referring to the same type of component within the broad generic category being considered. The use of the nested analysis of variance procedure then allowed us to identify the total variance from

$$\sigma^2 = \sigma^2 + \sigma_T^2 + \sigma_E^2 \quad (5)$$

where

- σ^2 = inherent uncertainty in each individual expert's fragility estimate,
- σ_T^2 = uncertainty resulting from the different groups of components within the generic category,
- σ_E^2 = uncertainty between experts whose data were combined in the same group.

In the analysis of Eq. (5), a weighted least squares approach was used in estimating $\hat{\sigma}^2$. The weights were assigned as a product of two factors: a factor for presumed expertise of the specialist providing the opinion and a factor for source of his opinion. In assigning weights, a differentiation between pressure boundary failure and functional failure was made to reflect a lesser degree of confidence in analytical methods for predicting functional failure.

It is at this point that data from the other sources (the SAFEGUARDS fragility data and the generic component design analyses previously described) were incorporated. These additional data were treated as independent expert opinions, with weight factors assigned based on our subjective evaluation of the quality of the data.

The final step in the development of a single fragility curve for a given generic category was to combine the fragility estimates (obtained from Eq. (2)) for each independent failure mode. This combination of modes was performed using the relation

$$F(r) = 1 - \pi \left[1 - F_i(r) \right] \quad (6)$$

where $F(r)$ is the single combined mode fragility curve and $F_i(r)$ are the fragility curves derived for the n failure modes identified for the generic category. This is the statistical union of failure modes and, in effect, produces a fragility curve which is nearly a lower bound.

3.2 Fragilities Used for Base Case Risk Assessment of Zion

Table 4 summarizes the numerical values of the fragilities used in the base case risk analysis. The fragilities are given in terms of the parameters of the lognormal distribution, i.e., the value of the median response at failure (m) and values of the standard deviation of the natural logarithm β_R and β_U . The uncertainty β_R represents that portion of the uncertainty which is due to inherent randomness in the failure level (due, say, to

Table 4. Summary of component fragilities.

Category	Fragility			Load Parameter	Frequency (Hz)	Damping % of Critical	Failure Mode ^a	Key for Table 5
	Median	β_R	β_U					
Reactor core assembly	2.06	0.24	0.12	Spectral accel. g	5-15	5	Deformation of guide tubes	A
Reactor pressure vessel	3.83	0.23	0.39	Spectral accel. g	5	5	Fracture of RPV outlet nozzle	A
Pressurizer	2.00	0.40	0.34	Spectral accel. g	20	5	Failure of support skirt bolting	A
Steam generator	2.45	0.24	0.37	Spectral accel. g	5	5	Support failure	B
Reactor coolant pump	2.64	0.24	0.37	Spectral accel. g	5	5	Support failure	A
Piping (master fragility)	2.44×10^6	0.18	0.33	Moment in - lb	--	--	Plastic collapse	C
Large vertical vessels w/formed heads	1.47	0.20	0.35	ZPA g	Rigid	--	Failure of anchor bolts	D
Large vertical tanks w/flat bottom	2.01	0.25	0.29	ZPA g	Rigid	--	Failure of anchor bolts	A
Large horizontal vessels	3.90	0.30	0.53	Spectral accel. g	12-20	5	Failure of anchor bolts	A
Small to Medium vessels & heat exchangers	1.85	0.25	0.45	Spectral accel. g	20	5	Failure of anchor bolts	E
Large vertical centrifugal pumps with motor Drive	2.21	0.22	0.32	Spectral accel. g	5	5	Failure of support connection	F
Motor driven pumps & compressors	3.19	0.21	0.27	Spectral accel. g	7	5	Impeller deflection	A

Table 4. (Continued)

Category	Fragility			Load Parameter	Frequency (Hz)	Damping % of Critical	Failure Mode ^a	Key for Table 5
	Median	β_R	β_U					
Large motor operated valves	4.83	0.26	0.60	Piping peak accel. g	Rigid	--	Distortion of extended operator	G
Large motor-operated valves	14.40	0.28	0.56	Piping peak	Rigid	--	Structural failure	R
Small motor operated valves	9.84	0.26	0.60	Piping peak accel. g	Rigid	--	Distortion of extended operator	A
Large hydraulic and air actuated valves	7.61	0.31	0.34	Piping peak accel. g	Rigid	--	Loss of control air	A
Large relief, manual, and check valves	8.90	0.20	0.35	Piping peak accel. g	Rigid	--	Internal damage	A
Miscellaneous small valves	12.50	0.33	0.43	Piping peak	Rigid	--	Internal damage	A
Horizontal motors	12.10	0.27	0.31	ZPA g	Rigid	--	Binding of rotating parts	H
Generators	0.65	0.25	0.31	Spectral accel. g	22	5	Shutdown valve trip	I
Battery racks	2.29	0.31	0.39	ZPA g	Rigid	--	Failure of battens	J
Switchgear	2.33	0.47	0.66	Spectral accel. g	5-10	5	Spurious operation of a protective relay	K
Dry transformers	2.78	0.28	0.30	Spectral accel. g	10	5	Failure of anchor bolts	L
Control panels and racks	11.50	0.48	0.74	Spectral accel. g	5-10	5	Dislodging or malfunction of component	A
Auxiliary relay cabinets	7.63	0.48	0.66	Spectral accel. g	5-10	5	Breaker trip	A

Table 4. (Continued)

Category	Fragility			Load Parameter	Frequency (Hz)	Damping % of Critical	Failure Mode ^a	Key for Table 5
	Median	β_R	β_U					
Local instruments	7.68	0.20	0.35	Spectral accel. g	5-35	5	Loosening of fasteners	M
Motor control centers	7.63	0.48	0.74	Spectral accel. g	5-10	5	Breaker trip	A
Communications equipment	5.00	0.33	0.35	Spectral accel. g	10-50	5	Dislodging of components	A
Light fixtures	9.20	0.14	0.14	Spectral accel. g	20-30	2	Dislodging of components	A
Inverters	15.60	0.76	0.35	Spectral accel. g	5-10	5	Relay trip	N
Cable trays	2.23	0.34	0.19	ZPA g	Rigid	--	Support system failure	O
Circuit breakers	7.63	0.43	0.74	Spectral accel. g	5-10	5	Breaker trip	P
Relays	4.00	0.48	0.75	Spectral accel. g	5-10	5	Relay chatter	A
Ceramic insulators	0.20	0.25	0.25	PGA g	2-8	5	Fracture of porcelain	Q
Air handling units	2.24	0.27	0.31	Spectral accel. g	5	5	Rubbing of fan on housing	A
Instrument racks and panels	1.15	0.48	0.66	Spectral accel. g	5-10	5	Relay chatter	A
Duct work	3.97	0.29	0.46	Spectral accel. g	5-10	7	Structural failure	A
Hydraulic snubbers and pipe supports	1.46	0.22	0.49	ZPA g	Rigid	--	Weld failure	A

^aOnly the most likely failure mode is listed, although the fragility may be based on a combination of modes.

Key to Table 4

Key from

Table 4

Basic event system and/or component

A	None (i.e., not used)
B	Steam generator and steam generator tubes
C	Piping and piping components
D	Tanks in the safety injection system and chemical and volume control system
E	Coolers in the reactor containment ventilation system, service water system, and residual heat removal system
F	Centrifugal pumps
G	Used for all valves except those with pressure boundary failure
H	Electrical heaters in the safety injection system
I	Generator, diesel generator, and diesel generator components in the main power and service water systems
J	Batteries in main electrical power system
K	Relays and protective switchgear
L	Dry transformers and battery chargers
M	Local instruments including sensors, detectors and controllers
N	Inverters in main electrical power system
O	Electrical conductors in both main power and instrumentation power systems
P	Circuit breakers, controller, starters and switches
Q	Loss of offsite power and turbine trip
R	Used for valves with pressure boundary failure

variations in material properties, dimensions, etc.) while β_U represents the uncertainty due to lack of knowledge. This would include uncertainty due to approximations in modeling, combination of loads in design analysis or uncertainty in exact configuration of any specific component in a generic category, for example. The crucial difference between these sources of uncertainty is that the random uncertainty β_R is not further reducible, while (in principle) the modeling uncertainty β_U could be reduced or eliminated by additional data, more testing, or more detailed analytical modeling. The calculations shown in this report utilize the random uncertainties β_R only. The modeling uncertainties β_U have been used in earlier SSMRP calculations to compute confidence bounds. This topic is discussed in Section 5.

3.3 Base Case Fragilities for the Steam Generator and Reactor Coolant Pumps

The fragilities for the steam generator and reactor coolant pumps at Zion were determined by the capacity of their supports. The supports for both components are similar, and are shown in Figs. 14 through 18. The vertical support for the steam generator consists of four column supports, each being pinned at the upper and lower ends. Thus, only axial loads (and perhaps some torsion) can be expected. The reactor coolant pump is supported by three vertical pipe column supports, also hinged at both ends.

The steam generator is laterally supported at two elevations (el. 588 and el. 618) by a sliding frame and snubber arrangement which permits thermal expansion displacements. Two lateral pipe supports are provided for the reactor coolant pump.

The fragilities for these two components were derived from the results of design calculations provided by Westinghouse (Ref. 13), modified by the results of the expert opinion survey.

Fig. 14 STEAM GENERATOR UPPER LATERAL SUPPORT STEEL

Fig. 15 STEAM GENERATOR LOWER LATERAL SUPPORT STEEL

Fig. 16 STEAM GENERATOR VERTICAL SUPPORT

FIG. 17. REACTOR COOLANT PUMP LATERAL SUPPORT

Fig. 18. REACTOR COOLANT PUMP VERTICAL SUPPORT

Steam Generator Capacity Based on Design Analysis

In Ref. 14 it is indicated that the NSSS component support stresses were limited to yield for static plus SSE loads. Information from Westinghouse indicated that for Zion the steam generator support columns are the most critically stressed item, with the static and SSE loads consuming 32 and 38% of the faulted condition allowable, respectively.

The construction material is ASTM A-588 with a 50 ksi minimum yield. Considering failure to occur when the yield stress is reached, and assuming the median yield strength to be about 1.25 times the specified design minimum yield stress, then applying Eq. (2) with the above stated stress levels, the strength factor is computed to be

$$F_S = \frac{1.25 - 0.32}{0.38} = 2.45.$$

The uncertainties in this strength factor are due to variability in the yield strength, which are taken as

$$\beta_R = 0.1$$

$$\beta_U = 0.1.$$

Reference 7 recommends that for design of members loaded primarily in compression the ductility (μ) should range from about 1.5 to 3.0. Since these are design values, 3.0 is considered to be a median value and 1.5 to be approximately a minus 2 logarithmic standard deviation value.

Applying Eq. (3), the median factor for ductility is

$$F_\mu = \sqrt{2\mu - 1} = 2.24.$$

Considering the range of ductility from 1.5 to 3 as representing 2 logarithmic standard deviations and considering the uncertainty in the application of Eq. (4), the variability in the ductility factor can be defined as

$$\beta_R = 0.10$$

$$\beta_U = 0.29.$$

Combining the median factors and logarithmic standard deviations, the overall capacity factor is specified by

$$F_C = F_S F_\mu = 5.5$$

$$\beta_R = 0.14$$

$$\beta_U = 0.31.$$

Multiplying the computed factor times the original design spectral acceleration for the SSE (0.6 g) results in a median capacity of 3.3 g at the 5 Hz fundamental NSSS system frequency. The input parameter is spectral acceleration at 5 Hz at the steam generator support at elevation 590 ft of the reactor building.

Reactor Coolant Pump Capacity Based on Design Analysis

Information from the NSSS supplier indicates that the pump support stresses from seismic and static loading are similar to the steam generator support stresses. The pump support fragility description is, therefore, identical to that of the steam generator. The median fragility level is 3.3 g at the 5 Hz NSSS system fundamental frequency. The applicable input parameter is spectral acceleration at the pump support/reactor building interface at elevation 591.

3.3.1 Fragilities of SG and RCP Used in Base Case Analysis

As described in Section 3.1, the fragilities are derived from data, design analysis and expert opinion, and when two or more data sources are available, the results are combined. In addition to the capacities of the SG and RCP based on design calculations described above, expert opinion data were available. This data (which is presented in Vol. 6 of Ref. 1) reduced the median capacities for the SG and RCP categories. The final fragility parameters used for the base case analysis presented herein, were

$$\text{Steam Generator} \quad A_F = 2.45g \quad \beta_R = 0.24 \quad \beta_U = 0.37$$

$$\text{Reactor Coolant Pump} \quad A_F = 2.64g \quad \beta_R = 0.24 \quad \beta_U = 0.37.$$

The input parameters are unchanged.

3.4 Degraded Capacity Fragilities for the Steam Generators and Reactor Coolant Pumps

As was shown in Sec. 3.3, the median capacity of the steam generators and reactor coolant pumps was obtained from

$$A_F = A_D F_S F_\mu,$$

where A_D is the spectral acceleration at the support computed for a SSE seismic design analysis. The aim of the calculations reported here is to assess, in some generic sense, the consequences of having less than as-designed fracture toughness in the SG and RCP supports. No attempt was made to compute exactly, from a linear elastic fracture mechanics (LEFM) viewpoint, the specific reduction in capacity of the supports. Rather, two cases were evaluated based on somewhat arbitrary choices of the product $F_S F_\mu$, given by:

$$\begin{array}{ll} \text{Case 1} & F_S F_\mu = 1.0 \\ \text{Case 2} & F_S F_\mu = 2.0. \end{array}$$

These two choices for the product $F_S F_\mu$ lead to the fragility descriptions:

Case 1	$A_F = 0.6g$	$\beta_R = 0.14$	$\beta_U = 0.37$
Case 2	$A_F = 1.2g$	$\beta_R = 0.14$	$\beta_U = 0.37.$

The same values were used for both the RCP and SG categories, although, of course, the input accelerations were somewhat different. Thus in Case 1, the supports will fail (with 50% probability) at the SSE level, while in Case 2, the failure level will be about twice the SSE level. It was felt* that Case 1 would represent (generically) the severely degraded case applicable to a wide range of RCP and SG support configurations.

3.4.1 Supporting Rationale for Degraded Support Fragility Values

As mentioned, the degraded values of the RCP and SG support fragility were not determined from a fracture mechanics analysis. However, two arguments are presented below which support the reasonableness of the degraded Case 1 and Case 2 values chosen for analysis.

The capacity of the support structure is quantified by use of two factors: the strength factor F_S which is a function of the materials' ultimate strength, and second, the ductility factor F_μ which is a function of the material's ability to deform prior to failure:

$$F_S = \frac{\sigma_{lim} - \sigma_{static}}{\sigma_{seismic}} = \frac{\frac{\sigma_{lim}}{\sigma_{design}} - \frac{\sigma_{static}}{\sigma_{design}}}{\frac{\sigma_{seismic}}{\sigma_{design}}}$$

$$F_\mu = \sqrt{2\mu - 1}, \quad \mu = \epsilon_{max} / \epsilon_y.$$

*Personal communication from R. Johnson, Generic Issues Branch, US NRC.

For the support material at Zion, $F_S F_\mu = 5.5$ based on the ratios

$$\frac{\sigma_{lim}}{\sigma_{design}} = 1.25 \quad , \quad \frac{\sigma_{static}}{\sigma_{design}} = 0.32 \quad , \quad \frac{\sigma_{seismic}}{\sigma_{design}} = 0.38$$

as described in Section 3.3. The question which we consider is how can these fragility values be altered to include the effects of fracture toughness and the effect of reducing toughness. We assume that the upper shelf fracture toughness for nominally good material is $75 \text{ ksi-in}^{1/2} \pm 20 \text{ ksi-in}^{1/2}$, while the lower shelf fracture toughness is taken as $25 \text{ ksi-in}^{1/2} \pm 5 \text{ ksi-in}^{1/2}$. These values were taken from data on A-36 and A-572 steel as shown in Fig. 13.

The first approach to include fracture is to normalize the fracture capacity to the same value as the tensile capacity. Thus, we define a fracture capacity factor similar to the tensile strength capacity factor:

$$F_K = \frac{K_{Ic} - K_{static}}{K_{seismic}}$$

where K_{Ic} is the experimental fracture toughness and K is the appropriate stress intensity factor. Equivalently we can write

$$F_K = \frac{\frac{K_{Ic}}{K_{design}} - \frac{K_{static}}{K_{design}}}{\frac{K_{seismic}}{K_{design}}}$$

Note that F_μ must be taken as 1.0 in the brittle fracture case since no plastic energy absorption is allowed.

Fig. 19. Experimental values of fracture toughness for A-36 (grade 50) and A-572 (grade 50) from Reference 16.

Now the stress intensity K can always be computed from an expression of the form

$$K = \sigma \sqrt{\pi a} f(a),$$

where σ is the nominal stress in the absence of a crack, $2a$ is the crack length and $f(a)$ is a function of crack length, geometry, curvature, etc. For most cases $f(a)$ is not too far from unity. Thus the fracture capacity can be expressed as

$$F_K = \frac{\frac{K_{Ic}}{K_{design}} - \frac{\sigma_{static} \sqrt{\pi a} f(a)}{\sigma_{design} \sqrt{\pi a} f(a)}}{\frac{\sigma_{seismic} \sqrt{\pi a} f(a)}{\sigma_{design} \sqrt{\pi a} f(a)}}$$

Then, since we are considering changes in toughness only (i.e., geometry and assumed crack sizes remain constant) we can reduce this equation to:

$$F_K = \frac{\frac{K_{Ic}}{K_{design}} - 0.32}{0.38}$$

since the nominal stress ratios will be the same. Setting F_K equal to the product of $F_S F_\mu = 5.5$.

$$\text{If } 5.5 = \frac{\frac{K_{Ic}}{K_{des}} - .32}{.38},$$

we can solve for the ratio

$$\frac{K_{Ic}}{K_{des}} = 2.41.$$

If we now degrade the toughness to 25 ksi-in

$$\frac{K_{Ic \text{ degraded}}}{K_{\text{design}}} = 2.41 \times \frac{25}{75} = .803,$$

the resulting equivalent fracture capacity becomes

$$F_K = \frac{(.803) - (.32)}{.38} = 1.27.$$

This method of maintaining the same value of capacity for both tensile failure and fracture failure for good material is equivalent to equating the energy required for failure in each case. This capacity factor of 1.27 lies between the values of 1.0 and 2.0 used to define the Case 1 and Case 2 degraded fragilities.

By setting the mean fracture capacity, F_K , equal to the stress and ductility capacity, $F_{S\mu} = 5.5$, we are essentially defining a mean flaw size for the structure. Assuming that the function of geometry in the stress intensity equation is equal to one, then, for the base case material property the flaw size is calculated as follows:

$$F_{S\mu} = F_K = \frac{K_{Ic} - (.32)(S_y) \sqrt{\pi a}}{(.38)(S_y) \sqrt{\pi a}}$$

$$5.5 = \frac{75^{\text{ksi-in}^{1/2}} - (.32)(50^{\text{ksi}}) \sqrt{\pi a}}{(.38)(50^{\text{ksi}}) \sqrt{\pi a}}$$

$$a = 0.123 \text{ in.},$$

where the material yield strength, S_y , is taken to be 50 ksi. This value falls into the range of mean flaw sizes (0.067 in., 0.084 in., and 0.24 in.) reported in the Load Combinations Program for three separate sources of crack occurrence data, (Ref. 15).

Section 4: Results of Degraded Support Seismic Risk Calculations

4.1 Summary of Results

As mentioned, two sets of calculations were performed to assess the change in risk due to two levels of reduced capacity of the steam generator and reactor coolant pump supports. The base case corresponds to the supports having as-designed capacities, with a median failure level of approximately four times the SSE level (0.17 g). Case 1 is the weakest case, where-in the support capacity was reduced to a median failure level equal to the SSE, while Case 2 is an intermediate case with the support capacities reduced to twice the SSE level.

The main result of interest is that the public dose assuming reduced capacity supports was found to be about 32 man-REM/year (Case 1) and 12 man-REM/year (Case 2), compared to the base case (normal support capacity) of 6.6 man-REM/year. These values are based on the seismic hazard at the Zion site, and include the effects of all sizes of earthquakes thought to be possible at the site, but taking into account the decreasing probability of occurrence of larger earthquakes. The estimated public dose (given a core melt) in the seven release categories for a PWR is given in Table 5. These values were taken from NUREG/CR-2800. It is a simple matter to modify these results for different seismic hazard curves or different exposure values.

The increase in dose per year is due entirely to the reduced capacities of the steam generators and reactor coolant pumps, but the changes in dose are not proportional to changes in capacity. To understand these results, we note that the probability of release (in any given release category) is obtained by summing up all the contributions from the Terminal Event Sequences (TES) which contribute to that release category. Each terminal event sequence probability is defined as

$$P(\text{TES}) = \sum_{i=1}^6 P(\text{earthquake level } i) \cdot P(\text{initiating event}) \\ \cdot P(\text{accident sequence}) \cdot P(\text{containment failure})$$

Table 5. Estimated public dose in WASH-1400 release categories (man-REM) from NUREG/CR-2800.

Release Category	Dose
PWR 1	5,400,000
PWR 2	4,800,000
PWR 3	5,400,000
PWR 4	2,700,000
PWR 5	1,000,000
PWR 6	150,000
PWR 7	23,000

There are many more terminal event sequences than accident sequences, since there are up to five different containment failure modes (with associated probabilities) associated with each accident sequence. Each terminal event sequence contributes to one, and only one, release category.

In the following, we discuss each of the four terms on the right-hand-side of the equation above, and how the terms vary when the capacities of the steam generator and reactor coolant pumps are changed.

4.2 Earthquake Level Probabilities

The risk of earthquakes at a given site is expressed in terms of the probability of exceeding any level of ground acceleration (i.e., a complementary cumulative distribution). Graphically, this plot is called the hazard curve. The hazard curve at Zion is shown in Fig. 1. The acceleration range on the abscissa has been discretized into six increments, called earthquake levels. The probability of having an earthquake which causes a ground acceleration in any range (level) can be taken from Fig. 1. The numerical probabilities associated with each earthquake level are listed in Table 6.

Note that the accelerations on the abscissa on Fig. 1 and in Table 6 are given as "rock outcrop" accelerations. This is the acceleration that the site would experience if the bedrock extended to the surface, i.e., if there were no soil overlaying the bedrock at the site. The rock outcrop acceleration is computed from a knowledge of the regional earthquake history (number and magnitudes of earthquakes) and the attenuation of earthquake ground motions in the local region (say, within 200-300 km of the site).

If soil overlays the bedrock under the power plant, then the surface ground acceleration is computed from the bedrock motion, and reflects the depth, composition and layering of the soil column. For example, the safe-shutdown earthquake (the design basis earthquake for Zion) was defined as 0.17 g at the ground surface. This corresponds to 0.10 g rock outcrop acceleration.

Table 6. Earthquake hazard at Zion.

Earthquake Level	Rock Outcrop Acceleration Range (g)	Probability per year
1	0.06 - 0.10	7.78×10^{-4}
2	0.1 - 0.2	3.78×10^{-4}
3	0.2 - 0.32	1.88×10^{-5}
4	0.32 - 0.42	2.30×10^{-6}
5	0.42 - 0.53	5.95×10^{-7}
6	0.53 - 0.69	2.30×10^{-7}

The hazard curve in Fig. 1 is our estimate of the median hazard curve for the Zion site. It was assembled from a combination of historical data and expert opinion, which provided 14 different hazard curves (each assumed equally likely), as shown in Fig. 20. The upper and lower curves are assumed to be 10% and 90% confidence limits. However, it should be noted that the median hazard curve was used in all the calculated values reported herein.

4.3 Probability of Initiating Events

In the event of a good-sized earthquake, it is assumed that the power plant will shut down (hot shutdown condition), even if only briefly. For small earthquakes (magnitude 3.5 Richter) this might be due to turbine vibration monitors or loss of off-site power, while for large earthquakes shutdown could result from damage to the plant.

Seven classes of initiating events have been identified for the Zion PWR, as listed in Table 1. These include the reactor vessel rupture (RPV) which is defined as any pipe or vessel break configuration for which the emergency core cooling system (ECCS) functions are ineffective, four sizes of loss-of-coolant accidents (LOCAs) and two classes of transients. Transient T2 is any abnormal thermal-hydraulic condition which requires SCRAM, but in which the power conversion system (PCS) is initially inoperative. Transient T1 is defined similarly, except the PCS is assumed to be initially available.

Given an earthquake of magnitude greater than 3.5 (Richter), it is assumed that one of the seven initiating events occurs, i.e., that the initiating event probabilities sum to 1.0. Further, it is assumed that, if sufficient pipe breaks occur such that the RPV initiating event has occurred, then all less severe LOCA initiating events are unimportant. Thus, the RPV and LOCA event probabilities are scaled so that the most severe RPV or LOCA situation which can occur will have the largest probability.

Fig. 20. Family of hazard curves for Zion site from which median curves and bounds were computed.

Failure of the steam generator and reactor coolant pump supports is assumed to cause complete severance of the 34 inch OD primary coolant piping. If only one failure occurs (either steam generator or pump support) then a large LOCA (LLOCA) occurs. Since there are four loops at Zion, there are eight single support failure events which can cause the LLOCA. If at least one support failure occurs in at least two of the four loops, then a RPV situation exists, since the ECCS is designed for single large breaks only. Thus there are 24 pairs of steam-generator support and/or reactor coolant pump support failures which can cause the RPV initiating event. Failure of the SG and/or RCP supports are assumed not to cause any of the other initiating events. (Note that other pipe break combinations contribute to the RPV and LLOCA initiating events, but even in the base case, the SG and RCP support failures are dominant.)

The computed median spectral accelerations at the supports of the SG and RCP are listed in Table 7 along with their logarithmic standard deviations (β). Note that these responses are the same for all three cases of support capacity reported herein. From the responses and the fragilities of the SG and RCP, the probability of failure of the SG and RCP supports are computed. These are shown in Table 8 for each of the six earthquake levels. It can be seen that both the SG and RCP have greater than 50% probability of failure at earthquake level 3 in Case 1, while this same degree of failure is not reached until level 5 for Case 2. For the base case, the probability of failure is less than 10% for all six earthquake levels.

The dominant initiating event probabilities are shown on Table 9. It can be seen that for Case 1, the dominant initiating event was the RPV for levels 2 and above, and thus for this case, the terminal event sequences resulting from the RPV initiating event completely determined the computed probability of release. Similarly, in Case 2, the RPV event dominated at earthquake level 3 through 6, while at levels 1 and 2, the transients T2 and T1 dominated. Thus the probability of release in this case came primarily from the RPV but significant contributions could be expected from T2 and T1. In the base case, the smaller LOCAs and the two transients were dominant, and the RPV and LOCA events were not significant contributors to the probability of radioactive release.

Table 7. Computed median (m) and log-standard deviation (β) of base spectral accelerations (ft/sec²) for steam generators and reactor coolant pumps.

		Earthquake Level					
		1	2	3	4	5	6
Steam	m	7.31	11.59	21.54	30.57	36.60	46.52
Generators	β	.308	.330	.283	.247	.347	.310
Reactor	m	8.73	12.93	23.81	31.60	36.50	46.20
Coolant Pumps	β	.327	.338	.417	.311	.371	.376

Table 8. Conditional Probabilities of failure of steam generator and reactor coolant pump at each earthquake level.

		<u>Steam Generator</u>		
Earthquake Level		Case 1	Case 2	Base Case
1		.002	0	0
2		.076	.0004	0
3		.631	.032	.0002
4		.945	.21	.003
5		.957	.44	.034
6		.995	.70	.087
		<u>Reactor Coolant Pump</u>		
Earthquake Level		Case 1	Case 2	Base Case
1		.0128	0	0
2		.139	.0014	0
3		.681	.136	.004
4		.925	.28	.006
5		.946	.44	.028
6		.985	.67	.086

Table 9. Initiating event probabilities (conditional) at each earthquake level.

<u>Case 1: Weak Supports</u>					
Earthquake Level	RPV	LLOCA	Other* LOCA's	T2	T1
1	.045	.061	0	.24	.65
2	.71	.18	.01	.09	.02
3	1.0	0	0	0	0
4	1.0	0	0	0	0
5	1.0	0	0	0	0
6	1.0	0	0	0	0
<u>Case 2: Intermediate Supports</u>					
Earthquake Level	RPV	LLOCA	Other* LOCA's	T2	T1
1	0	0	0	.27	.73
2	.005	.008	.01	.80	.17
3	.567	.22	.07	.15	.0001
4	.96	.037	0	.002	0
5	.99	.0001	0	0	0
6	1.0	0	0	0	0
<u>Base Case: Normal Supports</u>					
Earthquake Level	RPV	LLOCA	Other* LOCA's	T2	T1
1	0	0	0	.27	.73
2	0	0	.01	.82	.17
3	.004	.017	.23	.75	0
4	.003	.035	.57	.39	0
5	.11	.20	.58	.11	0
6	.40	.31	.29	.004	0

*Includes initiating events MLOCA, SLOCA, SSLOCA.

4.4 Probability of Accident Sequences and Containment Failure Modes

The accident sequences are assumed to be independent of the initiating events, and thus each accident sequence has the same probability of occurrence for all three cases considered. The relative importance of each accident sequence in contributing to the probability of radioactive release, however, is determined by the initiating event probabilities.

Each accident sequence is associated with one or more of the five containment failure modes. These containment failure modes are those defined in the WASH-1400 study (Ref. 4). Given a core melt (that is, an accident sequence which results in a core melt), it is assumed that some release always occurs. Thus the containment failure probabilities must always add to 1.0. The numerical values of the containment failure probabilities vary with each accident sequence, but are unaffected by the values of the initiating events. Thus the same values are used in all three cases discussed in this report. The definitions of the five containment failure modes are given in Table 10. The assignment of terminal event sequences to release categories is shown in the Appendix.

Table 11 lists the release category probabilities for all earthquake levels, for each of the three cases. Tables 12, 13 and 14 list the dominant accident sequences for the three cases. Core melt probability is the sum of all the release category probabilities, and for Case 1, the core melt probability (per year) is $2.8E-4$ as shown on Table 11. It can be seen that for Case 1, the release is dominated by categories 3 and 7 for earthquake levels 2 and 3 and release category 2 for earthquake levels 4 and 5. Examination of the terminal event sequences for these two categories showed that the release was due, almost entirely, to accident sequences RPV-1, RPV-6 and RPV-7. In accident sequence RPV-1, the RPV initiating event is followed by successful operation of the containment spray and fan cooler systems. Thus the release is due to the situation in which the RPV initiating event occurs due to SG or RCP support failure at very low acceleration levels, while all applicable safety systems still have insignificant probabilities of failure. For this sequence,

the containment failure probabilities assign 1% of the release to category 3 and 99% to category 7. For earthquake levels 4, 5, and 6 the dominant accident sequences are RPV-6 and RPV-7, because the containment sprays are assumed to have failed for these earthquake levels. For RPV-6 and RPV-7, 6% of the release goes to category 1 and 94% is assigned to category 2.

Table 1C. Definition of events used on the containment event tree.

Event	Name	Description
α	CR-VSE	<u>Containment Rupture - Vessel Steam Explosion.</u> Steam flashing caused by the interaction of the molten core with water in the bottom of the reactor vessel causes vessel overpressure and subsequent shattering of the vessel. Missiles resulting from the shattered vessel rupture the containment.
β	CL	<u>Containment Leakage.</u> Failure of the containment to completely isolate.
γ	CR-B	<u>Containment Rupture - Burning.</u> Hydrogen accumulated in the containment ignites, causing instantaneous overpressure, which ruptures the containment.
δ	CR-OP	<u>Containment Rupture - Overpressure.</u> Steam created in the core and released to the containment is not condensed by the containment ESF systems. The result is a slow buildup of containment pressure until overpressure occurs, which ruptures the containment.
ϵ	CR-MT	<u>Containment Rupture - Melt-through.</u> The molten core melts through the bottom of the reactor vessel and the containment basemat, thereby breaching the containment.

Table 11. Probability per year of release categories for each earthquake level - 3 cases.

Case 1: Weak Supports

Release Category	1	2	3	4	5	6	Total	Man-REM/yr
1	0	0	0	9.7E-8	2.5E-8	9.7E-9	1.32E-7	0.71
2	0	0	1.4E-9	1.5E-6	3.9E-7	1.5E-7	2.04E-6	9.79
3	4.5E-7	2.2E-6	1.3E-7	0	0	0	2.8E-6	15.12
4	0	0	0	0	0	0	0	0
5	0	0	0	0	0	0	0	0
6	0	0	0	0	0	0	0	0
7	4.5E-5	2.2E-4	1.3E-5	0	0	0	2.8E-4	6.4

Total = 2.85E-4 32.02

Case 2: Intermediate Supports

Release Category	1	2	3	4	5	6	Total	Man-REM/yr
1	0	0	0	9.34E-8	2.51E-8	9.7E-9	1.28E-7	0.69
2	0	0	0	1.46E-6	3.93E-7	1.52E-7	2.01E-6	9.65
3	6.5E-10	7.5E-8	1.09E-7	2.09E-9	0	0	1.87E-7	1.01
4	0	0	0	0	0	0	0	0
5	0	0	0	0	0	0	0	0
6	0	0	0	3.01E-8	0	0	3.01E-8	.005
7	6.44E-8	7.44E-6	1.08E-5	1.27E-8	0	0	1.83E-5	0.42

Total = 2.07E-5 11.77

Table 11. Probability per year of release categories for each earthquake level - 3 cases (continued).

Base Case: Normal Supports

Release Category	1	2	3	4	5	6	Total	Man-REM/yr
1	0	0	0	1.15E-8	6.93E-9	5.76E-9	2.42E-8	0.13
2	0	0	0	3.74E-7	1.08E-7	8.27E-8	5.65E-7	2.71
3	2.70E-13	4.69E-8	2.91E-8	4.41E-7	1.19E-7	2.67E-8	6.63E-7	3.58
4	0	0	0	0	0	0	0	0
5	0	0	0	0	0	0	0	0
6	0	0	0	2.15E-7	5.86E-8	6.61E-9	2.80E-7	.04
7	2.68E-8	4.64E-6	2.83E-6	3.15E-8	3.73E-8	2.27E-8	7.59E-6	0.17

Total = 9.12E-6 6.63

Table 12. Dominant accident sequences - Case 1

Release Category	Level	Terminal Event Sequences	Probability Per Year	Man-REM Per Year
1	4	RPV-7 α	6.7E-8	0.4
1	4	RPV-6 α	3.0E-8	0.2
1	5	RPV-7 α	2.1E-8	0.1
1	5	RPV-6 α	3.8E-9	
2	4	RPV-7 γ	1.0E-6	4.8
2	4	RPV-6 γ	4.5E-7	2.2
2	4	RPV-7 δ	3.4E-8	0.2
2	4	RPV-6 δ	1.5E-8	0.1
2	5	RPV-7 γ	3.2E-7	1.5
2	5	RPV-6 γ	5.7E-8	0.3
2	5	RPV-7 δ	1.1E-8	0.1
3	1	RPV-1 α	2.5E-7	1.4
3	1	LLOCA-13 α	1.9E-7	1.0
3	2	RPV-1 α	1.9E-6	10.3
3	3	RPV-1 α	1.3E-7	0.7
7	1	RPV-1 α	2.5E-5	0.6
7	1	LLOCA-13 α	1.9E-5	0.4
7	2	RPV-1 α	1.9E-4	4.4
7	3	RPV-1 α	1.3E-5	0.3

Table 13. Dominant accident sequences - Case 2

Release Category	Level	Terminal Event Sequences	Probability Per Year	Man-REM Per Year
1	4	RPV-7 α	6.4E-8	0.4
1	4	RPV-6 α	2.9E-8	0.2
1	5	RPV-7 α	2.1E-8	0.1
1	5	RPV-6 α	3.8E-9	
2	4	RPV-7 γ	9.8E-7	4.7
2	4	RPV-6 γ	4.3E-7	2.1
2	4	RPV-7 δ	3.2E-8	0.2
2	4	RPV-6 δ	1.4E-8	0.1
2	5	RPV-7 γ	3.2E-7	1.5
2	5	RPV-6 γ	5.7E-8	0.3
2	5	RPV-7 δ	1.1E-8	0.1
2	6	RPV-7 γ	1.4E-7	0.7
3	2	T2-4 α	4.6E-8	0.2
3	2	LLOCA-13 α	1.6E-8	0.1
3	2	RPV-1 α	1.3E-8	0.1
3	3	RPV-1 α	7.5E-8	0.4
3	3	LLOCA-13 α	2.9E-8	0.1
3	3	T2-4 α	4.4E-9	
7	3	RPV-1 ϵ	7.4E-6	0.2
7	3	LLOCA-13 ϵ	2.8E-6	0.1
7	3	T2-4 ϵ	4.3E-7	
7	3	SSLOCA-29 ϵ	8.8E-8	
7	3	LLOCA-2 ϵ	4.5E-8	
7	2	T2-4 ϵ	4.5E-6	0.1
7	2	LLOCA-13 ϵ	1.6E-6	
7	2	RPV-1 ϵ	1.3E-6	

Table 14. Dominant accident sequences - Base Case

Release Category	Level	Terminal Event Sequences	Probability Per Year	Man-REM Per Year
2	4	T2-4 δ	2.2E-7	1.1
2	4	T2-4 γ	9.4E-8	0.5
2	4	SSLOCA-21 γ	1.8E-8	0.1
2	4	SSLOCA-34 γ	1.5E-8	0.1
2	5	RPV-7 γ	5.6E-8	0.3
2	5	T2-4 δ	2.0E-8	0.1
2	5	RPV-6 γ	9.9E-9	0.1
2	5	T2-4 γ	8.5E-9	0.1
3	2	T2-4 α	4.6E-8	0.3
3	3	T2-4 α	2.0E-8	0.1
3	3	SSLOCA-29 α	4.1E-9	0
3	3	LLOCA-13 α	2.3E-9	0
3	4	SLOCA-21 δ	2.9E-7	1.6
3	4	SSLOCA-21 δ	1.3E-7	0.7
3	4	MLOCA-21 δ	2.7E-8	0.2
3	5	SLOCA-21 δ	1.0E-7	0.5
6	4	SSLOCA-34 ϵ	1.0E-7	
6	4	T2-4 ϵ	2.4E-8	
6	4	LLOCA-27 ϵ	2.9E-8	
6	5	LLOCA-27 ϵ	2.9E-8	
6	5	SSLOCA-34 ϵ	1.9E-8	
6	5	T2-4 ϵ	6.7E-9	
7	2	T2-4 α	4.6E-6	0.1
7	2	SSLOCA-29 α	6.2E-8	
7	3	T2-4 α	2.0E-6	0.1
7	3	SSLOCA-29 α	4.1E-7	
7	3	LLOCA-13 α	2.3E-7	
7	3	RPV-1 α	1.0E-7	

For Case 2, the intermediate support case, the core melt probability is $2.0E-5$ as shown on Table 11. The release is due almost entirely from release category 2 at levels 4, 5 and 6, and the release is coming from accident sequences RPV-7 and RPV-6. The release in category 2 is again due to the RPV initiating event with the containment sprays failing. The main difference between Case 2 and Case 1 is that, in Case 2, we have no RPV contributions at the lower three earthquake levels. In Case 1, RPV-1 at the lower earthquake levels in release category 3 were dominant. These are not present in Case 2, because the steam generator and reactor coolant pumps supports are significantly stronger.

For Case 3, the base case with normal support capacity, the total core melt probability is $9.1E-6$ per year, as shown on Table 11. The release is due primarily to release categories 2 and 3, coming from earthquake levels 4 and 5. As shown in the table of dominant accident sequences for this case (Table 14), the release is due primarily to transient T2-4 and SLOCA-21. The transient accident sequences occur due to failures of pairs of electrical components (relays and busses) in the auxiliary feedwater system, while the small LOCA sequence arises due to failures of pairs of pipes in the ECCS system which run between the auxiliary building and the containment.

Section 5: Estimation of Median Values and Confidence Limits

The computed probabilities of radioactive release presented in the last section were based only on inherent randomness in the random variables used in the computation (the values denoted as β_R). Thus each variable (fragility, response, etc.) was represented as a lognormal random variable with a median and random variance. The result of the calculations is a single ("point") value for the probability of release and dose to the public. If we wish to estimate confidence bounds on these point estimates, we must perform an uncertainty analysis.

5.1 Uncertainty Analysis

A probabilistic risk analysis (PRA) of a nuclear power plant involves the estimation of the probability of risk, e.g., core melt, associated with the occurrence of a damaging event such as an earthquake. The fact that the parameter of interest in a PRA is a probability recognizes that the damaging event, e.g., earthquake, as well as the consequence event, e.g., core melt, are uncertain events. Thus, the occurrence of these events are expressed in terms of probability.

It is important in a PRA to recognize the two sources of uncertainty associated with the assessment of risk. There is the inherent random uncertainty associated with the events of interest actually occurring. This uncertainty has been referred to as random variation or random uncertainty. This type of variation contributes to the level or value of probability that a PRA is attempting to estimate. The second source of variation in the assessment of risk is the uncertainty associated with the inputs, models and calculational techniques used to estimate the probability of risk. This source of uncertainty, due to the fact that no computer model or analysis technique can duplicate physical reality exactly, nor is there sufficient information available about rare events such as earthquakes and/or core melt to predict their occurrence, suggests that the outcome of a PRA should not only be a single (point) estimate of the probability of risk but a

distribution of values which reflects the uncertainties in the analysis methodology. We refer to this latter source of uncertainty as analysis or modeling uncertainty. A brief discussion of how these sources of variation relate to the design and methodology of the SSMRP Zion analysis is given here.

Random variation: This is the uncertainty associated with the occurrence of events, e.g., earthquakes, component failure, core melt, associated with the seismic risk. In the SSMRP probabilistic risk analysis methodology, random variation is associated with:

- the occurrence of earthquakes,
- soil/structure/piping parameters such as damping, frequency and soil shear modulus,
- component and subsystem strength at failure.

In the SSMRP, this random variation is described by associating probability distributions with each of these parameters. Specifically, the random variation is described as follows:

- Seismic hazard curve, the probability, per year, that at least one earthquake results in the peak ground acceleration at the Zion site exceeds level a , described as a function of a .
- Soil/structural/piping parameters are assumed to be lognormal random variables.
- Component/subsystem strengths are assumed to be lognormal random variables. The strength distribution is expressed in terms of a fragility curve, which is the probability that the strength is less than or equal to a level s , described as a function of s .

Given these distributions, the SSMRP risk analysis methodology described in Section 2 is used to compute a point value of frequency of core melt and radioactive release per year. It should be noted that the SMACS computations are based on a sample of time histories as well as sampled values of the soil/structure/piping parameters. The purpose of this sample is to estimate

the mean vector and variance-covariance matrix for the distribution for the 300+ modeled responses throughout the plant. The samples are chosen based on a Latin Hypercube design. Also, it is important to emphasize that this analysis is conditional on:

- the hazard curve,
- the values of the medians and coefficients of variation for the distribution of the response parameters,
- the values of the medians and coefficients of variation which characterize the fragility curves.

Analysis or modeling uncertainty: These are the uncertainties associated with the methods used to estimate the probability of core melt and release. The primary source of analysis uncertainties are:

- the models used in the analysis, e.g., linear response models, lognormal distributions for strengths, system models, etc.,
- the calculational methods, e.g., mathematical methods used for calculating responses, methods for combining probabilities, etc.,
- the assessment of the model parameters, e.g., value of soil/structure parameters, median strengths, etc.

An important source of uncertainty was the use of expert opinions and engineering judgement to assess the values of the model parameters. In the context of the SSMRP risk analysis methodology, this included:

- the elicitation of expert opinions about zonation, seismicity and ground motion models in the region surrounding Zion; these opinions were used to estimate the hazard curves,
- the use of expert opinion and test data to estimate the medians and coefficients of variation for the fragility curves,
- the use of engineering judgement to estimate the medians and coefficients of variation of the soil/structure/ piping parameters,

- the use engineering judgement to estimate the probabilities of failure for unmodeled basic events.

A convenient way to incorporate analysis or modeling uncertainties is to consider the input parameters themselves (e.g., the fragility medians) to be random variables. Two commonly used ways of doing this are:

- to consider a finite number of values for each parameter and associate a discrete probability distribution to the finite set or describe the uncertainty,
- to consider a continuum of values for each parameter and describe the uncertainty in terms of a cumulative distribution or probability density function.

Whichever way analysis uncertainties are described, an uncertainty analysis is highly desirable so that the affect of the analysis uncertainties on the output, e.g. probability of core melt, can be described. The results of such a study will describe the range of values for the probability of core melt, along with the likelihood, which is possible because of the uncertainties in the methodology. The SSMRP uncertainty analysis methodology is described in the next section.

5.2 SSMRP Uncertainty Analysis: Design and Methodology

The goal of the SSMRP uncertainty analysis was to describe the variation in the estimator of the probability of core melt and radioactive release due to the uncertainties associated with the risk analysis techniques. To describe the variation in the estimator of probability due to analysis uncertainty, certain parameters of the input random variables in the risk analysis were themselves considered to be random variables. Uncertainties were considered in the following inputs:

- Responses
 - soil shear modulus and damping,

- structure damping and natural frequency,
- piping system damping and natural frequency.
- Fragilities
 - fragility curves, i.e., component strength cumulative distribution functions.
- Hazard curves, i.e., the probability of at least one earthquake occurring, per year, resulting in the peak ground acceleration at the site exceeding a as a function of a
- Probability of failure for unmodeled basic events

Although the uncertainties in these input variables are discussed in other parts of this report, a brief outline of how the analysis uncertainties are modeled for each input is given to help make the discussion of the uncertainty analysis design and methodology more understandable.

5.2.1 Response Analysis Uncertainties

In the SSMRP Zion risk analysis each of the above mentioned inputs is considered to be subject to random variation. This source of variation was introduced in the risk analysis by considering each variable to have a lognormal distribution. The inputs were varied in the risk analysis by multiplying a nominal value, e.g., the nominal soil shear modulus, by a value chosen from a lognormal distribution with median 1 and coefficient of variation β_R . For the uncertainty analysis it was assumed that the analysis uncertainty associated with the response calculations could be accounted for by considering the nominal value for each input to be a random variable also. The variation introduced is that which results from using engineering or expert opinions about the nominal value, and the biases that may be introduced by the models and analysis methods used in making the response calculations. The analysis uncertainty was introduced in the uncertainty analysis by multiplying the nominal value by a factor chosen from a lognormal distribution with median 1 and coefficient of variation β_U .

5.2.2 Fragility Uncertainties

Fragility curves describe the probability that the strength of a component is less than or equal to response r as a function of r . Similar to the response inputs, the analysis uncertainties associated with the fragility curves was described by treating the median strength, denoted m_r , as a lognormal random variable. Again, this uncertainty was introduced in the analysis by multiplying the median strength by a value chosen from a lognormal distribution with median 1 and coefficient of variation β_U .

5.2.3 Hazard Curve Uncertainties

The principal source of uncertainty in the description of the seismic hazard at the Zion site was the differences of opinions in the zonation and seismicity of the region surrounding the site between the experts elicited to develop the seismic hazard at Zion. The variation in the seismic hazard due to different zonations and different levels of seismicity was incorporated into the uncertainty analysis by using the hazard curves based on the zonation and seismicity given by a subset of the experts used in the hazard analysis for the Zion site.

5.3 Experimental Design

Perhaps the optimal method for assessing the affect of analysis uncertainties on the estimator of the probability of core melt (or radioactive release) would be to analytically propagate the uncertainties through the analysis. Because of the complexity of the Zion risk analysis this is not possible. Alternatively, a Monte Carlo simulation procedure was chosen to assess the variation in the output probabilities due to the uncertainties in the inputs. Usually a simulation analysis involves a large number of iterations of the basic risk analysis. However, the number of iterations can be reduced by appropriately choosing the sets of inputs to be used for each iteration. The basis of one such method is to choose the values for each input such that the entire range of values for the input is represented in the sample of inputs

used. The design of the Zion uncertainty analysis is based on using such a method. Because of budget and other constraints it was determined that the entire risk analysis could only be replicated a limited number of times. The final number of iterations was limited to 14 assessments of the risk.

The experimental design used to select the parameter values for the inputs in the Zion risk analysis for the uncertainty study was a Latin Hypercube design. Since the uncertainty study was limited to 14 assessments of the risk, the range of each of the parameters (e.g., the median of the fragility curve of a component) was partitioned into 14 non-overlapping subintervals which span the range and which have equal probability (based on the uncertainty distribution of that parameter) associated with them. A value was chosen, at random, from each subinterval, thus giving 14 equi-probable values for each parameter. By randomly combining values from the different subintervals for all parameters, 14 sets of values for the parameters are specified. These are the design points that were used for the uncertainty study. Combining the parameter values in this way assures that the entire range for each parameter is represented and each subinterval of the range is included only once in the sample. Each of the 14 design points represents a subregion of equal probability in the space of possible values for the parameters of the inputs into the risk analysis.

Although a Latin Hypercube design assures the range of all parameters is represented in a sample, it must be recognized that a small sample size will still leave a good portion of the parameter space not represented in the sample. Although the Latin Hypercube design is more efficient than choosing points strictly at random, a sufficiently large sample must still be taken to assure a good representation of the variation in the output due to the uncertainties in the input parameters.

For the Zion uncertainty analysis, 8 response parameters and 20 generic fragility curves were considered to be subject to modeling uncertainty. Thus, a Latin Hypercube design was constructed for 8 x 20 parameters and 14 subintervals. Uncertainty in the seismic hazard was introduced by selecting

hazard curves based on the opinions of 14 different experts. Uncertainty in the probability of failure for the unmodeled basic events is incorporated into the uncertainty analysis by randomly choosing 14 values from the appropriate uniform distribution. Each combination of values for the response parameters, fragility curves, probability of failure of unmodeled basic events and a hazard curve represent one of the 14 design points used in the uncertainty study. Thus confidence bounds on the point estimates are obtained by making repeated calculations of the point estimates, while varying the median values of all variables according to the experimental design. This process is quite expensive for the seismic analysis, because new sets of responses must be computed for each repeated calculation, and the responses are obtained from detailed finite element structural analyses which are in themselves quite expensive.

5.4 Estimates of Confidence Bounds for the USI-A12 Results

An uncertainty analysis involving the 14 multiple calculations had earlier been performed to obtain confidence bounds for an SSMRP risk analysis of Zion for a slightly different base case than analyzed in this report. These results were presented at the 10th Water Reactor Safety Information Meeting. Because of time and dollar constraints, no confidence bounds were computed for the point estimates of radioactive release for the degraded support case presented in this report. However, using the earlier results (for which a report is currently being prepared) we have estimated confidence bounds for the USI-A12 results calculated in this study.

The case for which confidence bounds were previously computed differs in some respects from the cases analyzed in this report. The major differences are:

1. Structural failures of the crib house pump enclosure roof, diesel generator shear wall and inter-building piping failures due to rocking of the containment building were not included in the base cases analyzed in this report.

2. The tactic of using "feed and bleed" to cool the core was not considered in the cases analyzed in this report.

The "feed and bleed" capability was not included since it is not a universally accepted means of cooling the core, and requires effective operator interpretation and action which (following a large earthquake) may not be possible. The structural failures were not included since they were judged to be failure modes particular to the Zion plant, and their presence would unnecessarily limit any generic interpretations of the results.

The effect of the structural failures was assessed by a separate calculation, in which the structural failures mentioned above were added into Case 1, the fully degraded support case. Table 15 compares Case 1 with and without the structural failures. As can be seen, the dose increased from 32.2 man-REM/yr to 41.1 man-REM/yr. In spite of the differences between the cases analyzed in this report and the case for which confidence bounds were computed, the various point estimate cases evaluated recently in the SSMRP would tend to indicate that differences in assumptions as to structural failures, "feed and bleed" capability, etc., are not so great as to preclude use of the one set of calculated confidence bounds as estimates of bounds for other (similar) cases.

To estimate confidence bounds, we used Fig. 21 which shows the results of the 14 repeated calculations of the case which included structural failures and "feed and bleed." Each of these point estimates were considered equally likely. The solid circles on Fig. 21 are the 14 point values of core melt* frequency per year. The solid curve fit through these 14 points represents an approximation to the distribution function. Using this solid curve, the median (50% probability) core melt frequency is 2×10^{-5} per year. Also from the solid curve, the 90% probability value of core melt frequency is seen to be 4×10^{-4} per year. The point estimate core melt frequency based on random uncertainties only (analogous to the results computed in this study) was found to be 7.6×10^{-6} per year for the base case for which confidence bounds were computed.

*Defined here as the sum of all the release category probabilities.

Table 15. Probability per year of release categories for each earthquake level - with and without structural failures.

Case 1: Weak Supports (no structural failures)

Release Category	1	2	3	4	5	6	Total	Man-REM/yr
1	0	0	0	9.7E-8	2.5E-8	9.7E-9	1.32E-7	0.71
2	0	0	1.4E-9	1.5E-6	3.9E-7	1.5E-7	2.04E-6	9.79
3	4.5E-7	2.2E-6	1.3E-7	0	0	0	2.8E-6	15.12
4	0	0	0	0	0	0	0	0
5	0	0	0	0	0	0	0	0
6	0	0	0	0	0	0	0	0
7	4.5E-5	2.2E-4	1.3E-5	0	0	0	2.8E-4	6.4

Total = 2.85E-4 32.02

Case 1: Weak Supports (with structural failures)

Release Category	1	2	3	4	5	6	Total	Man-REM/yr
1	5.9E-11	1.6E-8	5.6E-8	9.7E-8	2.5E-8	9.7E-9	2.0E-7	1.10
2	1.7E-8	6.7E-7	8.8E-7	1.5E-6	3.9E-7	1.5E-7	3.9E-6	18.67
3	4.5E-7	2.2E-6	1.2E-7	0	0	0	2.8E-6	14.96
4	0	0	0	0	0	0	0	0
5	0	0	0	0	0	0	0	0
6	0	0	0	0	0	0	0	0
7	4.5E-5	2.2E-4	1.2E-5	0	0	0	2.8E-4	6.37

Total = 2.86E-4 41.10

Fig. 21. Probability of core melt frequency (per year) based on inclusion of systematic uncertainty.

Thus the ratio between the (random only) point estimate and the median core melt frequencies is

$$\frac{\text{Median Core Melt}}{\text{Point Estimate}} = \frac{2 \times 10^{-5}}{7.6 \times 10^{-6}} = 2.6.$$

The ratio between the 10% and 90% probability values of core melt frequency and the median core melt frequency is:

$$\frac{90\% \text{ Probability of Core Melt}}{\text{Median Core Melt}} = \frac{4 \times 10^{-4}}{2 \times 10^{-5}} = 20,$$

$$\frac{10\% \text{ Probability of Core Melt}}{\text{Median Core Melt}} = \frac{3 \times 10^{-7}}{2 \times 10^{-5}} = \frac{1}{66}.$$

5.5 Estimated Confidence Limits

Using the two ratios developed above, we can estimate confidence limits on the point estimates of dose computed for the degraded support capacity cases considered in this report. Thus we obtain:

Case 1

Point Estimate = 32.0 man-REM/year

Median = 32.0(2.6) = 83.2 man-REM/year

90% Upper Bound = 32.0(2.6)(20) = 1664 man-REM/year

10% Lower Bound = 32.0(2.6)(1/66) = 1.3 man-REM/year;

Case 2

Point Estimate = 11.8 man-REM/year

Median = 11.8(2.6) = 30.7 man-REM/year

90% Upper Bound = 11.8(2.6)(20) = 613 man-REM/year

10% Lower Bound = 11.8(2.6)(1/66) = 0.5 man-REM/year;

Base Case

Point Estimate = 6.6 man-REM/year

Median = $6.6(2.6) = 17.2$ man-REM/year

90% Upper Bound = $6.6(2.6)(20) = 343$ man-REM/year

10% Lower Bound = $6.6(2.6)(1/66) = 0.3$ man-REM/year.

Again, it is reiterated that the median, upper and lower bound values listed above are estimated from the results of previous (similar) calculations, and not the results of calculations performed for this study.

Section 6: 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. D. L. Bernreuter, Scaling and Estimation of Earthquake Ground Motion as a Function of the Earthquake Source Parameters and Distance, NUREG/CR-2103, April 1981.
4. 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.
5. D. A. Wesley, and P. S. Hashimoto, Seismic Structural Fragility Investigation for the Zion Nuclear Power Plant, Seismic Safety Margins Research Program (Phase I), Lawrence Livermore National Laboratory, Livermore, CA, UCRL-15380, NUREG/CR-2330 (1981).
6. Subsystem Hardness Assurance Report, U.S. Army Corps of Engineers, Huntsville Division, HNDDSP-72-156-ED-R (1975), Vols. I and II.
7. N. M. Newmark, "Inelastic Design of Nuclear Reactor Structures and Its Implications on Design of Critical Equipment," SMiRT Paper K 4/1, 1977 SMiRT Conf., San Francisco, CA.
8. A. S. Veletsos and N. M. Newmark, "Effect of Inelastic Behavior on the Response of Simple Systems to Earthquake Motions," Proc. 2nd World Conf. on Earthquake Engineering, Tokyo, Japan, 1960, pp. 895-912.
9. N. M. Newmark and W. J. Hall, "Building Practices for Disaster Mitigation," in Procedures and criteria for earthquake resistant design, National Bureau of Standards, Washington, D.C., Building Science Series 46, Vol. 1, February 1973, pp. 209-236.
10. N. M. Newmark, "A response spectrum approach for inelastic seismic design of nuclear reactor facilities," Trans. 3rd Intern. Conf. on Structural Mechanics in Reactor Technology, London, paper K5/1,1 Vol. 4, Part K, 1975.

11. R. Ridell and N. M. Newmark, Statistical Analysis of the Response of Nonlinear Systems Subjected to Earthquakes, Department of Civil Engineering Report UILU 79-2016, Urbana, IL (1979).
12. L. L. George and R. W. Mensing, "Using Subjective Percentiles and Test Data for Estimating Fragility Functions," paper presented at DOE Statistical Symp., October 29-31, 1981, Berkeley, CA.
13. WTD-SM-74-097, 51 Series Steam Generator, Generic Seismic Analysis, Westinghouse Electric Corp., Nuclear Energy Systems, Large Components Division, Tampa, FL, Dec. 1974.
14. Final Safety Analysis Report (FSAR), Zion Station, Commonwealth Edison Company.
15. D. O. Harris, et al., Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant, Vol. 5, NUREG/CR-2189, 1981.
16. J. M. Barsom, "Development of the AASHTO Fracture Toughness Requirements for Bridge Steels," Engineering Fracture Mechanics, Vol. 7, 1975, pp. 605-618.

A P P E N D I X A

Table A-1. RVR Release Category Assignments

Seq. No.	α	β	γ	δ	ϵ
1	3(.01)				7(.99)
2	3(.01)				7(.99)
3	3(.01)			2(.49)	7(.50)
4	3(.01)			2(.49)	7(.50)
5	3(.06)				7(.94)
6	1(.06)	2(*)	2(.91)	2(.03)	
7	1(.06)	2(*)	2(.91)	2(.03)	

*Fraction determined by calculated isolation valve failure. Less than 0.01.

Table A-2. LLOCA Release Category Assignments

Seq. No.	α	β	γ	δ	ϵ
2	3(.01)	5(*)			7(.99)
4	3(.01)	5(*)			7(.99)
6	1(.01)	2(*)	2(.12)	2(.08)	6(.78)
7	1(.01)			3(.99)	
8	3(.01)	5(*)		7(.49)	7(.50)
9	3(.01)	5(*)			7(.99)
10	3(.01)	5(*)			7(.99)
11	3(.01)	2(*)			6(.99)
12	3(.01)	4(*)		7(.39)	7(.60)
13	3(.01)	5(*)			7(.99)
14	3(.01)	5(*)			7(.99)
15	3(.01)	5(*)			6(.99)
16	3(.01)	4(*)		7(.39)	7(.60)
18	3(.01)	5(*)			7(.99)
20	1(.01)	5(*)	2(.24)	2(.56)	6(.19)
21	1(.01)			3(.99)	
22	3(.01)	5(*)		7(.60)	7(.39)
23	1(.01)	4(*)			7(.99)
24	1(.01)	4(*)	2(.12)	2(.04)	6(.82)
25	1(.01)	4(*)		7(.39)	7(.60)
26	1(.01)	4(*)			7(.99)
27	1(.01)	4(*)	2(.12)	2(.04)	6(.82)
28	1(.01)	4(*)		7(.49)	7(.50)

*Fraction determined by calculated isolation valve failure. Less than 0.01.

Table A-3. MLCCA Release Category Assignments

Seq. No.	α	β	γ	δ	ϵ
2	3(.01)				7(.99)
4	3(.01)				7(.99)
6	1(.01)		2(.12)	2(.08)	6(.73)
7	1(.01)			3(.99)	
8	3(.01)	5(*)		7(.49)	7(.49)
9	3(.01)				7(.99)
10	3(.01)				7(.99)
11	3(.01)				6(.99)
12	3(.01)			3(.39)	7(.60)
13	3(.01)				7(.99)
14	3(.01)				7(.99)
15	3(.01)				6(.99)
16	3(.01)			3(.39)	7(.60)
18	3(.01)				7(.99)
20	1(.01)		2(.24)	2(.56)	6(.19)
21	1(.01)			3(.99)	
22	3(.01)			7(.60)	7(.39)
23	1(.01)			3(.39)	7(.60)
24	1(.01)		2(.12)	2(.04)	6(.83)
25	1(.01)			3(.39)	7(.60)
26	1(.01)			3(.39)	7(.60)
27	1(.01)		3(.12)	2(.04)	6(.83)
28	1(.01)			7(.49)	7(.50)
29	3(.01)				7(.99)
30	3(.01)				7(.99)
31	3(.01)	2(*)			6(.99)
32	3(.01)	4(*)			7(.99)
33	1(.01)				7(.99)
34	1(.01)	2(*)		2(.39)	7(.99)
35	1(.01)	4(*)		3(.39)	7(.99)

*Fraction determined by calculated isolation valve failure. Less than 0.01.

Table A-4. SLOCA Release Category Assignments

Seq. No.	α	β	γ	δ	ϵ
2	3(.01)				7(.99)
4	3(.01)				7(.99)
6	1(.01)		2(.12)	2(.08)	6(.78)
7	1(.01)			3(.99)	
8	3(.01)	5(*)		7(.49)	7(.49)
9	3(.01)				7(.99)
10	3(.01)				7(.99)
11	3(.01)				6(.99)
12	3(.01)			3(.39)	7(.60)
13	3(.01)				7(.99)
14	3(.01)				7(.99)
15	3(.01)				6(.99)
16	3(.01)			3(.39)	7(.60)
18	3(.01)				7(.99)
20	1(.01)		2(.24)	2(.56)	6(.19)
21	1(.01)			3(.99)	
22	3(.01)			7(.60)	7(.39)
23	1(.01)			3(.39)	7(.60)
24	1(.01)		2(.12)	2(.04)	6(.82)
25	1(.01)			3(.39)	7(.60)
26	1(.01)			3(.39)	7(.60)
27	1(.01)		2(.12)	2(.04)	6(.82)
28	1(.01)			7(.49)	7(.49)
29	3(.01)				7(.99)
30	3(.01)				7(.99)
31	3(.01)	2(*)			6(.99)
32	3(.01)	4(*)			7(.99)
33	1(.01)				7(.99)
34	1(.01)	2(*)	3(.39)		6(.60)
35	1(.01)	4(*)			7(.99)

*Fraction determined by calculated isolation valve failure. Less than 0.01.

Table A-5. SSLOCA Release Category Assignments

Seq. No.	α	β	γ	δ	ϵ
2	3(.01)	5(*)			7(.99)
4	3(.01)	5(*)			7(.99)
6	1(.01)	2(*)	2(.12)	2(.08)	6(.78)
7	1(.01)			3(.99)	
8	3(.01)	5(*)		7(.04)	7(.95)
9	3(.01)	5(*)			7(.99)
10	3(.01)	5(*)			7(.99)
11	3(.01)	2(*)		3(.04)	6(.95)
12	3(.01)	4(*)		3(.39)	7(.60)
13	3(.01)	5(*)			7(.99)
14	3(.01)	5(*)			7(.99)
15	3(.01)	2(*)		3(.04)	6(.95)
16	3(.01)	4(*)		3(.39)	7(.60)
18	3(.01)	5(*)			7(.99)
20	1(.01)	6(*)	2(.24)	2(.56)	36(.19)
21	1(.01)	6(*)	2(.12)	3(.99)	
22	1(.01)			7(.60)	7(.39)
23	1(.01)	2(*)	2(.12)	2(.04)	6(.83)
24	1(.01)	2(*)	2(.12)	2(.04)	6(.83)
25	1(.01)	2(*)	2(.12)	2(.04)	6(.83)
26	1(.01)	2(*)	2(.12)	2(.04)	6(.83)
27	1(.01)	2(*)	2(.12)	2(.04)	6(.83)
28	1(.01)	2(*)	2(.12)	2(.04)	6(.83)
29	3(.01)	5(*)			7(.99)
30	3(.01)	5(*)			7(.99)
31	3(.01)	2(*)		2(.04)	6(.95)
32	3(.01)	4(*)		4(.39)	7(.60)
33	1(.01)	2(*)	2(.12)	2(.04)	6(.83)

*Fraction determined by calculated isolation valve failure. Less than 0.01.

Table A-5. SSLOCA Release Category Assignments (continued)

Seq. No.	α	β	γ	δ	ϵ
34	1(.01)	2(*)	2(.12)	2(.04)	6(.83)
35	1(.01)	2(*)	2(.12)	2(.04)	7(.83)
36	3(.01)	5(*)	5(.12)	5(.04)	7(.83)
37	3(.01)	5(*)	5(.12)	5(.04)	7(.83)
38	2(.01)	3(*)		3(.04)	6(.95)
39	3(.01)	4(*)			7(.99)
40	1(.01)	2(*)		2(.04)	6(.95)
41	1(.01)	2(*)		2(.04)	6(.95)
42	1(.01)	2(*)		2(.04)	6(.95)

*Fraction determined by calculated isolation valve failure. Less than 0.01.

Table A-6. T1 Release Category Assignments - Earthquake Levels 1, 2, 3.

Seq. No.	α	β	γ	δ	ϵ
7	3(.01)	5(*)			7(.99)
8	3(.01)	5(*)			7(.99)
9	3(.01)	5(*)			7(.99)
10	3(.01)	5(*)			7(.99)
14	3(.01)	5(*)			7(.99)
15	3(.01)	5(*)			7(.99)
16	3(.01)	5(*)			7(.99)
20	3(.01)	5(*)			7(.99)
21	3(.01)	5(*)			7(.99)
22	3(.01)	5(*)			7(.99)
23	3(.01)	5(*)			7(.99)
24	3(.01)	5(*)			7(.99)

*Fraction determined by calculated isolation valve failure. Less than 0.01.

T1 Release Category Assignments - Earthquake Levels 4, 5, 6.

Seq. No.	α	β	γ	δ	ϵ
7	1(.01)		2(.24)	2(.56)	6(.19)
8	1(.01)		2(.24)	2(.56)	6(.19)
9	1(.01)		2(.24)	2(.56)	6(.19)
10	1(.01)		2(.24)	2(.56)	6(.19)
14	1(.01)		2(.24)	2(.56)	6(.19)
15	1(.01)		2(.24)	2(.56)	6(.19)
16	1(.01)		2(.24)	2(.56)	6(.19)
20	1(.01)		2(.24)	2(.56)	6(.19)
21	1(.01)		2(.24)	2(.56)	6(.19)
22	1(.01)		2(.24)	2(.56)	6(.19)
23	1(.01)		2(.24)	2(.56)	6(.19)
24	1(.01)		2(.24)	2(.56)	6(.19)

Table A-7. T2 Release Category Assignments - Earthquake Levels 1, 2, 3

Seq. No.	α	β	γ	δ	ϵ
4	3(.01)	5(*)			7(.99)
5	3(.01)	5(*)			7(.99)
6	3(.01)	5(*)			7(.99)
7	3(.01)	5(*)			7(.99)
11	3(.01)	5(*)			7(.99)
12	3(.01)	5(*)			7(.99)
13	3(.01)	5(*)			7(.99)
14	3(.01)	5(*)			7(.99)
15	3(.01)	5(*)			7(.99)

*Fraction determined by calculated isolation valve failure. Less than 0.01.

T2 Release Category Assignments - Earthquake Levels 4, 5, 6.

Seq. No.	α	β	γ	δ	ϵ
4	1(.01)		2(.24)	2(.56)	6(.19)
5	1(.01)		2(.24)	2(.56)	6(.19)
6	1(.01)		2(.24)	2(.56)	6(.19)
7	1(.01)		2(.24)	2(.56)	6(.19)
11	1(.01)		2(.24)	2(.56)	6(.19)
12	1(.01)		2(.24)	2(.56)	6(.19)
13	1(.01)		2(.24)	2(.56)	6(.19)
14	1(.01)		2(.24)	2(.56)	6(.19)
15	1(.01)		2(.24)	2(.56)	6(.19)

DISCLAIMER

This document was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

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

DO NOT MICROFILM
THIS PAGE

Available from
GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
and
National Technical Information Service
Springfield, Virginia 22161

Technical Information Department · Lawrence Livermore Laboratory
University of California · Livermore, California 94550

DO NOT REMOVE
THIS PAGE
FROM FILE