NUREG/CR-3300
SAND83-1118
Vol. 1

Review and Evaluation of the Zion Probabilistic Safety Study

Plant Analysis

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U.S. Nuclear Regulatory
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Manuscript Completed: March 1984
Date Published: May 1984

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NRC FIN A1125

ABSTRACT

This report describes the review of the internal and external event plant analyses of the Zion Probabilistic Safety Study (ZPSS). The review was conducted by Sandia National Laboratories. The purpose of the review was to search for areas in the ZPSS where omissions and critical judgments were made which could impact the quantitative results. The review identified several of these areas.

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ACKNOWLEDGMENT

The authors wish to thank Scott Newberry of the U.S. Nuclear Regulatory Commission for his comments and guidance during this program. We also wish to thank Emily Preston for her help in typing and assembling this report.

Acknowledgment is also due to Commonwealth Edison and Pickard, Lowe, and Garrick personnel whose comments led to correction of several errors in our analysis. It should be noted that substantial areas of disagreement remain between us and the above-mentioned parties; some of these are addressed in Appendix E of Volume I. It should not be inferred that they concur with any specific part of this report.

SUMMARY

S.1 Introduction

This summary describes the Sandia National Laboratories review of the Zion Probabilistic Safety Study (ZPSS) for the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission (NRC). The review addressed the ZPSS systems analysis and external events analysis (i.e., plant analysis). The review of the ZPSS containment and consequence analysis was conducted by Brookhaven National Laboratory. Brookhaven's review is published under a separate cover.

The primary purpose of the review was to search for significant omissions and critical judgments in the ZPSS and to evaluate the impact of these on the results of the study. The evaluation resulted in revised estimates for some plant damage state frequencies. The inherently negative focus of the review is reflected in this report. However, it should be noted that, in general, the ZPSS was considered to be a comprehensive and competent analysis.

The reader should be aware that disagreements exist between the ZPSS analysts and the reviewers. The draft of this document was given to Commonwealth Edison and its consultants and they responded with many helpful comments which resulted in our amending the review. There are points, however, over which the two teams disagree. Those items of disagreement pertaining to the plant analysis are presented in Appendix E in which the relevant sections of the comments of the ZPSS analysts to the draft are reproduced, and our response to them is given.

S.2 Uncertainties in Results

The uncertainties involved in both the ZPSS and in our review are reflected in uncertainty bounds placed on numerical estimates, in the discussion of assumptions, conservatisms and unconservatisms in the analysis, and in the review of sensitivity issues. Most of the uncertainty treatment is omitted in this summary, in the interest of brevity. However, the reader should be aware that such uncertainties exist and should view the quantitative information provided accordingly. The major sources of uncertainty in a probabilistic risk assessment are:

- Limitations in the modeling and analysis methods.
- Limitations in the data available on nuclear power plant component and system failure rates.

- Lack of information regarding plant and operator responses to various accident conditions.

- Assumptions made to facilitate the analysis in light of these limitations.

S.3 Areas of Review

The ZPSS, as any Probabilistic Risk Assessment (PRA), is composed of several interrelated tasks. A review of a PRA is not complete unless the information and analysis which comprises each task is examined. The ZPSS PRA tasks are depicted in Figure S-1. Also shown are the report sections which summarize our review of a task. Tasks reviewed by Brookhaven National Laboratories (BNL) are so noted. As can be seen, we did not review the first task, "initial information collection." Our review assumes that the ZPSS has collected accurate Zion design and operations information; e.g., correct piping and instrumentation layouts, etc.

The findings of our review are ultimately expressed quantitatively in terms of the effect they have on ZPSS damage state frequencies. Damage states are, in essence, functional classifications of core-melt accidents. Classification of core-melt accidents functionally is necessary to perform the containment and consequence analysis performed by BNL. The ZPSS defined 21 plant damage states. These can be grouped as follows: (1) SEFC, SEC, SEF, AEFC, AEC, AEF, TEFC, TEC, TEF; (2) SE, TE, AE; (3) SLFC, SLC, SLF, ALFC, ALC, ALF; (4) SL, AL; (5) V. The nomenclature is: S or A denote small or large LOCA and T denotes transient, E or L denote early or late core melt, F and C denote fans and sprays working respectively, and V denotes an interfacing systems LOCA. These five groups of plant damage states can be qualitatively described as follows: (1) early core melt with containment cooling; (2) early core melt without containment cooling; (3) late core melt with containment cooling; (4) late core melt without containment cooling; and (5) containment bypass before core melt.

S.4 Initiating Events

The initiating events covered in the ZPSS seemed to be generally complete. Comparisons were made to other PRAs (e.g., ANO IREP [2], etc.), an NRC list of concerns about potentially omitted initiating events, and EPRI NP 801. [1] In addition, several initiating events were identified by NRC as being of particular interest. Review of this list of potential ZPSS initiating events has indicated that pressurized thermal shock, shutdown events, and loss of component cooling water due to a pipe break appear to be the only potentially significant initiating events omitted in

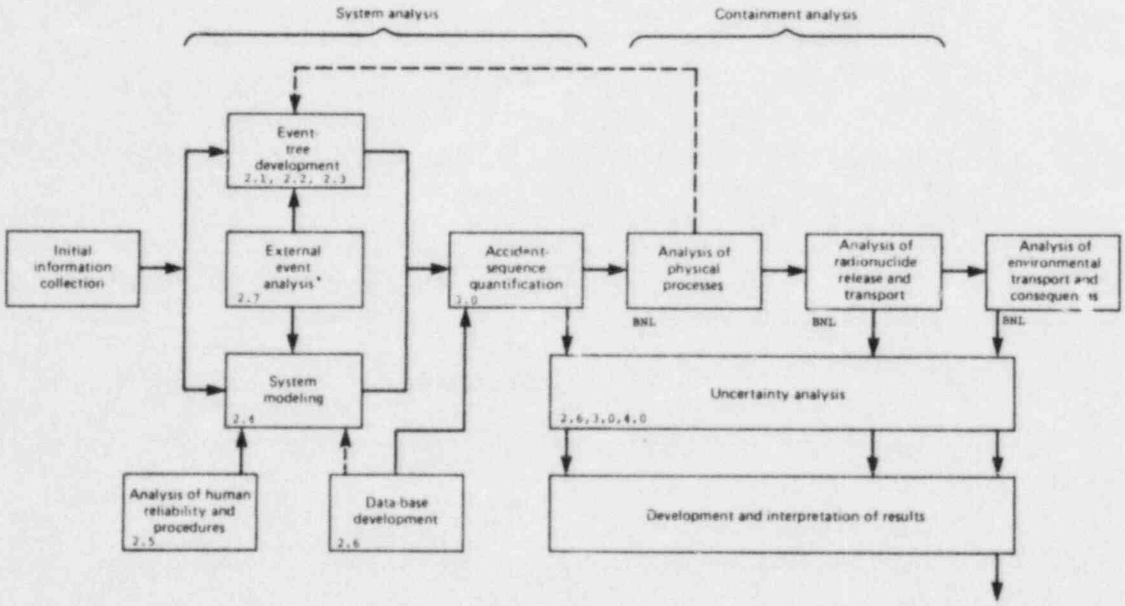

Figure S-1. **Risk Assessment Procedure** the ZPSS. Also the ZPSS treatment of a DC power bus initiating event appears to be inappropriate.

It should be noted that seven external initiating events (seismic, fire, flood, wind, aircraft accidents, explosions from transportation of hazardous materials, and turbine missiles) were considered, which is more than most PRAs have attempted.

S.5 Event Trees

The ZPSS constructed 14 event trees to model the plant system response to the internal initiating events. We reviewed these trees for validity. The event tree findings are briefly discussed in this section and are categorized according to the topics underlined below.

Core-Melt/Safety System Interactions

The interdependencies incorporated into the Zion event trees imply that the containment spray and fan cooler systems may be utilized during a core-melt accident. This is an important assumption since the Zion analysis predicts that the operation of these systems can significantly reduce the risk associated with a core-melt accident. Since this issue is currently being addressed in several NRC and Sandia equipment qualification research programs, we did not attempt to resolve this assumption. Rather, we performed a sensitivity analysis in order to gauge the possible effect this could have on the final quantitative results. The results of this investigation are presented later in this summary.

Sodium Hydroxide Addition

All event trees model the additions of sodium hydroxide to the containment spray water. Discussions with ZPSS personnel revealed that analysis performed late in the study indicated that sodium hydroxide addition had a negligible effect on the assessment of plant damage states and release categories. All event trees could therefore be simplified by removal of the sodium hydroxide addition event.

Main Feedwater System

The Zion study conservatively assumed that the main feedwater system was unavailable for purposes of removing post-shutdown decay heat following all internal and external initiating events analyzed. We assessed that this conservatism was unwarranted and thus gave credit for main feedwater restoration for many sequences.

Core Melts Caused By Containment Overpressure Failure

The Zion event trees do not model core melts caused by containment overpressure failure. These sequences have been shown to be important in other PRAs (e.g., the S_2C sequence in WASH-1400). However, an assessment of these potential sequences indicates that for Zion the impact is negligible.

Transient Induced Pressurizer Safety Valve Demands

The ZPSS event trees do not model the demand of the pressurizer safety valves in response to a transient. Based on an estimate of the frequency of this event, however, it is not believed that any important accident sequences were missed.

Event Tree 11c--Turbine Trip Due to a Loss of Service Water and Event Tree 13b--Reactor Trip Due to a Loss of Component Cooling Water

The ZPSS used the turbine trip and reactor trip event trees to model the plant response to a loss of service water and loss of component cooling water initiating events respectively. These event trees do not completely model the plant response to these initiating events for the following reasons:

1. The trees do not allow for a reactor coolant pump (RCP) seal LOCA to occur following a sustained loss of component cooling or service water.

2. The safety systems which respond to a seal LOCA are not fully modeled.

3. Station blackout initiated by a loss of service water followed by a loss of off-site power is not modeled.

If a loss of component cooling occurs, the RCP seals will lose cooling due to failure of the charging pumps [3] and cooling to the thermal barrier heat exchanger. The ZPSS predicts a 1200 gpm seal LOCA will occur approximately 30 minutes following a loss of seal cooling. Since component cooling also cools the safety injection pumps, they would be expected to subsequently fail. A core melt would ensue leading to an SEFC plant damage state. The ZPSS omitted quantification of such a sequence. This omission was found to have a significant effect on the quantitative results. This can be noted via examination of our revised dominant sequences presented in Section S.11 of this summary, i.e., the sequence with the highest frequency is caused by a loss of the component cooling water system.

S.6 Mitigating Systems Success Criteria

In response to LOCA and transient initiating events, various Zion core cooling and containment systems are called upon to bring the plant to a safe shutdown condition. If core cooling is unsuccessful and a core melt ensues, the containment systems may be able to reduce the consequences of the accident by maintaining the containment boundary and thus isolating the core melt from the environment. The combinations of plant systems required to cool the core and maintain the containment boundary constitute the Zion mitigating system success criteria. A review of the success criteria employed in the ZPSS indicates that they are consistent with criteria employed in PRAs of similar plants.

In addition to the major core cooling and containment system success criteria discussed above, the ZPSS developed a variety of support system success criteria. These support systems must succeed to allow successful operation of the core cooling and containment systems. Support systems include pump cooling systems, electric power systems, and the plant operators. We reviewed these criteria with the aid of the FSAR, previous PRA analyses, and discussions with the ZPSS analysts. Some apparent problems were identified and are discussed below in Section S.7.

S.7 Fault Trees

The system fault trees presented in the ZPSS were reviewed for accuracy and completeness. The review of Zion fault trees included an examination of the trees themselves, the supercomponent arrangement and definition, and the system calculations. For the most part, these appear to be complete and correct. However, a few important exceptions are noted below in comments on the individual fault trees. (The most important exception was found in the component cooling water system fault tree.)

<u>Emergency Electric Power System</u> - Except for the treatment of loss of DC power as an initiating event, the analysis was found to be appropriate.

<u>Reactor Protection System</u> - The ZPSS analysis of this system and the resulting failure probability were found to be appropriate.

<u>Safeguards Actuation System</u> - The analysis of this system was found to be appropriate. However, in the system application to the small LOCA event tree, credit was given for actuation from high containment pressure, which does not appear to be a viable actuation mode for small LOCAs.

High Pressure Injection System - The ZPSS analysis appears to be correct except that common mode pump failure was considered negligible. We believe Atwood [4] common mode factors are appropriate in this case and applied them in a reanalysis.

Feed and Bleed - Subsequent to the ZPSS analysis, the PORV block valves at Zion were changed from normally closed to normally open. A substantial portion of the failure probability for feed and bleed in the ZPSS was based on failure of the block valves to open. Consequently, the feed and bleed failure estimate was reduced in our analysis.

Low Pressure Injection System - The ZPSS analysis of this system was found to be appropriate. A different common mode factor was used in our reevaluation, resulting in a minor change to the results.

Recirculation Systems - The analysis of the high pressure, low pressure and recirculation systems was found to be appropriate.

Containment Spray Injection System - The ZPSS analysis of this system was found to be generally appropriate. A common mode factor was applied to the pump trains in our reevaluation, resulting in a minor change to the results.

Containment Fan Cooling System - The ZPSS analysis of this system was found to be appropriate except that a single manual valve in the return line of the fan cooling coils was not modeled. Also, the analysis assumed that the system could function in a post core-melt environment. We agree with this assumption, but addressed it as a sensitivity issue.

Component Cooling Water System - The ZPSS analysis for this system assumed that three component cooling water pumps were required for system success, except for reactor pump seal cooling (for which only one pump was assumed to be necessary). We believe that two pumps would be required. This is an important assumption since component cooling water appears in many dominant accident sequences. Therefore, we performed a reevaluation. It was assumed in the ZPSS, that component cooling water is not required for charging and safety injection pump cooling. We found from Reference 5 that these pumps do need cooling from the component cooling system. This has a large impact on core-melt frequency. This can be realized via examination of our revised dominant sequences presented in Section S.11 of this summary, i.e., many of these sequences involve failure of the component cooling water system.

Service Water System - The ZPSS analysis of this section assumed that three service water pumps would be required for system success. Based on information in the Zion FSAR, we assume two pumps are sufficient, and reevaluated the system accordingly.

Auxiliary Feedwater System - The ZPSS analysis of this system was found to be poorly described, but generally correct. The ZPSS analysis assumed that common mode pump failure was negligible. We included common mode failure in our reanalysis. This had a relatively minor impact on the results.

S.8 Human Reliability Analysis

The human reliability analysis in the ZPSS was impressive in terms of its scope and level of effort. Nevertheless, several situations were found in which the human error estimates were judged to be higher or lower than we believe to be appropriate. The net effect is believed to be overall optimism in human error treatment. A complete evaluation was not possible because of insufficient documentation. Suggested revisions to the ZPSS human error rate estimates are presented in the main report.

S.9 Estimation Methods

The treatment of uncertainty associated with estimates from existing data sources appears to be inconsistent. Generally 5 percent and 95 percent bounds from WASH-1400 were used as 20 percent and 80 percent limits in ZPSS. Notable exceptions to this were the treatment of interfacing system LOCAs, pressure vessel rupture, and pipe ruptures. In all three cases, substantially higher estimates would have been obtained had their general rule been followed. The results are highly sensitive to this assumption. The Bayesian methodology used to estimate accident sequence rates was evaluated. Where Zion data exist and are used to modify the ZPSS's prior probability distributions, the effect of the prior distributions is generally unimportant with respect to the estimated accident sequence rates. Where Zion data are not available or used, the estimates are quite sensitive to the assumed prior distribution. Suggested revisions to the ZPSS uncertainty ranges are presented in the main report.

S.10 External Events

The ZPSS treatment of external events is more comprehensive than most PRAs. Events addressed include seismic, floods, tornados, fire, turbine missiles, aircraft accidents, and explosions from transportation of hazardous materials. The following comments arise from our review of the external events sections:

Seismic - The seismic analysis was, in general, difficult to review due to lack of clear documentation. Among the concerns noted were:

- The choice of boundaries of seismogenic zones and rate of seismic activity.

- The imposition of an upper bound on effective peak acceleration.

- The definition of damage effective ground acceleration.

- The treatment of seismic events only as opposed to combinations of seismic and nonseismic events.

However, the results were considered acceptable, within the limits of the uncertainties which apply to this type of analysis.

Fire - The ZPSS fire analysis was reviewed and several problems were noted. Specifically, the fire analysis

- Only analyzed the auxiliary equipment room and the cable spreading room. Other important areas such as the auxiliary building zone and the component cooling water pump room were either assessed qualitatively or not addressed at all.

- Did not address seal LOCA events caused by loss of component cooling water.

- Did not consider the loss of service water or component cooling water components by fire in conjunction with loss of redundant components due to maintenance.

The resolution of these problems was deemed to be beyond the scope of the review. A reanalysis of this subject is recommended.

Other External Events - The ZPSS did not identify any specific external events other than seismic and fire which had a significant effect on risk. We found no reason to disagree with this position.

S.11 Accident Sequence Analysis

In this section, sequences we identified as dominant are discussed. These include the sequences which, by our estimates, dominate core-melt frequency or plant damage state frequency. We identified 14 such sequences (Table 1). Of these sequences five are on the list of dominant accident sequences presented in the ZPSS, i.e., the sequences in Table S-1 with the asterisk. The remaining nine are sequences which did not appear on the ZPSS list either because they were not interpreted as leading to core melt or because the frequency calculated in the ZPSS was not high enough to consider them dominant. The plant damage state used in the tables is: S or A denote small or large LOCA and T denotes transient, E or L denote early or late core melt, F and C denote fans and sprays working respectively.

We must stress that the accident sequences involving loss of component cooling water are based on a system success criterion of two pumps operating. (Such sequences also use a service water success criterion of two pumps operating.) We have been given information by Commonwealth Edison that suggests that one CCW pump is sufficient but that three SW pumps are required. In this report, we consider these alternate criteria as a sensitivity issue (summarized in Section S.12). Furthermore, as this report goes to press, Zion personnel are reexamining these success criteria for specific situations. Thus, the reader should fully realize that the accident sequence frequencies presented here are potentially transient in nature.

S.11.1 Failure of Component Cooling Water (CCW), SEFC

The complete loss of component cooling water frequency is given in the ZPSS as 9.4(-4) per reactor year. This frequency was derived by a two stage Bayesian analysis. To date, no such events have occurred at Zion or any other plants considered in the ZPSS data base. No further information is provided by the ZPSS.

The ZPSS analysts assumed, based on information available, that failure of component cooling water did not lead directly to core melt, without additional system failures. Specifically, it was believed that loss of component cooling water did not cause failure of the safety injection pumps and charging pumps during the injection phase. Consequently, it was concluded that loss of component cooling water as an initiating event would result in core melt only if it were combined with independent failure of these pumps (or associated hardware).

Subsequent information from Commonwealth Edison [5] is that both charging pumps and safety injection pumps will fail "in a short period of time," given loss of component cooling water. On this basis the following sequence is applicable:

Table S-1

Revised Zion Dominant Accident Sequences

Rank with Respect to Core Melt	Sequence	Plant Damage State	Annual Frequency
1	CCW Failure (causing failure of all charging and SI pumps, seal LOCA)	SEFC	~2(-4)
2	Loss of off-site power: failure of component cooling water: failure to recover off-site power in 4 hours	SEFC	4.6(-5)
3	Loss of off-site power: failure of component cooling water: failure to recover off-site power in 1 hour	SEFC	4.0(-5)
4	Loss of off-site power, failure of component cooling water, failure to recover off-site power in 8 hours, failure of containment fans	SEC	1.8(-5)
5*	Small LOCA, failure of recirculation cooling	SLF	1.6(-5)
6	Loss of off-site power, failure of component cooling water, failure to recover off-site power in 8 hours	SEFC	7.9(-6)
7	Failure of DC Bus 112 (causing failure of 1 PORV and loss of AC Bus 149), failure of auxiliary feedwater	TEFC	~7(-6)
8*	Seismic: loss of all AC power	SE	5.6(-6)
9*	Large LOCA: failure of recirculation cooling	ALF	4.9(-6)
10*	Medium LOCA: failure of recirculation cooling	ALF	4.9(-6)

Table S-1 (continued)

Revised Zion Dominant Accident Sequences

Rank with Respect to Core Melt	Sequence	Plant Damage State	Annual Frequency
11	Loss of off-site power, failure of component cooling water, failure to recover off-site power in 8 hours, failure of containment sprays and fan coolers	SE	4.7(-6)
12*	Large LOCA: failure of low pressure Injection	AEFC	1.4(-6)
13	Loss of off-site power: failure of auxiliary feedwater: failure of feed and bleed: failure to restore off-site power in 4 hours	TEFC	1.1(-6)
14	Loss of off-site power: failure of auxiliary feedwater: failure of feed and bleed: failure to restore power in 1 hour	TEFC	1.0(-6)
	Interfacing system LOCA**	V	1.1(-7)

*Sequences identified by the ZPSS to be dominant.
**Included here because of its potential impact on con sequence analysis, not one of the dominant core-melt sequences.

1. Component cooling water is lost with consequent loss of cooling to the reactor coolant pump seal thermal barriers.

2. The two centrifugal charging pumps fail. We estimate that each pump would fail in about 5 minutes based on information received from Consolidated Edison for similar pumps during our Indian Point Safety Study Review. Since the pumps would be operated in succession, seal cooling would be lost approximately 10 minutes after CCW failure.

3. All four reactor coolant pump seals fail in about 30 minutes after loss of seal cooling with maximum loss of coolant through each seal of 300 gallons per minute (total 1,200 gallons per minute).

4. Both safety injection pumps are actuated by low reactor coolant pressure, and fail due to loss of cooling in about 5 minutes.

5. With loss of makeup capability through either the charging or safety injection pumps, core uncovery will ensue. A core-melt accident will be assured unless cooling to the safety injection pumps is restored in about 45 minutes.

An important assumption in the analysis of this sequence is the number of CCW pumps required for system success. The ZPSS indicated that only one pump would be required. However, based on a review of the CCW system loads in this situation, we believe that two pumps would be required. The frequency of this sequence is calculated to be ~2(-4).

S.11.2 Loss of Off-site Power: Loss of Component Cooling Water: Failure to Restore Power in 4 Hours. SEFC

In this sequence the initiating event, loss of off-site power, is followed by loss of component cooling water with failure to restore power in 4 hours. Given loss of component cooling water, a series of events leading to a seal LOCA with loss of makeup capability and thus to core melt will occur, as described in Section S.11.1 of this summary. The frequency of the events in this sequence is 4.6(-5).

S.11.3 Loss of Off-site Power: Loss of Component Cooling Water: Failure to Restore Power in 1 Hour SEFC

In this sequence the initiating event, loss of off-site power, is followed by loss of component cooling water, with failure to restore power in 1 hour. Given loss of component cooling water, a series of events leading to a seal LOCA with loss of makeup capability, and, thus, to core melt will occur, as described in Section S.11.1.

The calculation of this sequence frequency is the same as that described in Section S.11.2, except for the probability of failure to restore off-site power. The calculated frequency is 4.0(-5).

S.11.4 Loss of Off-site Power: Failure of Component Cooling Water: Failure to Restore Off-site Power in 8 Hours, Failure of Containment Fans, SEC

This sequence is the same as that described in Section S.11.6 except that it also includes containment fan system failure. Since the success criterion for the containment fan system is three of five fan coolers operating, the dominant cause of fan system failure in this sequence is loss of power from two of the three unit 1 AC buses. The frequency of this sequence is calculated as 1.8(-5).

S.11.5 Small LOCA: Failure of Recirculation Cooling, SLF

By Zion's estimates this accident is the most probable cause of core melt. Zion's dominant sequence occurs when AC power is available at all three buses and recirculation cooling fails (R-2). Zion's mean value for the probability of R-2 is given as 4.55(-4). Multiplying by the mean Small LOCA rate (3.54(-2) per year) and by the probability of power at all AC buses (~1) yields 1.6(-5) per year as the estimated core-melt rate for this sequence. As discussed elsewhere, the initiating event estimates (given as posterior means and variances) are reasonably consistent with the data presented and we agree with their estimate.

S.11.6 Loss Off-site Power: Loss of Component Cooling Water: Failure to Restore Power in 8 Hours, SEFC

In this sequence, the initiating event, loss of off-site power, is followed by loss of component cooling water, with failure to restore power in 8 hours. Given loss of component cooling water, a series of events leading to a seal LOCA with loss of makeup capability and thus to core melt will occur, as described in Section S.11.1. The calculated frequency of this sequence is 7.9(-6).

S.11.7 Failure of DC Bus 112, Failure of Auxiliary Feedwater, TEFC

Failure of DC Bus 112 would cause loss of main feedwater and reactor trip. It would also remove DC power from one of the two PORV's. Since Bus 112 provides control power for AC Bus 149, the auxiliary feedwater system would lose the availability of one motor-driven pump.

The sequence of interest in this case is failure of DC Bus 112, loss of main feedwater, reactor trip, loss of auxiliary feedwater, and failure of feed and bleed capability due to loss of one PORV. The sequence leads to core melt. The frequency of occurrence of the events in this sequence is calculated as ~7(-6).

S.11.8 Seismic: Loss of All AC Power, SE

In this sequence, a seismic event large enough to fail off-site power and the service water pumps occurs. Failure of the service water pumps causes subsequent failure of the diesel generators, due to lack of cooling. A loss of all AC power results followed by failure of RCP seal cooling and a RCP LOCA. Since safety injection and containment systems require AC power, a core melt ensues that results in damage state SE. The ZPSS frequency estimate for this sequence is 5.6(-6) per year. We conclude that this estimate is reasonable.

S.11.9 Large LOCA: Failure of Recirculation Cooling, ALF

For internally initiated accidents, the ZPSS assessed this sequence to be the second leading contributor to core melt. Zion's posterior distribution for the Large LOCA rate has a mean of 9.4(-4) per year and a variance of 5.7(-6), which seem consistent with the available data. Recirculation failure pertains to low pressure recirculation. The ZPSS assessed failure probability for this system also appeared reasonable. We therefore concur with their estimated rate for this sequence, i.e., 4.9(-6).

S.11.10 Medium LOCA: Failure of Recirculation Cooling, ALF

The ZPSS analysis and results for this sequence are identical to their treatment of the Large LOCA: Failure of Recirculation Sequence. Our comments are the same.

S.11.11 Loss of Off-site Power: Failure of Component Cooling Water: Failure to Restore Off-site Power in 8 Hours: Failure of Containment Sprays and Fan Coolers, SE

This sequence is similar to that described in Section S.11.4 except that it also involves containment sprays and containment fan system failure. The dominant cause of failure of both fans and sprays in this sequence is loss of power from AC buses in Unit 1. The frequency of this sequence is calculated as 4.7(-6).

S.11.12 Large LOCA: Failure of Low Pressure Injection, AEFC

The large LOCA initiating event estimates have been previously discussed. Failure of the low pressure injection system is given a mean value of 1.39(-3). The ZPSS analysis of this sequence appears to be appropriate. The frequency is calculated as 1.4(-6).

S.11.13 Loss of Off-site Power: Failure of Auxiliary Feedwater: Failure of Feed and Bleed: Failure to Restore Off-site Power in 4 Hours, TEFC

In this sequence, the initiating event, loss of off-site power, is followed by loss of auxiliary feedwater and loss of feed and bleed capability, with failure to restore power in 4 hours. The loss of auxiliary feedwater eliminates the capability for secondary cooling, since without off-site power the main feedwater pumps have tripped and cannot be restored. The loss of feed and bleed capability removes the remaining option for core cooling. The frequency of this sequence is calculated as 1.1(-6).

S.11.14 Loss of Off-site Power: Failure of Auxiliary Feedwater: Failure of Feed and Bleed: Failure to Restore Off-site Power in 1 Hour, TEFC

This sequence is similar to S.11.13, above. The frequency is calculated as 1.0(-6).

S.11.15 Event V: The Interfacing LOCA

Event V leads to release category 2 which, by Zion's risk estimates, is one of the dominating releases. The dominant V sequence is the joint failure of two motor-operated valves in the RHR suction path. The frequency calculated in the ZPSS is 1.1(-7). Though we disagree with the ZPSS model used to calculate this frequency, we concur with the frequency estimate. This is because our alternate model also predicted 1(-7) as the frequency estimate.

S.11.16 Other ZPSS Dominant Sequences

A number of accident sequences which appeared in the ZPSS dominant accident sequence list (ZPSS Table 8.10-1) have been omitted from the foregoing discussion because their importance to the plant damage state frequencies has diminished. This is primarily due to the fact that they have been supplanted by sequences which in our analysis, have higher frequencies of occurrence. Included in this group are the sequences numbered as 7, 8, 10, 11, 12, 13, 14, and 15 in the above-mentioned ZPSS table.

S.12 Special Issues

A number of special issues were addressed in the review. Generally, these represented assumptions made in our reanalysis of the ZPSS. To illustrate their impact, plant damage state frequencies based on each assumption were compared with the frequencies which would have resulted if the particular assumption had not been made. The special issues addressed, and a brief synopsis of the results of the investigation, are discussed in the following paragraphs:

1. Core Melt/System Interaction

The ZPSS, as well as our reanalysis of the ZPSS assumed that the containment fan cooler system would not fail due to the harsh environment expected within the containment following a core-melt accident. If it is assumed the fan coolers do fail, all accident sequences that were assessed to lead to the "late core melt with containment cooling" plant damage state are transferred to the "late core melt without containment cooling" plant damage state.

2. Feed and Bleed Core Cooling

Our reanalysis and the ZPSS gave credit for post shutdown decay heat removal via feed and bleed core cooling. If credit is not given for feed and bleed, the "early core melt with containment cooling" plant damage state increases by 48 percent.

3. Reactor Coolant Pump (RCP) Seal LOCA

The ZPSS and our reanalysis assumed that a 1200 gpm LOCA occurred via the RCP seals if all RCP seal cooling systems failed. If it is assumed a LOCA does not occur, the overall Zion core-melt frequency is reduced by approximately one order of magnitude.

4. Pump Room Cooling System Test

During our review we discovered that the Zion procedures did not include inspection or testing of pump room cooling system function. We were advised by plant personnel that a test procedure would be put in place. Our reanalysis therefore assumed that the test was in place. If it is assumed that room cooling is not tested, the overall Zion core-melt frequency could conceivably increase by a factor of 50. It should be noted, however, that pessimistic assumptions concerning operator recovery are necessary to yield this factor.

5. ZPSS Fire Analysis

We noted in our review that the ZPSS fire analysis appeared incomplete and, in some areas, inappropriate. We were unable to accurately reanalyze fire accidents at Zion due to lack of information. We performed a bounding analysis and showed that the "early core melt without containment cooling" plant damage state could conceivably increase by a factor of 7.

6. Anticipated Transients Without Scram (ATWS)

The ZPSS, as well as our reanalysis of the ZPSS, assessed that a turbine trip circuit is installed at Zion that can mitigate the affects of an ATWS. If it is assumed that this trip circuit is incapable of performing its designated function, the overall Zion core-melt frequency increases a factor of 2.

7. Success Criteria

Our reanalysis of the ZPSS assumed that two component cooling pumps and two service water pumps were required to prevent a RCP seal LOCA and subsequent failure of core cooling pumps. Zion personnel suggest that one component cooling pump and three service water pumps are required. It was found that the Zion assumption yields approximately the same plant damage state frequencies as we predict in this report.

S.13 Summary and Conclusions

S.13.1 General

In general, we found the systems analysis portion of the study to be consistent in scope and detail with ongoing probabilistic risk assessments. The scope of the external events analysis represents an advancement over what has been done in the past. We also commend the ZPSS analysis team for the utilization of plant-specific data in their analysis.

Section S.13.2 presents our recommended estimates of plant damage state frequencies for use in the containment and consequence analysis. These estimates reflect, to the degree possible given the limited scope of our review, our best judgment of these frequencies. Section S.13.2.1 summarizes our findings for the internal events, and Section S.13.2.2 summarizes our findings for the external events. Section S.13.3 combines the findings for internal and external events.

S.13.2 Estimated Plant Damage State/Release Category Frequencies and Sensitivity Issues

S.13.2.1 Internal Events

Table S-2 summarizes the effect that the findings discussed in the previous sections have on the Zion internal event plant damage states and release category frequencies.

The first column is a listing of 21 plant damage states defined in the ZPSS. The nomenclature is: S or A denote small or large LOCA, T denotes transient, V denotes interfacing systems LOCA, E or L denote early or late core melt, F and C denote fans and sprays working, respectively. Also appearing in column one are the mean frequencies of those damage states as calculated in the ZPSS.

The second column represents the revised estimates of the ZPSS plant damage states. It can be noted that a dash appears instead of a frequency estimate in several places. A dash denotes that we did not attempt to recalculate a frequency because these damage states were found to have a small impact on risk as calculated in the ZPSS.

The third column represents the revised "NRC-defined" plant damage states. The "NRC-defined" states consist of the sum of ZPSS damage states listed to the left.

Also listed in columns two and three are the upper and lower 95 percent confidence limits for the damage states. These were obtained by estimating the sum of the accident sequence rates for those dominant sequences that make up each damage state, using the Maximus methodology.

S.13.2.2 External Events

Table S-3 summarizes the effect that the findings discussed in the previous sections have on the Zion external event plant damage states. (The ZPSS did not report the external event plant damage state frequencies. They were deduced by comparing ZPSS, Tables 8-2 and 8.10-1 and Figure 8.10-1 presented in ZPSS, Section 8 for external events.)

S.13.3 Combined Internal and External Events

Table S-1 (presented previously) listed the revised dominant core-melt internal and external accident sequences. Table S-4 summarizes the effect that the internal and external event findings have on the "NRC-defined" plant damage state frequencies. The frequencies listed in Table S-4 were obtained by summing the frequencies listed in Tables S-2 and S-3.

Table S-2

Zion Internal Event Results (Events/Reactor Year)

ZPSS Plant Damage States		Revised Plant Damage States				Revised NRC Defined Plant Damage States			
	Mean		Point Estimate	L95	U95		Point Estimate	L95	U95
SEFC	7.4(-6)	SEFC	~3.0(-4)	4(-6)	1(-3)				
AEFC	1.7(-6)	AEFC	1.9(-6)	0	3(-6)				
SEC	1.8(-8)	SEC	1.9(-5)	8(-7)	7(-5)	Early core melt with containment cooling	3.3(-4)	2(-5)	2(-3)
AEC	8.2(-9)	AEC	--*	--	--				
TEFC	8.3(-7)	TEFC	~1(-5)	8(-7)	2(-3)				
TEC	9.3(-7)	TEC	--	--	--				
SEF	1.3(-9)	SEF	--	--	--				
AEF	1.9(-10)	AEF	--	--	--				
TEF	1.6(-9)	TEF	--	--	--				
SE	6.5(-10)	SE	4.7(-6)	1(-7)	1(-5)	Early core melt without containment cooling			
TE	2.3(-7)	TE	7.7(-7)	1(-7)	2(-6)		5.5(-6)	1(-8)	3(-5)
AE	1.1(-11)	AE	--	--	--				
SLFC	1.9(-5)	SLFC	0			Late core melt with containment cooling			
ALFC	9.8(-6)	ALFC	0						
SLC	1.9(-6)	SLC	0				2.6(-5)	3(-8)	3(-5)
ALC	4.0(-10)	ALC	0						
SLF	4.7(-9)	SLF	1.7(-5)	3(-8)	7(-5)				
ALF	7.3(-10)	ALF	9.8(-6)	0	3(-5)				
SL	1.3(-8)	SL	1.0(-7)			Late core melt without containment cooling			
AL	2.5(-13)	AL	--	--	--		1.0(-7)	--	--
V	1.1(-7)	V	1.0(-7)	0	1(-7)	Bypass	1(-7)	0	1(-7)

*--Denotes frequency was not recalculated because of small impact on risk.

Table S-3

Zion External Event Results (Events/Reactor Year)
(Excluding Fire)

ZPSS Plant Damage States		Revised Plant Damage States*		Revised NRC Defined Plant Damage States*	
(Mean)		(Point Estimate)		(Point Estimate)	
AEFC	<1(-7)	AEFC	--		
AEF	<1(-7)	AEF	--		
AEC	<1(-7)	AEC	--		
SEFC	<1(-7)	SEFC		Early core melt with containment cooling	<1.0(-7)
SEC	<1(-7)	SEC	--		
TEFC	--	TEFC	--		
TEF	--	TEF	--		
TEC	--	TEC	--		
AE	<1.0(-7)	AE	--		
SE	5.6(-6)	SE	5.6(-6)	Early core melt without containment cooling	5.6(-6)
TE		TE	--		
SLF	<1.0(-7)	SLF	<1.0(-7)	Late core melt without containment cooling	--

*Reflects seismic contribution only. See Section S.8 for discussion of fire analysis.

Table S-4

Revised Zion Combined Internal and External Event Results

NRC Defined Damage State	ZPSS Frequency (Mean)	Revised Frequency (Point Estimate)
Early core melt with containment cooling	1.5(-5)	3.3(-4)
Early core melt without containment cooling	5.8(-6)	1.0(-5)
Late core melt with containment cooling	3.1(-5)	2.7(-5)
Containment bypass prior to core melt	1.1(-7)	1.1(-7)

As can be seen, the revised damage state frequency estimates are within a factor of two of the ZPSS estimates except for "Early Core Melt With Containment Cooling." In the field of PRA, factors of two are usually not considered a significant disagreement.

The difference in the "Early Core Melt With Containment Cooling" category is due primarily to the inclusion of sequences involving loss of component cooling and a DC power initiated sequence in our revised frequency estimate. The ZPSS did not identify such sequences. (Refer to Table S-1.)

REFERENCES

1. ATWS: A Reappraisal--Part III, Frequency of Anticipated Transients, EPRI NP-801, Interim Report, July 1978.

2. Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One - Unit 1 Nuclear Power Plant, NUREG/CR-2787, June 1982.

3. Summary of NRC Staff and Consultants Questions on the Zion Probabilistic Safety Study, Commonwealth Edison Response, 1982.

4. Corwin Atwood, Common Cause Fault Rates for Pumps: Estimates Based on Licensee Event Reports at U. S. Commercial Nuclear Power Plants, January 1, 1972 - September 30, 1980, EGG-EA-5289, August 1982.

5. Summary of NRC Staff and Consultants Questions on the Zion Probabilistic Safety Study, Commonwealth Edison Response, 1982.

6. Zion Probabilistic Safety Study, Commonwealth Edison Company, September 1981.

REVIEW AND EVALUATION
OF THE PLANT ANALYSIS PRESENTED IN
THE ZION PROBABILISTIC SAFETY STUDY

1. Introduction

This volume documents the Sandia National Laboratories review of the system analysis and external events analysis (i.e., plant analysis) of the Zion Probabilistic Safety Study[1-1] (ZPSS) for the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission (NRC). The review was conducted by Sandia personnel with contractor support.

Each major topic area of the plant analysis portion of the study was reviewed: initiating events, event trees, success criteria, fault trees, human reliability analysis, component data, and uncertainty. The treatment of external events including seismic, fires, floods, missiles, wind, and aircraft crashes were also reviewed. Not every topic was reviewed in detail. Emphasis was placed on those portions of the analysis which appeared most important to the results of the Zion study.

In addition to each topical area, the important accident sequences from the study were reviewed in detail. The sequences dominating risk were reviewed in detail, as well as sequences important to the core-melt probability but that contributed little to risk due to the low consequences anticipated for these accidents. The intent of the sequence review was to evaluate the analysis of the Zion study and to determine the changes in the estimated frequencies of the sequences which could arise from differences in assumptions and the treatment of data.

Several issues and assumptions were evaluated in addition to the sequences. The issues were chosen as a result of interest on the part of NRC or because of their having been important in other risk assessments. Several of these issues, such as feed and bleed capability and interactions between core melt and containment systems, are issues for which assumptions must be made which may differ significantly between analysts. Other issues, such as anticipated transients without scram, are generic, unresolved safety issues. Still others, such as the treatment of reactor coolant pump seal LOCAs, arose because of assumptions used in ZPSS. These issues were generally treated in the manner of a sensitivity study. Assumptions were varied to see what the effects on the results could be. Often, this took the form of a bounding calculation.

It should be noted that the primary emphasis of the review was to search for significant omissions and critical judgments in the ZPSS. We therefore did not keep close account of small differences (e.g., those that affect the core-melt frequency or risk by approximately less than a factor of two).

The results of our review are presented in the following sections. The review of the systems analysis and external event topics are presented in Section 2. Section 3 presents the review of selected accident sequences. Section 4 details the review of selected issues. Section 5 summarizes the principal findings and presents estimates of plant damage state frequencies for use in the containment and consequence calculations in Volume II.

REFERENCE

1-1. Zion Probabilistic Safety Study, Commonwealth Edison Company of Chicago, Fall 1981.

2. Areas of Review

The ZPSS, as any Probabilistic Risk Assessment (PRA), is composed of several interrelated tasks. A review of a PRA is not complete unless the information and analysis which comprises each task is examined. The ZPSS PRA tasks are depicted in Figure 2-1. Also shown are the Volume 1 report sections which summarize our review of a task. Tasks reviewed in Volume II are so noted. As can be seen, we did not review the first task, "initial information collection." Our review assumes that the ZPSS has collected accurate Zion design and operations information; e.g., correct piping and instrumentation layouts, etc.

The findings of our review are ultimately expressed quantitatively in terms of the effect they have on ZPSS damage state frequencies. Damage states are, in essence, functional classifications of core-melt accidents. Classification of core-melt accidents functionally is necessary to perform the containment and consequence analysis presented in Volume II. The ZPSS defined 21 plant damage states. These can be grouped as follows: (1) SEFC, SEC, SEF, AEFC, AEC, AEF, TEFC, TEC, TEF; (2) SE, TE, AE; (3) SLFC, SLC, SLF, ALFC, ALC, ALF; (4) SL, AL; (5) V. The nomenclature is: S or A denote small or large LOCA and T denotes transient, E or L denote early or late core melt, F and C denote fans and sprays working respectively, and V denotes an interfacing systems LOCA. These five groups of plant damage states can be qualitatively described as follows: (1) early core melt with containment cooling; (2) early core melt without containment cooling; (3) late core melt with containment cooling; (4) late core melt without containment cooling; and (5) containment bypass before core melt.

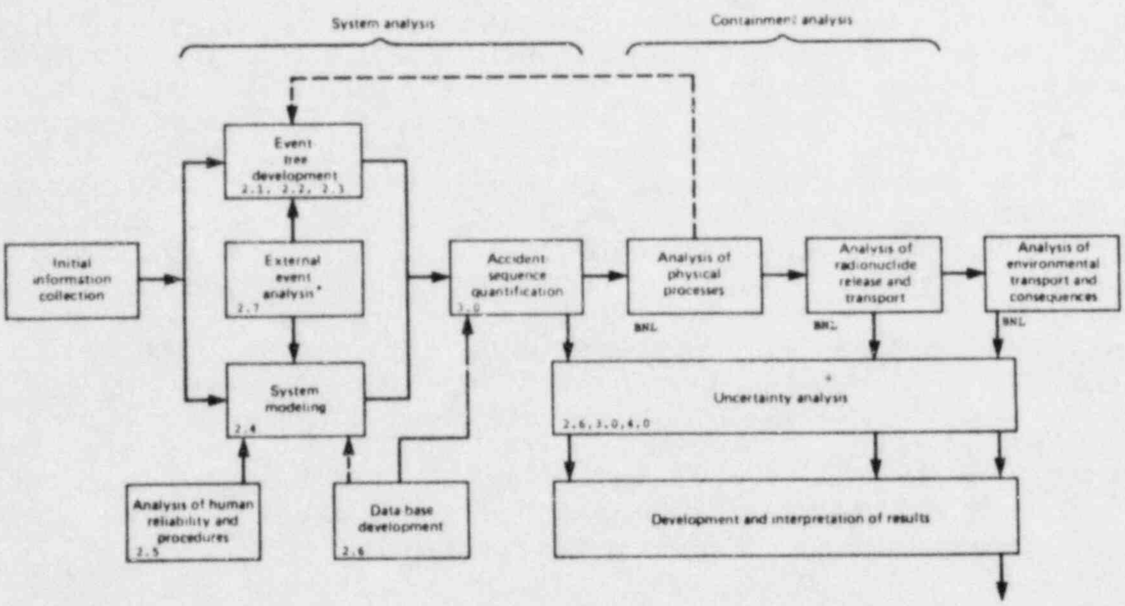

Figure 2-2. **Risk Assessment Procedure**

2.1 Initiating Events

The initiating events covered in the ZPSS seem to be relatively complete compared to those addressed in previous PRAs. The initiating event categories analyzed were identical for both Zion units. ZPSS Table 1.5.1-31, which summarizes the initiating events considered, is reproduced on page 2-4 for reference. The treatment of these initiating events is discussed in other sections of this review. Comparisons were made to other PRAs, an NRC list of concerns about potentially omitted initiating events, and EPRI NP 801.[2-1] (It should be noted that the ZPSS used data contained in NP801 to quantify the ZPSS initiating event frequencies.) In addition, several initiating events were identified by NRC as being of particular interest. These are discussed below.

1. Excess Letdown or Decreased Charging

The result of this potential initiating event is lowering the reactor coolant inventory without detection to a level that would require reactor trip and mitigation via closure of the letdown line by the operator. Although not addressed explicitly in the ZPSS, this event would be included in the EPRI NP-801 data used to quantify the reactor trip event (ZPSS subcategory 13).

2. Insufficient Letdown or Increased Charging

This potential initiating event would cause RCS overpressure and thus falls under ZPSS subcategory 13, which is included in the EPRI NP-801 data used to quantify the initiating event.

3. Pressurized Thermal Shock

This is a safety issue not addressed by the ZPSS or any of the current or past PRAs. It is a complex issue which requires very detailed plant specific probabilistic, thermohydraulic, and fracture mechanics analysis. Due to the time limitations placed on this review, we were not able to evaluate this initiating event.

4. Failure of the Pressurizer Sprays or Heaters

This initiating event results in loss of RCS pressure control and thus falls under ZPSS subcategory 13, which is included in the EPRI NP-801 data used to quantify the initiating event.

5. Inadvertent Containment Spray Operation

This initiating event was not treated explicitly in the ZPSS or in previous PRAs. The apparent concern is actuation

TABLE 1.5.1-31

Zion Initiating Event Subcategories

1. Large Loss of Coolant Accidents (LOCAs)

2. Medium LOCAs

3. Small LOCAs

   a. Pressurizer relief or safety valve opening
   b. Miscellaneous small LOCAs

4. Steam Generator Tube Rupture

5. Steam Pipe Rupture Inside the Containment

6. Steam Pipe Rupture Outside the Containment

7. Loss of Feedwater Flow

   a. Loss/reduction of feedwater flow in one steam generator
   b. Loss of feedwater flow in all steam generators
   c. Feedwater flow instability--operator error
   d. Feedwater flow instability--mechanical causes
   e. Loss of one condensate pump
   f. Loss of all condensate pumps
   g. Condenser leakage
   h. Miscellaneous secondary leakage

8. Full or Partial Closure of One Main Steam Isolation Valve (MSIV)

9. Loss of Primary Flow

   a. Loss of primary flow in one loop
   b. Loss of primary flow in all loops

10. Core Power Increase

    a. Uncontrolled rod withdrawal
    b. Boron dilution--chemical and volume control system malfunction
    c. Cold water addition

TABLE 1.5.1-31 (continued)

Zion Initiating Event Subcategories

11. Turbine Trip

    a. Turbine trip

        1. Closure of all main steam isolation valves
        2. Increase in feedwater flow in one-steam generator
        3. Loss of condenser vacuum
        4. Loss of circulating water
        5. Throttle valve closure/electro-hydraulic control problems
        6. Generator trip or generator caused faults
        7. Increase in feedwater flow in all steam generators

    b. Turbine trip due to loss of off-site power
    c. Turbine trip due to loss of service water

12. Spurious Safety Injection Activation

13. Reactor Trip

    a. Reactor trip

        1. Control rod drive mechanism problems and/or rod drop
        2. High or low pressurizer pressure
        3. Spurious automatic trip--no transient condition
        4. Automatic/manual trip--operator error
        5. Manual trip due to false signal
        6. Spurious trip--cause unknown
        7. Primary system pressure, temperature, power imbalance
        8. Loss of power to necessary plant systems

    b. Reactor trip due to loss of component cooling water of the spray system during shutdown while on RHR. This could lead to a loss of RCS inventory since the spray headers take suction from the RHR lines. Neither the ZPSS nor we analyzed possible events during shutdown.

6. Inadvertent Containment Isolation

If this potential initiating event were to occur, the reactor may be shutdown and therefore challenge the safety systems. This would most likely lead to a loss of main feedwater, initiating event 7.

7. Loss of Instrument and Control Power

This refers to loss of instrument and control power independent of total AC or DC failure. Loss of individual AC or DC buses was analyzed in the ZPSS in Section 1.3.4.13.5. A ZPSS bounding analysis indicated that these buses did not offset the plant core-melt frequency significantly. We disagreed with this assessment and identified a dominant sequence initiated by a DC bus failure. This sequence is discussed in Section 3.

8. Events Occurring During Cold Shutdown

None of the PRAs to date, the ZPSS, or our review addressed these events.

9. Reactor Coolant Pump and Other Internal Missiles

Turbine missiles were considered under external events; however, RCP or other internal missiles apparently were not and would appear to be logically considered as a failure mode of components in the vicinity of components potentially producing such missiles. This does not appear to be a significant exclusion.

10. Reactor Coolant Pump Seal Failure

The RCP seal failure initiating event should be considered a small LOCA. However, it is not included in the data base. While it is true they were not included, the small LOCA frequency of approximately .035 quoted in the report for Zion is a reasonable estimate. The NRC has conducted a study of RCP rupture LOCAs[2-2] which suggests their frequency to be approximately .02. Conceivably, the Zion small LOCA frequency could be .055. However, upon review of the data comprising the ZPSS small LOCA frequency, it was noted that many of the small LOCA events involved stuck-open pressurizer PORVs. It is generally known that some of these events were recovered by the operators in a few minutes via closure of the PORV block valves. The ZPSS did not consider recovery and thus probably overpredicted the frequency of PORV LOCAs. It is felt that this overprediction would tend to cancel the underprediction of RCP seal failure and thus the small LOCA frequency estimate of .035 is reasonable

11. Loss of Component Cooling Water Due to a Pipe Break

This potential initiating event could conceivably lead to core melt unless judicious operator recovery actions are performed within about an hour. Assuming no operator recovery, a large pipe break in the component cooling system would cause a reactor trip, and could eventually cause a reactor coolant pump seal LOCA, and failure of the pumps which provide makeup to the reactor coolant system.[2-3] It should be noted that the ZPSS analyzed a "loss of pump flow" induced loss of component cooling water initiating event. However, the ZPSS did not analyze one induced by a pipe break. The system responses are somewhat different for the two cases.

In conclusion, review of the NRC list of potential ZPSS initiating event omissions has indicated that pressurized thermal shock, shutdown events, and loss of component cooling water due to a pipe break appear to be the only potentially significant events omitted in the ZPSS. We also found the ZPSS treatment of a DC power bus initiating event to be inappropriate.

The loss of component cooling water due to a pipe break and loss of a DC bus are evaluated in Section 3 of this review. As stated earlier, an evaluation of pressurized thermal shock or shutdown events does not appear in this review.

It should be noted that six external initiating events (seismic, fire, flood, wind, aircraft accidents, and turbine missiles) were considered, which is more than most PRAs have attempted. The external event review appears in Section 2.7.

<u>Initiating Event Quantification</u>

Estimated initiating event frequencies are expected to vary from plant to plant depending on the plant characteristics, design, and its specific data base. The ZPSS initiating event data were compared to the data used in the Arkansas Nuclear One (ANO) IREP[2-4] analysis. (The reason for choosing ANO is because it is a recently completed NRC-sponsored PRA.) The purpose of the comparison was to look for potential differences in judgment or calculation. The mean values from the ZPSS are:

Initiating Event Category	Occurrences/Year
1. Large LOCA	9.4×10^{-4}
2. Medium LOCA	9.4×10^{-4}
3. Small LOCA	3.5×10^{-2}
4. Steam Generator Tube Rupture	2.4×10^{-2}
5. Steam Break Inside Containment	9.4×10^{-4}
6. Steam Break Outside Containment	9.4×10^{-4}
7. Loss of Main Feedwater	5.2
8. Trip of One MSIV	2.5×10^{-1}
9. Loss of RCS Flow	3.6×10^{-1}
10. Core Power Excursion	2.3×10^{-2}
11a. Turbine Trip	3.7
11b. Turbine Trip--Loss of Off-site Power	5.8×10^{-2}
11c. Turbine Trip--Loss of Service Water	9.4×10^{-4}
12. Spurious Safety Section	6.4×10^{-1}
13a. Reactor Trip	3.8
13b. Reactor Trip--Loss of Component Cooling	9.4×10^{-4}
Reactor Trip--Loss of DC Bus	2.8×10^{-1}
V. Interfacing System LOCA	1.1×10^{-7}

The ANO PRA utilized WASH-1400[2-5] data for breaks greater than 2 inches. For breaks less than 2 inches, WASH-1400 data was added to the 2×10^{-2} reactor coolant pump seal rupture data discussed in 10 above. The following compares ANO, WASH-1400, and ZPSS LOCA frequency data:

	ZION	ANO	WASH-1400
1. Large LOCA >6"			
5%	3.3(-5)	1(-5)	1(-5)
Median	3.4(-4)	1(-4)	1(-4)
Mean	9.4(-4)	2.7(-4)	2.7(-4)
95%	3.5(-3)	1(-3)	1(-3)
2. Medium LOCA 2"-6"			
5%	3.3(-5)	3(-5)	3(-5)
Median	3.4(-4)	3(-4)	3(-4)
Mean	9.4(-4)	8(-4)	8(-4)
95%	3.5(-3)	3(-3)	3(-3)
3. Small LOCA <2"			
5%	1.3(-2)	--	1(-4)
Median	3.1(-2)	--	1(-3)
Mean	3.5(-2)	2.1(-2)	2.7(-3)
95%	7.4(-2)	--	1(-2)
4. Interfacing Systems LOCA			
Mean	1.05(-7)	<1(-6)	4(-6)

It can be noted that the ZPSS mean frequencies are significantly greater for the large LOCA. The reason the means are greater is due to the ZPSS Bayesian methodology used to establish the probability distributions and 5 percent and 95 percent bounds. Because of these differences, the ZPSS mean values are skewed higher than the WASH-1400 means. The interfacing systems LOCA has a smaller estimate in the ZPSS because of more frequent testing of the low pressure injection check valves than the Surry Plant in WASH-1400. (Due to the more frequent testing, the dominant Zion interfacing systems LOCA location is in the RHR suction path.)

Transients are subdivided differently at ANO but six are directly related.

		ZION		ANO
7.	Loss of Main Feedwater	5.2		1.0
11b.	Turbine Trip-- Loss of Off-site Power	5.8×10^{-2}		3.2×10^{-1}
11c.	Turbine Trip-- Loss of Service Water	9.4×10^{-4}		2.6×10^{-3}
11a.	Turbine Trip--	3.7	7.5	7.1
13a.	Reactor Trip--	3.8		
13b.	Reactor Trip-- Loss of a DC Bus	2.8×10^{-1}		1.8×10^{-2}

The ZPSS transient initiating event frequencies appear reasonable: the differences are the result of the influence of plant specific data or operations practices, e.g., several DC bus failures caused by switching errors have occurred at Zion. These have a higher frequency of occurrence than ANO due to the Zion practice of inter-unit DC bus cross-tying. (The other ZPSS initiating events were not explicitly analyzed at ANO because they were either (a) not applicable, (b) were not identified to be significant, or (c) grouped with other transients.)

The reactor vessel rupture LOCA (R) cannot be mitigated and thus leads to core melt by itself. The ZPSS concluded that the frequency of such an event is small compared to other events leading to the same plant damage state; e.g., large LOCA followed by failure of low pressure injection. This conclusion is questionable since the ZPSS did not analyze a vessel rupture sequence initiated by pressurized thermal shock.

In summary, these frequencies appear to be consistent with what would be expected from experience and from what was used in the ANO PRA. The most significant difference is in the large LOCA frequency estimate. However, this difference can be attributed to the use of Bayesian methodology in the ZPSS. The Bayesian methodology used to quantify the initiating events is reviewed in Section 2.6.4 of this report.

Initiating Event/Safety System Interdepencies

One of the most important tasks in a PRA is to search for system or component failures which can simultaneously cause a reactor trip and failure of safety systems. These types of initiating events have occurred in the nuclear industry (e.g., Rancho Seco, Crystal River) and have also been shown to be important contributors to risk in some PRAs (e.g., ANO). The ZPSS search for such initiators is documented in Sections 1.3.4.11.4 and 1.3.4.13.5. Failures of service water, component cooling, and electrical buses were identified as initiating events.

We reviewed how these interdependencies were treated in the quantification process. Our findings are documented in Section 2.2.2 (Event Tree 13b, and 13c) and Section 3 of this report.

2.2 Event Trees

The ZPSS constructed 14 event trees to model the plant system response to the initiating internal events discussed in Section 2.1. We reviewed these trees for validity. During the review, several questions were generated which could not be answered by information or analysis presented in the text. These questions were, for the most part, answered during meetings and conversations between personnel from Sandia Laboratories and ZPSS personnel. The findings are of two types. General findings are those that apply to all or several of the event trees. Specific findings are those that apply to a particular event tree. These findings and the impact they may have on the ZPSS results will now be discussed.

2.2.1 General Event Tree Findings

Containment Spray System Analysis

There are two containment spray systems installed at a Zion unit. The containment spray injection system (CSIS) consists of two pump trains which take suction from the refueling water storage tank (RWST). Upon depletion of the RWST, the CSIS pumps are shut down. During the recirculation phase, the containment spray recirculation system (CSRS) is utilized. The CSRS is a two-train system which utilizes the same pumps as the low pressure recirculation system (LPRS). A portion of the LPRS flow is diverted to the CSRS spray headers. During recirculation, the LPRS pumps take suction from the containment sump.

Though not explicitly stated in the ZPSS, no credit was given on the event trees for operation of the CSRS. Referring to event tree 2 (ZPSS Figure 1.3.4.2-1, event tree 2), for example, it can be seen that on sequences 40 and 44, the CSRS is defined to be operating, yet the plant damage state (AEF and AE, respectively) implies that sprays are <u>not</u> operating. This is a conservatism adopted in the Zion analysis and may be justified for the following reasons: (1) In the vast majority of core-melt sequences the PRA analyzed, the LPRS is unavailable. Since the LPRS and CSRS share much of the same equipment, the CSRS would most likely also be unavailable. (2) During a core-melt accident, the LPRS/CSRS pumps may fail since their sump water supply could be clogged with core-melt debris.

The CSIS, on the other hand, is given more credit on the event trees than may be justified. Upon close examination of event tree sequence plant damage states (e.g., sequence 2, event tree 2) and the CSIS event definitions, it is noted the ZPSS assumes the CSIS is available during the recirculation phase if it was successful during the injection phase. In order for the CSIS to be available during recirculation, the RWST must be refilled by the operators. The validity of giving credit for the CSIS during recirculation is questioned since there is no mention of refilling the RWST in the Zion LOCA emergency procedures. The ZPSS analysis team feels RWST refill should be given credit (Reference 2-3) because about 12 hours would be available to refill the tank. They argue emergency support personnel will be available to perform the actions. Though 12 hours seems more than adequate to perform the refill, it is not clear that the operators would be cognizant that they should do this. The ZPSS event trees imply that many of the refills would be performed after the onset of core melt. It is felt that the confusion in the control room at that point would be extreme and if the operators were doing anything constructive, they would probably be trying to restore core cooling.

If one assumes that the CSIS will not be available during the recirculation phase, all core melts that are initiated during the recirculation phase would not have sprays available to mitigate the consequences of the accident. This implies that plant damage states characterized by C (spray injection operating) and L (core melt initiated in the recirculation phase) are not possible. Thus, ZPSS damage states SLFC, ALFC, SLC, and ALC become SLF, ALF, SL, and AL, respectively. (We have recently received preliminary information that Zion intends to develop a RWST refill procedure. See Section 4.1 for more details.)

Core Melt/Safety System Interactions

The interdependencies incorporated into the Zion event trees imply that the containment spray and fan cooler systems may be utilized during a core-melt accident. This is an important assumption since the Zion analysis predicts that the operation of these systems can significantly reduce the risk associated with a core-melt accident. This topic is discussed more fully in Section 4.1.

Sodium Hydroxide Addition

All event trees model the additions of sodium hydroxide to the containment spray water. This was modeled because it was thought to enhance the radioactive material scrubbing capability of the spray water during a core-melt accident. Discussions with ZPSS personnel revealed that analysis performed late in the study indicated that sodium hydroxide addition had a negligible effect on the assessment of plant damage states and release categories. All event trees could therefore be simplified by removal of the sodium hydroxide addition event. This is consistent with the findings of WASH-1400 with respect to sodium hydroxide addition.

Main Feedwater System

The Zion study assumed that the main feedwater system was unavailable for purposes of removing post-shutdown decay heat following all internal and external initiating events analyzed. This is a conservatism adopted by the ZPSS which we do not feel is justified. Discussions with Zion personnel and review of Zion procedures indicated that the main feedwater system can be restored following many reactor trips not caused by a loss of off-site power.

Core Melts Caused By Containment Overpressure Failure

The Zion event trees do not model core melts caused by containment overpressure failure. These sequences have been shown to be important in other PRAs (e.g., the S_2C sequence in WASH-1400).

We have assessed the effect of this potential sequence in the ZPSS and found it to be negligible. A review of the Zion core cooling and containment heat removal systems indicated that it is almost completely assured that if core cooling during the recirculation phase is provided, so also will containment heat removal. This is because the core cooling and one of the containment heat removal systems share most of the same equipment (i.e., pumps and support systems). Because of this dependence, the probability of having core cooling and no containment heat removal is negligible.

Transient Induced Pressurizer Safety Valve Demands

The ZPSS event trees do not model the demand of the pressurizer safety valves in response to a transient. This raised a concern that the study may have missed some important accident sequences. We feel the ZPSS has not missed important accident sequences for the following reasons:

- With off-site power available, stuck-open relief valve sequences are adequately modeled by considering them as small LOCA initiating events. Via review of the ZPSS small LOCA initiating event data, it is noted that pressurizer relief valve failures are included as initiating events.

- Following a loss of off-site power (LOP), it is very questionable whether the safety valves would be demanded. NUREG-0611[2-6] quotes two instances of PORV demands in Westinghouse plants following a LOP in approximately 150 reactor years.*

Based on this data, we roughly estimate the challenge rate of the PORVs to be approximately .07 following a LOP. However, it should be noted that these represent PORV rather than safety valve challenges and, since the safety valves open at a higher setpoint, the safety valve challenge rate would be lower. We conservatively calculated the most likely core-melt accident involving a LOP and stuck-open safety valve using a challenge rate of .1 and found that frequency to be much less than 1×10^{-6}. This value is small compared to the overall core-melt frequency calculated in the ZPSS and the frequency of the plant damage state in which this sequence would be placed.

2.2.2 Specific Event Tree Findings

All of the event trees were reviewed in detail. The following subheadings delineate the significant event tree specific findings. If nothing is written for a particular event tree, this means there are no specific findings. The general findings delineated in Section 2.2.1 do apply to these trees, however.

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*It should be noted that the Zion control room simulator indicates that the PORVs will be demanded following a LOP. We pursued this possible inconsistency with Commonwealth Edison. They responded that the simulator did not reflect the correct thermal-hydraulic response. We were unable to verify this contention within the scope of this review. We have assumed that Commonwealth Edison is correct and that the PORVS will not be demanded following every loss of off-site power.

Event Tree 1 - Large LOCA

Sequences 31, 32, 34, 35, 37, 38, 43, and 44 contain decision branches for CF-1 (Fan Coolers) given failure of SA-1 (Safety Injection Actuation Signal). Decision branches for CF-1 appear inappropriate for these sequences since SA-1 failure causes failure of CF-1. This leads to sequences 31, 34, 37, and 43 being eliminated from the tree, and sequences 32, 35, 38, and 44 having a CF-1 unavailability of 1.0. This does not significantly affect the plant damage states.

Event Tree 2 - Medium LOCA

The comments regarding SA-1/CF-1 which were made with respect to event tree 1 also apply to event tree 2. Zion claims that if SA-1 fails, the operator would have ample time for this size LOCA to start the fan coolers manually prior to containment failure. Their fan coolers analysis, however, does not analyze a manual initiation. If one were to assume the operator does not initiate the fans upon SA-1 failure, the plant damage states are not significantly affected.

Event Tree 4 - Steam Generator Tube Rupture

Sequence 28 involves failure of events SA-2 and OP-5. If these two events occur, the high-pressure injection system is unavailable (SA-2) to make up for the inventory lost through the steam generator, and the operator is unable to isolate the steam generator and stop the inventory loss (OP-5). These two events would lead to core uncovery and thus core melt. The event tree, however, incorrectly labels this sequence as a success. The event data presented for SA-2 in Table 1.3.4.4-2 are also incorrect. The data implies that the probability of event SA-2 is unaffected by the loss of AC power. Since SA-2 includes the high-pressure injection system, which requires AC power, the data for SA-2 should closely resemble the data presented for HH-2 in Table 1.3.4.3-2. These errors, however, do not appear to significantly affect the plant damage state frequencies.

The event tree also does not address steam generator tube ruptures accompanied by stuck-open secondary safety valves. These may be potentially high-risk accidents, if core melt ensues, since a direct path from inside containment to the atmosphere would exist.

We performed a rough quantification of core-melt scenarios involving a single steam generator tube rupture and a stuck-open secondary safety valve and found the frequency of such sequences to be small, i.e., less than 1×10^{-7}. One of the primary reasons for this is because we feel the probability of demanding the secondary safety valves is small. In order for a secondary safety valve to be demanded, the atmospheric dump would have to fail to open. Review of the Zion steam generator tube rupture procedure indicated that the dump valve would most likely be available since the operator is not directed to close the dump valve blocking valve.

Event Tree 11c--Turbine Trip Due to a Loss of Service Water and Event Tree 13b--Reactor Trip Due to a Loss of Component Cooling Water

The ZPSS used the turbine trip and reactor trip event trees to model the plant response to a loss of service water and loss of component cooling water initiating event, respectively. These event trees do not appear to adequately model the plant response to these initiating events for the following reasons:

a. The trees do not allow for a reactor coolant pump (RCP) seal LOCA to occur following a sustained loss of component cooling or service water.
b. The systems which respond to a seal LOCA are not adequately modeled.
c. Station blackout initiated by a loss of service water and followed by a loss of off-site power is not modeled (station blackout initiated by a LOP followed by a loss of service water is modeled on 11b).

If a loss of component cooling occurs, the RCP seals will lose cooling due to failure of the charging pumps[2-3] and cooling to the thermal barrier heat exchanger. The ZPSS predicts a 1200 gpm seal LOCA will occur approximately 30 minutes following a loss of seal cooling. Since component cooling also cools the safety injection pumps, they will subsequently fail.[2-3] A core melt would ensue leading to an SEFC plant damage state. The ZPSS omitted quantification of such a sequence. We quantified this sequence in Section 3.

Service water cools component cooling water via three heat exchangers. If service water to the heat exchangers fails, the component cooling system would gradually heat up. If service water to the heat exchangers is not restored in the long term, RCP seal cooling and the safety injection pumps could fail. This would most probably lead to a LOCA followed by core melt leading to an SEC plant damage state. This loss of service water sequence is similar to the loss of component cooling water sequence described in the previous paragraph, except it takes much longer to lead to core melt. The operators would thus have a higher probability of recovering from this sequence. Because of this, we will only quantify the loss of component cooling water event in Section 3; loss of service water would have a second order effect on core melt-frequency.

Service water also cools the diesel generators. If service water fails, followed by a loss of off-site power, the diesels will fail followed by station blackout. If AC power is not restored within approximately an hour, a seal LOCA could occur followed by core melt. If AC power is not restored within several hours, a containment overpressure failure leading to an SE plant damage state could occur. Based on an abbreviated analyses we performed for this omitted sequence, we found the sequence frequency estimate to be small compared to other sequences which appear in this plant damage state. The reason for this is because the probability of a loss of off-site power following this initiating event is small ($\sim 10^{-3}$).

Event Tree 13--ATWS

ZPSS personnel constructed a new ATWS event tree based on our findings described in an earlier review.[2-7] We reviewed the revised event tree. The main problem we have with it is the inclusion of event SO. This event represents failure of the high-pressure injection system given an RCS pressure of greater than 3200 psi. Failure of the system is postulated to occur via inoperability of the injection line check valves. The ZPSS analysis subjectively estimated the probability of this event to be 1×10^{-2} based on information presented in WCAP 8330. We do not find this information to be conclusive and therefore conservatively assume that the probability of this event is 1.0. However, the ATWS sequences are not significant contributors to core-melt probability because of the arrangement of turbine trip circuits, as discussed in Section 4.7.

2.3 Mitigating Systems Success Criteria

In response to LOCA and transient initiating events, various Zion core cooling and containment systems are called upon to bring the plant to a safe shutdown condition. If core cooling is unsuccessful and a core melt ensues, the containment systems may still be able to reduce the consequences of the accident by maintaining the containment boundary and thus isolating the core melt from the environment. The combinations of plant systems required to cool the core and maintain the containment boundary constitute the Zion mitigating system success criteria. We have reviewed the validity of the success criteria employed in the ZPSS. We have judged the success criteria to be consistent with criteria employed in PRAs of similar plants.

Table 2.3-1 summarizes the LOCA and transient success criteria employed in the ZPSS.

TABLE 2.3-1

ZPSS LOCA and Transient Mitigating System Success Criteria

LOCA

SIZE	Emergency Core Cooling Early (RWST)	Emergency Core Cooling Late (SUMP)	Containment Overpressure Protection	Radioactivity Removal
0-2"	1/4 High-Pressure Pumps (HP) and 1/3 Auxiliary Feedwater Pumps (AFWS) OR 1/4 HP and 2/2 PORVs	1/4 HP and 1/2 RHR	1/3 Containment Spray Pumps OR 3/5 Containment Fans	1/3 Containment Spray Pumps
2-6"	2/4 HP and 1/2 RHR	1/2 RHR	Same	Same
6"	3/4 Accumulators and 1/2 RHR Pumps	1/2 RHR Pumps	Same	Same

TRANSIENTS

Emergency Core Cooling Early (Secondary or RWST)	Emergency Core Cooling Late (Secondary or SUMP)	Containment Overpressure Protection	Radioactivity Removal
1/3 AFWS OR 1/4 HP and 2/2 PORVs	1/3 AFWS OR 1/4 HP and 1/2 RHR	Same	Same

The ZPSS employed the containment system success criteria given in the FSAR, i.e., containment overpressure protection can be provided by one of three spray system trains or three of five fan cooler system trains. It is important to note that the FSAR criteria applies to protecting the containment following a LOCA and not a core melt. The ZPSS has assessed that the same criteria is also valid following core melt. We investigated this assessment (see Section 4.1). Other PWR PRAs, however, have typically used the post-LOCA criteria as the post core-melt criteria.

We could not find in the Zion FSAR an explicit statement of the core cooling success criteria in response to the full range of potential LOCA break sizes and transient initiating events. The ZPSS apparently made use of some Westinghouse documents and the FSAR in establishing the criteria employed in the report. The report contains references for some of the criteria used (e.g., definition of L-1 success on Pages 1.3-34 and 1.3-113), but omits them for others. The ZPSS gave credit for "feed and bleed" core cooling during transients and small LOCAs following failure of the auxiliary feedwater system. Feed and bleed cooling is still an open question (see Section 4.2), but recent TRAC computer analysis at Los Alamos[2-8] has suggested that it is a viable core cooling option. Though we could not validate the entire core cooling success criteria employed in the ZPSS, it is our opinion that it is reasonable since it is fairly similar to that used in the Reactor Safety Study.

In addition to the major core cooling and containment system success criteria discussed above, the ZPSS developed a variety of support system success criteria. These support systems must succeed to allow successful operation of the core cooling and containment systems. Support systems include pump cooling systems, electric power systems, and the plant operators. The support system criteria which influenced the ZPSS results the most are listed in Table 2.3-2. We reviewed these criteria with the aid of the FSAR, previous PRA analyses and discussions with the ZPSS analysts.[2-3] Some problems were identified which are discussed in Section 2.4 for each individual system.

TABLE 2.3-2

Important ZPSS Support System Criteria

Major System/ Components	Support System(s)	Support System Criteria
Safety Injection and Charging Pumps	Component Cooling Water System (CCW), Electric Power	Operating pumps will fail in a short time if CCW is not supplied to oil coolers.*
Containment Fan Coolers	Service Water and Electric Power	Fans coolers cannot prevent a containment overpressure if support systems do not succeed within approximately 3 hours.
Core Cooling Systems	Electric Power and/or Operator Action	Following a transient initiating event these core cooling support systems must succeed within one hour to prevent core melt.
Reactor Coolant Pumps (RCPs)	CCW or Seal Injection System	Failure of these RCP seal cooling support systems will cause a 300 gpm LOCA per RCP after 30 minutes.
Residual Heat Removal Pumps	CCW and Service Water	CCW pump operation and service water cooling of CCW is required to cool the RHR pumps.
Containment Spray System	Electric Power	Sprays cannot prevent a containment overpressure if support systems do not succeed within approximately 3 hours
Main Feedwater System	Operator Action	Operator must recover within a short time.

*This criteria does not appear in the ZPSS. The ZPSS analysis assumes that these pumps do not require CCW cooling during the injection phase. The ZPSS analysts changed this assumption and adopted the above criteria in Reference 2.

2.4 Review of the ZPSS Fault Trees

The system fault trees presented in ZPSS Section 1.5 were reviewed for accuracy and completeness. The findings of this review are presented in Section 2.4.1. In Section 2.4.2 we compare our revised system unavailability estimates with estimates for similar systems given in other PRAs and NUREG/CR-2497.[2-9]

2.4.1 Fault Tree Analysis

The review of Zion fault trees included an examination of the fault trees, the super-component arrangement and definition, and the system calculations. The ZPSS analyzed systems for the case where all electric power was available, and various degraded power states which represent loss of off-site power combined with failures of some or all of the emergency diesel generators. In our review of the degraded power state cases, we found substantial differences existed between the system unavailabilities presented in the ZPSS and the unavailabilities resulting from our calculations. Since these differences proved to be important in the loss of off-site power accident sequences, we summarize our results in Table 2.4-1.

For purposes of comparison, the unavailabilities used in the ZPSS are summarized in Table 2.4-2. The analyses leading to the unavailabilities in Table 2.4-1 are described, for each system, in the following sections.

2.4.1.1 Emergency Electric Power System Fault Tree

The ZPSS emergency electric power systems analysis was reviewed. Eight different fault trees for the system were developed in the ZPSS, one for each "power state," with each state being defined as having AC power either available or unavailable at combinations of three 4160V buses: 147, 148, and 149 (247, 248, and 249 for Unit 2). The reason for the eight separate analyses is embedded in the methodology used in the ZPSS in that the support systems, such as electric power, are modeled explicitly at the event tree level. Hence, each specific accident sequence, e.g., is analyzed eight times, one for each power state.

In addition, each power state calculation is subdivided into four specific cases, two of which consider that off-site power is initially available, and two which consider that the loss of off-site power is the initiating event. The availability (and unavailability) of off-site power is divided into two separate situations: neither unit has an engineered safeguards (ES) signal, or one unit has such a signal. The necessity for separating the ES from the non-ES situations results from the fact that, although the two Zion units have six major AC buses (three each), there are only

TABLE 2.4-1

Degraded Power State Event Unavailabilities - Review Results

Event / Power on	All Buses	Buses 147 148	Buses 147 149	Buses 148 149	Bus 147	Bus 148	Bus 149	No Buses
HH-2	2.1(-8)	6.8(-7)	1.4(-7)	1.2(-5)	1.2(-5)	2.1(-3)	5.6(-3)	1.0
Feed and Bleed	3.0(-3)	3.0(-3)**	3.0(-3)	3.0(-3)**	3.0(-3)**	5.1(-3)**	8.6(-3)**	1.0
AFW Event L-1	3.4(-5)*	2.3(-4)	2.3(-4)	3.4(-5)	0.039	2.3(-4)	2.3(-4)	0.039
CS	6.3(-5)	1.2(-4)	4.6(-4)	4.6(-4)	6.8(-3)	6.8(-3)	6.8(-2)	1.0
CF	3.1(-6)	1.1(-2)	1.1(-2)	9.2(-5)	1.0	1.0	1.0	1.0
CCW	4.3(-5)	2.5(-3)	2.5(-3)	2.5(-3)	3.2(-2)	3.2(-2)	3.2(-2)	0.17
SW	2.0(-8)	1.1(-5)	1.1(-5)	1.1(-6)	2.5(-3)	2.5(-4)	2.5(-4)	1.3(-2)

*For sequences other than loss of electric power (AC or DC) or loss of main feedwater, credit for main feedwater restoration is given and the event L-1 is evaluated as 2.2(-7).

**These values become 1.0 if the initiating event is loss of DC Bus 112.

TABLE 2.4-2

Degraded Power State Event Unavailabilities - ZPSS Results

Event / Power on	All Buses	Buses 147 148	Buses 148 149	Bus 149	Bus 147	Bus 148	Bus 149	No Buses
HH-2	7.4(-5)	2.9(-7)	2.9(-7)	8.0(-6)	8.0(-6)	8.2(-4)	8.0(-4)	1.0
Feed and Bleed	6.1(-3)	1.0	6.1(-3)	6.1(-3)	1.0	1.0	6.1(-3)	1.0
AFW Event L-1	4.2(-6)	3.7(-4)	3.7(-4)	4.2(-6)	4.9(-2)	3.7(-4)	3.7(-4)	4.9(-2)
CS	2.2(-4)	8.7(-3)	8.7(-3)	8.7(-3)	8.9(-3)	8.9(-3)	7.0(-2)	1.0
CF	6.1(-7)	1.1(-2)	1.1(-2)	4.4(-5)	1.0	1.0	1.0	1.0
CCW	2.9(-6)	2.9(-6)	2.9(-6)	2.9(-6)	4.2(-2)	4.2(-2)	4.2(-2)	1.0
SW	2 2(-8)	2.2(-8)	2.2(-8)	2.2(-8)	2.2(-8)	2.2(-8)	2.2(-8)	1.0

five diesel generators available. Buses 147 and 247 can be powered by a swing diesel generator which loads onto the bus of the unit experiencing an ES condition. If no ES condition exists, ZPSS assesses, and we concur, that it is equally likely that the swing diesel generator will load onto 147 or 247. (It should be noted that six batteries are available, one supplying control power to each AC bus, except as noted below.)

The review of the calculations, relevant procedures, and drawings verified that all the AC failures, appropriate to each situation, were considered in the ZPSS. Loss of control power to an AC bus (or diesel generator) is not explicitly modeled, but a bounding analysis is provided that shows that the unavailability of DC power, <u>subsequent</u> to another initiating event, is highly unlikely. We concur with this analysis. We disagree, however, with the ZPSS analysis concerning the loss of a DC bus as an initiating event. Quarterly, each battery is placed on an equalization charge during which time the DC bus it powers is manually tied to a DC bus from the other unit. Each unit has experienced reactor trips due to improper switching of the buses. The alternative analysis for the loss of a DC bus as an initiating event proposed by this review is presented in Section 3.2.2.

(It should be further noted that on August 17, 1979, both Zion units tripped due to a severe lightning strike. This is of potential concern because several plants have experienced inverter failures as the result of lightning strikes. At Zion, however, no inverter failures ensued. In fact, no emergency electric system failures occurred at all.)

As to the actual failure probabilities used in the analysis, the ones of most interest are those concerning the diesel generators because of their potential impact on loss of off-site power sequences. Below are presented ZPSS values and those used in the TAP A-44 Study:

	ZPSS	TAP A-44
Diesel Generator, Failure to Start	1.8-2	2.5-2
Diesel Generator, Unavailable Due to Test and Maintenance	3.4-2	6.0-3
Common Mode Failure of Two Diesel Generators (Cooling Water Required)	1.1-4	6.6-4

Although the common mode value of two diesel generators used in the ZPSS is a factor of six lower than the generic value, the ZPSS failure probability of two diesel generators is only 20 percent lower than that suggested by the generic information. Primarily, this results from the higher maintenance unavailability of the diesel generators at Zion.

The above example is true in general. Specific probabilities in the ZPSS electric power analysis are different than those used in other studies and PRAs. The overall result, however, appears reasonable in comparison to the overall results of the other studies with the exception, as noted above, of the initiating event caused by the loss of a DC bus.

Two other points need mentioning in reference to the electric power system. First, values appearing in ZPSS Table 1.5.2.2.1-2D are repeated here. They are the mutually exclusive conditional probabilities for a given electric power state following a loss of off-site power initiating event. We reproduce them here because of their importance in our accident sequence review of Section 3.

Power Available at Unit 1 Bus	Conditional Probability of This Power State, Given LOP
147, 148, 149	0.38
147, 148	3.61(-2)
147, 149	3.61(-2)
148, 149	0.451
147	3.22(-3)
148	4.52(-2)
149	4.52(-2)
none	3.86(-3)

These probabilities, however, denote little with respect to the availability of power at Unit 2. The only information we do know about Unit 2 power is that, in those states above where Bus 147 is available, then Bus 247 cannot be. Hence only two buses, at most, can be available at Unit 2. If 147 is not available, though, the contrary does not automatically hold, that 247 is available. This is the second additional point we must address: given that 147 is not available following a loss of off-site power, what is the probability that 247 will be? The answer to this question is important to several subsequent calculations we will make.

Previously in this section, we gave the diesel generator failure to start datum as 1.8(-2) and its maintenance unavailability as 3.4(-2). (To simplify matters, we neglect diesel failure to run, given start). Thus, 5.2 percent of the time the swing diesel will be on neither bus, and 94.8 percent of the time it will be on 147 or 247. If we assume, as did the ZPSS, that given the diesel has started, its feeding 147 or 247 is random, then 47.4 percent of the time it is feeding 147 and 47.4 percent of the time, 247. (The selection is not random if one of the units has a safety injection signal present.) Therefore, the probability that 247 has power, given that 147 does not, is

$$P = \frac{0.474}{0.474 + 0.052} = 0.90 \quad .$$

Thus, for those Unit 1 power states where 147 is unavailable, 90 percent of the time Unit 2 potentially has three buses available (and definitely has one, 247), and 10 percent of the time, it can have at most two buses available.

2.4.1.2 Reactor Protection System Fault Tree

The failure probability of the reactor protection system (RPS) is calculated to be 1.8(-4) in section 1.5.2.2.2 of the ZPSS. After review of appropriate documents and numerous discussions, this value is found to be acceptable. The failure probability has three major contributors:

Random coincident failures of two trains of the trip system	1.7(-4)
Random failure of one train while other is in test or maintenance	6.2(-6)
Failure of rod control cluster assemblies to enter the core	3.0(-6)

Therefore, the largest contributor to RPS unavailability is that of random, coincident hardware failures in both trains. Failure of the two RPS trip breakers to open on demand comprises more than 80 percent of this value. It must be noted that the trip breakers at Zion open solely on undervoltage relays; there are no shunt coils in the circuit design.

The analysis of the Zion RPS is noteworthy in five respects:

1. Failure probability is higher than that calculated in other PWR PRAs, but comparable to that calculated in Reference 2-10.
2. The independent failure of two circuit breakers to open dominates the system unavailability.
3. Westinghouse DB-50 trip breakers are used at Zion, the same breaker type as that at Salem.
4. Zion has experienced five trip breaker failures during test.
5. The only RPS common cause failure analyzed in the ZPSS is that of instrument miscalibration.

As listed on p. 1.5-54 of the ZPPS, the five breaker failures are:

Date	Breaker	Cause
9/17/76	Unit 1 B Breaker	Dirty contacts
3/27/77	Unit 2 A Breaker	Unspecified
5/31/77	Unit 1 B Breaker	Undervoltage relay failure
5/08/79	Unit 2 A Breaker	Undervoltage relay plunger misadjusted
10/9/79	Unit 2 A Breaker	Undervoltage trip lever binding

These five could possibly result from the same improper maintenance and lubrication which caused the Salem ATWS.

Discussions were held with Commonwealth Edison personnel on this matter, and in addition a copy of their on-site review concerning the Salem event was obtained and reviewed (Reference 2-11).

Apparently, the maintenance and lubrication procedures developed by Westinghouse for the DB50 breakers were not originally received by Zion (NCD-ELEC-18 called out on p. 5-2 of Reference 2-12 and letter 74-2 denoted on p. F-51 of the same document). They were brought to the attention of Zion by Westinghouse during a meeting in late 1979 to discuss the above breaker failures. On December 13, 1979, Zion implemented these procedures (see Zion maintenance procedures ED15-1 and E000-3, attached to Reference 2-11), and no trip breaker failures have occurred since.

The fact that this problem appears to have been rectified at Zion, however, does not mean that other potential problems have also been rectified, but that the circuit breaker failure contribution to RPS failure from this failure mode is now negligible. We believe that the RPS failure probability of 1.8(-4) may very well be high when compared to that found in other PRAs, but without a detailed reanalysis, we have no specific basis for lowering its value. In addition, the Zion RPS unreliability is essentially the same as that used in Reference 2-10. Several sensitivities of potential ATWS core-melt sequences are analyzed in Section 4.7 of this report.

2.4.1.3 Safeguards Actuation System Fault Tree

The analysis of the Zion engineered safeguards actuation system, presented in Section 1.5.2.2.3 of the ZPPS, appears appropriate. Its application to the accident sequence analysis, however, appears wrong in two respects. First, an erroneous value is used to quantify event trees 1 and 2 (sequences initiated by large and medium LOCAs respectively), and secondly, for event tree 3, sequences initiated by a small LOCA, credit is given for actuation from high containment pressure sensing trains when that does not appear to be a viable actuation mode for small LOCAs.

As to the first problem, the event SA-1 in event trees 1 and 2 is the failure of the safeguards actuation system to generate a signal. The failure probability given to quantify these two trees is 2.9(-7) (See ZPSS Tables 1.3.4.1-2 and 1.3.4.2-2). On page 1.5-327 of the ZPSS, however, the failure probability is calculated to be 2.7(-6). The major contributors to this unavailability are random hardware failures in one train with the other train in test, 2.1(-6), random hardware failures in both trains, 3.0(-7), and the RCS pressure sensor signal blocked with coincident random hardware failures in the building sensor trains, 2.8(-7). These failure probabilities appear to be calculated correctly, and hence the value used in the event tree analyses is assessed to be in error.

As to the second problem identified above, the quantification of event tree 3 uses 2.8(-7) as the failure probability of the engineered safeguards actuation system, given a small LOCA initiating event (Table 1.3.4.3-2). As above, for consistency with the systems analysis, this number should be 2.7(-6). Building pressure sensors are assumed in the event tree quantification to be capable of actuating the necessary equipment following a small LOCA (page 1.3-113 of the ZPSS). On page 1.5-326, the ZPSS states that should the low RCS pressure signals fail, "For a large or medium LOCA, backup actuation signals are available from the containment pressure transmitters." Nothing is mentioned about small LOCAs. By implication, the high building pressure bistables will not be tripped in a sufficiently short time to actuate the necessary core-cooling equipment following a small LOCA, especially because containment fans are normally running.

Furthermore, other PRAs of PWRs with large dry containments and a high building pressure trip setting of 4 psig or above (e.g., ANO-1) have not given credit for this actuation method for core cooling during a small LOCA. The setpoint at Zion is 4.5 psig, and its containment is very large (nearly 3 million cubic feet). Without supporting analysis in the ZPSS, it is difficult to justify such an actuation possibility.

Lastly, the small LOCA initiating event frequency used in the ZPSS is dominated by the opening of a PORV. At Zion, such a small LOCA would egress to the quench tank and not the building atmosphere. Thus, there would be no pressure buildup in the containment from this initiating event until such time as the rupture disc on the quench tank functioned (which should be shortly after the initiating event).

If the safeguards actuation system fault tree (Figure 1.5.2.2.3-4) is requantified for the small LOCA case (i.e., high building pressure actuation fails with a probability of unity), the unavailability of SA-1 for a small LOCA is 2.2(-3). This value is almost exclusively the result of human error. When coming up to power at Zion, the low pressurizer pressure signal must be manually unblocked. The failure to do this is analyzed on page 1.5-326 of the ZPSS.

The failure probability of 2.2(-3) is unrealistic for small LOCAs, however, in that the time to core uncovery is not immediate and thus the operator can manually actuate the equipment. NUREG-1278 gives the probability of misdiagnosis at 30 minutes after an event to be 0.01 and at 60 minutes, 0.001. Hence, for small LOCA initiating events a maximum failure probability for SA-1 of 2.2(-5) appears more appropriate.

Actuation from the building pressure sensors alone is estimated in the ZPSS to fail with a probability of 1.6(-4). We concur with the analysis.

2.4.1.4 Zion High-Pressure Injection System Fault Tree

The Zion high-pressure injection (HPI) system fault tree was reviewed. In one operating mode or another HPI is part of every ZPSS event tree except ET1, the tree for a large LOCA initiating event. Following are comments regarding the system in general after which the medium and small LOCA success criteria cases will be examined.

The high-pressure injection system includes the safety injection system and the charging system. The safety injection system comprises two standby safety injection pumps and associated piping and valves. The charging system, which is normally in use during plant operation, comprises two centrifugal pumps and one positive displacement pump and associated plumbing. The positive displacement pump is not considered in the high-pressure injection system analysis.

The system is analyzed by segmenting it into "supercomponents." Supercomponents I, J, K, and L (see ZPSS Figure 1.5.2.3.1-1) are the valves in the safety injection lines to the reactor coolant system cold legs, each of which consists of two valves (see Figure 1.5.2.3.1-2). These are manual valves 9013 A, B, C, and D; and check valves 9012 A, B, C, and D. The Zion piping and instrumentation drawings indicate that an additional check valve, 9001 A, B, C, and D, exists in these lines and should have been included in the super-component analysis. Since the 9001 check valves are also common to the cold-leg injection paths for the low-pressure injection system, a single valve failure could preclude both high- and low-pressure injection in its associated valve train--a situation which is not reflected in the system models. However, because of the piping arrangements, the failure of the 9001 valves would appear in only higher order terms (triple or quadruple element cut sets). Therefore, their omission has no significant impact. With this minor exception, we believe that the modeling of the high-pressure injection system in the ZPSS is appropriate.

The ZPSS discussion of human error states that monthly and quarterly tests appear to minimize human error. However, we do not believe that this is the case. We note that pump tests are not "staggered" at Zion, and are, in practice, performed by the same people on the same day. Consequently, we believe that there is a strong human error dependency possible for this system.

Because of this testing policy, common mode failures needed to be examined in this review. The literature was searched, and a report by Corwin Atwood (EGG-EA-5289)[2-13] on the historic ß-factor associated with pumps in the nuclear power industry was used in the review. The draft of the precursor study was also examined, and the HP failures presented there are apparently accounted for in the Atwood work. (This appears to be true for the pumps of other systems as well.)

HPI for Medium LOCAs

Event tree ET2 requires HPI with success criteria of two of four pumps injecting into two of four headers (Event HH-1). The ZPSS indicates an unavailability for this system mode of 1.37(-6). The analysis allows for dependence among the pump trains by adopting a subjective ß-factor of 0.014. As described above, literature was searched to determine the applicability of this value. Data presented in Atwood indicate that, for failures and command faults of ESF standby pumps, a ß-factor of 0.165 should be used for a system which requires one of two standby pumps to operate and has monthly testing. A ß-factor of 0.090 should be used for the requirement of one of two alternating pumps. Applying these factors provides the following results:

Common mode failure of SI pumps = 7.21(-4) x .165
= 1.2(-4)

Common mode failure of charging pumps = 7.21(-4) x .090
= 6.5(-5)

where 7.21(-4) is the PSS value for random failure of the pumps.

With the application of the above ß-factors, the system unavailability (Event HH-1) was recomputed to be 2.7(-6) for the condition in which all electric power is available. The unavailability value is higher for various degraded electric power states. However, these were found to be unimportant in accident sequence evaluation.

This probability is potentially nonconservative for two reasons. First, the dependency parameters taken from the referenced report by Atwood are medians, not means, and hence the data presented biases the results obtained to the low side. The distribution of the data, however, indicates that the means and medians are quite close. In addition, the ß-factor is a ratio of common mode failures to all failures, and as a ratio, the median-mean difference is less. Secondly, the data given in Atwood are taken from the whole nuclear industry, not just from Zion. Hence, for example, the effects of staggered and nonstaggered tests are included while Zion does not stagger tests, as noted above, and might therefore have stronger dependencies among the pump trains. However, we feel that the Zion situation is adequately represented with the use of the generic data.

HPI for Small LOCAs

Event tree ET3 requires HPI with success criteria of one of four pumps injecting into one of four headers (Event HH-2). In addition, this event is part of many other event trees, either as HH-2 or by its inclusion as part of events SA-2 or OP-1 through OP-5. As in the case of HPI for medium LOCAs, described above, we believe that the ß-factor values from Atwood are more appropriate than the value used in the ZPSS. With this change we calculate an unavailability for HH-2 to be 2.1(-8) as compared to the ZPSS value of 7.4(-9).

HPI for Loss of Off-site Power

Due to the fact that HPI appears in many accident sequences, particularly those initiated by loss of off-site power (as part of OP2), it was found necessary to calculate unavailabilities for various degraded electric power states. Each of these calculations was based on the configuration of pump trains potentially available for a given combination of available AC power buses. The calculations are described below:

All AC power available - This is the basic calculation described above. The value is 2.1(-8).

Power on Buses 147 and 148 - Two safety injection pumps and one charging pump receive power from these buses. The failure probability is dominated by common mode failure of the SI pumps, 1.2(-4), and random failure or unavailability due to maintenance of the remaining charging pump train, 5.6(-3). The product of these values is 6.8(-7).

Power on Buses 147 and 149 - Two charging pumps and one safety injection pump receive power from these buses. The failure probability is dominated by common mode failure of the charging pumps, 6.5(-5), and random failure or unavailability due to maintenance of the SI pump train, 2.1(-3). The product of these terms is 1.4(-7).

Power on Buses 148 and 149 - One charging pump and one SI pump receive power from these buses. The product of random failure plus maintenance unavailability for the charging pump, 5.6(-3), and the same unavailabilities for the SI pump, 2.1(-3), is 1.2(-5).

Power on Bus 147 - One charging pump and one SI pump receive power from this bus. The unavailability is the same as that calculated for the Bus 148 and 149 case above, i.e., 1.2(-5).

Power on Bus 148 - Only one SI pump receives power from this bus. The value for HH-2 is the unavailability of this SI pump due to random failure or maintenance, i.e., 2.1(-3).

Power on Bus 149 - Only one charging pump receives power from Bus 149. The value for HH-2 is the unavailability of this charging pump due to random failure or maintenance, i.e., 5.6(-3).

Failure of All AC Buses - With no power available all pumps fail and the value of HH-2 is 1.0.

Feed and Bleed

Since the high-pressure injection system is involved in the feed and bleed process, we present our review of this topic here. The basic feed and bleed process is one in which the operator provides primary system cooling when other cooling options are not available. As described in the ZPSS, the operator must do the following:

a. Recognize that auxiliary feedwater and secondary heat removal have failed.

b. Start a charging pump/safety injection pump (if pressure is low enough) and establish valve lineup if needed.

c. Open both power-operated relief valves (PORVs).

d. Verify that adequate heat removal is taking place.

The ZPSS identifies the following contributors to feed and bleed failure:

Human Error	1.30(-4)
PORV Fails to Open	1.44(-3)
Block Valve Fails to Open	1.55(-3)

Because there are two PORVs and two block valves, the ZPSS calculation is

$$2[1.44(-3) + 1.55(-3)] + 1.3(-4) = 6.11(-3) \quad .$$

for the case where all electric power is available. For the case where no power is available at AC Bus 149 the conditional failure probability is given as 1.0, since the normally-closed block valves were powered from this bus.

Subsequent to publication of the ZPSS, the PORV block valves at Zion were changed from normally closed to normally open. Due to this change, the failure of block valves to open (reflected in the calculation above) is no longer applicable, and the basic failure probability for feed and bleed when AC power is available becomes

$$2[1.44(-3)] + 1.3(-4) = 3.0(-3) \quad .$$

Note that in these calculations the probability of failure of high head injection (charging and safety injection pumps) is not reflected, because their failure probability is not large enough to significantly affect the results. Given a loss of off-site power, however, unavailability of high-pressure injection becomes a significant factor. Since the PORVs are not AC power dependent, their contribution to feed and bleed failure remains the same for all degraded AC power states. The probability of human error also remains the same. To calculate the feed and bleed unavailability for degraded power states we applied the unavailability of high-pressure injection for each power state (described elsewhere in this section) to the human error and PORV unavailability.

The results are as follows:

Power at all AC buses	3.0(-3)
Power at Buses 147 and 148	3.0(-3)
Power at Buses 147 and 149	3.0(-3)
Power at Buses 148 and 149	3.0(-3)
Power at Bus 147	3.0(-3)
Power at Bus 148	3.0(-3) + 2.1(-3) = 5.1(-3)
Power at Bus 149	3.0(-3) + 5.6(-3) = 8.6(-3)
No AC bus power	3.0(-3) + 1.0 = 1.0

While the PORVs are independent of AC power, they do require DC power. Specifically, one of the PORVs requires power from DC Bus 111 and the other from DC Bus 011-1. Since both PORVs are required to open for success of feed and bleed, the probability of feed and bleed failure given loss of either of these buses is 1.0.

2.4.1.5 Zion Low Pressure-Injection Fault Tree

The Zion low pressure-injection (LPI) system fault tree was reviewed and found to be appropriate. This tree was used to evaluate events LP-1 (Large LOCA) and LP-2 (Medium LOCA). In the ZPSS evaluation of the fault tree a common cause ß-factor of .014 was used for the residual heat removal pumps. As noted in the discussion of the high-pressure injection system fault tree, we feel that the ß-factor developed by Atwood for the two pump standby system (.165) is more appropriate. Results of calculations using the Atwood value are discussed below.

<u>LPI for Large LOCA</u>

Event tree ET1 requires LPI (Event LP-1) with success criteria of one of two pumps and three of four accumulators. The ZPSS calculates a value for this event of 4.65(-4) for LPI plus 9.5(-4) for accumulators = 1.4(-3). Our evaluation assumed a common mode ß-factor of .165 rather than the .014 value used in the ZPSS. The resulting calculations produced a value for LPIS unavailability of 5.6(-4) and a value for LP-1 of 5.6(-4) plus 9.5(-4), for the accumulators = 1.5(-3).

<u>LPI for Medium LOCA</u>

Event tree ET2 requires LPI (Event LP-2) with success criterion of one of two pumps (accumulators not required). Based on the calculations described in LPI for Large LOCA above, the ZPSS value for event LP-2 is 4.65(-4) and our calculation is 5.6(-4).

2.4.1.6 Zion Accumulator System Fault Tree

The fault tree constructed for the Zion accumulator system and the evaluation were found to be correct. The results are included in the discussion of the LPI fault tree above.

2.4.1.7 ZPSS Recirculation System Fault Trees

The hardware portions of the recirculation systems analysis were herein reviewed. The human error contributions, which in the ZPSS are the dominant causes of failure, are examined in Section 2.5 of this report. In the ZPSS, there are three types of recirculation considered: high pressure, low pressure, and containment spray.

High-Pressure Recirculation (HPR)

The mean failure probability for this system is calculated as 3.9(-4) in the ZPSS of which 1.6(-4) results from human error and 2.3(-4) results from hardware failure. For HPR the RHR portion of the LPRS must be operating, as well as component cooling water. These dependencies are explicitly modeled on the fault tree. The calculation of system unavailability included the use of a β-factor of .014 for common mode failures. As noted above, Atwood β-factors of .165 for the RHR and safety injection pumps and .090 for the charging pumps are considered to be more appropriate values. However, the common mode factor in the ZPSS was applied to both pumps and motor-operated valves. The common mode contribution to system unavailability is nearly the same using either approach. With this minor exception we believe the modeling and calculations for high-pressure recirculation are appropriate.

Low-Pressure Recirculation (LPR)

The failure probability for LPR at Zion is given as 5.16(-3) of which 4.8(-3) is human error and 3.6(-4) is hardware. As in the case of the high-pressure recirculation analysis, we differ with the β-factor for common mode quantification, but the result of recalculation using a higher factor is not significantly different. In all other respects the modeling and calculations for low-pressure recirculation appear to be appropriate.

Containment Spray Recirculation (CSR)

The failure probability of the CSR system is estimated in the ZPSS as 1.6(-3) of which 94 percent is human error. Since the availability of core-cooling recirculation is assumed in the ZPSS, that portion of the CSR system upstream of the heat exchangers is assumed to succeed. Therefore, the only significant hardware-related failure is for two motor-operated valves to fail closed. We found the modeling and analysis of containment spray recirculation to be appropriate.

2.4.1.8 Zion Containment Spray Injection Fault Tree

The fault tree analysis for the Zion containment spray injection system appears to be inconsistent with the analysis of other systems in the treatment of common mode failure. The ZPSS states that because the sprays are a standby system and because of system diversity (two motor-driven pumps and one diesel-driven pump) common cause failures are insignificant relative to other causes. To be consistent with the treatment of other systems, common mode failures should have been considered in the analysis.

For the two motor-driven pumps, the common mode β-factor, from Atwood, is .165. The failure probability of these pumps is then .165 times the failure to start value, 7.2(-4) or 1.2(-4). The product of motor-driven pump common mode, 1.2(-4), and the unavailability of the diesel-driven pump, 6.8(-2) is 8.1(-6). The addition of this value to the other system failure modes results in an overall system unavailability of 6.3(-5) rather than 5.5(-5) as indicated in the ZPSS. Although the difference is only about 15 percent, the consistent treatment of common mode is considered desirable.

Because the containment spray system appears in many accident sequences (as Event CS) associated with loss of off-site power, it was necessary to calculate system unavailabilities for various degraded electric power states. Each of these calculations was based on the configuration of pump trains potentially available for a given combination of available AC power buses (with one pump required for system success). The calculations are described below:

<u>All AC Power Available</u> - This is the basic calculation for CS described above. The value is 6.3(-5).

<u>Power on Buses 147 and 148</u> - The two motor-driven pumps receive power from these buses. The failure probability is dominated by the common mode failure of the pumps, 1.2(-4).

<u>Power on Buses 147 and 149</u> - One motor-driven pump receives power from Bus 147. The diesel-driven pump train is controlled by Bus 149. The system failure probability in this case is dominated by failure of these two trains, 4.6(-4).

<u>Power on Buses 148 and 149</u> - One motor-driven pump receives power from Bus 148. The diesel-driven pump

train is controlled by Bus 149. The system unavailability is the same as that calculated for the Bus 147 and 149 case, above, 4.6(-4).

Power on Bus 147 - One motor-driven pump receives power from Bus 147. The system unavailability is dominated by the failure of this pump train, i.e., 6.8(-3).

Power on Bus 148 - One motor-driven pump receives power from Bus 148. The system unavailability is 6.8(-3).

Power on Bus 149 - The diesel-driven pump train is controlled by power from this bus. The system unavailability is dominated by the failure of this train, i.e., 6.8(-2).

Failure of All AC Buses - With no power available, all pump trains fail and the value of CS is 1.0.

Unavailability of the Sodium Hydroxide Addition System was included in the ZPSS, and an analysis was presented in the fault tree section. Discussions with Zion personnel led to the conclusion that this system is not required for success of the containment sprays. Consequently, no fault tree analysis for this system is included here.

2.4.1.9 Zion Containment Fan-Cooling System Fault Tree

The containment fan-cooling system comprises five motor-driven fans with associated cooling coils which are normally operating as necessary to maintain containment pressure within appropriate limits. Following a LOCA, the fans are switched automatically to the accident mode. Manual switching to this mode is also possible. According to the ZPSS, successful operation in the accident mode requires that at least three of the five fans switch to the accident mode.

The fault tree constructed for the fan cooling system omits the failure contribution of SW 0767, a single, manually-operated valve in the return line of the fan-cooling coils. The contribution of this valve is calculated by assuming a 1.0(-7) failure rate per hour for plugging of the normally-open manual valve, for the 24-hour period of the analysis. This contribution, 2.4(-6), added to the other contributors identified in the ZPSS, results in a system unavailability of 3.1(-6) rather than the ZPSS value of 6.1(-7).

Three of the assumptions used in analysis could alter the calculated system unavailability. Two of these would decrease the value and one would increase it. Of the former type, the success criterion assumed in the analysis is that of the Zion FSAR, that three of the five fan-cooling units must operate to achieve system success. Other PRAs (such as ANO-1) have found that the success criteria for fan systems which are reported in safety analysis reports can be conservative, instead of realistic. Hence, the calculated fan system failure probability may be conservative. The second conservative assumption is that the analysis does not give credit for manual actuation of the system; only automatic actuation is considered. Because of the relatively long time available for operator recovery actions to restore system function and prevent containment overpressurization, manual actuation is viable. Failure of automatic actuation, however, is a small contributor to the overall system unavailability.

The assumption which is potentially nonconservative is that the charcoal filter beds will not plug with airborne debris during the course of the accident. This assumption has been made in other PRAs (again, such as ANO-1) but has been a subject of sensitivity studies in them because the phenomenology is not currently well-defined. The sensitivity of the overall risk to the assumption that the fan coolers will operate was not examined in the ZPSS, but is investigated in Section 4.1 of this report.

Since the containment fan-cooling system appears in many accident sequences associated with loss of off-site power (as Event CF), it was necessary to calculate system unavailabilities for various degraded electric power states. Each of these calculations was based on the configuration of fan-cooler trains potentially available for a given combination of available AC power buses and the system success criterion (three out of five fans required). The calculations are described below:

<u>All AC Power Available</u> - This is the basic calculation for CF described above. The value is 3.1(-6).

<u>Power on Buses 147 and 148</u> - Three fan-cooler units receive power from these buses. Therefore, a single unit failure would result in system unavailability. The value of CF in this case is 1.1(-2), as calculated in ZPSS Section 1.5.2.3.6.4.

<u>Power on Buses 147 and 149</u> - Three fan-cooler units receive power from these buses. The system unavailability is the same as for the Bus 147 and 148 case, i.e., 1.1(-2).

<u>Power on Buses 148 and 149</u> - Four fan-cooler units receive power from these buses, therefore, the failure of two of these units would cause system unavailability. The equation for failure of two out of four units is given in ZPSS Section 1.5.2.3.6.4. However, correction

of an arithmetic error plus the addition of a term for failure of valve SW 0767 results in a value for this case of 9.2(-5).

<u>Power on Bus 147 or 148 or 149</u> - The availability of power on only one of these three buses would mean that only one fan-cooler unit (Bus 147) or two fan-cooler units (Bus 148 or 149) would receive power. Since system success requires three units, the unavailability of the system (Event CF) would be equal to 1.0.

2.4.1.10 Zion Component Cooling Water System Fault Tree

The component cooling water (CCW) system at Zion comprises five pump trains which supply a common discharge header which is the source of component cooling water for both units. Component cooling water from cooled components is also returned to a common pump suction header. Components cooled by the CCW system include the charging pumps, the safety injection pumps, the residual heat removal pumps, the residual heat removal heat exchangers, the reactor coolant pump seal thermal barriers, and the spent fuel pit cooling heat exchangers, as well as several other miscellaneous loads.

For the case where Zion Unit 1 experiences an initiating event other than loss of off-site power and Unit 2 is operational, the ZPSS analysis assumes that a minimum of three operating CCW pumps are required for system success. From a review of the CCW loads (see ZPSS Figure 1.5.2.3.7-2) we conclude that only two pumps would be needed. In fact we believe that if the spent fuel pit cooling load were isolated following an initiating event, the remaining CCW loads could be supplied by one pump. However, we do not find any provisions for isolating spent fuel-pit cooling in the Zion procedures. The number of pumps required is an important factor in the analysis because component cooling water system failure is a contributor to many important accident sequences. In the following analysis we assume that two pumps are required for component cooling water system success.

<u>Loss of Component Cooling Water in LOCA Sequences (Event CC)</u>

The equation for system unavailability, given that three pumps are required, is found on page 1.5-614 of the ZPSS. We modify the equation as follows to reflect the two-pump criterion:

$$Q_{pumps} = P_{(2S)} \left(3\, P_{(OP)}^{2}\right) \cdot P_{(STBY)}^{2} + 2\left(P_{(OP)}^{3} \cdot P_{(STBY)}\right)\Big]$$

$$+ P_{(1S)} \left[4\left(P_{(OP)}^{3} \cdot P_{(STBY)}\right) + P_{(OP)}^{4}\right] + P_{(OP)} \left[5\, P_{(OP)}^{4}\right]$$

With the use of the component values given in the ZPSS, this results in a value of 9.3(-12).

The foregoing, however, assumes (as did the Zion analysts) that common mode failure of the CCW pumps was negligible. Our conclusion is that common mode pump failure should be included in the calculations. We believe that the Atwood common mode ß-factors are appropriate. Consequently, we add the following to the above results, based on the component failure rates presented in the ZPSS:

Common Mode Failure of Four Out of Five Pumps (β = .021)

Q_{cm} = Pump Failure Probability · ß-factor for failure of four pumps out of five

= 4.2(-5) · .021 = 8.9(-7)

where 4.2(-5) is the value for a pump failing to run for 24 hours, (from the ZPSS).

Common Mode Failure of Three Out of Four Running Pumps (β = 0.04)

If we assume one pump in maintenance, a maximum of four pumps could be running. Then the common mode of failure of three out of four running pumps would be

4.2(-5) x .04 = 1.7E-6

and

1.7E-6 x 3.2-2 (maintenance unavailability of 1 pump)

= 5.4(-8) .

Common Mode Failure of Two Out of Three Running Pumps (β = 0.09)

If we assume two pumps in maintenance, a maximum of three pumps could be running. Then the common mode failure of two out of three running pumps would be

$$4.22(-5) \times .09 = 3.8(-6)$$

and

$$3.8(-6) \times 7.7(-3) = 2.9(-8)$$

where 7.7(-3) is the ZPSS frequency of two pumps in maintenance.

Summing these common mode probabilities we obtain

$$8.9(-7) + 5.4(-8) + 2.9(-8) = 9.7(-7)$$

which is our estimate for component cooling-water system unavailability in LOCA sequences (Event CC).

Loss of Component Cooling Water Given LOP (Event LS)

In the ZPSS analysis of accident sequences initiated by loss of off-site power (LOP), the frequency of loss of reactor coolant pump seal cooling appears as event LS. This frequency is calculated in the ZPSS by assuming that one CCW pump is required for system success and that pump failure probability is equivalent to the failure probability of the AC bus which provides its electric power. We differ from the ZPSS with regard to both of these assumptions. We assume that two CCW pumps are required for system success and that the pump failure probability arises from pump failure to start (following loading of the emergency diesel generators on their AC buses) and pump unavailability due to maintenance, as well as AC bus failure. Our calculations are based on the combinations of AC bus failures and pump failures which would lead to event LS. The contributors to these combinations and the values given in the ZPSS are as follows:

CCWP - Failure of CCW pump to start	7.2(-4)	ZPSS page 1.5-612
CCWM - CCW pump in maintenance	0.032	ZPSS page 1.5-606
CCW2M- Two CCW pumps in maintenance	7.7(-3)	ZPSS page 1.5-606
DGFS - Failure of diesel generator to start	0.018	ZPSS page 1.5-238
DGM - Diesel generator in maintenance	0.034	ZPSS page 1.5-191

Using these values we examine the CCW system failure probability for the initiating event loss of off-site power to both units. Note that in this event both units are tripped and all components must restart. The degraded power states examined are:

Case 1: Power available at no buses of Unit 1
Case 2: Power available at 1 bus of Unit 1
Case 3: Power available at 2 buses of Unit 1
Case 4: Power available at 3 buses of Unit 1

Case 1: The diesel generators at AC Buses 147, 148, and 149 have failed. Two component cooling water pumps are needed; the diesels for Buses 248 and 249 and both pumps powered by these buses must therefore succeed. Thus, a single bus (diesel) or pump failure will result in event LS. For this probability we calculate

Q_{LS} = 1 of 2 diesel generators failing or 1 of 2 CCW pumps failing

$$= 2(\text{DGFS} + \text{DGM}) + 2(\text{CCWP} + \text{CCWM}) = 0.17 \quad .$$

Case 2: One diesel generator (at AC Bus 147, 148, or 149) has succeeded and the other two have failed. We, again, have the possibility of pumps available on Buses 248 and 249. If two pumps fail (due either to pump unavailability or loss of power), event LS will occur. Thus, two diesel generators may fail, or one diesel generator and one pump, or two pumps.

$$Q_{LS} = \text{DGFS}^2 + 2(\text{DGFS} \cdot \text{DGM}) + 4(\text{DGFS} + \text{DGM})(\text{CCWP} + \text{CCWM})$$
$$+ 3\ \text{CCWP}^2 + 6\ \text{CCWP} \cdot \text{CCWM} + 3\ \text{CCW2M} = 3.2(-2) \quad .$$

Case 3: Two diesel generators are available at AC Buses 147, 148, or 149 and the other diesel generator has failed. We have the possibility of pumps available on Buses 248 and 249. If three pumps fail (due either to pump unavailability or loss of power), event LS will occur. Thus, two diesel generators and one CCW pump may fail, or one diesel generator and two pumps, or three pumps.

$$Q_{LS} = 2(\text{CCWP} + \text{CCWM})(\text{DGFS}^2 + 2\ \text{DGFS} \cdot \text{DGM})$$
$$+ (2)(\text{DGFS} \cdot \text{DGM})(3)(\text{CCWP}^2 + 2\ \text{CCWP} \cdot \text{CCWM} + \text{CCW2M})$$
$$+ 4\ \text{CCWP}^3 + 12\ \text{CCWP}^2 \cdot \text{CCWM} + 12\ \text{CCWP} \cdot \text{CCW2M}$$
$$= 2.5(-3) \quad .$$

Case 4: Three diesel generators are available at AC Buses 147, 148, and 149. We have the possibility of CCW pumps available on all five buses. If four pumps fail (due either to pump unavailability or loss of power) event LS will occur. Thus, two diesel generators and two pumps may fail, or one diesel generator and three pumps, or four pumps.

$$Q_{LS} = (DGFS^2 + 2\ DGFS)(3\ CCWP^2 + 6\ CCWP \cdot CCWM + 3CCW2M)$$

$$+ 2(DGFS + DGM)(4\ CCWP^3 + 12\ CCWP^2 \cdot CCWM$$

$$+ 12\ CCWP \cdot CCW2M) + 5\ CCWP^4 + 20\ CCWP^3 \cdot CCWM$$

$$+ 30\ CCWP^2 \cdot CCW2M = 4.3(-5) \quad .$$

Note that this frequency, 4.3(-5) is greater than the frequency calculated for the sequences initiated by a LOCA because loss of off-site power results in a shutdown and restart of all CCW pumps whereas in the LOCA case three pumps are running and continue to run.

The discussion in this section has dealt with component cooling water system unavailability following a LOCA or loss of off-site power. Loss of component cooling water as an initiating event is a separate issue which is discussed in Section 3.2.1.

2.4.1.11 Zion Service Water System Fault Tree

The Zion service water (SW) system consists of six pumps supplying two main crosstie headers which in turn are connected to the service water loads at both units. The headers are normally connected so that any combination of pumps can serve both units. During normal operation two pumps are operating per header. The ZPSS states that, during abnormal operation, any combination of three operating pumps constitutes system success. However, the Zion Final Safety Analysis Report indicates that the system will perform satisfactorily with two operating pumps. Therefore, two pumps appear to be adequate for system success.

The components requiring service water during abnormal plant conditions are the five emergency diesel generators, the three component cooling water heat exchangers, the five containment ventilation coolers of each unit, the containment spray pump diesel engine oil coolers, the motor-driven auxiliary feedwater pump coolers and the pump room coolers. In addition the service water system operates as a backup water supply to the auxiliary feedwater pumps.

Although the ZPSS states that three pumps are necessary for service water system success, the fault tree for the service water system appears incorrect in that it implies that failure of all SW pumps would be required in order for the system to fail. The ZPSS reliability block diagram is also inaccurate in that it suggests that two of three pumps feeding one header and one of three pumps feeding the other header are required for system success. However, the system success criterion contained in the note on the reliability block diagram (Figure 1.5.2.3.8-1) is consistent with the stated success criterion of three pumps. We believe, however, that common mode failure of the service water pumps should have been included in the analysis.

In the "No Loss of Off-site Power" scenario analyzed in the ZPSS, four pumps are assumed to be running at the time of an initiating event. An additional pump will be started by the safeguards actuation signal. If it is assumed that the other standby pump would be actuated by the operator if needed, the common mode failure of five of the six pumps would result in system failure. The Atwood tables for this type of pump does not provide a ß-factor for common mode failure of five out of six pumps, so we adopt the .014 factor used by the ZPSS for common mode failure of pump trains in other systems. On this basis, common mode failure of SW is

$$Q_{cm} = 3.3(-5) \cdot 0.014 = 4.6(-7)$$

where 3.3(-5) is the value for a pump failing to run for 24 hours given in the ZPSS.

Other combinations of common mode failure (of fewer than five pumps) with maintenance, human error and pump failure to start were examined. However, they did not contribute significantly to the overall common mode failure frequency.

Addition of the 4.6(-7) common mode failure frequency to the 2.2(-8) system failure frequency calculated in the ZPSS for other causes, results in an overall value for the service water system of 4.8(-7), in the "No Loss of Off-site Power Case."

<u>Service Water for Loss of Off-site Power</u>

Due to the fact that the service water system has some impact on loss of off-site power sequences, it was found necessary to calculate unavailabilities for various degraded electric power states. Each of these calculations was based on the configuration of pump trains potentially available for a given combination of available AC power buses. The calculations are described below:

Service water pump power supplies are arranged as follows:

Pump 1A	AC Bus 147
Pump 1B	AC Bus 148
Pump 1C	AC Bus 149
Pump 2A	AC Bus 247
Pump 2B	AC Bus 248
Pump 2C	AC Bus 249

Since only five emergency diesel generators exist, one diesel generator will be connected to either Bus 147 or Bus 247, leaving the other of these two buses unpowered. Consequently, given loss of off-site power, only five pumps and five AC buses are available. Under these circumstances any combination of pump and bus failures which would render four of the five pumps unavailable would cause system failure. The contributors to these combinations and the values given in the ZPSS are as follows:

SWP	- Failure of SW pump to start	7.2(-4)	ZPSS page 1.5-656
SWM	- SW pump in maintenance	2.31(-3)	ZPSS page 1.5-650
DGFS	- Failure of diesel generator to start	0.018	ZPSS page 1.5-238
DGM	- Diesel generator in maintenance	0.034	ZPSS page 1.5-191

Using these values, we examine the SW system failure probability for the initiating event loss of off-site power to both units. Note that in this event both units are tripped and all components must restart. The degraded power states examined are:

Case 1: Power available at no buses of Unit 1
Case 2: Power available at Bus 147 of Unit 1
Case 3: Power available at Bus 148 or 149 of Unit 1
Case 4: Power available at Bus 147 and one other bus of Unit 1
Case 5: Power available at Buses 148 and 149 of Unit 1
Case 6: Power available at three buses of Unit 1

The analysis is rendered more complicated by the fact that power may be absent from Bus 147 of Unit 1 for two reasons -- the diesel which serves this bus may have failed or it may be connected to Bus 247 of Unit 2. We do not know for any initiating event, the position of the "swing" diesel. However, we calculate that, given there is no power on Bus 147, there is a .10 probability that the "swing" diesel is unavailable and a .90 probability that it is connected to Bus 247. To reflect this situation, we examine two cases for each power state involving absence of power from Bus 147 -- two buses potentially available at Unit 2 (assuming the "swing" diesel has failed), and three buses potentially available at Unit 2 (assuming the "swing" diesel is connected to Bus 247). Values from these calculations are then combined probabilistically. The degraded power states examined are:

Case 1A: No diesels at Unit 1, two at Unit 2. Two SW pumps are required for system success, so the diesels at 248 and 249 must start and both pumps powered by these diesels must succeed. A single bus (diesel) failure or pump failure will result in event SW. For this probability, we calculate

Q_{1A} = 1 of 2 diesel generators or 1 of 2 SW pumps failing

$$.9\ [2(DGFS + DGM) + 2(SWP + SWM)] = 1.1(-2) \quad .$$

Case 1B: No diesels at Unit 1, three at Unit 2. We have the possibility of three buses at Unit 2. If two pumps fail (due either to pump unavailability or loss of power) event SW will occur. Thus two diesel generators may fail, or one of three diesel generators and one of two pumps, or two of three pumps.

$$Q_{1B} = 0.9[(DGF^2 + 2(DGF)(DGM) + 6(SWP + SWM)(DGF + DGM)$$
$$+ 3SWP^2 + 6(SWP + SWM)] = 2.25(-3)$$

Case 1 total = 1.1(-2) + 2.25(-3) = 1.3(-2)

Case 2: Power available at Bus 147 (Unit 1). One diesel generator at Unit 1 (Bus 147) has succeeded and the other two have failed. Two SW pumps are needed. Thus, two buses at Unit 2 may fail, or one of two buses at Unit 2 and one of two pumps may fail, or two of three pumps may fail.

$$Q^2 = (DGF^2 + 2(DGF)(DGM) + 6(SWP + SWM)(DGF + DGM)$$
$$+ 3\ SWP^2 + 6(SWP)(SWM) = 2.5(-3) \quad .$$

Case 3A: Power available at Bus 148 or 149 (Unit 1) and potentially available at two buses of Unit 2.

This is equivalent to Case 2, above so

$$Q_{3A} = 0.1(2.5-3) = 2.5-4 \quad .$$

Case 3B: Power available at Bus 143 or 149 and potentially available at three buses of Unit 2. Two pumps are needed. Thus, in this case, all three buses at Unit 2 may fail, or two of three buses and one of two pumps may fail, or one of three buses and two of three pumps may fail or three of four pumps may fail.

$$Q_{3B} = 0.9[6(SWP^2 + 2\ SWP \cdot SWM)(DGF + DGM)$$
$$+ 2(DGF^2 + 2\ DGF \cdot DGM)(SWF + SWM)$$
$$+ 4\ SWP^3 + 12\ SWP^2 \cdot SWM] = 9.5(-6) \quad .$$

The combined case of 3A and 3B is

$$2.5(-4) + 9.5(-6) = 2.6(-4) \quad .$$

Case 4: Power available at Buses 147 and 148 or at Buses 147 and 149. In this case power is available at two buses of Unit 1 and potentially available at two buses of Unit 2. In this case two buses at Unit 2 and one of two pumps may fail, or one bus at Unit 2 and two of three pumps may fail, or three of four pumps may fail.

$$Q_4 = 2(DGF^2 + 2\ DGF \cdot DGM)(SWP + SWM)$$
$$+ 6(SWP^2 + 2\ SWP \cdot SWM)(DGF + DGM) + 4\ SWP^3$$
$$+ 12\ SWP^2 \cdot SWM = 1.1(-5)$$

Case 5A: Power available at 148 and 149, two buses at Unit 2. In this case two buses are available at Unit 1 and two buses are potentially available at Unit 2. This is the same as Case 4 above, so

$$Q_{5A} = 1.1(-5) \cdot 0.1 = 1.1(-6)$$

Case 5B: Power available at 148 and 149, three buses at Unit 2. Two buses are available at Unit 1 and three buses potentially available at Unit 2. In this case three buses and one of two pumps may fail, or two buses and two of three pumps may fail, or one bus and three of four pumps may fail or four of five pumps may fail.

$$Q_{5B} = 0.9[(DGF^2 + 2\ DGF \cdot DGM)(3SWP^2 + 6SWP \cdot SWM)$$
$$+ 2(DGF + DGM)(4\ SWP^3 + 12\ SWP^2 \cdot SWM) + 5\ SWP^4$$
$$+ 12\ SWP^3 \cdot SWM] = 1.75(-8) \quad .$$

The combined case of 5A and 5B is

$$1.1(-6) + 1.75(-8) = 1.1(-6)$$

Case 6: Power available at all three buses of Unit 1 and potentially available at two buses of Unit 2. In this case two buses and two of three pumps may fail, or one bus and three of four pumps may fail, or four of five pumps may fail.

$$Q^6 = (3\ DGFS^2 \cdot SWP^2) + 8(DGFS + DGM)(SWP^3)$$
$$+ 5\ SWP^4 = 2.0(-8) \quad .$$

2.4.1.12 Auxiliary Feedwater System Fault Tree

The Zion Auxiliary Feedwater (AFW) system was difficult to review. The lack of discussion of specific basic event frequencies and cause frequencies makes the analysis difficult to follow. There is no simplified schematic of the system which would have been helpful. There are no equations presented in the text which would correspond to the fault trees. There are no equations for the supercomponent analysis. In general, this section of the report concedes nothing to the reader. There are several minor discrepancies in the AFW analysis which do not have a significant impact on the results. Some of these are described below. The omission of common mode pump failure in the analysis, which does have a significant influence on results, is also discussed.

The fault tree for the auxiliary feedwater system appears incorrect in that it indicates that failure of a check valve between each AFW output line and a corresponding main feedwater line would reduce the flow from the AFW output line by more than 50 percent. We believe that other valves in the main feedwater system would prevent this from being the case. In other respects, the fault tree appears to be correct, and this apparent error does not significantly impact the system analysis.

Each of the output lines from either the motor-driven pump (MDP) or turbine-driven pump (TDP) headers to the steam generators has an air-operated valve which was not modeled on the fault tree.

Rupture or flow blockage in the discharge line from motor-driven pump C is considered (PPP0018L) but not so for the motor-driven pump B. This would be common to both pumps. This event while on the fault tree is not in the list of basic events (Table 1.5.2.3.9-4). Similarly, plugging of the discharge line is not considered for pump A. Perhaps this is insignificant, but there is no corresponding discussion.

The AFWS turbine pump fault tree branch P includes many failure modes of that component to start but does not consider the turbine failure. This is probably insignificant relative to the other failures, but nevertheless should be on the fault tree for completeness.

Two comments regarding the supercomponents. First, the layout for supercomponent G is drawn incorrectly at the junction of the pump outputs and header to the steam generator. In any case we cannot reproduce the calculation to get the mean value of 4.64×10^{-5}. The equation used would be very helpful to the reader. Second, supercomponents H and I refer to loss of Bus 148 and 149 respectively. As it turns out these supercomponents also correspond to the single motor-driven pump trains C and B respectively and are used in Table 1.5.2.3.9-7 that way. This should be discussed for the sake of clarity.

The Service Water System shows SW to the motor-driven AFWS pumps for cooling (also reference Section 1.5.2.3.8.2.1 page 1.5-645). Verbal communication from ZPSS personnel implies that SW cooling of the AFWS pump is not necessary; nevertheless, we cannot find supporting analysis for this. This is a potential common mode. Another AFWS problem resulting from the SWS is found in ET-11c where the failure of the SWS is the initiating event. Therefore, no credit should have been given for SW backup to the CST.

As noted above, the analysis of the AFW system included an assumption that common mode failure of the pumps was negligible. Our conclusion is that common mode pump failure should be included in the calculations. The Atwood common mode β-factor for motor-driven pumps appears to be applicable. The calculation proceeds by applying the β-factor of 0.15, for two motor-driven AFW pumps, to the individual pump failure rate of 5.81(-3) in Table 1.5.2.3.9-6c

$$5.81(-3) \cdot 0.15 = 8.7(-4)$$

which provides the common mode failure frequency of the motor-driven pumps. This is then combined with the sum of random failure and maintenance outage frequencies of the AFW turbine-driven pump

$$8.7(-4) \cdot 0.039 = 3.4(-5)$$

a result significantly larger than the 4.2(-6) frequency given in the ZPSS.

The following values for the turbine-driven AFW pump are given in Table 1.5.2.3.9-1A:

Hardware "random" failure	0.0131
Test and maintenance	0.0358
Human error	0.0005
Total	0.0494

The test and maintenance contribution given was based primarily on maintenance outage data for the years 1975 through 1980. Data for the years 1981 and 1982 have recently become available. Analysis of the complete maintenance record indicates that an appropriate estimate for test and maintenance unavailability should be 0.026, which, when combined with the hardware failure and human error frequencies would yield an overall turbine-driven pump unavailability of 0.039.

AFW Unavailability Values for Loss of Off-site Power

Due to the fact the AFW appears in accident sequences initiated by loss of off-site power, it is necessary to calculate unavailabilities for various degraded electric power states. Each of these calculations was based on the configuration of pump trains potentially available for a given combination of available emergency AC power buses. Note that motor-driven pumps B and C receive power from AC buses 148 and 149 respectively, and that turbine-driven pump A has no AC power dependency. The calculations are described below.

All AC Power Available - This is the basic calculation for AFW described above. The value is 3.4(-5).

Power on Buses 147 and 148 - In this case, motor pump C and the turbine-driven pump are available. Unavailability of the motor-driven pump is 5.89(-3) and the turbine-driven is 0.039. The combined unavailabilities are:

$$5.89(-3) \cdot 0.039 = 2.3(-4)$$

Power on Buses 147 and 149 - Motor pump B and the turbine-driven pump are available. The calculated system failure frequency is the same as for power on Buses 147 and 148, i.e. 2.3(-4).

Power on Buses 148 and 149 - In this case power is supplied to both motor-driven pumps, so all pumps are available. As in the case of power present at all AC buses, the system unavailability is 3.4(-5).

Power on Bus 147 - In this case no power is supplied to the motor-driven pumps, so only the turbine-driven pump is available. The unavailability of the turbine-driven pump, analyzed earlier in this section is 0.039.

Power on Bus 148 - With no power on Bus 149, only one motor-driven pump and the turbine-driven pump are available. This is the same as the case for power on Buses 147 and 148, i.e., 2.3(-4).

Power on Bus 149 - With no power on Bus 148, only one motor-driven pump and the turbine-driven pump are available. This is the same as the case for power on Buses 147 and 148, i.e., 2.3(-4).

Failure of All AC Buses - With no AC power present, only the turbine-driven pump is available. The unavailability of the turbine-driven pump analyzed earlier in this section is 0.039.

2.4.2 System Unavailability Comparison

Presented in Table 2.4-3 is a comparison of the major system unavailabilities calculated in the ZPSS, our revised estimates, and estimates for similar systems given in other NRC-sponsored PRA's and NUREG/CR-2497, "Precursers to Potential Severe Core Damage Accidents."

The revised figures listed in the table are within the range of values calculated for similar systems in previous PRA's and in the precursor study, except for the high-pressure recirculation, containment spray system, and containment fan-cooler system. These are discussed below.

The low failure probability for high-pressure recirculation is primarily due to the redundancy and diversity of the pump trains; two trains of charging pumps which are normally in operation and two trains of safety injection pumps on standby. The diversity of these high-pressure injection sources reduces the probability of common mode

TABLE 2.4-3

System Unavailability Comparisons

	ZPSS	Revised* ZPSS	Previous NRC PRAs	Precursor Study
HI Pressure Injection (medium LOCA)	1.37(-6)	5.4(-6)		
High-Pressure Injection (small LOCA)	7.4(-9)	2.2(-3)	5(-2)-1(-3)	2(-3)-8(-4)
LPI/Accumulators	1.4(-3)	1.5(-3)	1(-1)-2(-3)	
High-Pressure Recirculation	4.6(-4)	4.1(-4)	1(-2)-5(-3)	2(-3)-6(-4)
Low-Pressure Recirculation	5.2(-3)	5.2(-3)	1(-1)-4(-3)	
Containment Spray Injection	2.2(-4)	2.4(-4)	5(-2)-2(-3)	
Containment Spray Recirculation	1.6(-3)	1.8(-3)	8(-3)-1(-4)	
Auxiliary Feedwater	4.2(-6)	3.6(-5)	6(-4)-1(-5)	1.1(-3)
Containment Fan Coolers	6.1(-7)	1.6(-4)	5(-3)-1(-3)	

*The figures in this column do not match the results elsewhere in this report. This is because it was necessary to include the failure frequency of actuation circuits (without recovery) in this table to effect a comparison with previous NRC PRA's and the precursor study. In the ZPSS and in our review, actuation failures are treated separately.

β factors which would impact the high-pressure supply. In the case of containment fan coolers the redundancy of the coolers (five coolers; three required) and of the service water system which supplies the coolers (six pumps; two required) contribute to the high availability. The low unavailability for the containment spray injection function results from the diversity of the pump trains, with one diesel-driven pump and two motor-driven pumps.

Comparison of our revised estimates with NUREG/CR-2497 reveals one difference, namely, the auxiliary feedwater system unavailability. We reviewed the unavailability calculation in NUREG/CR-2497 to determine if it was applicable to the Zion AFWS design. The 1.1×10^{-3} estimate was derived from eight events in the nuclear industry. Of these eight, six could not occur at Zion due to design differences and two could possibly occur but did not significantly impact our revised AFWS unavailability.

Of the six that could not occur at Zion, (1) two were due to clogged suction strainers, (2) one resulted from a non-nuclear instrumentation failure, (3) two were due to failure of an AFWS consisting of solely turbine-driven pumps, and (4) one resulted from open full flow test lines. These four classes of events cannot occur at Zion because (1) suction strainers have been removed, (2) the non-nuclear instrumentation failure was peculiar to older Babcock and Wilcox plants only, (3) Zion has one turbine-driven and two motor-driven pumps, and (4) the Zion design utilizes mini flow test lines.

The two events that occurred that did not impact our revised estimate are (1) failure of pumps to auto start due to failure to install fuses and (2) failure to deliver flow due to closed valves. The first event did not impact our revised estimate because following TMI the motor pumps are now required to have their auto start circuits tested regularly and even if the fault did occur, it is recoverable by starting the pumps from the control room. The second event occurred early during the TMI accident and involved the inadvertent closure of two valves at the discharge of two pump trains. For a similar event to occur at Zion, eight valves at the discharge of three pump trains would have to be inadvertently closed. At Zion, then, one would expect a smaller probability of such an event occuring because of the larger number of human errors which would have to be committed. The ZPSS assessed the probability of such an error to be about 10 percent of our total revised AFWS estimate of $\sim 3 \times 10^{-5}$. We found no reason to arrive at a markedly different estimate of this failure mode.

In conclusion, we find our revised system unavailability estimates to be similar with values reported in other NRC sponsored PRAs and NUREG/CR-2497. Any difference that does exist can be reconciled when one considers Zion plant operation and design differences, the use of plant specific data in our revised estimates, and the uncertainties in the PRA process.

2.5 Human Reliability Analysis

2.5.1 Scope of the HRA Review

The human reliability analysis (HRA) portions of the Zion PRA were reviewed and evaluated by a Sandia human reliability analyst.

Because of the large number of human activities analyzed in the ZPSS and the limited time available to perform our review, it became necessary to focus on a subset. The initial subsets chosen were those human activities identified by the system's analysis to have a major impact on the dominant accident sequences. (See Table 2.5-1 at end of this section.) After completion of the review of these activities, other less important human actions were reviewed as time permitted. Thus the review of the HRA portions of the ZPSS was a limited review of selected topics rather than a comprehensive analysis integrated into the overall analysis of accident sequences.

The ZPSS HRA is impressive and obviously reflects considerable work and effort. It is hoped that the comments in this section of the Sandia review will be taken in the spirit intended: to indicate how the HRA part of the Zion PRA could be improved.

The use of NUREG/CR-1278[2-14] as the basis for many of the estimated human error probabilities (HEP's) made it easy to find sources of such estimates. However, it was not possible to fully understand and evaluate the HRA by reading only those sections clearly labeled as "human reliability," "human error," or "human factors." Because of the lack of documentation and the difficult-to-follow format, it was sometimes difficult or impossible to evaluate the estimates of some HEP's and to track the translation of these HEP's into equations which combined both equipment failure and human error terms.

The first conclusion reached was that HRA parts of the PRA should be documented in some systematic and reproducible manner, for example, as shown in NUREG/CR-2254[2-15]. It would have been helpful if human error terms had been shown separately just prior to being combined with equipment failure terms, as was done in the fault trees in WASH-1400. Despite these problems, the Zion PRA provides better documentation in general of human error terms than was done in WASH-1400. The point is that considerable improvement is possible in the Zion PRA and is necessary if the HRA is to be fully evaluated by an independent assessor.

2.5.2 Findings

The following is a list of 11 such findings. The next section provides a description of each.

1. Incomplete and incorrect documentation of the HRA.

2. Use of large uncertainty bounds in the HRA.

3. Use of undue optimism in assessment of credit for human redundancy.

4. Use of optimistic assessments of human performance under stress, especially for the case of multiple problems.

5. Use of persons to estimate operator performance in place of simple measurements.

6. Lack of documentation on how expert opinion was used.

7. Incomplete documentation of data sources used for estimated human performance.

8. Use of optimistic assessments of dependence among tasks done by the same person.

9. Apparent nonconsideration of some possibilities for common-cause failures from human errors.

10. Possible insufficient consideration of errors in restoring safety components after test, maintenance, or calibration.

11. Frequent use of conservatism in the HRA.

2.5.3 Description and Qualitative Assessment of the Importance of the Areas of Disagreement

This section discusses each of the 11 areas identified in the previous section. The next section discusses some more specific quantitative impacts.

1. <u>Incomplete and Incorrect documentation of the HRA.</u>

Comments on the difficulty of tracing the HRA were made in Section 2.5.1 above. Part of the problem is the unevenness in treatment of underlying assumptions and performance-shaping factors. In some cases a reasonably complete justification was given for certain estimated HEP's; e.g., the rationale for use of a .05 basic HEP for responding to a large LOCA. But in other cases, estimates were given without enough information for an independent assessment.

Errors in the report (i.e., incorrect documentation) made evaluation of the HRA difficult. For example, on Page 0.15-1 of Vol. 2 and Page 1.5-129 of Vol. 3, it is stated that the upper bound for each HEP will be represented by the 90th percentile rather than the 95th percentile as suggested in NUREG/CR-1278. The report gives an example on Page 0.15-2 of Vol. 2 which does use the 90th percentile. However, all of the calculations thereafter are based on the 95th percentile value of 1.645 for the equations used to calculate the mean human error rates (HER's) (which they call alpha) and the variances (which they call beta squared). This discrepancy was not discovered until too late to do more than try to check the estimated HEP's (i.e., the medians) in the report without checking the means and variances.

2. <u>Use of Large Uncertainty Bounds in the HRA.</u>

The size and propagation of uncertainty bounds is a controversial issue in general and no less so with estimates of human error rates. The human error uncertainty bounds given in NUREG/CR-1278 are based on judgment and limited data on the distribution of human performance. In the ZPSS (Pages 0.15-1, Vol. 2, and 1.5-129, Vol. 3) it is stated that they intend to use more conservative (i.e., wider bounds) than those given in NUREG/CR-1278. Thus, it is stated they will use the 90th percentile rather than the 95th percentile for the upper bound of the median estimate (i.e., the HEP's). We agree with the ZPSS rationale for using wider bounds: and add that because the Zion PRA team did not include anyone with specialized training in human performance technology, this measure of conservatism would have been worthwhile. However, as noted earlier, when the calculations were done, the 95th percentile was used for the upper bound.

This relative nonconservatism is compensated for, at least in part, by the Zion PRA's use of larger error factors for HEP's than are used in NUREG/CR-1278. For example, on Page 0.15-4, Vol. 2, and Page 1.5-131, Vol. 3, they use an error factor of 5 for the joint probability of error for more than one person.* An error factor of 5 corresponds to

*With symmetrical error bounds, as were employed in the Zion PRA, EF = 5 means that the lower bound is the basic HEP + 5, and the upper bound is the basic HEP x 5.

a total range ratio (between the 5th and 95th percentile HEP's) of 25, as compared with the range ratio of 10 most often used in NUREG/CR-1278.

For the small LOCA (Page 1.5-456, Vol. 3) and large LOCA (Page 1.5-472) they assign, respectively, error factors of 20 and 30. In NUREG/CR-1278, error factors of 10 are suggested thus, the Zion PRA allows for more uncertainty and thus, extra conservatism.

With regard to propagation of uncertainty bounds on human errors, as near as can be ascertained, when several human errors in sequence were required to produce some estimated HER's to be included in system sequence calculations, the Zion PRA used final uncertainty bounds of a magnitude chosen on the basis of conservatism. That is, they did not try to propagate uncertainty bounds through a series of human error events; they merely assigned large bounds to the final error event to go into the system sequence calculations. This general approach is being used in other ongoing PRA's.

3. Use of Undue Optimism in Assessment of Credit for Human Redundancy.

The Zion PRA notes (Page 1.5-453) that for the Zion 2-Unit Control Room there are on duty three reactor operators (RO's) and a supervisor (the shift engineer - SE). Furthermore, there is a shift technical advisor (STA) who, like the SE, is a senior reactor operator (SRO). For certain major transients, the report makes the reasonable assumption that one-half hour into the event there should be four people present who could be monitoring the control panel of the errant unit for various indications and to effect certain inputs via switches and other manual controls. (One of the three RO's is presumed to be involved with the other unit.) When they assume that all four people are indeed present, they assume a high level of dependence (HD) between the two RO's, moderate dependence (MD) between the SE and the two RO's, and low dependence (LD) between the STA and the others. They use the equations for these levels of dependence as given in NUREG/CR-1278.

The above levels of dependence are not unreasonable for some required human actions after a transient has been recognized. However, the report is sometimes optimistic in that (1) all four people are presumed to be present in as little time as 10 minutes into a transient, (2) it is assumed that the STA will be actively involved in almost all tasks of the two RO's, and (3) low levels of dependence are used when it appears that higher levels would be more appropriate.

For the small and large LOCA's, the Zion PRA assumes that there would be four people present, all of whom would have to fail to remember when to establish switchover to recirculation. In view of the time involved and the amount of simulator practice on LOCA's, this assumption does seem reasonable. However, the same four people are presumed to take an active interest in every detail in carrying out the many activities in coping with the small and large LOCA's. For example, one of the failure modes in the SNL review (see Table 2.5-1) is that of failure to open MOV's CC9412A and B after a large LOCA. While the Zion PRA conservatively (appropriately, we believe) uses a .1 basic HEP rather than the .05 value, it is assumed that even the STA would be actively involved to such an extent that nearly 9 times in 10 he would catch the operators' failure to manipulate these switches in a timely fashion. No rationale was given for this seemingly optimistic assumption. In their analysis they conservatively assumed that the two operators would miss one of the places stated in the emergency operating procedures (EOP-9) where they are supposed to open the MOV's. Therefore, in our analysis, we judged that this conservatism balanced the optimism of assuming four operators, and we accepted their estimate (as indicated in Table 2.5-1).

4. <u>Use of Optimistic Assessments of Human Performance under Stress, Especially for the Case of Multiple Problems.</u>

In the ZPSS, two basic models of human response under stress are given, based on NUREG/CR-1278. A high-pressure recirculation model (for the small LOCA) is described beginning on Page 1.5-452, and a low-pressure recirculation model (for the large LOCA) is described beginning on Page 1.5-469. The assumptions made for these models, including the numbers of people present, are reasonable. It is assumed for the HP recirculation model that a moderately high level of stress is present, as defined in NUREG/CR-1278. A higher level of stress is presumed for the LP recirculation model, and the estimated basic HEP is taken from NUREG/CR-1278, but is divided by 2 to reflect the considerable amount of simulator training of operators with the large LOCA. These assumptions all seem reasonable.

One problem is that for various transients, either of these models is applied without any apparent justification for doing so. This is especially a problem for transients which may not have been associated with a substantial amount of simulator practice for each operator. Even more questionable is the application of either of these models to the case where the operating personnel must respond to a combination of problems. In such cases, no additional degradation of human performance is assessed. The total degradation of human performance in response to the combination of problems should be greater than the degradation for either problem considered singly. NUREG/CR-1278 provides no guidance on this type of multiple problem, but unless the Zion personnel can show that the operators at that plant are well practiced in such multiple events, the use of the LP recirculation model without any upward adjustment of estimated HEP's can be considered to be overly optimistic.

Another problem is that the application of the LP or the HP recirculation model is sometimes made for response to events when considerably less time is available for successful operator intervention than was assumed for these two models. The HP recirculation model is based on there being a minimum of two hours into a small LOCA before recirculation is required, and the LP recirculation model is based on a minimum time of 30 minutes before recirculation is required (after a large LOCA). If either model is applied to time stresses of greater magnitude, upward adjustment of the estimated HEP's should be made.

5. Use of Persons to Estimate Operator Performance in Place of Simple Measurements.

In several cases, estimates of response times were obtained by interviewing operating personnel when it would have been possible to take actual measurements. Skilled personnel typically underestimate how much time it will take them to perform various tasks. For example, on Pages II.4-96 to 98, Vol. 1, and 1.3-14 and 15, Vol. 2, histograms are given of time for operators to travel to the diesel room and to the 345 kV relay house. Apparently, these histograms are based merely on asking operating personnel how long it would take them. It would have been easy to make some realistic in-plant simulations of the various activities involved in responding to loss of off-site power. Expert opinion is merely opinion; measurement of human performance should be used when possible.

6. Lack of Documentation of How Expert Opinion was Used.

Proper use of expert opinion is not entirely an art; it should make use of accepted and long-used methods of psychological scaling. Nowhere in the Zion report is there a description of the methods used, but one can infer that the "method" used was merely to ask people their opinions and to calculate some estimate of central tendency, probably the mean. If this approach was in fact taken, one cannot have any confidence in such an informal method of obtaining expert opinion. This skepticism especially pertains to the response time estimates (such as those involved in restoration of off-site power) and the combinations of response times and probabilities (given in the above-referenced pages). No mention is made of the qualifications of people who developed the psychological scaling used. Until such information is available, it is reasonable to view estimates based on undocumented use of expert judgment as not credible. The ZPSS response to our questions concerning this does not answer our basic criticism that insufficient information was presented to enable us to understand the psychological scaling methodology employed. As can be noted in Table 2.5-1, we decided to use generic data to model off-site power recovery.

7. Incomplete Documentation of Data Sources Used for Estimated Human Performance.

For many of the estimated HEP's for individuals and teams of individuals, tables and other information from NUREG/CR-1278 were used. Moreover, in nearly all cases, proper use was made of the tabled HEP's from that document. In the one case where the proper table was not used, the HEP obtained from the incorrect table was the same as the value from the correct table. As acknowledged in NUREG/CR-1278, considerable judgment is required in making use of its tabled data. In our opinion, a good job was done in the use of this information, although some of the applications were optimistic.

The major problem in data sources was the undocumented use of expert opinion, as indicated in the previous item (6). The use of histograms as cited above where estimated probability is on the ordinate and estimated time is on the abcissa is questionable, considering the probable method by which this information was obtained.

8. Use of Optimistic Assessments of Dependence Among Tasks Done by the Same Person.

In addition to the optimistic assumptions about dependence among team members (see Item 3), on Page 0.15-5, Vol. 2, the Zion PRA estimates that the level of dependence between the restoration of the first two valves of several is moderate, and for all the remaining valves is complete. This general rule could lead to extreme optimism for cases where the true level of dependence for operator errors is complete. This can often happen, especially for errors of omission. That is, for certain valve configurations (as described in Chapter 13 of NUREG/CR-1278), it is very likely that if an operator fails to restore one valve, he will always fail to restore another. If these two valves represent equipment redundancy and if we assume a basic error probability of omission of .003, the application of the above Zion general rule would result in an estimated joint HEP of

$$.003 \times \frac{1 + 6(.003)}{7} = 4 \times 10^{-4}$$

whereas the correct estimate would be $.003 \times 1.0 = 3 \times 10^{-3}$, nearly a factor of 10 higher.

This Zion general rule for dependence in valve restorations was not used. Its presence in the report is misleading.

Countering the above optimistic assessment of dependence are other estimates in the Zion PRA which are definitely conservative. For example, on Page 1.5-208, Vol. 3, it is stated that because the diesels are serviced by the same plant maintenance staff and are based on the same maintenance procedures and testing procedures, a high degree of dependence is assumed for maintenance activities.

9. Apparent Nonconsideration of Some Possibilities for Common-Cause Failures from Human Errors.

We were unable to make a detailed assessment, but were left with the impression that for several so-called redundant trains, no dependence of human actions was considered, or it was dismissed without sufficient documentation to evaluate. For example, beginning on Page 1.5-298, common cause failure due to human error is discussed. It is noted that during periodic calibration of certain instrument channels, tests are usually performed sequentially among identical channels, but they dismiss this dependence by stating that most calibration activities do not result in an instrument that fails to provide a trip. They further note that during monthly logic channel testing, a single logic channel failure could cause the RPS to fail and that both trains of logic are tested sequentially; again, a possibility for common-cause human errors. They ignore this possibility because the logic testing does not involve the changing of trip set points of logic arrangements. In both of these cases, insufficient documentation is given for an analyst to evaluate their conclusions. We wonder about the possibility for common-cause influence from failure of technicians or operators to restore circuits or components to the normal status after disruption of the normal status to permit such calibration.

10. Possible Insufficient Consideration of Errors in Restoring Safety Components After Test, Maintenance, or Calibration.

It is not clear if sufficient consideration was given to the possibilities for unavailability of safety components due to restoration errors after maintenance, calibration, or testing. We have the impression that considerable optimism may have occurred. But the lack of discussion in this area did not permit an accurate assessment.

For example, in discussing the availability of the low-pressure injection system (Page 1.5-400, Vol. 3), it is noted that an important human error is forgetting to open isolation valves MOV8812A and B after testing every 3 months, combined with failure of control room personnel to discover the wrong positions of the valves from looking at the two pairs of green and red indicator lamps. (The switches for these two MOV's are labeled, "RHRS SUC FROM RWST.") The original error would be the probability of an error of omission during the restoration procedures. The Zion PRA assess this error rate as a mean of 2.2×10^{-3}, based on the basic HEP from NUREG/CR-1278 of .001 (median). The failure of recovery was assessed as 7×10^{-2} (median), and modified to a mean of 8.28×10^{-2}. Their total unrecovered failure rate is assessed as 3.64×10^{-4} for the pair of MOV's.

It was not possible to assess the reasonableness of the above estimates using only the Zion PRA and drawings of the control boards. Did the operators restoring the valves use checklists in which each item is supposed to be checked off? For each shift check of the control board, was a checkoff list used which specifically called out the pair of MOV's in question? If dependence instead was placed on the usual unstructured scan of the control boards, almost no credit for recovery could be given. Without such a recovery factor and assuming the usual 50-50 split between correct and incorrect use of checklists, our recalculated mean would be approximately a factor of 70 higher than that in the Zion PRA.

Therefore, a phone call was made to the plant to obtain information not found in the Zion PRA. Based on information obtained, our best estimate is that their mean estimate could be an underestimate by about a factor of 2, or possibly even a factor of 7. The revised estimate, using the factor of 2, was inserted into the sequence calculation described in Section 3.2.B, and it made no difference. It appears that even a factor of 7 would make no difference.

The Zion PRA basic HEP of .001 for the original error or failure to restore is based on the assumption (from the table on Page 1.5-134) of a nonpassive task, with a short list (i.e., 10 or fewer items), and 100 percent correct use of the checkoff provision. If, on the other hand, the list is actually a long one, and if people use the checkoff feature correctly, only about 50 percent of the time (our usual assumption), the HEP would be .007, a factor of 7 higher than that of the Zion PRA. If the list really is a short list and still assuming the 50-50 split between correct and incorrect use of the checkoff feature, the HEP would be .002. In the phone conversation, we could not obtain the necessary information about the design of the checklist, so we assumed that the Zion analysts were correct in assuming a short list.

The phone call enabled us to check on the Zion estimate of 2.1 days mean time to recover from the original human error. The control room personnel (an auxiliary operator) do use a checklist at the beginning of each shift, and one of the items on the checklist is to note that the above two MOV's are in the correct open position (i.e., red lights showing rather than green). The Zion estimate of 2.1 days is based on some undocumented histogram mentioned on Page 1.5-406. This analysis, using Item 2 from Table 20-7 of NUREG/CR-1278, assesses the error of each shift checker as a median of .01. Thus, the probability of success in 2 days (6 shifts) would be $.9^6 = .53$, which appears to agree well with the median of 2.1 days assumed in the Zion report. Therefore, it was assumed that their estimated time for recovery and the associated HER's were approximately correct, and our revisions to the sequence calculations were made primarily on the factor of 2 described above.

11. Frequent Use of Conservatism in the HRA.

Apart from specific comments above on the possibility of undue optimism in the Zion PRA for certain analyses, it was apparent that in several cases the Zion PRA team did incorporate appropriate measures of conservatism in other analyses. For example, on Page 1.5-177, Vol. 3, the report notes that no credit is given for operators to start the emergency diesel generators without DC control power because of "the lack of personnel experience in performing the required operations and the lack of specific station procedures for the operations."

This and other similar manifestations of deliberate conservatism (without going to extremes) gave us the impression that those responsible for the HRA part of the Zion PRA did attempt to avoid undue optimism in assessing the effects of human performance. Their use of some inappropriate optimism (in the opinion of this analyst) reflects honest errors of judgment in their analyses.

2.5.4 Quantitative Evaluation of Potential Change to Sequence and/or Plant Damage State Frequencies

Table 2.5-1 lists the suggested revisions in probabilities that were supplied for input to the dominant accident sequences found in Section 3.2. Several other Zion PRA estimates were evaluated; still others cannot be evaluated without further information.

2.5.5 Summary of the Review of the Human Reliability Analysis

The major problem in completely reviewing the HRA is the lack of documentation. While this is also a problem for the PRA as a whole, it is a much bigger problem for a review of an HRA. HRA deals with the most difficult component of a system to understand and to quantify.

While the Zion PRA does not deliberately appear to be optimistic in its assessments of human errors, assumptions made regarding the credit to be given for more than one person in the performance of several tasks did have that effect. Furthermore, the development of only two stress models and the misapplication of these models probably had the net result of underestimating the effects of human errors in responding to unusual events, especially in the case where there is more than one unusual event.

The above optimism is countered, at least for some analyses, by a deliberate decision not to take full credit for certain recovery factors, and by the use of rather wide uncertainty bounds.

2.6 Estimation Methodology

2.6.1 Introduction

In this section we examine the Zion Probabilistic Safety Study (ZPSS) estimates of initiating event rates and the failure probabilities and unavailabilities of components and systems. Our emphasis is on identifying the strengths, weaknesses, and potential effects of the methodology used. The comments thus apply to ZPSS and to other studies that may adopt the same methodology. Contributions of the methodology to specific accident sequence estimates are addressed in Section 3.

Future events, such as human errors, the failure of reactor components and systems, and the resulting consequences cannot be foretold exactly. However, by careful modeling of the occurrence of these events as the outcome of random processes, this unpredictability can be gauged and assessed. Developing these models is an essential activity in a probabilistic risk assessment (PRA).

TABLE 2.5-1

Suggested Revisions to ZPSS Human Error Rate Estimates for Selected Failure Events in the Revised Dominant Accident Sequences

SNL Section	Event	Human Error	Revision
3.2.2	Failure of DC Bus Failure of AFW	Human Causes Failure of DC Bus	Zion estimate accepted
3.2.6	Small LOCA, failure of Recir. Cooling	Failure to Initiate Switchover	Zion estimate accepted
3.2.9 3.2.10	Large or Med. LOCA, failure of Recir. Cooling	Failure to Initiate Switchover	1) Zion estimate accepted 2) Zion estimate accepted even though 4 people assumed. Other conservatism made up for optimistic assumption of 4 people.
3.2.12	Large LOCA, failure of low pressure inj.	1) Leave either MOV-8812A or B closed after testing	1) Increase mean HEP to 4.4×10^{-3} (factor of 2)
		2) No recovery in control room.	2) Double mean HEP to 10^{-3} NOTE: These changes had no material influence on the sequence calculation.

TABLE 2.5-1 (Continued)

Suggested Revisions to ZPSS Human Error Rate Estimates for Selected Failure Events in the Revised Dominant Accident Sequences

SNL Section	Event	Human Error	Revision
3.2.3, 3.2.4, 3.2.5, 3.2.7, 3.2.11,	Loss of off-site power sequences	Failure to restore off-site power	Use generic industry data regarding off-site power recovery.
3.2.13, 3.2.14	Feed and bleed sequences	Failure to initiate feed and bleed	Zion estimate accepted.

The numbers that go into a probability model, e.g., failure rates and probabilities, component availabilities, and human error probabilities, are not known exactly. Indeed, since they are quantities in a model which is only an approximation to reality, the notion that they exist and are knowable as is the case for a physical constant such as the speed of light, is somewhat ephemeral. Nevertheless, within the context of the specified model, it is necessary to estimate these quantities. Obtaining estimates, substantiating them, and conveying the possible errors--the uncertainty--present in these estimates pose considerable problems for a risk analysis. The authors of the ZPSS (whom we shall refer to as Zion) approached these problems using Bayesian methodology. Under this approach the study team represented, probabilistically, their prior beliefs about the rates and probabilities of interest, then modified these beliefs by historical data obtained from Zion's experience (if available), and convoluted them to yield a probability distribution representing their posterior beliefs about the frequency and consequences of various accidents.

Bayesian methodology is controversial, but our intent in this review is not to add fuel. Rather, we attempt to identify the effect of this methodology on the estimates obtained. That is, we undertake a limited sensitivity study which the ZPSS authors did not do. If the ZPSS estimates are to be convincing, one needs to know the assumptions made and the extent to which the results depend on them.

Bayesian methodology applied to risk assessment is also new. Readers of the Zion PSS might, therefore, be overwhelmed, or mystified by it, so we begin this review by making some general comments about Bayesian methodology and the ZPSS rendition of it.

2.6.2 Bayesian Methodology

Consider a component that either succeeds or fails on demand. Assume that in a sequence of n demands the result on each demand--success or failure--is independent of the results on the other demands and assume that a constant, unknown failure probability, p, underlies the sequence. That is, assume a coin-tossing model. Then the probability of observing k failures in n demands is

$$P(k;\ n,\ p) = \frac{n!}{k!(n-k)!}\, p^k (1-p)^{n-k}$$

the binomial distribution. The problem is to estimate p, given data of k failures in n demands. Conventional statistical methodology yields point estimates and confidence intervals based on this model.

The Bayesian, however, seeks to incorporate other information about p. He expresses his state of belief about p by a probability distribution, g(p). In principle, this distribution is specified prior to observing the data, to maintain independence, and so is called the prior distribution (Zion calls it the generic distribution). By Bayes' Theorem (which is a straightforward manipulation of conditional probabilities) the data are used to modify the prior distribution, the result being called the posterior distribution of p (Zion calls it the updated distribution). To wit,

$$g(p|k,n) = \frac{p(k;n,p)\ g(p)}{\int_0^1 p(k;n,p)\ g(p)dp}$$

One then presents this distribution or selected moments and percentiles to summarize his posterior degree or belief about p.

The appeal of this analysis is that people cognizant of the component surely know more about p than just what is embodied by the data, and it is advantageous to incorporate that information. A difficulty is in determining g(p). One has to translate his knowledge and beliefs to probability. He has to say, "What I know about p is equivalent to knowing that it was generated at random from g(p)." This translation is difficult. Whether one can justify such precision is open to question. Also, one can question whether the updated quantified beliefs of some person or persons are of much value to those who may not share those beliefs. In the following sections we examine how Zion handled these difficulties. First, though, some comments about terminology.

In the preceding and subsequent discussions we use the term "probability" as a parameter in a model; e.g., the parameter p above, or a parameter calculated from a model, such as the probability of no failures in τ hours, given the constant failure rate model with parameter λ. One can think of a model as a mathematical representation of what would happen in infinite repetitions of some hypothetical experiment, but that's not a requirement. We use the term "personal probability," or "Zion's probability," to denote probabilities calculated to reflect degree of belief. We also distinguish between failure rates, which are dimensioned failures per unit time, and failure probabilities, which are dimensionless.

Zion calls both of the latter "frequencies," and defines these as the outcome of an experiment involving repeated trials, either an actual experiment or a "thought experiment," (Page 0.4-1). Thus, rates and probabilities are not distinguished (so we see a "probability" of 4.11 on Page 1.5-161), nor are estimates of probabilities or rates, which result from a finite number of repeated trials, distinguished from the parameter being estimated, which correspond to infinite repetitions. Zion uses "probability" variously as quantified degree of belief, confidence, or knowledge (which are not all the same). In the following sections we consider the estimation of component failure rates and probabilities, initiating event rates, and maintenance unavailability, and then combining these estimates to estimate system failure probabilities.

2.6.3 Treatment of Component Failure Data

Zion's estimates of component failure rates and probabilities were obtained from the following sources:

Zion site-specific experience, as given by LER's and other station records

Industry-wide LER summaries on valves, pumps, and diesel generators published by EG&G

WASH-1400

IEEE-500 estimates of electrical component failure rates and probabilities

The last three sources were used to develop prior distributions, which were then modified by the Zion data, using Zion's DPD (discrete probability distribution) arithmetic to arrive at the posteriors. The means and variances of these distributions are reported in the ZPSS Table 1.5.1-5 (reproduced on the following pages).

From the authors' Bayesian orientation one would expect their prior probability distributions, regardless of how they are developed, to be described only as their prior degree of belief about the unknown Zion parameters. But they make the much stronger claim (p. 0.14.3) that these are "frequency distributions," the "known results of experiments on populations." They are said to represent the "variation of performance of individual components within the population." This is a presumptuous claim and unnecessary from the Bayesian viewpoint. It is unclear why Zion made it. They contradicted this claim when they subsequently assumed that individual components of a given type, e.g., all motor-operated valves at Zion, all have the same constant failure rate, rather than individually different rates.

Most of Zion's prior distributions are based in part on WASH-1400. It is not at all clear from WASH-1400 how the lognormal distributions given there are to be interpreted, but there is no basis to regard them as the results of (infinite) "experiments on populations." In fact, the nuclear plant data in WASH-1400 amount to one year's worth (1972) of (what are now called) LER's. For Zion to regard the distributions supplied by WASH-1400, even after they are stretched out so that the 5th and 95th percentiles become the 20th and 80th, as known frequency distributions, and to call them "generic" is unwarranted.

One consequence of assuming that Zion's prior distributions are the frequency distributions of plant-to-plant variability is that in order to proceed with the derivation of the posterior distribution you must next assume that the Zion plant is a random sample from the population of plants. This too seems difficult to support.

What seems most plausible is to regard Zion's prior distributions as their representation of their prior personal belief, or knowledge, of the failure rates and demand probabilities for classes of components at Zion. These priors, rather than being obtained by careful introspection and elicitation of the knowledge possessed by the study team or the Zion personnel, as one would expect Bayesians to do, were obtained by applying ad hoc prescriptions to the numerical results published in the above sources. As we shall see, the effect of this approach is quite uneven. Also, as we shall see in our Sections 2.6.6 and 3, there are important, unannounced exceptions to Zion's treatment of WASH-1400's 5th and 95th percentiles as 20th and 80th.

Regardless of whether one accepts, rejects, or ignores the claims made by Zion for their prior distributions, the important question remains as to what effect these distributions had on their estimates. Looking at Table 1.5.1-5 does not tell you. Some light can be shed on this question by pretending the "updated results" are based on a statistical (as opposed to Bayesian) analysis. In a statistical analysis, given data consisting of f failures in T hours and assuming a constant failure rate one would estimate that failure rate by $*\lambda = f/T$, where the asterisk denotes an estimate. Under the assumption that T is fixed and known, the variance of λ^* would be estimated by $var^*(\lambda^*) = f/T^2$. Zion provides a posterior mean (their point estimate) and variance. If we equate these to f/T and f/T^2, respectively, and solve for f and T, then we obtain

Table 1.5.1-5

SPECIALIZED COMPONENT HARDWARE FAILURE DATA

Component Description and Failure Mode	Plant-Specific				Generic		
	No. of Failures	Service Hours or Demands	Updated		Mean [λ20]	λ80/λ20 [λ80]	Comments - Data Sources
			Mean	Variance			
1) System: All Component Type: Manual Valves, Motor-Operated Valves Failure Mode: Transfer Closed	0	1.11(7) hours	5.28(-8) /hour	2.82(-15) /(hour)^2	[2.8(-8)] /hour	[2.8(-7)] /hour	W-1400. Fail to remain open. Plugged. λ5 = 3(-5)/dem λ95 = 3(-4)/dem used 1 dem/45 days to convert to 1 hour. λ5 = 2.8(-8)/hr λ95 = 2.8(-7)/hr
2) System: All Component Type: Manual Valves Failure Mode: Transfer Open/Excessive Leakage Through Valve			(NO DATA FOUND)		2(-8) /hour	100	N-1363. Manual valves. External leakage PWR's $\bar{X}$ = 2(-8)/hr. W-1400. MOV's. External leakage/Rupture λ5 = 1(-9)/hr λ95 = 1(-7)/hr
3) System: All Component Type: Check Valves Failure Mode: Failure to Open on Demand	0	6.968(3) demands	4.32(-5) /demand	2.41(-9) /(demand)^2	1(-4) /demand	10	N-1363 Check valves. Fail to open. PWRs. $\bar{X}$ = 1(-4)/dem W-1400 Check valves. Failure to open. λ5 - 3(-5)/dem λ95 = 3(-4)/dem
4) System: All Component Type: Check Valves Failure Mode: Failure to Seat/Excessive Leakage	0	6.08(5) hours	8.38(-7) /hour	7.29(-13) /(hour)^2	3(-6) /hour	10	N-1363 Check valves. Internal leakage. PWRs. $\bar{X}$ = 3(-6)/hr W-1400 Check valves. Reverse leak. λ5 = 3(-6)/hr λ95 = 3(-5)/hr
5) System: All Component Type: Relief/Safety Valves Failure Mode: Premature Opening or Leakage	2	6.19(5) hours	1.63(-6) /hour	1.81(-12) /(hour)^2	5(-7) /hour	10	N-1363 PWR safety valves. Premature opening. $\bar{X}$ = λ5(-7)/hr W-1400 Relief valves (all Rx types). Premature opening. λ5 = 1(-7)/hr λ95 = 1(-6)/hr
6) System: All, Except Containment Spray and Chemical and Volume Control Component Type: Motor Operated Valves Failure Mode: Failure to Operate on Demand	14	1.131(4) demands	1.55(-3) /demand	6.30(-8) /(demand)^2	4(-3) /demand	10	N-1363 PWR - Remote operated plus MOV. No command faults. Fail to operate. $\bar{X}$ = 4(-3)/dem W-1400 Failed to operate. λ5 = 3(-4)/dem λ95 = 3(-3)/dem

NOTE: 1.23(4) indicates 1.23×10^4
W-1400: WASH-1400, Table III 2-1.
N-1363: NUREG/CR-1363, Table 23, page 63.
N-1205: NUREG/CR-1205, Table 14, page 35.
N-1362: NUREG/CR-1362, Table 20, page 51.

Table 1.5.1-5

SPECIALIZED COMPONENT HARDWARE FAILURE DATA

Component Description and Failure Mode	Plant-Specific				Generic		
	No. of Failures	Service Hours or Demands	Updated Mean	Updated Variance	Mean [λ20]	λ80/λ20 [λ80]	Comments - Data Sources
7) System: All Component Type: Motor Operated Valves Failure Mode: Transfer Open/Excessive Leakage Through Valve	0	6.95(5) hours	3.14(-8) /hour	1.56(-14) /(hour)^2	1(-7) /hour	100	N-1363 PWR Remote + MOV - No command fault. External leakage $\bar{X}$ = 1(-7)/hr W-1400 MOV's External leakage/rupture λ5 = 1(-9)/hr λ95 = 1(-7)/hr
8) System: All Component Type: Air Operated Valves Failure Mode: Failure to Operate on Demand	3	1540 demands	1.44(-3) /demand	7.93(-7) /(demand)^2	9(-4) /demand	10	N-1363 PWR AOV Fail to operate. No command faults. $\bar{X}$ = 9(-4)/dem W-1400 AOV. Fails to operate λ5 = 1(-4)/dem λ95 = 1(-3)/dem
9) System: All Component Type: Air Operated Valves Failure Mode: Transfer Closed, Plugged	0	2.13(6) hours	1.12(-7) /hour	1.84(-14) /(hour)^2	[2.8(-8)] /hour	[2.8(-7)] /hour	W-1400 AOV. Plug. Failure to remain open. Used 1 dem 45 days to convert to 1 hour. λ5 = 3(-5) = 2.8(-8)/hr λ95 = 3(-4) = 2.8(-7)/hr
10) System: All Component Type: Air Operated Valves Failure Mode: Transfer Open, External Leakage			(NO DATA FOUND)		1(-7) /hour	100	N-1363 AOV. External leakage $\bar{X}$ = 1(-7)/hr W-1400 AOV. External leakage/rupture λ5 = 1(-9)/hr λ95 = 1(-7)/hr
11) System: All, Except Auxiliary Feedwater Component Type: Pumps-Motor Driven Failure Mode: Failure to Start on Demand	3	3.138(3) demands	7.21(-4) /demand	1.91(-7) /(demand)^2	5(-4) /demand	10	N-1205 Standby system. Does not start. No command faults $\bar{X}$ = 5(-4)/dem W-1400 Electric motor. Failure to start. λ5 = 1(-4)/dem λ95 = 1(-3)/dem
12) System: Auxiliary Feedwater Component Type: Turbine Driven Auxiliary Feedwater Pump Failure Mode: Failure to Start on Demand	6	2.31(2) demands	2.29(-2) /demand	7.60(-5) /(demand)^2	4(-3) /demand	100	N-1205 Standby System. No command faults. Does not start. $\bar{X}$ = 4(-3)/dem λ80/λ20 based on engineering judgment.

NOTE: 1.23(4) indicates 1.23×10^4
W-1400: WASH-1400, Table III 2-1.
N-1363: NUREG/CR-1363, Table 23, page 63.
N-1205: NUREG/CR-1205, Table 14, page 35.
N-1362: NUREG/CR-1362, Table [illegible] page 51.

Table 1.1.1-5 (continued)

SPECIALIZED COMPONENT HARDWARE FAILURE DATA

Component Description and Failure Mode	Plant-Specific				Generic		
	No. of Failures	Service Hours or Demands	Updated Mean	Updated Variance	Mean [λ20]	λ80/λ20 [λ80]	Comments - Data Sources
13) System: Safety Injection Component Type: Safety Injection Pumps Failure Mode: Fail During Operation	0	4.6(1) hours	1.55(-5) /hour	2.96(-8) $/(\text{hour})^2$	2(-5) /hour	100	N-1205 Alternating System. Does not operate given start $\bar{X}$ = 2(-5)/hr. W-1400 pump (w/o motor) Failure to run λ5 = 3(-6)/hr λ95 = 3(-4)/hr
14) System: Residual Heat Removal Component Type: Residual Heat Removal Pumps Failure Mode: Fail During Operation	0	3.25(4) hours	2.53(-6) /hour	3.39(-11) $/(\text{hour})^2$	2(-5) /hour	100	Same as No. 13
15) System: Component Cooling Component Type: Component Cooling Pumps Failure Mode: Fail During Operation	0	7.6(4) hours	1.76(-6) /hour	9.62(-12) $/(\text{hour})^2$	2(-5) /hour	100	Same as No. 13
16) System: Service Water Component Type: Service Water Pumps Failure Mode: Fail During Operation	0	1.52(5) hours	1.32(-6) /hour	3.74(-12) $/(\text{hour})^2$	2(-5) /hour	100	Same as No. 13
17) System: Containment Spray Component Type: Containment Spray Pumps Motor-Driven Failure Mode: Fail During Operation	0	6.6(1) hours	1.50(-5) /hour	1.96(-8) $/(\text{hour})^2$	2(-5) /hour	100	Same as No. 13
18) System: Auxiliary Feedwater Component Type: Motor Driven Auxiliary Feedwater Pumps Failure Mode: Fail During Operation	1	3.8(3) hours	9.87(-5) /hour	1.98(-8) $/(\text{hour})^2$	2(-5) /hour	100	Same as No. 13
19) System: Auxiliary Feedwater Component Type: Turbine Driven Auxiliary Feedwater Pump Failure Mode: Fail During Operation	0	1.9(3) hours	7.63(-6) /hour	1.20(-9) $/(\text{hour})^2$	2(-5) /hour	100	Same as No. 13. Turbine-driven pump failure during operation similar to motor driven pump during operation.

NOTE: 1.23(4) indicates 1.23×10^4
W-1400: WASH-1400, Table III 2-1.
N-1363: NUREG/CR-1363, Table 23, page 63.
N-1205: NUREG/CR-1205, Table 14, page 35.
N-1362: NUREG/CR-1362, Table 20, page 51.

TABLE 1.5.1-5 (continued)

SPECIALIZED COMPONENT HARDWARE FAILURE DATA

Component Description and Failure Mode	Plant-Specific				Generic		
	No. of Failures	Service Hours or Demands	Updated Mean	Updated Variance	Mean [λ20]	λ80/λ20 [λ80]	Comments - Data Sources
20) System: Containment Fan Coolers Component Type: Containment Fan Coolers Failure Mode: Failure to Start on Demand-Motor Failures	2	1.155(3) demands	1.17(-3) /demand	7.60(-7) /(demand)^2	[1(-4)] /demand	[1(-3)] /demand	W-1400 Electric motor failure to start. λ5 = 1(-4)/dem λ95 = 1(-3)/dem
21) System: Containment Fan Coolers Component Type: Containment Fan Coolers Failure Mode: Fail During Operation	0	1.52(5) hours	3.53(-6) /hour	1.33(-11) /(hour)^2	[1(-4)] /hour	[1(-2)] /hour	W-1400 Electric motor. Failure to run. Extreme environment. λ5 = 1(-4)/hr λ95 = 1(-2)/hr
22) System: All Component Type: Heat Exchangers Failure Mode: Leakage	0	2.36(5) hours	7.13(-7) /hour	1.13(-12) /(hour)^2	4.56(-6) /hour	100 /hour	NPRDS page 34 $\bar{X}$ = 4.56(-6)/hr λ80/λ20 based on engineering judgment.
23) System: All Component Type: Heat Exchangers Failure Mode: Plugged (Tube Side)	0	8.35(4) hours	ε	ε	ε	ε	Negligibly small failure rate. Estimated on the basis of engineering judgment.
24) System: All Component Type: Heat Exchangers Failure Mode: Plugged (Shell Side)	0	8.35(4) hours	ε	ε	ε	ε	Negligibly small failure rate. Estimated on the basis of engineering judgment.
25) System: Diesel Generators Component Type: Diesel Generators Failure Mode: Failure to Start on Demand	30	1.693(3) demands	1.82(-2) /demand	1.26(-5) /(demand)^2	3(-2) /demand	10	N-1362 Monthly testing. Diesel generator fails to start. No command faults. $\bar{X}$ = 3(-2)/dem W-1400 Diesel generator. Failure to start λ5 = 1(-2)/dem λ95 = 1(-1)/dem
26) System: Diesel Generators Component Type: Diesel Generators Failure Mode: Fail During Operation	6	1.34(3) hours	5.97(-3) /hour	4.05(-6) /(hour)^2	2(-2) /hour	100	N-1362 Monthly testing. Diesel generator. Does not continue to run. No command faults. $\bar{X}$ = 2(-2)/hr W-1400 diesel generator. Failure to run. λ5 = 3(-4)/hr λ95 = 3(-2)/hr

NOTE: 1.23(4) indicates 1.23 x 10^4
W-1400: WASH-1400, Table III 2-1.
N-1353: NUREG/CR-1353, Table 23, page 63.
N-1205: NUREG/CR-1205, Table 14, page 35.
N-1362: NUREG/CR-1362, Table [illegible] page 51.

TABLE 1.5.1-5 (continued)

SPECIALIZED COMPONENT HARDWARE FAILURE DATA

Component Description and Failure Mode	No. of Failures	Plant-Specific			Generic		
		Service Hours or Demands	Updated Mean	Updated Variance	Mean [λ20]	λ80/λ20 [λ80]	Comments - Data Sources
27) System: AC Electric Power Component Type: Bus Feed Breakers Failure Mode: Failure to Close on Demand	5	3.12(3) demands	1.26(-3) /demand	2.67(-7) /(demand)^2	[1.0(-7)] /demand	[1(-5)] /demand	IEEE-500 Interior design. AC breaker fails to close. Page 148 Rec = 1(-6)/dem Max = 1(-5)/dem
28) System: AC Electric Power Component Type: Bus Feed Breakers Failure Mode: Failure to Open on Demand	1	3.12(3) demands	5.31(-4) /demand	9.73(-8) /(demand)^2	[2.27(-5)] /demand	[2.27(-3)] /demand	IEEE-500 Interior design. AC breaker fails to open. Page 148 Rec = 2.27(-4)/dem Max = 2.27(-3)/dem
29) System: AC Electric Power Component Type: Bus Feed Breakers Failure Mode: Transfer Open	0	9.10(5) hours	2.32(-7) /hour	3.03(-13) /(hour)^2	[3.08(-9)] /hour	[6.0(-7)] /hour	IEEE-500 Interior design. AC breaker spurious operation. Page 148. Rec = 4.3(-8)/hr Max = 6(-7)/hr
30) System: AC Electric Power Component Type: Transformers Failure Mode: Fail During Operation	1	3.04(5) hours	1.73(-6) /hour	3.17(-12) /(hour)^2	[1.44(-7)] /hour	[1.51(-6)] /hour	IEEE-500 Transformers (601v-15kv). All modes. Pg. 300 Rec = 4.67(-7)/hr Max = 1.51(-6)/hr
31) System: AC Electric Power Component Type: Inverters Failure Mode: Fail During Operation	3	3.04(5) hours	1.09(-5) /hour	3.04(-12) /(hour)^2	[3(-7)] /hour	[3(-5)] /hour	W-1400 Solid state device. High power application. Fails to function. λ5 = 3(-7)/hr λ95 = 3(-5)/hr
32) System: DC Electric Power Component Type: Batteries Failure Mode: Fail During Operation	0	2.02(5) hours	7.61(-8) /hour	4.27(-14) /(hour)^2	[4.95(-9)] /hour	[8.74(-8)] /hour	IEEE-500 Batteries-lead-acid. Page 104. Stationary types for float service. Catastrophic failure. Rec = 2.08(-8)/hr Max = 8.74(-8)/hr
33) System: DC Electric Power Component Type: Battery Chargers Failure Mode: Fail During Operation	0	2.02(5) hours	5.54(-7) /hour	6.94(-12) /(hour)^2	[1.78(-8)] /hour	[1.25(-4)] /hour	IEEE-500 Rectifiers. Stationary type. All modes. Page 90. Rec = 1.49(-6)/hr Max = 1.25(-4)/hr

NOTE: 1.23(4) indicates 1.23×10^4
W-1400: WASH-1400, Table III 2-1.
N-1363: NUREG/CR-1363, Table 23, page 63.
N-1205: NUREG/CR-1205, Table 14, page 35.
N-1362: NUREG/CR-1362, Table 20, page 51.

Table 1.5.1 (continued)

SPECIALIZED COMPONENT HARDWARE FAILURE DATA

Component Description and Failure Mode	Plant-Specific				Generic		
	No. of Failures	Service Hours or Demands	Updated Mean	Updated Variance	Mean [$\lambda 20$]	$\lambda 80/\lambda 20$ [$\lambda 80$]	Comments - Data Sources
34) System: Containment Spray Component Type: Motor-operated valves Failure Mode: Failure to Operate on Demand	10	1647 demands	5.73(-3) /demand	1.84(-6) /(demand)^2	4(-3) /demand	10	N-1363 PWR. Remote operated plus MOV. No command faults. Fail to operate. $\bar{X}$ = 4(-3)/dem W-1400 MOV fails to operate. $\lambda 5$ = 3(-4)/dem $\lambda 95$ = 3(-3)/dem
35) System: Chemical and Volume Control Component Type: Motor Operated Valves Failure Mode: Failure to Operate on Demand	7	1720 demands	3.72(-3) /demand	2.71(-6) /(demand)^2	4(-3) /demand	10	N-1363 PWR. Remote operated plus MOV. No command faults. Fail to operate $\bar{X}$ = 4(-3)/dem W-1400. MOV. Fails to operate. $\lambda 5$ = 3(-4)/dem $\lambda 95$ = 3(-3)/dem
36) System: Auxiliary Feedwater Component Type: Pumps; Motor-Driven Failure Mode: Failure to Start on Demand	4	462 demands	5.02(-3) /demand	9.26(-6) /(demand)^2	5(-4) /demand	10	N-1205 Standby system. Does not start. No command faults $\bar{X}$ = 5(-4)/dem W-1400 Electric motor. Fail to start. $\lambda 5$ = 1(-4)/dem $\lambda 95$ = 1(-3)/dem
37) System: Containment Spray Component Type: Diesel-Driven Containment Spray Failure Mode: Failure to Start on Demand	1	183 demands	4.23(-3) /demand	1.31(-5) /(demand)^2	5(-3) /demand	10	N-1205 Standby. Diesel pump. Does not start. $\bar{X}$ =5(-3)/dem W-1400 Diesel plant. Failure to start $\lambda 5$ = 1(-2)/dem $\lambda 95$ = 1(-1)/dem
38) *System: Containment Spray Component Type: Diesel-Driven Containment Spray Failure Mode: Fail During Operation	2	33 hours	2.91(-2) /hour	5.53(-4) /(hour)^2	1.41(-2) /hour	VARIANCE 3.05(-1) /(hour)^2	W-1400 Pumps (w/o motor). Failure to run. Normal environment $\lambda 5$ = 3(-6)/hr $\lambda 95$ = 3(-4)/hr W-1400 Diesel (engine only). Failure to run. $\lambda 5$ =3(-5)/hr $\lambda 95$ = 3(-3)/hr

NOTE: 1.23(4) indicates 1.23 x 10^4
W-1400: WASH-1400, Table III 2-1.
N-1363: NUREG/CR-1363, Table 23, page 63.
N-1205: NUREG/CR-1205, Table 14, page 35.
N-1362: NUREG/CR-1362, Table 20, page 51.

*Calculated the mean and variance of each of the two W-1400 $\lambda 5$, $\lambda 95$ values as if they were $\lambda 20$, $\lambda 80$ and then added the resulting means and variances together.

Table 1.5.1-5 (continued)

SPECIALIZED COMPONENT HARDWARE FAILURE DATA

Component Description and Failure Mode	Plant-Specific				Generic		
	No. of Fail-ures	Service Hours or Demands	Updated Mean	Updated Variance	Mean [$\lambda 20$]	$\lambda 80/\lambda 20$ [$\lambda 80$]	Comments - Data Sources
39) System: Reactor Containment Fan Coolers Component Type: Dampers Failure Mode: Failure to Operate on Demand	3	1155 demands	1.85(-3) /demand	1.39(-6) /$(\text{demand})^2$	9(-4) /demand	10	N-1363 AOV. Failed to operate. PWR. $\bar{X}$ = 9(-4)/dem W-1400 AOV. Fails to operate $\lambda 5$ = 1(-4)/dem $\lambda 95$ = 1(-3)/dem
40) System: Chemical and Volume Control Component Type: Centrifugal Changing Pumps Failure Mode: Fail During Operation	0	7.6(4) hours	1.76(-6) /hour	9.62(-12) /$(\text{hour})^2$	2(-5) /hour	100	N-1205 Alternating System does not operate given start. Motor-driven. $\bar{X}$ = 2(-5)/hr W-1400 Pumps. Failure to run Extreme environment $\lambda 5$ = 1(-4)/hr $\lambda 95$ = 1(-2)/hr
41) System: All Electric Power Component Type: Bus, Metal-Enclosed Failure Mode: Open Circuit	0	3.03(6) hours	1.91(-8) /hour	2.22(-15) /$(\text{hour})^2$	[6.44(-10)] /hour	[2.74(-8)] /hour	IEEE-500 Metal-enclosed bus. Open circuit. Page 188 Rec = 4.2(-9)/hr Max = 2.74(-8)/hr
42) System: All Electric Power Component Type: Manual Transfer Devices (Switches) Failure Mode: Transfer Open		*	ϵ	ϵ	ϵ	ϵ	Negligibly small failure rate. Estimated on the basis of engineering judgment.
43) System: Reactor Protection Component Type: Control Cable Failure Mode: Line-to-Line Short		*	3.22(-6) /hour	8.96(-11) /$(\text{hour})^2$	[2.93(-7)] /hour	[3.68(-6)] /hour	IEEE-500 Copper conductor. Line-to-line short. Page 524. Rec = 1.038(-6)/hr Max = 3.675(-6)/hr
44) System: Reactor Protection Component Type: Control Cable Failure Mode: Line-to-Ground Short		*	7.52(-6) /hour	4.88(-10) /$(\text{hour})^2$	[6.84(-7)] /hour	[8.58(-6)] /hour	IEEE-500 Copper conductor. Line-to-ground short. Page 524 Rec = 2.422(-6)/hr Max = 8.575(-6)/hr

NOTE: 1.23(4) indicates 1.23×10^4
W-1400: WASH-1400, Table III 2-1.
N-1363: NUREG/CR-1363, Table 23, page 63.
N-1205: NUREG/CR-1205, Table 14, page 35.
N-1362: NUREG/CR-1362, Table 29, page 51.

* Generic data was used to calculate the mean and variance. No plant data was used.

Table 1.5.1-5 (continued)

SPECIALIZED COMPONENT HARDWARE FAILURE DATA

Component Description and Failure Mode	Plant-Specific				Generic		
	No. of Fail-ures	Service Hours or Demands	Updated Mean	Updated Variance	Mean [λ20]	λ80/λ20 [λ80]	Comments - Data Sources
45) System: All Electric Power Component Type: Trip Relay Failure Mode: Fails to Open on Demand		*	6.28(-6) /demand	2.49(-11) /(demand)^2	[2.73(-6)] /demand	[8.85(-6)] /demand	IEEE-500. Auxiliary relays. Fail to open. Page 164 Rec = Not given. An RF = $\sqrt{10}$ was used based on W-1400 Max = 8.85(-6)/dem
46) System: All Pipes (3" In Diameter) Component Type: Pipe Section Failure Mode: Ruptures/Plugs		*	8.60(-9) /hour	6.00(-15) /(hour)^2	λ5 3(-11) /hour	λ95 3(-8) /hour	W-1400 Pipes (< 3") High quality section as defined in W-1400. λ5 = 3(-11)/hr λ95 = 3(-8)/hr
47) System: All Pipes (3" In Diameter) Component Type: Pipe Section Failure Mode: Ruptures/Plugs		*	8.60(-10) /hour	6.00(-17) /(hour)^2	λ5 3(-12) /hour	λ95 3(-9) /hour	W-1400. Pipes (> 3") High quality section as defined in W-1400. λ5 = 3(-12)/hr λ95 = 3(-9)/hr
48) System: Auxiliary Feedwater Component Type: Condensate Storage Tank Failure Mode: Does Not Supply Water		*	8.48(-10) /hour	5.10(-17) /(hour)^2	λ50 1(-10) /hour	RF 30	W-1400. Fault tree event PTKCOWOF Table II 5-5 Page II-207 λ50 = 1(-10)/hr RF = 30
49) System: Reactor Protection Component Type: Breakers Failure Mode: Fail to Open on Demand	5	612 demands	9.79(-3) /demand	4.39(-5) /(demand)^2	[2.97(-5)] /demand	[2.97(-3)] /demand	IEEE-500 Indoor design. AC breakers. Catastrophic failure. Page 148 Rec = 2.97(-4)/dem. Max = 2.97(-3)/dem.
50) System: Safeguards Actuation Component Type: Fuse, DC Power Failure Mode: Opens Prematurely		*	8.32(-7) /hour	1.08(-9) /(hour)^2	[2.15(-9)] /hour	[2.05(-7)] /hour	IEEE-500. Fuse. DC power. Opens prematurely. Page 193 Rec = 2.1(-8)/hr Max = 2.05(-7)/hr

NOTE: 1.23(4) indicates 1.23 x 10^4
W-1400: WASH-1400, Table III 2-1.
N-1363: NUREG/CR-1363, Table 23, page 63.
N-1205: NUREG/CR-1205, Table 14, page 35.
N-1362: NUREG/CR-1362, Table 20, page 51.

Generic data was used to calculate the mean and variance. No plant data was used.

TABLE 1.5.1-5 (continued)

SPECIALIZED COMPONENT HARDWARE FAILURE DATA

Component Description and Failure Mode	Plant-Specific				Generic		
	No. of Failures	Service Hours or Demands	Updated Mean	Updated Variance	Mean [λ20]	λ80/λ20 [λ80]	Comments - Data Sources
51) System: Safeguards Actuation Component Type: Master Relay Latch and Failure Mode: Fails to Energize on Demand		*	1.15(-5) /demand	3.38(-9) /(demand)^2	[4.92(-7)]	[1.03(-5)]	IEEE-500. Relay. Fails to close. Page 154 Rec = 2.25(-6)/dem Max = 1.03(-5)/dem
52) System: Safeguards Actuation Component Type: Relay General Failure Mode: Contacts Open		*	2.43(-7) /hour	3.26(-13) /(hour)^2	[3(-8)]	[3(-7)]	W-1400. Relay. Normally closed contacts open. λ5 = 3(-8)/hr λ95 = 3(-7)/hr
53) System: Safeguards Actuation Component Type: Bistable Switch Failure Mode: Failure to Open on Demand		*	3.88(-7) /demand	1.47(-13) /(demand)^2	[1.38(-7)] /demand	[5.52(-7)] /demand	IEEE-500. Bistable Switch. Failed to function when signaled. Page 483. Rec = 2.76(-7)/dem Max = 5.52(-7)/dem
54) System: Safeguards Actuation Component Type: Transmitters Failure Mode: Fail to Provide Proper Output		*	1.66(-6) /hour	6.28(-12) /(hour)^2	λ50 9.18(-7) /hour	RF 6	IEEE-500. Transmitter. Catastrophic failure. Page 428. Rec = 9.18(-7)/hr RF = 6 $\sqrt{max/low}$
55) System: Safeguards Actuation Component Type: Switches Failure Mode: Contacts Short Across		*	4.28(-7) /hour	3.36(-10) /(hour)^2	[1(-9)] /hour	[1(-7)] /hour	W-1400. Switches. Contacts short across. λ5 = 1(-9)/hr λ95 = 1(-7)/hr

NOTE: 1.23(4) indicates 1.23 x 10^4
W-1400: WASH-1400, Table III 2-1.
N-1363: NUREG/CR-1363, Table 23, page 63.
N-1205: NUREG/CR-1205, Table 14, page 35.
N-1362: NUREG/CR-1362, Table 20, page 51.

* Generic data was used to calculate the mean and variance. No plant data was used.

pseudodata effectively corresponding to the information assumed by Zion in estimating a failure rate. Alternatively, one can do a Bayesian analysis beginning with some uninformative or "flat" prior distribution, then modify it by f and T to obtain a posterior distribution which would have (at least approximately) a mean and variance equal to f/T and f/T^2. Also, this correspondence between f/T and the posterior mean is consistent with Zion's practice of equating the value of f/T in the EG&G reports to their prior mean, so we are not doing anything inconsistent by this transformation. If Zion had followed conventional Bayesian practice by choosing a "natural conjugate" prior distribution, in this case a gamma distribution, then the parameters of the posterior distribution, which, fortunately, is also a gamma distribution, are directly interpretable as effective data--number of failures and number of hours. Zion used discretized lognormal distributions for their prior distributions, so we cannot make this correspondence exactly. But, and this is one saving feature of a Bayesian analysis, with enough data the prior distribution does not matter too much, so approximating a discretized lognormal distribution by a gamma distribution should be reasonably adequate.

Thus, the failure rate posterior means and variances in the ZPSS Table 1.5.1-5 can be converted to effective data, say f_{POST} failures in T_{POST} hours. The Zion specific f and T are given so we can subtract them from the posterior effective f and T to determine the effective f and T associated with the prior distribution:

$$f_{PRIOR} = f_{POST} - f_{ZION}$$

$$T_{PRIOR} = T_{POST} - T_{ZION}$$

For example, consider the first entry in Table 1.5.1-5 which is reproduced at the end of this section. The posterior and variance, labeled, "Updated," are 5.28(-8)hr and 2.82(-15)/hr^2. Equating these to f/T and f/T^2 yields

$$T_{POST} = \frac{5.28(-8)}{2.82(-15)} = 1.87(7) \text{ hrs.}$$

$$f_{POST} = 5.28(-8) \times 1.87(7) = 1$$

That is, Zion's posterior mean and variance correspond to what one would estimate given only the data of one failure in 18.7 million hours (mhrs). The Zion experience consists of zero failures in 11.1 mhrs. Thus, the difference, which is Zion's rendering of the non-Zion information, amounts to 1 failure in 7.6 mhrs. (We note in passing that expressing prior information as being equivalent to 1 failure in 7.6 mhrs is much more scrutable than being told it is equivalent to a lognormal distribution with a 20th percentile of 2.8(-8)hr and an 80th percentile of 2.8(-7)hr.) From the Zion data alone the upper 95 percent statistical confidence limit on the underlying failure rate would be 2.7(-7)hr. From the effective posterior data, the upper 95 percent statistical confidence limit is 2.5(-7)hr, so in this case, and from this view, the prior does not have a marked effect.

For demand probabilities, given data of f failures in n demands, one would obtain the estimate, $p^* = f/n$, and the estimated variance, $var^*(p^*) = p^*(1-p^*)/n$. These can be equated to Zion's posterior mean and variance to solve for an effective f and n. For small p^*, these solutions correspond to those for λ^* with n replacing T.

Table 2.6-1 gives the effective prior data for all the entries in the ZPSS Table 1.5.1-5. The contributions of the priors to the final results vary considerably. In many cases the prior denominator, n or T, is roughly the same size as that for the Zion data; e.g., components 1, 3, 4, so the effect is roughly to decrease the variance by a factor of two. The precise effect depends on the numerator. In several cases the prior leads to a smaller and more precise estimate than would be obtained from the Zion data alone by effectively subtracting from the numerator while adding to the denominator (components 5, 8, 11, 18, 20, 30, 36, 38, and 39). In other cases (components 13, 17, 31, and 32), the prior denominator is about ten times that for Zion alone, so considerable additional precision is imparted. There are three cases (components 25, 35, and 49) where the prior leads to less precision than the Zion data alone would yield by subtracting from both numerator and denominator. Whether or not the contributions of the prior distributions are fair and just, depends on the actual information contained in the source documents. Whether this question is worth worrying about in the ZPSS depends on where the various component events occur in the system models. We address this question in Section 3.

It should be noted that the preceding analysis, and Zion's, is predicated on the Zion data given in the report. We have no way of validating the data, of determining the accuracy of the reported numerators and denominators. Section 1.5.1 of the Zion study indicates a good deal of care in collecting component data.

TABLE 2.6-1

Plant-Specific and Effective Prior Failure Data

	Plant Specifics			Effective Prior		
Component*	f	T(Hrs)	n (Demands)	f	T(Hrs)	n (Demands)
1	0	1.11(7)		1	7.6(6)	
2	(NO DATA FOUND)					
3	0		6968	.8		1.1(4)
4		6.08(5)		1	5.4(5)	
5	2	6.19(5)		-.5	2.8(5)	
6	14		1.131(4)	24		1.3(4)
7	0	6.95(5)		.1	1.3(6)	
8	3		1540	-.4		276
9	3	2.13(6)		.7	4.0(6)	
10	(NO DATA FOUND)					
11	3		338	-.3		637
12	6		231	.7		63
13	0	46		0	478	
14	0	3.25(4)		.2	4.2(4)	
15	0	7.6(4)		.3	1.1(5)	
16	0	1.52(5)		.5	2.0(5)	
17	0	66		0	700	
18	1	3.8(3)		-.5	[illegible]	
19	0	1.9(3)		0	[illegible]	
20	2		1155	[illegible]		385
21	0	1.52(5)		[illegible]	1.1(5)	
22	0	2.35(5)		[illegible]	4.0(5)	
23	(Assumed Negligible)					
24	(Assumed Negligible)					
25	30		1693	-4.2		-275
26	6	1.34(3)		2.8	1.3(2)	
27	5		3.12(3)	.5		1.3(3)
28	1		3.12(3)	1.9		2.3(3)

TABLE 2.6-1 (continued)

Plant-Specific and Effective Prior Failure Data

Component*	Plant Specifics f	Plant Specifics T(Hrs)	Plant Specifics n (Demands)	Effective Prior f	Effective Prior T(Hrs)	Effective Prior n (Demands)
29	0	9.10(5)		.2	-1.4(5)	
30	1	3.04(5)		-.1	2.4(5)	
31	3	3.04(5)		36	3.3(6)	
32	0	2.02(5)		.1	1.6(6)	
33	0	2.02(5)		0	-1.2(5)	
34	10		1647	7.8		1449
35	7		1720	-1.9		-352
36	4		462	-1.3		80
37	1		183	.4		142
38	2	33		-.5	20	
39	3		1155	-.5		174
40	0	7.6(4)		.3	1.1(5)	
41	0	3.03(6)		.2	5.6(6)	
42	(Assumed Negligible)					
43	(NO PLANT DATA)			.1	3.6(4)	
44	(NO PLANT DATA)			.1	1.5(4)	
45	(NO PLANT DATA)			1.6	2.5(5)	
46	(NO PLANT DATA)			0	1.4(6)	
47	(NO PLANT DATA)			0	1.4(7)	
48	(NO PLANT DATA)			0	1.7(7)	
49	5	612		-2.8		-391
50	(NO PLANT DATA)			0	770	
51	(NO PLANT DATA)			0		3400
52	(NO PLANT DATA)			.2	7.5(5)	
53	(NO PLANT DATA)			1		2.6(6)
54	(NO PLANT DATA)			.4	2.6(5)	
55	(NO PLANT DATA)			0	1.3(3)	

*See Table 1.5.1-5 for component definitions.

The ZPSS analysis is also based on the assumption of constant (across time and similar components) failure rates and probabilities. This is standard in risk assessments, but the reader should be aware that it may be the source of substantial errors that are not quantifiable except by Bayesian extremists (and Zion does not go that far). Aging effects may be present and failures may cluster due to imperfect repair. Modeling such effects can be difficult and is often impossible to do with meaningful precision because of limited data. The result of the Zion study is not "the risk from the Zion plant," as Section II,2.1 claims, but is an estimate of the Zion risk--an estimate built from a variety of simplifying assumptions and models.

2.6.4 Estimation of Initiating Event Rates

The initiating event frequency data are given in ZPSS Table 1.5.1-48, Page 1.5-159, by reactors (PWR's), and the operation years covered by these data are given in Table 1.5.1-49. The analysis used is called a two-stage Bayesian analysis, but details are not given. (An unpublished paper is referenced which we obtained from the author.) In this analysis, the transient data are used to (Bayesianly) estimate a frequency distribution for the plant-to-plant variation in underlying initiating event rates, then this distribution is modified by the Zion data to yield posterior distributions which are summarized in the attached Table 1.5.1-50 erroneously labeled Initiating Event Occurrence Probability, (by either their definition or ours). We will not delve deeply into this analysis, but we will examine its effect, as before, by equating the posterior mean and variance to f/T and f/T^2, which one would obtain from a statistical analysis based on f occurrences in T years.

Table 2.6-2 gives the Zion data, and the effective prior and posterior data calculated this way. It shows reasonable results. For initiating events for which there is little evidence of plant-to-plant variation, the non-Zion data count much more heavily than when there is considerable variation among plants. For example, events 1, 2, 5, 6, 10, 11c, and 13b have not occurred in the 131 operational years covered by the data. Zero out of 131 is not much different from the effective .15/164 that Zion uses. (It is a little surprising, from a Bayesian point of view, that the states of knowledge for these events are identical.) Events 7, 11a, 12, and 13a show considerable variability across plants and in these cases the prior contribution is negligible.

If we calculate an upper 95 percent statistical confidence limit on an initiating event rate, based on data of .15/164, the resulting bound is .02/yr (based on linear interpolation in a chi squared table). The Table 1.5.1-50 upper 95th posterior percentile for the Zion specific data

TABLE II.4-12 and 1.5.1-50

ZION UNITS 1 AND 2
INITIATING EVENT OCCURRENCE PROBABILITY PER PLANT YEAR

Initiating Event Category	Zion 1 Plus Zion 2 Plant-Specific						PWR Population Generic	
	5%	Median	95%	Range Factor	Mean	Variance	Mean	Variance
1	3.33-5	3.44-4	3.55-3	1.03+1	9.40-4	5.74-6	1.01-3	6.37-6
2	3.33-5	3.44-4	3.55-3	1.03+1	9.40-4	5.74-6	1.01-3	6.37-6
3	1.27-2	3.07-2	7.40-2	2.41	3.54-2	4.17-4	2.69-2	2.23-1
4	2.84-3	1.48-2	7.68-2	5.20	2.44-2	1.03-3	8.75-2	1.40-1
5	3.33-5	3.44-4	3.55-3	1.03+1	9.40-4	5.74-6	1.01-3	6.37-6
6	3.33-5	3.44-4	3.55-3	1.03+1	9.40-4	5.74-6	1.01-3	6.37-6
7	4.14	5.13	6.35	1.24	5.17	4.55-1	3.41	2.02+1
8	9.36-2	2.20-1	5.18-1	2.35	2.52-1	1.97-2	6.00-1	1.07+1
9	1.90-1	3.37-1	5.98-1	1.78	3.58-1	1.66-2	3.21-1	9.57-2
10	4.65-3	1.68-2	6.07-2	3.61	2.28-2	4.37-4	4.77-2	8.13-2
11a	2.84	3.65	4.69	1.29	3.69	3.21-1	4.00	1.29+1
11b	8.72-3	3.84-2	1.69-1	4.40	5.76-2	4.15-3	4.04-1	3.59
11c	3.33-5	3.44-4	3.55-3	1.03+1	9.40-4	5.74-6	1.01-3	6.37-6
12	3.29-1	5.96-1	1.08	1.81	6.36-1	5.62-2	1.59-1	4.02-1
13a	2.94	3.73	4.74	1.27	3.77	3.03-1	4.11	1.00+1
13b	3.33-5	3.44-4	3.55-3	1.03+1	9.40-4	5.74-6	1.01-3	6.37-6

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

TABLE 2.6-2

Initiating Event Zion and Effective Prior and Posterior Data Table Entries are (No. of Occurrences)/(No. of Operating Yrs.)

Initiating Event Category*	Zion	Prior	Posterior
1. Large LOCA	0/11	.15/153	.15/164
2. Medium LOCA	0/11	.15/153	.15/164
3. Small LOCA	1/11	2/74	3/85
4. SIG Tube Rupture	0/11	.6/13	.6/24
5. Steam Break Inside Cont.	0/11	.15/153	.15/164
6. Steam Break Outside Cont.	0/11	.15/153	.15/164
7. Loss of Feedwater Flow	58/11	.7/.4	58.7/11.4
8. Closure of one MSIV	3/11	.2/1.8	3.2/12.8
9. Loss of Primary Flow	5/11	2.7/10.6	7.7/21.6
10. Core Power Increase	0/11	1.2/41	1.2/52
11a. Turbine Trip	41/11	1.4/.5	42.4/11.5
11b. T.T., Loss of Off-site Power	0/11	.8/2.9	.8/13.9
11c. T.T., Loss of Serv. Water	0/11	.15/153	.15/164
12. Spurious Safety Inj.	8/11	-.8/.3	7.2/11.3
13a. Reactor Trip	42/11	4.9/1.4	46.9/12.4
13b. Reactor Trip, Loss of Cooling Water	0/11	.15/153	.15/164

is .00355, lower than the posterior bound by about a factor of six. This indicates the distribution is somewhat tighter than would be expected from the mean and variance, perhaps because of the DPD arithmetic (which is not explained in enough detail to reproduce). An alternative way to translate Zion's results into effective data is the following. The ratio of upper and lower 95 percent statistical confidence limits, given data of

$$f/T, \text{ is } \chi^2_{95}\ (2f + 2)/\chi^2_{05}\ (2f)\ ,$$

where

$$\chi^2_p\ (n)$$

denotes a percentile on the chi squared distribution with n degrees of freedom. If we equate the ratio of Zion's posterior 95th and 5th percentiles to this ratio, we can solve for f. Then equating the Zion posterior mean to f/T yields T. Table 2.6-3 gives effective prior and posterior data based on this analysis. The pattern in Table 2.6-3 is much the same as that in Table 2.6-2, but the prior counts more heavily, particularly for those events with few occurrences. Now the events that have not occurred are counted as 1/1064 when the data across all plants considered by Zion are 0/131. Still, the Zion estimates seem plausible or conservative. An alternative way to analyze these data would be to group the plants in clusters which have apparently homogeneous occurrence rates, then estimate the rate for Zion using all the plants in its cluster. It appears that such an analysis would yield larger denominators than the posterior results in Table 2.6-3, except for those events that have not happened.

An assumption underlying Zion's analysis here, as in their analysis of component failure data, is that of a constant occurrence rate across time. No analysis is given to support this assumption, though the referenced source of transient data (EPRI NP-801) should permit such an analysis. There may be aging trends that need to be considered for transients such as steam generator tube rupture.

2.6.5 The Treatment of Maintenance Data

Zion models the unavailability of a component due to maintenance as the rate at which maintenance actions occur (actions per component hour, excluding cold shutdown hours) times the mean duration of a maintenance. Prior distributions for both are developed, modified by the Zion data

TABLE 2.6-3

Initiating Event Zion and Effective Prior and Posterior Data Based on Percentiles: (No. Occurrences)/ (No. Operational Yrs.)

Initiating Event Category*	Zion	Prior	Posterior
1	0/11	1/1053	1/1064
2	0/11	1/1053	1/1064
3	1/11	3.5/116	4.5/127
4	0/11	1.5/50.5	1.5/61.5
5	0/11	1/1053	1/1064
6	0/11	1/1053	1/1064
7	58/11	7/1.6	65/12.6
8	3/11	2/8.8	5/19.8
9	5/11	5/17	10/28
10	0/11	2.5/99	2.5/110
11a	41/11	6/1.7	47/12.7
11b	0/11	2/24	2/35
11c	0/11	1/1053	1/1064
12	8/11	1.5/4	9.5/15
13a	42/11	9/2.5	51/13.5
13b	0/11	1/1053	1/1064

to yield posterior distributions, then the distribution of the product is obtained. Consider first the rate of maintenance actions.

The ZPSS results on maintenance frequency are summarized in ZPSS Table 1.5.1-29 (attached). The analysis is just the same as that for component failure rates, except for the source of the priors. Similarly, we can equate the posterior mean and variance to f/T and f/T^2 to identify the effect of the assumed prior distributions. Because, for the most part, there have been several maintenance actions on the different components, the priors contribute little to the final estimate. One exception is the turbine-driven auxiliary feedwater pumps. The Zion data are 41 events in 6.2(4) hours. Thus, the prior effectively diminishes both numerator and denominator, an indication that prior and data are not too consistent.

Zion's analysis is based on the assumption that the maintenance rate is constant. The annual maintenance data, given in Tables 1.5.1-17 to 1.5.1-28 (PP. 1.5-91ff), in some cases (most notably the turbine-driven AFWS pumps), cast doubt on this assumption. We will consider this case in Section 3.

For their maintenance duration analysis Zion assumed maintenance duration to be lognormally distributed with parameters μ and σ^2. (Individual maintenance times are not given in the report so we cannot check this assumption. To develop a prior distribution for μ and σ^2, Zion specified a discrete set of 5th percentiles on maintenance time and assigned subjective probabilities to each. They did the same for 95th percentiles. Then they took all pairs and derived the values of μ and σ^2 that result. For example, six 5th percentiles and six 95th percentiles result in 36 (μ, σ^2) pairs and corresponding subjective probabilities. The mean of a lognormal distribution is $\exp(\mu + \sigma^2/2)$.

From the prior distribution of (μ, σ^2), the prior distribution of $\exp(\mu + \sigma^2/2)$ can be derived. It is not necessary to do so, but Zion did and the results are shown (as continuous distributions) in Tables 1.5.1-10, 12, 14, and 16. (This analysis is not described in the report; we conjecture that this must be the method and this was confirmed by the authors.) A problem with these tables is that the components to which they apply are not identified. Apparently, though, each of these even-numbered tables is paired with the preceding odd-numbered table and there the applicable components are identified. These odd numbered tables are also called prior distributions, but they are not. They merely give a distribution of maintenance duration corresponding to one (μ, σ^2) pair--the one assigned the highest subjective probability. Also, Tables

TABLE II.4-8 and 1.5.1-29

SPECIALIZED COMPONENT MAINTENANCE FREQUENCY DATA

Component	SITE-SPECIFIC DATA		PRIOR DISTRIBUTION	SPECIALIZED FREQUENCY DISTRIBUTION*	
	Events	Service Hours	Table	Mean (Events/Hour)	Variance
Motor-Driven Auxiliary Feedwater Pumps	26	1.24×10^{5}	1.5.1-7	1.87×10^{-4}	1.24×10^{-9}
Turbine-Driven Auxiliary Feedwater Pumps	41	6.20×10^{4}	1.5.1-7	5.33×10^{-4}	1.54×10^{-8}
Centrifugal Charging Pumps	17	1.24×10^{5}	1.5.1-8	1.45×10^{-4}	1.07×10^{-9}
Component Cooling Pumps	46	1.90×10^{5}	1.5.1-8	2.36×10^{-4}	9.29×10^{-10}
Containment Spray Pumps	11	1.86×10^{5}	1.5.1-6	6.54×10^{-5}	2.44×10^{-10}
Residual Heat Removal Pumps	6	1.24×10^{5}	1.5.1-6	6.12×10^{-5}	2.87×10^{-10}
Safety Injection Pumps	1	1.24×10^{5}	1.5.1-6	4.04×10^{-5}	1.50×10^{-10}
Service Water Pumps	24	1.86×10^{5}	1.5.1-8	1.35×10^{-4}	6.74×10^{-10}
Fan Cooler Units	14	3.10×10^{5}	1.5.1-8	5.85×10^{-5}	1.57×10^{-10}
Diesel Generators	137	1.62×10^{5}	1.5.1-8	8.09×10^{-4}	3.76×10^{-9}
Sodium Hydroxide Addition Lines	7	1.86×10^{5}	1.5.1-6	5.13×10^{-5}	1.74×10^{-10}
Service Water Lines to Auxiliary Feedwater Pumps	5	1.86×10^{5}	1.5.1-7	5.24×10^{-5}	1.74×10^{-10}

*Distributions are lognormal.

1.5.1-9 and 11 are said to be applicable to <u>no</u> Zion components, but for some reason they are in the report. The accompanying text is an exception to the generally lucid prose in the report.

At any rate, given a prior for (μ, σ^2), and individual maintenance times, the posterior distribution of (μ, σ^2) can be obtained and from that the posterior distribution of the mean duration, $\exp(\mu + \sigma^2/2)$ can be obtained. These are shown as continuous distributions in Figs. 1.5.1-2 to 13 (PP. 1.5-105ff). Convoluting these with the posterior distributions of maintenance frequency yields the posterior distributions of unavailability shown in Figs. 1.5.1-14 to 25.

Not having the individual maintenance times, we cannot separate the contributions to these distributions of the priors and of the data. We can, however, compare the posterior means to those calculated from just the Zion data (in Tables 1.5.1-17 to 28). This comparison is given in Table 2.6-4. The most notable disagreement is for Safety Injection Pumps, but there only one observation is available and hence the prior dominates.

Zion's posterior distributions of unavailability represent their uncertainty about what might be called average annual unavailability. Since the objective of the study is to estimate annual risks, one might instead consider the variation of annual unavailability. The tables of maintenance data appear to cover about five full years. Thus, roughly, one could multiply the variances of the posterior distributions of unavailability by five in order to get distributions which incorporate the year-to-year variability of unavailability. Visually, imagine stretching the distributions in Figs. 1.5.1-14 to 25 by a factor of the square root of 5.

With the exception of the question of what variation should be included, the ZPSS estimates of unavailability are consistent with the Zion data. There are enough data to overcome their priors, so their method of constructing the priors was really unnecessary. It should be noted that a more conventional analysis could have been done. In this analysis one considers the succession of up and down times for each component, looks for trends or similarities, and estimates the average up time and average down time (perhaps Bayesianly), then estimates unavailability by

$$\mu = \frac{\text{Ave. Down Time}}{(\text{Ave. Up Time}) + (\text{Ave. Down Time})}$$

TABLE 2.6-4

Comparison of Maintenance Estimated Means

Component	Posterior			Zion Data			
	Freq. (per hr.)	Dur. (hrs.)	Unavail.	Freq. (per hr.)	Dur. (hrs.)	Unavail.	n*
Motor-Driven AFWS Pumps	1.9(-4)	55	1.0(-2)	2.1(-4)	47	1.0(-2)	26
Turbine-Driven AFWS Pumps	5.3(-4)	67	3.6(-2)	6.6(-4)	58	3.8(-2)	41
Cent. Chg. Pumps	1.5(-4)	34	4.9(-3)	1.4(-4)	35	4.7(-3)	17
Comp. Cool. Pumps	2.4(-4)	136	3.2(-2)	2.4(-4)	200	4.8(-2)	46
Cont. Spray Pumps	6.5(-5)	19	1.3(-3)	5.9(-5)	15	8.7(-4)	11
RHR Pumps	6.1(-5)	63	3.8(-3)	4.8(-5)	77	3.7(-3)	6
Safety Inj. Pumps	4.0(-5)	40	1.6(-3)	8.1(-6)	24	1.9(-4)	1
Serv. Water Pumps	1.4(-4)	17	2.3(3)	1.3(-4)	16	2.0(-3)	24
Fan Cooler Units	5.9(-5)	16	9.3(-4)	4.5(-5)	32	1.4(-3)	14
Diesel Generators	8.1(-4)	43	3.4(-2)	8.4(-4)	36	3.0(-2)	137
Sod. Hyd. Add. Lines	5.1(-5)	88	4.5(-3)	3.8(-5)	45	1.7(-3)	7
Serv. Water Lines to AFWS Pumps	5.2(-5)	445	2.3(-2)	3.2(-5)	1680	5.4(-2)	6

*n = Number of Maint. Events

2.6.6 Data-Free Estimates

As discussed in 2.6.3 above, to obtain prior distributions Zion either equated WASH-1400 5th and 95th percentiles to their 20th and 80th, or they took the ratio of WASH-1400's 5/95 percentiles as their 20/80 ratio. This can result in quite skewed and elongated distributions for which the mean and variance do not provide a very good description. Fortunately, the amount of data available from Zion and the DPD arithmetic can effectively chop off these long tails in the most extreme cases. There are, however, numerous probabilities and rates for which no data are available. Most of these pertain to human errors, but some pertain to hardware failures. With respect to the latter we have encountered some instances in which Zion accepted WASH-1400 bounds as their own 5/95 percentiles, rather than stretch them out of 20/80 as they did in those cases in which data were available. These are:

Rupture of a motor-operated valve. As discussed in Section 3.d, rupture of two MOV's leads to an interfacing systems LOCA and one of the more serious releases. If Zion had stretched out the WASH-1400 bounds, the estimated probability of this event would increase by four orders of magnitude. If they had modified this distribution by Zion data, their estimate would have been a factor of six larger.

Pressure vessel rupture. By citing WASH-1400 bounds on the occurrence rate of this event, Zion dismissed it as a potential LOCA. If they had stretched these bounds, the contribution would not have been negligible.

Pipe rupture. For pipes exceeding 3" diameter the WASH-1400 bounds are 3(-12) and 3(-9) pipe failures per hr. Equating these to lognormal 5th and 95th percentiles yields a mean of 8.6(-10)/hr. Equating these to the 20/80 percentiles yields a mean of 4.5(-7), an increase by a factor of 500. Thus, for example, in the component cooling water system the ZPSS identified 30 piping sections and thus estimates the failure probability as 2.58(-8) over a one-hour period. If they had used 20/80 assumptions, this probability would have been estimated as 1.4(-5). Similar results pertain to the service water system in which 25 pipe sections are considered.

The point of this discussion is not to claim one estimate is right, the other wrong, nor is it to insist that Zion should have been consistent in their treatment of WASH-1400 bounds. As Bayesians they can specify any prior distributions they feel represent their state of knowledge. One wishes, though, the reader would be told why in some cases, WASH-1400 bounds are acceptable and why, in others, they should be stretched out. The main point of these examples is that the results can be quite sensitive to what would seem to be minor differences in assumptions. This point is more than academic because of the dominant role of the interfacing systems LOCA in estimating risk.

As noted above, the DPD arithmetic can chop off the tails of highly skewed lognormal distributions. An example of this effect, discussed in our Sec. 3, occurs in Zion's estimate of recirculation cooling failure. Two first order terms are $1.25Q_{H1}$ and Q_{SUMP}, where H1 is a human error and SUMP denotes a clogged sump. Both probabilities are estimated strictly subjectively. The variance of Q_{H1} is given as 4.54(-7) and that of Q_{SUMP} is 6.4(-6). Thus the system variance should exceed $(1.25)^2$ 4.54(-7) + 6.4(-6) = 7.1(-6), because there are also other terms contributing to system failure. However, when all of these are convoluted by DPD arithmetic, the system variance is only 1.7(-7) which is smaller by a factor of 40 or more than what would be obtained without discretizing. In effect, DPD increased the (apparent) available information pertaining to recirculation failure by a factor of 40. No explanation is given in the report as to how a distribution was discretized--how many discrete values and which ones--and conversations with the authors indicate no consistent algorithm was used. Zion should have reported the mean and variance of the distributions actually used in the calculation.

2.6.7 System Quantification

If one defines a system as a specified arrangement of components by a fault tree or a reliability block diagram, a mathematical model can be developed which expresses the system failure probability as a function of the component failure probabilities and rates. Given posterior probability distributions for these component parameters, and prior distributions where no data are available, the resulting posterior distribution of the system failure probability can then be derived or approximated. The approximation method used by Zion is their DPD arithmetic.

In Section 3 we consider the results of this analysis for some specific systems. As in the cases of component and initiating event estimates, it is possible to express Zion's analysis in terms of effective data and a conventional statistical analysis and thus assess the impact of their prior distributions and analysis methodology on their system results. Here we consider a general point.

In Section 0.16 the ZPSS authors make the excellent point (couched in Bayesian terms) that if a system contains two or more components whose failure probabilities are estimated by the same data, then this fact must be accounted for in estimating the system failure probability. Thus, for example, for two identical components in series for which the posterior mean and variance are α and β^2, respectively, the system failure probability has a posterior mean and variance of 2α and $4\beta^2$. If the two estimates were incorrectly assumed to be independent, the derived variance would be $2\beta^2$, which is too small. For two parallel components, the failure probability is p^2, say, which has a mean value of $\alpha^2 + \beta^2$. This is correct, but as a point estimate of p^2, this mean value can be very conservative.

Suppose one begins with a noninformative prior and modifies it with data, x/n, so that the posterior distribution has a mean of $p^* = x/n$ and variance = $p^*(1-p^*)/n$. Then the posterior mean which is the Zion point estimate of p^2 is

$$\alpha^2 + \beta^2 = p^{*2} + p^*(1-p^*)/n \quad .$$

The expected value of this estimate (with respect to the sampling distribution of p^*) is (approximately)

$$E(\alpha^2 + \beta^2) = p^2 + 2p(1-p)/n \quad .$$

This result shows that, unless $(1-p)/n$ is much less than p, the Zion posterior mean value, regarded as an estimator of p^2, could be seriously biased (but in a conservative direction). This problem affects Zion's estimate of the probability of an interfacing system LOCA, which is one of their dominating contributors to risk.

From a Bayesian viewpoint one could argue that both p and p^2 should not be estimated by their posterior means. In full-blown Bayesian analyses, a point estimate is selected on the basis of a loss function. If squared error loss is chosen, which means the penalty for estimating p by p^* is $(p-p^*)^2$, the posterior mean is the resulting estimator. However, squared error for p is not equivalent to squared error for p^2, so a Bayesian indiscretion occurs. Straightening this out is beyond the scope of this review. Section 0.16 of the ZPSS creates the unfortunate impression that if one has selected a point estimate, say p^*, with or without encumbering that estimate with lognormal connotations, then p^{*2} is unacceptable as a point estimate of p^2. Not so by either Bayesian or statistical arguments.

2.6.8 Completeness

Another concern in risk estimation is completeness. What about accident sequences not covered in the report? In Section 0.19 of the Zion report, the authors discuss completeness. They argue that all possible initiating events are included in their list, that all possible resulting plant damage states have been identified, that the requisite system failures that lead to a damage state, given an initiating event, are known, and that the combinations of component failures that fail a system are known. Thus, there is no set of damage-causing circumstances omitted from the study. The only thing conceivably incomplete is the set of causes by which multiple component failures might occur. But, because Zion can put a number on this everything is covered.

As an example, consider a system consisting of two identical trains. It can fail if (a) there are two independent train failures, (b) one train is out of service for maintenance and the other fails, or (c) one train has been disabled due to a human error and the other fails. Additionally, there may be (d) a human error or errors that disable both trains and there may be (e) support system failures that disable one or both trains. Zion considers all of these by conditioning on the state of a support system, generally electric power for which eight states are defined, then estimating the conditional probability of a through d. Even so, it is recognized that there may be "other" causes of joint failure of the two trains. For example, there may be human or physical links not explicitly recognized. Zion estimates system failure probabilities for these situations in a variety of ways:

1. Inclusion of a β-factor.
2. Linkage to another estimate.
3. Judgment leading to a conclusion of "negligible."

Consider these in turn.

The β-factor is in effect a factor to account for possible dependence between failure events. In the above example, if q denotes the failure probability of one train, then inclusion of a β-factor leads to system failure probability of $q^2 + \beta q$, ignoring other terms in the system failure model. If we write this as $q(q + \beta)$, then $q + \beta$ corresponds to the conditional failure probability of the second train given failure of the first. In principle, β can be estimated from data, but it is not in the ZPSS. Zion specifies their personal probability distribution for β as a lognormal distribution with a mean of .014 and a variance of 6.1(-4), which corresponds to 5th and 95th percentiles of

.001 and .05. This "state of knowledge" is the same everywhere it is used, including the following systems.

Low-Pressure Injection

One train, or supercomponent, consists of a single pump, motor-operated valve, check valve, an air-operated valve, and two manual valves. Zion's mean failure probability for this train is 8.61(-4), so multiplying by .014 gives an "other" contribution to system unreliability of 1.2(-5). The system unreliability estimate is 4.5(-4), dominated by a human error, so the impact of "other" is negligible.

Hi-Head Recirculation

The ß-factor of .014 is applied to one pair of pumps and four pairs of MOV's in the two trains, and results in a contribution of about one-fourth of the estimated system unreliability of 3.9(-4).

Lo-Head Recirculation

The ß-factor of .014 applied to one pair of pumps and three pairs of MOV's contributes a failure probability of 7.6(-5) to the total of 5.2(-3) and so is negligible.

Containment Spray Recirculation

Among the minimal cutsets, one is the failure of two MOV's; six are failures of three MOV's. Therefore, Zion takes $7\beta Q_{MOV}$ to be the "other" failure probability. Even so, this probability is estimated as 1.5(-4) and so is only about 10 percent of the estimated system unreliability of 1.6(-3).

There is one case where the ß-factor appears to be improperly applied.

High-Pressure Injection

Here, the ß-factor of .014 is multiplied by the total estimated probability of triple failures, which is 2.1(-9) and already negligible, thus resulting in an "other" contribution of 2.9(-11). To be consistent with their other "other" analyses, and the definition of a ß-factor, Zion should have counted up all pairs and trios of like components and multiplied their single failure probabilities by .014.

The basis for Zion's assumed personal probability distribution for the ß-factor is vague. A typical statement is the following:

"Most of the observed coupled failures in the industry involved motor-operated or air-operated valves that had to change position on demand. The frequent partial tests and the full annual system test indicate that an <u>unforseen</u> (sic) common cause failure is of low frequency. We express this state of knowledge by taking a ß-factor with range."

$$1.0 \times 10^{-3} \text{ to } 5.0 \times 10^{-2} \text{ which yields Mean:}$$

$$1.4 \times 10^{-2} \quad . \quad \text{(P. 1.5-375.)}$$

Of course any two beings possessing the same knowledge would do the same. (See P. 0.4-2.) It would have been more straightforward for the authors to say, "We will model explicitly those dependencies we are aware of and deem important, such as by conditioning on electric power, and omit any others, because we feel they have negligible probability."

<u>Linkage to Another Estimate</u>

In some cases, Zion argued that the system failure probability due to "other" causes was quantitatively linked to that due to some explicit cause. One example of linkage was for the <u>Electric Power System</u>. The identified failure causes are hardware failures, maintenance, fuel oil of the ZPSS supply, diesel fires, and human error (See e.g., P. 1.5-263 of the ZPSS). All of these are estimated assuming independence. The report says, "We are extremely confident that the total contribution to each system failure state from these unidentified causes is much less than that presented by the least contributing identified cause. For conservatism, however, we set the 95th percentile of the "other causes" distribution equal the mean of the smallest identified contributor (and) allow a range of three orders of magnitude between the 5th and 95th percentiles of the assumed lognormal distribution" (P. 1.5-211).

The result of this is to add another 30 percent of the minimum term to the system failure probability and thus no impact results.

In two other systems, "other" failures are not discussed, but an explicit common cause failure is called out.

<u>Reactor Protection System</u>

The common cause failure considered is miscalibration of similar instruments in a particular set. The calibration is done sequentially by a single technician or group of technicians so "extremely close coupling" results. But it is said that most calibration errors would not render the instrument unable to signal a reactor trip. Zion then sets the failure probability of a set of instruments due to miscalibration equal to the failure probability of a single instrument channel by other causes. Since this is a small number, 2.66(-4), and it takes failure of two sets of instruments to fail the system, the resulting contribution is negligible.

Engineered Safeguards Actuation System

The treatment of instrument miscalibration is the same as for the Reactor Protection System. Thus, again, the quantitative treatment of "other" system failure causes could just as well have been omitted, as it was for some systems.

Judgement Leading to a Conclusion of Negligible

There are six systems for which it is argued that common cause failures, not explicitly accounted for, have negligible probability:

- Containment Spray
- Sodium Hydroxide Addition
- Containment Fan Cooling
- Component Cooling Water
- Service Water
- Auxiliary Feedwater

In general, these systems involve two or more trains of similar components and so would be candidates for application of the universal ß-factor of .014. We have not investigated the effect of doing so and do not advocate this analysis, but it looks as though some appreciable changes could result. In Section 3 we point out that the AFWS unreliability could double if the ß-factor were included. The containment fans and sprays are critical in mitigating various accidents so it is certainly desirable that there be no overlooked cause of multiple component failures. In our reanalysis, we include common cause factors for these systems.

2.7 External Events

2.7.1 Seismic Effects

In this section, the seismic external event is reviewed. The material in Sections 2.7.1.2 to 2.7.1.7 is based on four draft reports prepared under contract to Sandia National Laboratories. These reports are contained in the Appendices B through E. In addition, the comments given in References 2-16 through 2-18 were used in the review process. The comments given in Sections 2.7.1.2 to 2.7.1.7 are presented by the section being reviewed in the ZPSS.

2.7.1.1 Seismic Logic Model

The approach to system modeling used started with the determination of the fragility of all major components of the applicable plant safety-related systems and structures. All but those components or structures which were in the range of possible ground acceleration were then eliminated. The initial fluid system/component list seems to be reasonably complete with the possible exception of a number of valves, and the main feedwater pumps which are presumably assumed to fail due to loss of off-site power. The generic electrical components seem to be well represented in ZPSS Table 7.2-2, but not necessarily with respect to specific applications. An example of this is that cable trays are only listed under Electric Power (480V), but apply to other electrical systems as well.

Given that a seismic event will cause loss of off-site power before any other failures, the seismic initiator becomes essentially a turbine trip due to loss of off-site power or a seismically-induced small LOCA due to RCP seal failure also with loss of off-site power.

The core-melt model is then represented entirely by a fault tree which was apparently developed from the event tree/fault tree model used in the internal events analysis by eliminating all events not affected by seismic failures.

The resulting fault tree seems reasonable on this basis, although the ZPSS procedure was not reconstructed. Some events not expected to fail seismically were included for some reason in the fault tree only to be eliminated when the Boolean equation is subsequently written. A potential shortcoming of this type of analysis is the exclusion of all components/events not subject to seismic failure which could fail due to random hardware failures, human errors, or test and maintenance in combination with seismically-caused failures. It is possible that cutsets from the seismic failures only may dominate the core-melt frequency due to the seismic initiating event cause, but that was not discussed in the documentation nor evaluated in the revision.

The equation for M_S, found on ZPSS Page 7.2-7, differs from that found on Page II.7-15. In the development of M_S in Section II, component 15, the safety injection pumps, is erroneously included.

The ZPSS seismic analysis was entirely separate from the other parts of the system analysis until the application of the site matrix. An alternative method which considers nonseismic and seismic failures together would be to apply the seismic cause directly to the basic internal event tree/fault tree model. As it is, the ZPSS fault tree appears to be sufficient in detail to instill confidence in the reader and may fail to identify some important cutsets involving combinations of seismic and nonseismic events.

2.7.1.2 Review of Section II.7.1, Seismic

Section II.7.1.1, Seismicity

1. Page II.7-1, fifth from last line: "Such a curve... would adequately characterize the seismic activity at the site, were we able to draw it." This is an overstatement. The curve would provide no information, for example, about duration of shaking or frequency content, although these may have a significant impact on seismic response, performance or damage.

2. This summary (and also that in Section 7.2.2) does not faithfully restate the conclusions and reproduce the results of Section 7.9.1. Nowhere in the analysis given in Section 7.9.1 are rigid bounds imposed on effective peak acceleration; this asymptotic behavior at low risk levels is, however, the single most striking feature of seismicity curves of Fig. II.7-1 (or Fig. 7.2-1). The last sentence in Section II.7.1.1 does not adequately explain the logic which led from Section 7.9.1 to the exceedance curves used to evaluate core damage probabilities and final seismic "frequency-of-probability" curves.

Section II.7.1.2, Fragility

3. Figure II.7-2: It is preferable to label the different fragility curves with fractions (which sum to one) rather than with cumulative frequencies. The format of display in Fig. II.7-4 is correct in this regard. The use of cumulative frequencies is especially confusing if there is a chance that the different curves in a family might overlap. (This could easily happen if the fragility curves are permitted to have significantly different "β" values.)

4. The last paragraph of ZPSS Section II.7.1.2 makes it clear that acceleration is not necessarily the best response parameter in terms of which to define fragility curves; for example, relative displacement might be superior in some cases.

5. The choice of the lognormal distribution is expedient but not necessarily consistent with available information. Seismic response is more nearly normal than lognormal. (Seismic excitations are approximately normal, with mean zero, and any linear system preserves this normality; hence, the response time histories are normal.) The absolute maximum of the random response of a linear system follows an extreme value distribution about which much is known. Hence, the sweeping assumption of lognormality is justified mainly by analytical convenience (i.e., it facilitates analysis of products of independent random variables).

Section II.7.1.3, Plant Logic

6. Figure II.7-4. There is inadequate explanation of how the core-melt fragility curves in the figure were obtained. In theory, there should be one such fragility curve for each combination of mean component resistances (parameter a for each component) and seismic hazard curve in Fig. II.7-1. All these (ideally infinite) curves should then be grouped (e.g., into five curves as in the case in Fig. II.7-4). No such work is documented or mentioned in any of the sections reviewed. In any case, the grouping of Fig. II.7-4 is too coarse, especially at the low-resistance end.

7. Page II.7-15: The expression for M_8 should not include component 15.

Section II.7.1.4, Seismic Core Melt Frequencies

8. Figure II.7-5: Again, intermediate steps of calculation are not shown. In each of the 45 combinations of the nine seismic hazard functions in Fig. II.7-1 and the five fragility curves in Fig. II.7-4, there is one value of the

annual frequency of core melt. It would be helpful to show the histogram of these 45 values (or more values if Fig. II.7-4 is revised to include more refined grouping) and compare this histogram with the smooth fits of Fig. II.7-5.

Section II.7.1.5, Initial Assembly Leading to Containment Tree Entry States

9. Table II.7-2: The discretization of acceleration levels is coarse and may lead to nonnegligible errors in the calculation of the frequency of core melt. More refined convolution should be made of the rate density of EPA with the conditional core-melt probability, for low to moderate acceleration values. It appears necessary to extend calculations beyond 0.65 g since according to Fig. II.7-1, accelerations of this magnitude cannot occur.

2.7.1.3 Review of Section 7.2, Seismic Events

1. In general, experience indicates that relationships such as

$$M_b = 0.5\ (I_o + 3.5)$$

should be avoided when doing site-specific studies because of the weak correlation between m_b and I_o as compared to m_b and other types of intensity data; i.e., m_b and the fall off-of-intensity with distance, m_b and the total felt area associated with the event.

2. As discussed in the Review of Section 7.9.1, the maximum historical earthquake could have been as much as a 5.6 event, and it would be more appropriate if the predicted maximum m_b was chosen as 6.0 rather than 5.8.

3. We disagree with the portion of the section which utilizes the Wisconsin Arch and Wisconsin Arch-Michigan Basin seismogenic zones proposed in Section 7.9.1.

4. Section 7.2.4 is reviewed in Section 3 of this report.

2.7.1.4 Review of Section 7.9.1, Seismic Ground Motion Hazard at Zion Nuclear Power Plant

1. Evaluation of Overall Methodology

The overall methodology used in considering the seismic hazard at the Zion Nuclear Power Plant is an appropriate approach. There is disagreement, however, with that portion of the study dealing the seismogenic zones, the maximum historical earthquake, and the rate of activity.

In particular, it does not appear that the proposed Wisconsin Arch or Wisconsin Arch-Michigan Basis seismogenic zones can be justified on the basis of the known seismicity on the deep-seated geological structures. It is also felt that the maximum historical earthquake in the area could have been a 5-1/2 m_{bLg} magnitude event, and that the epicenter of that event was appreciably closer to the power plant site than that which was apparently used in the study.

2. Seismogenic Zones

We disagree with this section of the report because of the suggested "Wisconsin Arch" and "Wisconsin Arch-Michigan Basin" seismogenic zones. Seismogenic zones in the central United States should be based on the observed distribution of seismicity or the deep-seated structures involving the crystalline basement within which most of the earthquakes in the central United States occur.

Figure 2.7.1-1 is a plot of the seismicity listed by Nuttli (1979) for the area bounded by the latitudes of 41° and 45°N and the longitudes of 84° and 92°W. The earthquakes in the figure are plotted to scale in accordance to their epicentral intensities. Two earthquakes not shown in the figure, but which appear in Nuttli's (Reference 2-24) catalogue, are the events of February 9, 1899, and May 19, 1906. The first event is listed in Barstow, et all, (Reference 2-19) as not being an earthquake, while the second event was determined to be 800 kegs of blasting powder exploding at Pleasant Prairie, Wisconsin.

Another difference between the seismicity plotted in Figure 2.7.1-1 and that listed by Nuttli is the epicentral location of the May 26, 1909, event. After reviewing the distribution of the earthquake effects, it is felt that the epicenter near Aurora, Illinois, as suggested by Docekal (Reference 2-21) is more appropriate than 42.5°N/89.0°W used by Nuttli (1979) and Coffman and von Hake (Reference 2-20). The results of the review of this event are discussed below.

Illustrated along with the seismicity in Figure 2.7.1-1 is the outline of the proposed Wisconsin Arch seismogenic zone. The proposed zone does a poor job of accounting for the known seismicity in the area and does not seem to be justified.

Figure 2.7.1-2 illustrates the same seismicity shown in Figure 2.7.1-1, but with an outline of the proposed Wisconsin Arch-Michigan Basin seismogenic zone. We have two problems with this zone. Firstly, as with the Wisconsin Arch zone, the proposed Wisconsin Arch-Michigan Basin zone does not correlate well with the known seismicity. And secondly, based on the gravity data of the region about

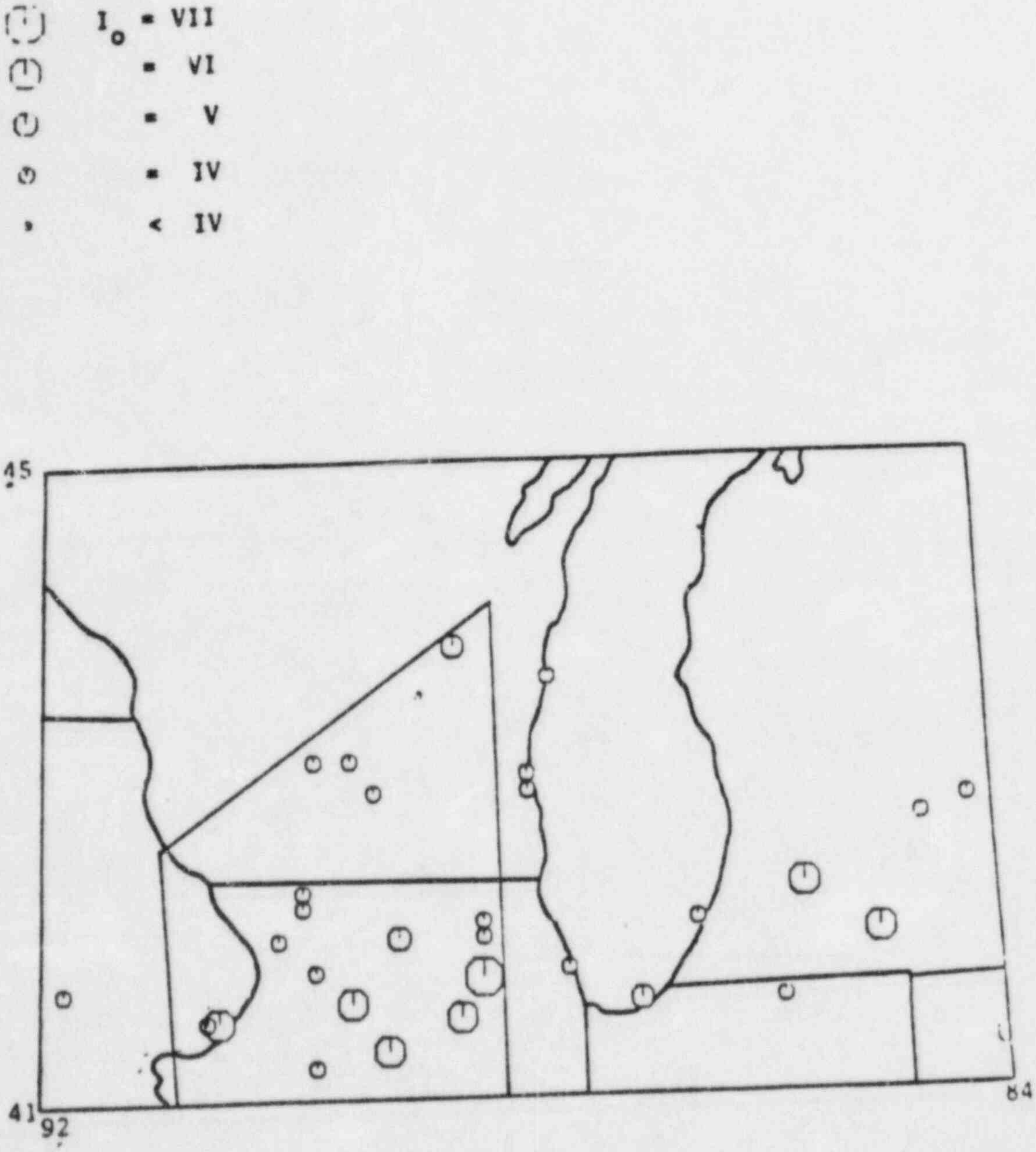


Figure 2.7.1-1. Seismicity Based on Nuttli (Reference 2-24)

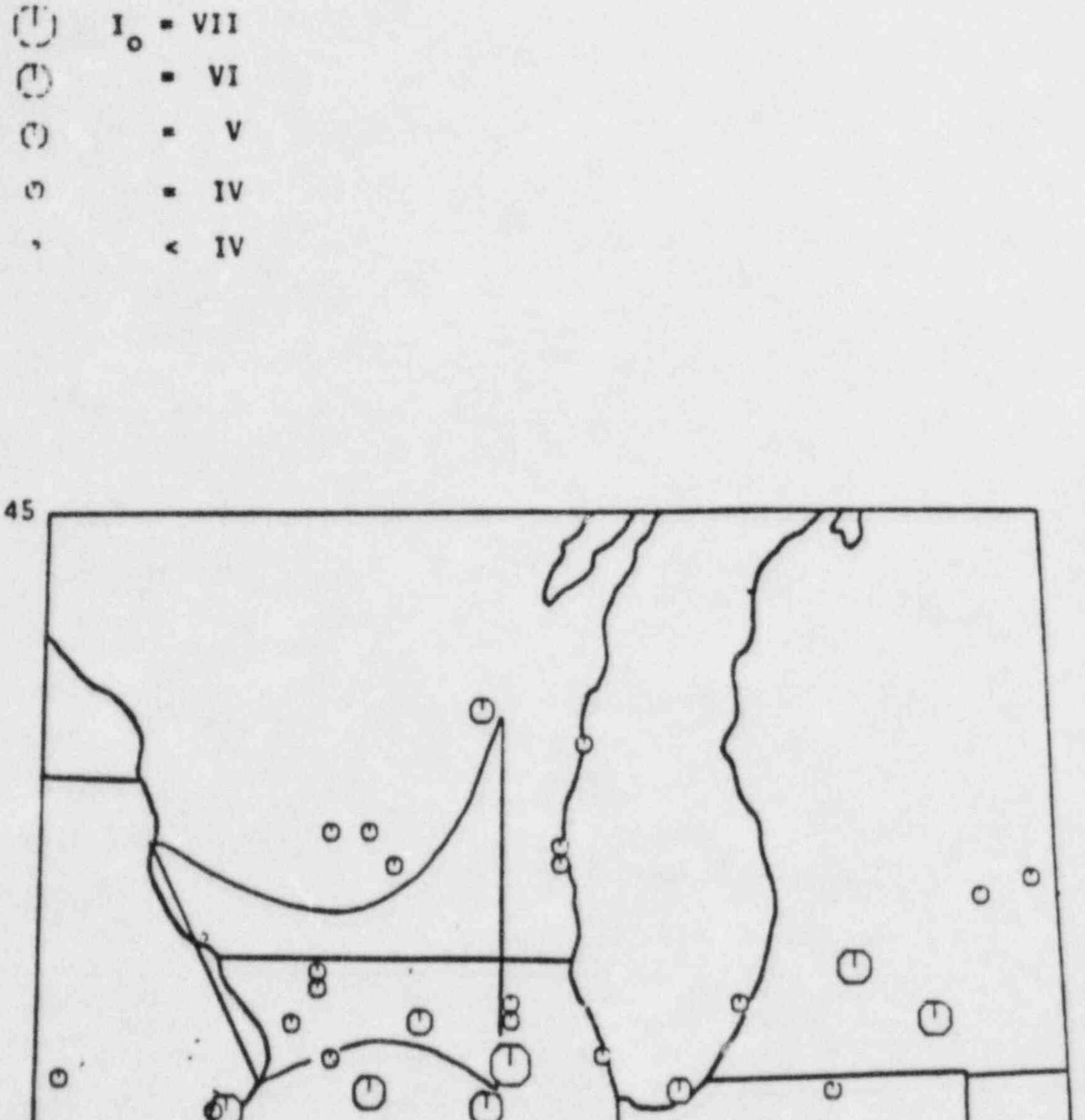


Figure 2.7.1-2. Wisconsin Arch-Michigan Basin

northeastern Illinois and southern Michigan, the proposed seismogenic zone cuts across a major basement structure and suggests that such a zone is unlikely.

As an alternative to the proposed Wisconsin Arch and Wisconsin Arch-Michigan Basin seismogenic zones, it is suggested that a zone more on the order of the one outlined in Figure 2.7.1-3 be used. This zone is similar to the Northern Illinois zone proposed by Nuttli and Herrmann (Reference 2-23), but unlike their zone, the outlined zone in the figure has been extended northwards to include the seismic activity that seems to be spatially associated with the southern border of the Wisconsin Dome. In addition, the outlined zone correlates somewhat with the proposed Wisconsin Arch zone except that by considering a larger area--particularly in the southerly direction--all of the significant seismic activity has been indicated.

3. Seismic Parameters

a. Seismic Activity Rate

The rate of the seismic activity depends upon the choice of the boundaries of the seismogenic zones. And since we disagree with the seismogenic zones proposed in the previous section of the study, we also arrive at a different cumulative magnitude-recurrence curve.

Using the earthquakes in Table 2.7.1-1, which represent the seismic activity that occurred within the seismic zone shown in Figure 2.7.1-3 during the 95-year period of 1880-1975, and the method of plotting the observed cumulative rates of activity at the lower end of 0.5 unit magnitude intervals for those events of $m_{bLg} \geq 4.0$, we get a cumulative magnitude-recurrence curve very similar to that determined by Nuttli and Herrmann for their proposed Northern Illinois source area. The open circles superimposed on Figure 2.7.1.4 (taken from Nuttli and Herrmann, 1978), indicate the data points that were determined.

b. Maximum Magnitude

In this part of Section 7.9.1 it is stated that the maximum historical earthquake to have occurred in the area had an estimated m_b magnitude of 5.3. The event apparently being referred to is the May 26, 1909, earthquake that Nuttli (Reference 2-24) lists as a 5.3 event.

Figure 2.7.1-5 illustrates the distribution intensity data for the May 26, 1909, event based on a review of the newspaper articles for this event. Superimposed on the figure is the interpretation of where the various isoseisms

TABLE 2.7.1-1

Date Day-Mo.	Year	Magnitude m_{bLg}
27-05	1881	4.7
28-11	1907	3.8
28-11	1908	3.8
26-05	1909	5.3
22-10	1909	4.0
02-01	1912	4.7
25-09	1912	3.6
17-10	1913	3.6
07-10	1914	3.8
31-05	1916	3.0
22-02	1918	3.8
07-07	1922	4.2
03-03	1925	3.2
23-01	1928	3.8
10-06	1931	4.2
18-10	1931	3.4
07-12	1933	4.2
12-11	1934	4.7
05-01	1935	4.2
05-01	1935	3.4
12-02	1938	4.2
08-11	1938	3.0
08-11	1938	3.0
08-11	1938	3.0
24-11	1939	3.2
01-03	1942	4.0
16-03	1944	3.4
16-03	1947	3.6
06-05	1947	4.0
15-01	1948	3.9
20-04	1948	3.8
08-01	1957	3.6
15-09	1972	4.4

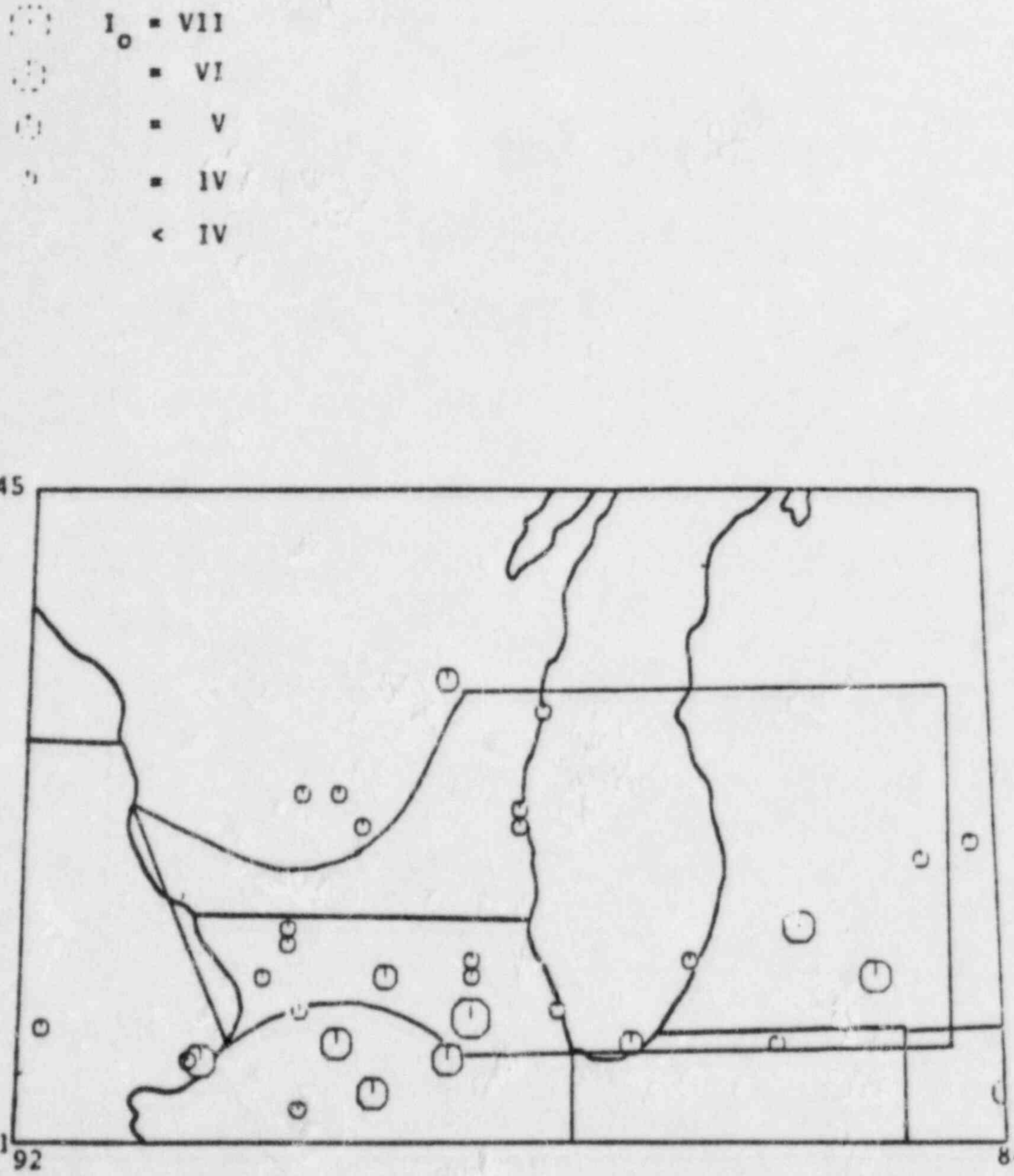


Figure 2.7.1-3. Proposed Seismogenic Zones

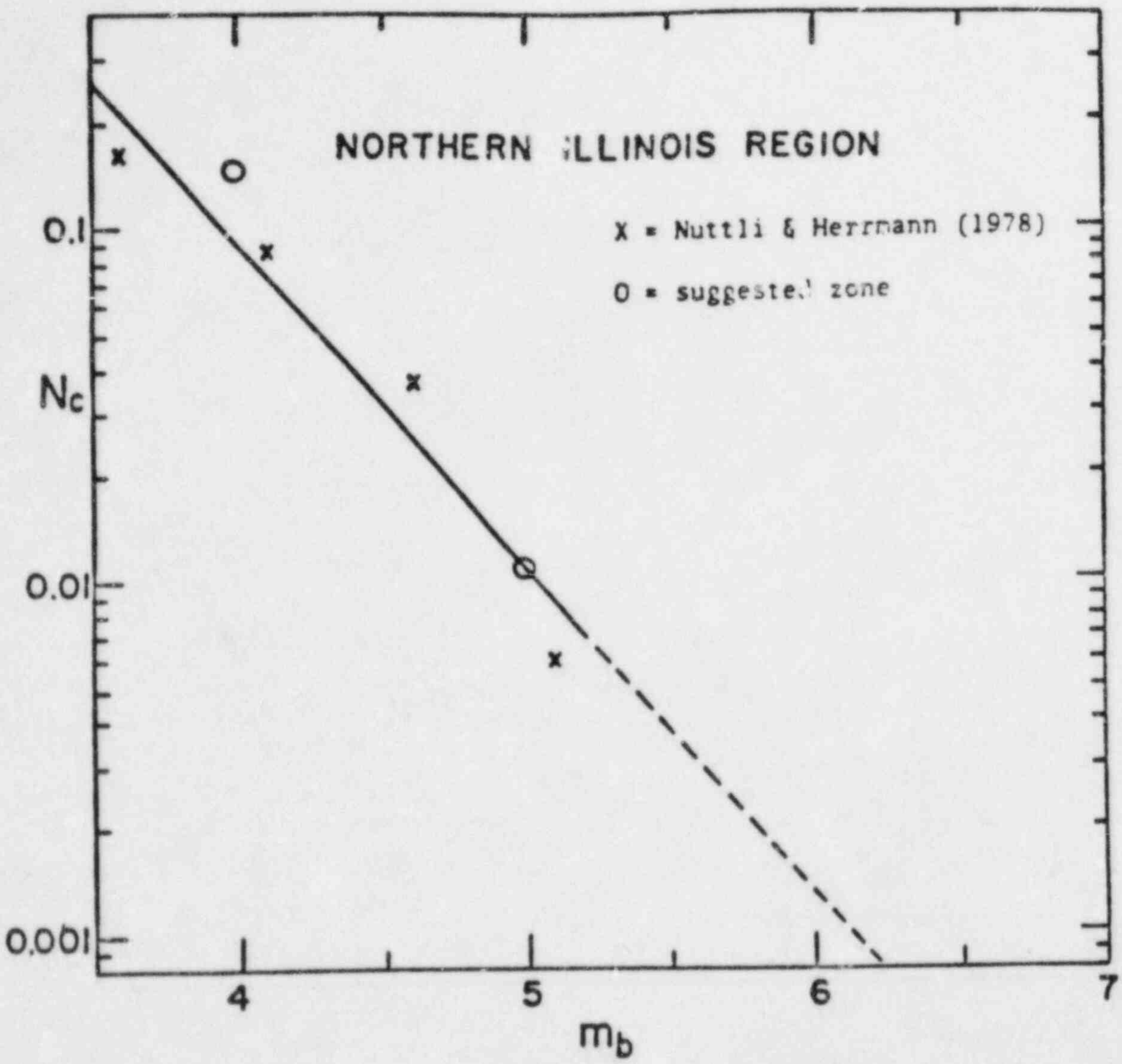


Figure 7. Cumulative magnitude-recurrence curve for the Northern Illinois region

(taken from Nuttli and Herrmann, 1978)

Figure 2.7.1-4

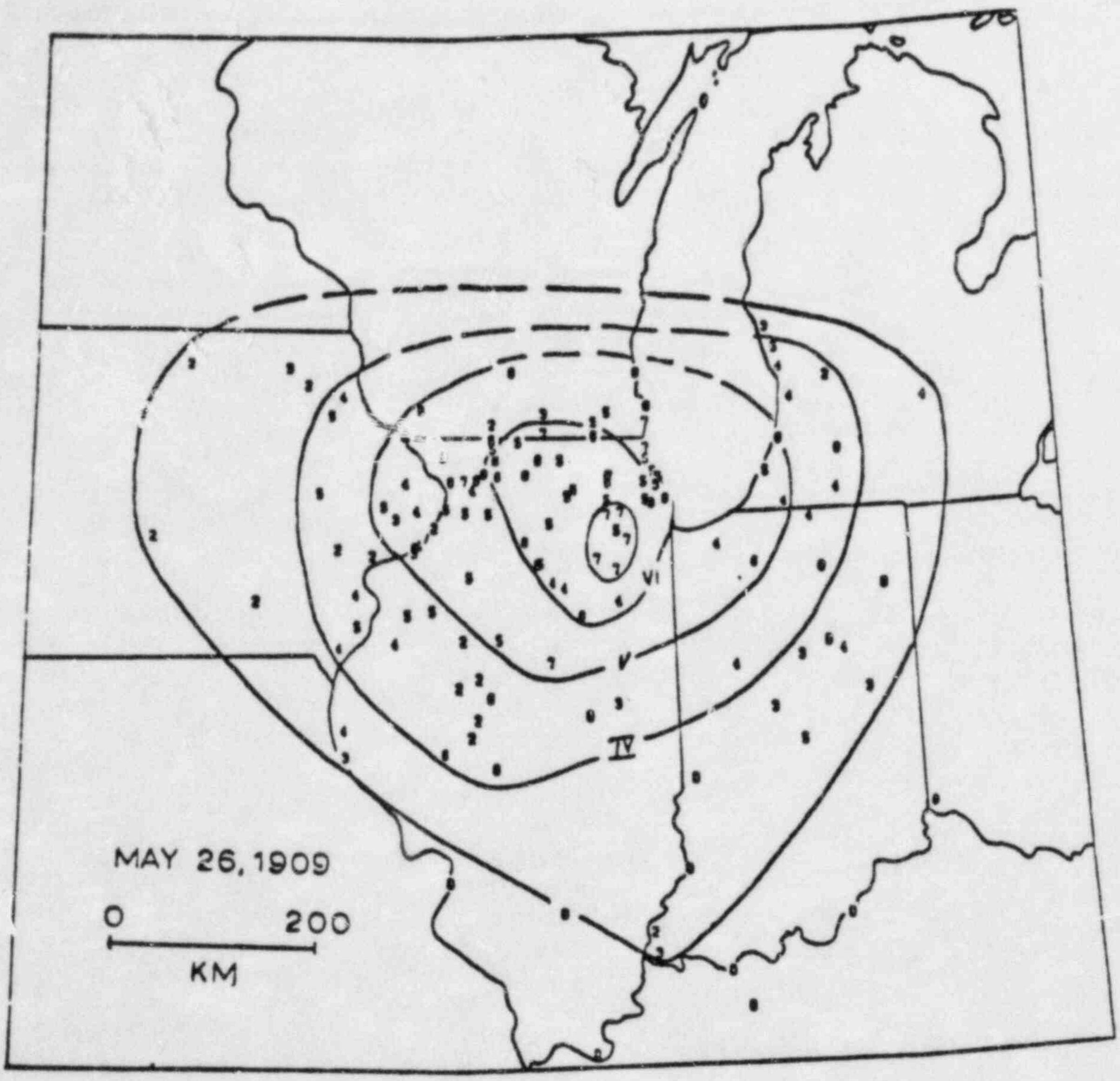


Figure 2.7.1-5

should be drawn. The isoseisms, along the northern portion of the map, are dashed to indicate the uncertainty resulting from the lack of information. Note that the greatest level of concentration occurred in and about the Aurora, Illinois area and it is for this reason that Docekal's (Reference 2-21) epicentral coordinates were chosen rather than those of Nuttli (Reference 2-24) and Coffman and von Hake (Reference 2-20).

Given the distribution of the MM intensity data for an event, there are a number of empirical techniques that have been developed for the purpose of estimating m_b magnitudes for earthquakes in eastern North America. Using Nuttli's (Reference 2-22) fall-off-of-intensity with distance technique, the m_b magnitude of the event is estimated to be 5.6. Using the area within the intensity IV isoseism (184,000 km^2) and the results of Nuttli, et all, (Reference 2-25), the m_b magnitude of the event is determined to be 5.5 ± 0.23 and using the 800,000 km^2 felt area listed by Nuttli (Reference 2-24) and the results of Street and Lacroix (Reference 2-26), the m_b magnitude of the event is determined to be 5.4 ± 0.30. On the other hand, it is estimated that the felt area to have been more on the order of the 445,000 km^2 given by Docekal (1970), which by Street and Lacroix is equivalent to a m_b of 5.1 ± 0.30.

In summary, we do not disagree with the 5.3 maximum historical earthquake, but there is a distinct possibility the m_b magnitude of the May 26, 1909, was as large as 5-1/2. And as a consequence, it is suggested that the best estimate of $m_{b,max}$ should be raised from 5.8 to 6.0

4. Estimation of Seismic Ground Motion

The approach used in this section of the study to estimate peak acceleration as a function of earthquake magnitude and distance seems to be appropriate. The conclusions in this section, however, are dependent on the acceptance of Parts 1 and 2 of Section 2.7.1.4.

5. Additional Comments

a. Page 3, Seismic Hazard Model, Item 3, "...local soil conditions." Local soil conditions at the Zion site are not explicitly accounted for as has been common in nuclear plant seismic design. Recently however, at a number of nuclear plant sites, successful attempts have been made to identify and isolate the systematic amplification effect which local soil has on incoming seismic waves.

b. The assignment of uncertainty to the attenuation laws ($\sigma_{\ell n a} = 0.6$) is reasonable. Alternate assumptions could have been tested (with appropriate weights attached), but this would not be expected to have much impact on the final results. The same may be said about the choice of the lower limit on magnitude ($m_b = 4$).

c. The comment (on Page 6, Item 2, line 4), "...even if peak accelerations are high..." is revealing. It implies recognition that accelerations are indeed highly variable. Many seismologists and earthquake engineers would say that this is equally true at high as at low values of m_b (or Mercalli Intensity), and that any rigid upper bound on peak acceleration is unrealistic.

d. Uncertainty about "b-value" (on Page 7): The three-valued discretization (mean and mean $\pm$ one standard deviation) appears inadequate as it obviously does not cover the tails of the distribution.

e. Discretization of $m_{b,max}$ (on Page 7 and 8): The double-triangular distribution has an upper bound of 6.2: it is then converted into a three-valued probability mass function whose largest value is $m_{b,max} = 6$. The resulting error in seismic risk calculations may not be negligible (in the low probability range) if the rigid bound on effective acceleration were to be relaxed.

f. It is stated on Page 8 that, "It was felt by the seismological consultant that there is some negative correlation between b-values and values of $m_{b,max}$." This is the apparent justification for assuming complete probabilistic dependence between b and $m_{b,max}$. It would be interesting to see some results based on the assumption that b and $m_{b,max}$ vary independently. Also, it might have been preferable to quantify the seismological consultant's judgment in terms of a (discretized) joint probability distribution implying partial correlation.

g. Consideration of alternative attentuation laws (Equations 5 and 6) is adequate.

h. Nuttli's data in Figure 4 indicate that the 1.37 value for the ratio of sustained to peak acceleration applies to the magnitude range $m_b \geq 6.0$. The 1.37 value is in fact adopted for all magnitudes. Note, however, that the upper magnitude bound adopted in the study equals $m_{b,max} = 6.0$ (with probability 0.28), while the m_b magnitude follows a truncated exponential distribution; it follows that the condition $m_b \geq 6.0$ (to which the 1.37 value corresponds) is the fact assigned zero probability of occurrence. The 1.37 value is therefore subject to question. The 0.9 factor mentioned on Page 10 of Section 7.9.1 (leading to the factor $1.37 \times 0.9 = 1.233 \approx 1.25$ in Section 7.9.3) is acceptable.

i. The influence of the choice of a_{max} is understated, for example on Page 13 in Section 7.9.1: "In general, the variation in hazard resulting from the use of alternate estimates of peak acceleration is within the variation resulting from different hypotheses on seismogenic zones." It is quite obvious from Table 3 in Section 7.9.1 that calculated probabilities are more sensitive to a_{max} than to zonation in the critical "high acceleration-low probability" range of the seismicity curves. It is this range of the curves which most influences the calculated risk of earthquake-induced core damage.

2.7.1.5 Review of Section 7.9.2, Conditional Probabilities of Seismic-Inducted Failures for Structures and Components for the Zion Nuclear Generating Station

Due to the large number of comments, details regarding the review of this section are given in Appendix D. Listed below are the major areas of concern:

1. The definition of damage effective ground acceleration used in the analysis may not be appropriate for electrical and mechanical equipment, buried pipe, and equipment which depends on functional operation as opposed to ductile strength capacity.

2. Design and construction errors and aging should be considered.

3. The possibility of a LOCA followed by an aftershock, or the occurrence of a moderate earthquake, when some safety-related equipment is unavailable, should be considered.

4. The effects of variability in SDOF (single degree of freedom) models for MDOF (multi-degree of freedom) structures for determining the contribution of inelastic behavior should be included in the analysis.

5. The basis for the variability split into randomness and uncertainty components should be documented for critical structures and components.

6. Piping systems and cable trays may have less capacity because of the numerous series components present and the potential lack of dependence.

7. The fragility curves for the batteries and racks should be recalculated based on more detailed information for these components.

8. The coarseness of the data points for the hazard and fragility curves may affect the accuracy of the tails of the probability density function for frequency of core melt, although we do not feel that the tails are particularly meaningful, except in a qualitative sense.

9. Information for the basis of the service water pump capacity should be documented since this is the most critical component.

10. The decision to eliminate the electrical components from further consideration should be reevaluated in light of the comments made in this report.

11. The development of the damage effective ground acceleration value in Section 7.9.3 should consider the effect of a best estimate site-specific ground response spectrum relative to the broad-banded spectrum used in the analysis.

12. Documentation of the bases for assumptions should be provided.

13. A sensitivity section in the ZPSS report should be included to inform the reader concerning the effect of changes in values of significant parameters on the frequency of core-melt analysis.

14. When the SSMRP study is completed for Zion, a comparison of the two approaches should be conducted as a check.

2.7.1.6 Review of Section 7.9.3, Comments on Effective Ground Acceleration Estimates

1. Definition of Effective Peak Acceleration (EPA)

a. The damage of systems in the frequency range from 2 to 10 Hz is best correlated with spectral ordinates obtained by anchoring the response spectrum of strong, broad-band motions to an "effective peak acceleration" EPA, defined as (Equation 2 in Section 7.9.3):

$$EPA = \frac{1.25}{F} A_{3F} \qquad (1)$$

in which A_{3F} is the acceleration at the top of Page 3 of Section 7.9.3 and F is a quantity that depends on magnitude and distance.

b. A_{3F} in Equation (1) can be satisfactorily replaced with sustained peak acceleration (SPA), as defined by Nuttli (third highest acceleration peak). Similarity of the factor 1.25 in Equation (1) with the factor 1.23 in Equation (3) of Section 7.9.1 makes one believe that this replacement is in terms of SPA for a generic horizontal direction, not for the worst of two orthogonal directions.

c. For the Zion site, a conservative single-value estimate of F is taken to be F = 1.25. Therefore, a conservative definition of EPA for the Zion site is (Equation 4 of Section 7.9.3)

$$EPA = SPA \qquad (2)$$

d. Seismic hazard is calculated in terms of EPA = SPA using Nuttli's median attentuation function in Equation (2) of Section 7.9.1.

e. Fragility curves in terms of EPA are obtained by assuming three to five cycles of linear response near the value of the response spectrum for long, broad-band earthquakes with peak acceleration EPA (i. e., with the spectrum anchored to the EPA). See first paragraph of Section 7.9.3.

f. If both hazard and fragility curves are in terms of Nuttli's SPA, then we see no reason why one should relate SPA to instrumental peak acceleration (IPA); i.e., the comments in the last part of Page 9 in Section 7.9.1, starting from, "To estimate peak acceleration," are irrelevant.

g. If, as stated at the top of Page 10 in Section 7.9.1, Equation (2) in that section refers to the larger SPA for the two horizontal components of motion, then Equation (2) should be corrected by multiplying the right-hand sides by 0.9. We find no evidence that this was actually done.

h. The comment in the second paragraph of Page 14 in Section 7.9.1, that the smaller damage potential of low-magnitude short-duration events is accounted for by limiting peak acceleration, is in contrast with our understanding from Section 7.9.3 that the difference in earthquake damage potential is the reason for replacing IPA with EPA, not for constraining the values of acceleration.

i. We agree with searching for quantities such as EPA that correlate with structural damage better than IPA.

However, one must then face the problem of having to work with two different earthquake intensity measures, one for acceleration-sensitive equipment, the other for EPA-sensitive structures. If two such categories of components exist (and we believe they do), then seismic hazard should be defined jointly in terms of IPA and EPA. This should be commented on.

j. The report should show how seismic fragilities have indeed been obtained in terms of EPA and not, for example, in terms of IPA. How was the fact that three-to-five peaks occur near the maximum response value taken into consideration?

2. Upper Bound EPA

a. We cannot follow the argument that imposes limits to the EPA based on limits on I_{mm}. That argument is especially tenuous if EPA is defined as EPA = A_{3F} (Equation 4 of Section 7.9.3). In this case, the statement at the top of Page 12 in Section 7.9.1 does not hold. We also find it objectionable to use damage to masonry construction in order to obtain limits on EPA for nonmasonry structures.

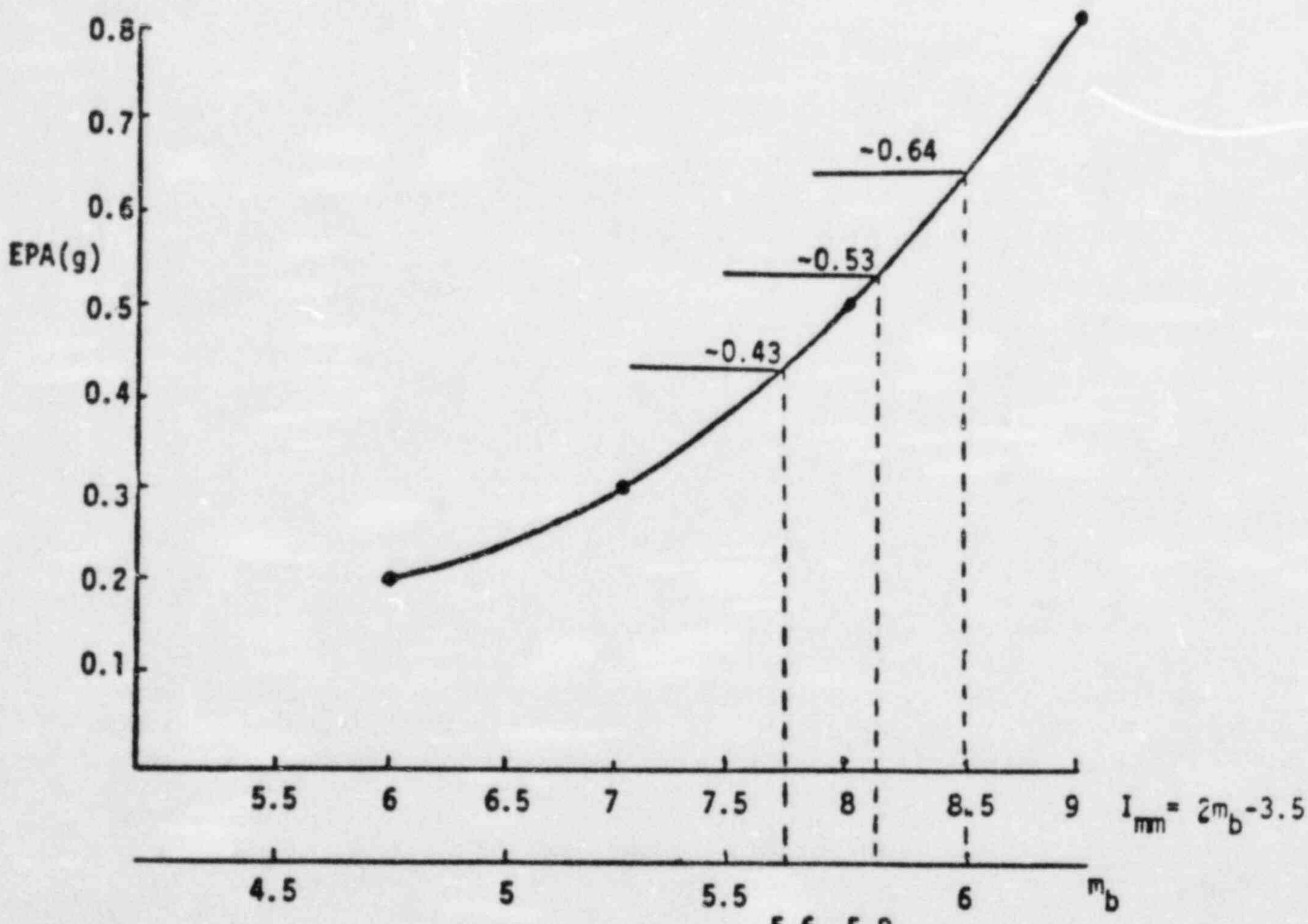
b. If an upper bound to EPA exists, such bound should be included through truncation of the attenuation error distribution, not through correction of the final hazard curves.

c. There is no evidence in the report that the curves of Figure II.7.1 have actually been calculated.

d. We have tried to reconstruct the method used to obtain the upper bounds on EPA in Figure II.7.1. It seems that upper bound intensity values have been found from upper bound magnitudes using the relationship $I_{mm} = 2m_b - 3.5$ and that the associated upper bounds EPA have been calculated by interpolation of the values on Page 6 of Section 7.9.3. (See Figure 2.7.1-6.) However, the above relationship between I_{mm} and m_b was obtained by fitting dispersed data and does not apply to upper bounds.

e. Large uncertainty exists on the maximum value of EPA. It would therefore be appropriate to consider alternative values of this parameter through different truncations of the attenuation error distribution.

f. In any case, the presence of these acceleration correction factors and imprecise bounds point to the urgent need to implement improved earthquake ground motion



Upper bound EPA values in the Zion Probabilistic Safety Study (Fig. II.7.1)

Figure 2.7.1-6

descriptions which explicitly account for duration (in addition to a measure of intensity such as peak acceleration) and to apply analysis procedures which predict seismic response measures more directly correlated with performance and damage. Much of this is within the state-of-knowledge of earthquake engineering.

3. Conclusions

a. Implementation of the methodology appears inappropriate in two major aspects:

1. The EPA upper bound should be incorporated in the seismic hazard analysis through truncation of the attenuation error distribution. This seems to be an important parameter and should be subjected to sensitivity analysis.

2. Calculation of the fragility curves for core melt should be documented and curve grouping should be more detailed, especially in the range of low EPA values. Because no calculation detail is given in the Zion report, it is difficult to anticipate the effect of a more accurate work. However, we believe that this may lead to higher risk values.

2.2.1.7 Review of Section 8.8.1, Seismic Risk

We feel that the frequency of acceleration values at different probability levels given in Table 8.8-1 are at too coarse a spacing to give stable frequency of core-melt values in the tails of the probability density function for core melt. (See ZPSS Figure 7.2.5.)

Figures 2.7.1-7 to 2.7.1-9 show plots of these values (with obvious corrections to typographical errors in Table 8.8-1) as compared to the data obtained from Section 7.9.1 which have been shifted by a factor of 1.23 and truncated for maximum acceleration values. We expected the corresponding pairs of curves to coincide. The differences are about 30 percent. The effect on the mean frequency of core melt would be small. However, the effect on the tails of the probability density function would be much larger.

2.7.2 Fire

This topic is addressed as a special issue. Section 4.6.

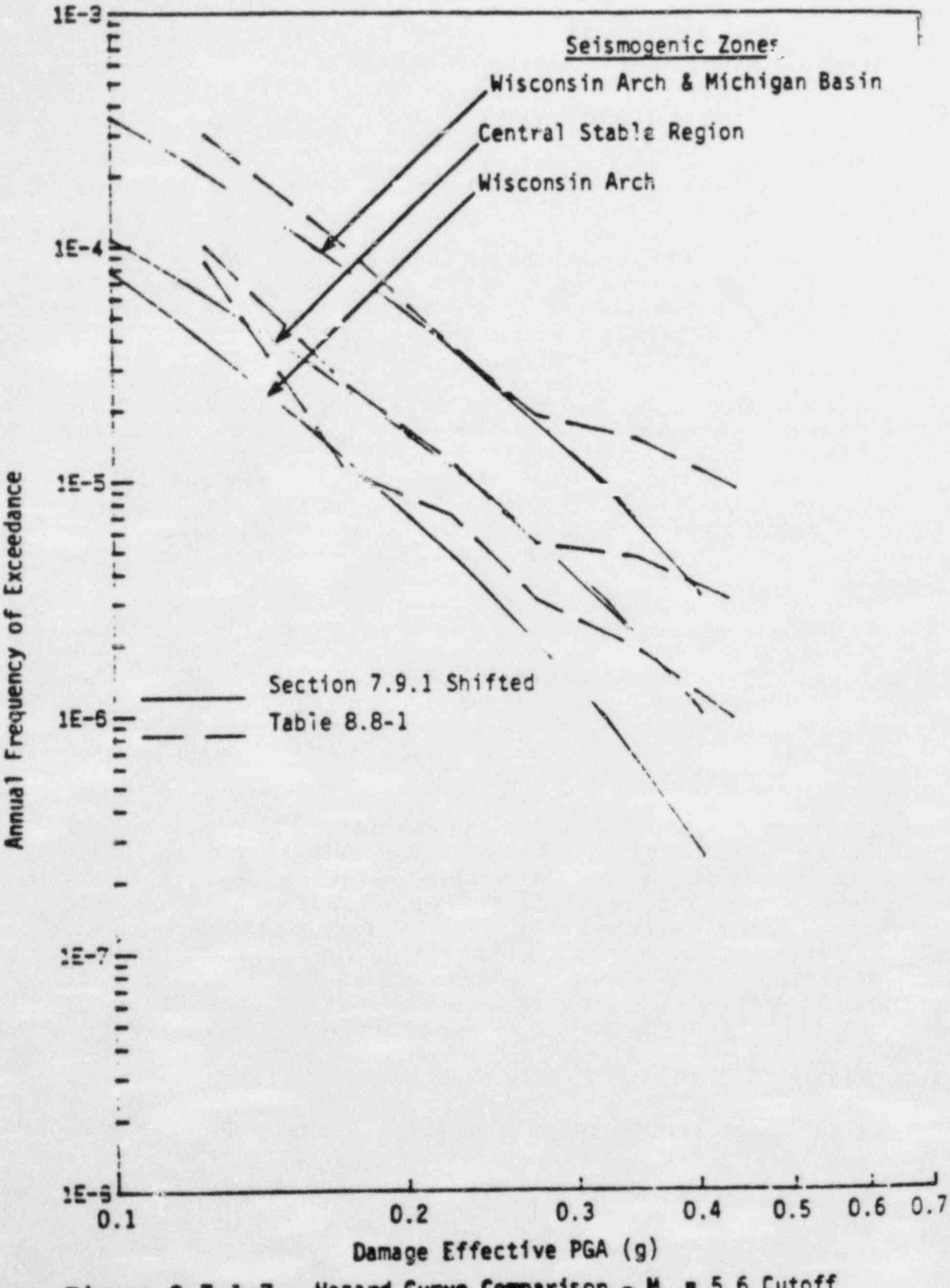


Figure 2.7.1-7 Hazard Curve Comparison - M_b = 5.6 Cutoff

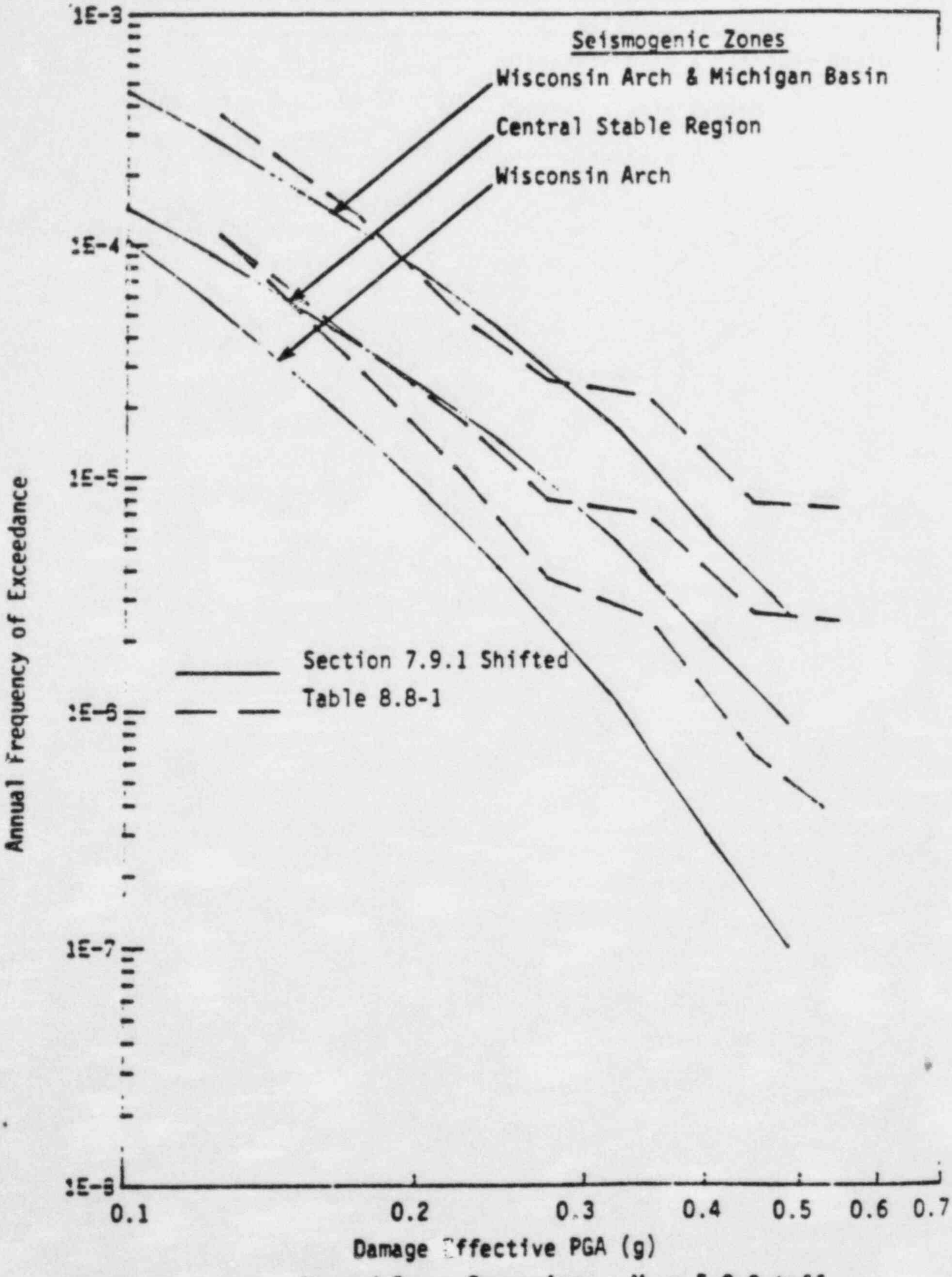


Figure 2.7.1-8 Hazard Curve Comparison - M_b = 5.8 Cutoff

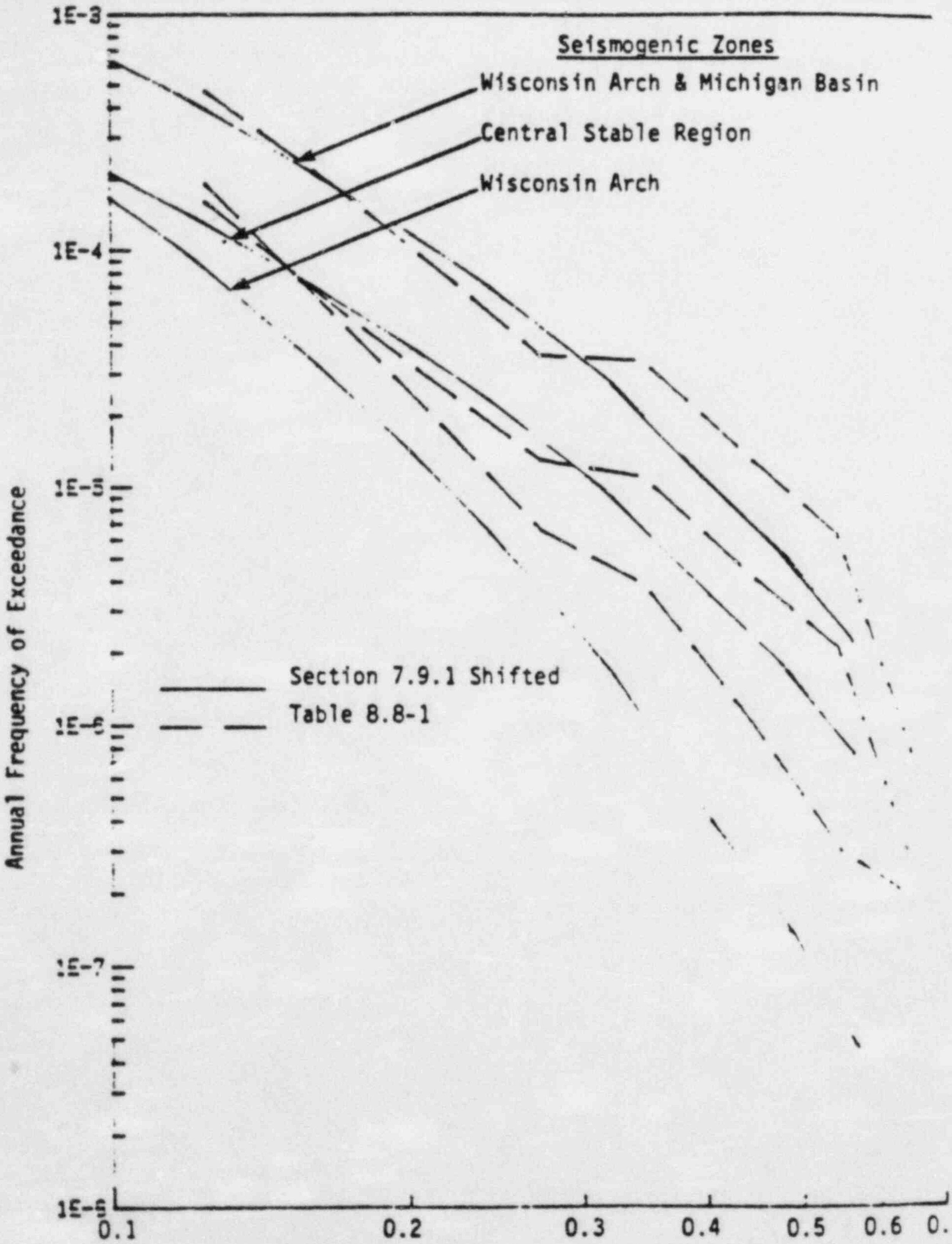


Figure 2.7.1-9 Hazard Curve Comparison - M_b = 6.0 Cutoff

2.7.3 Flooding*

2.7.3.1 Review of ZPSS Section 7.4.1, External Flood Sources

It was concluded in the Zion PRA report that the contribution to the frequency of core melt due to external flood sources is insignificant. This conclusion was based on a deterministic analysis of assessing flood levels. The basis for the conclusion, including a description of the methodology and pertinent data, was not provided in the report. We feel that the approach presented is not appropriate nor adequate for purposes of a probabilistic risk assessment. We discuss below pertinent issues that should be addressed. Note, at this time, no comments can be made concerning the conclusion presented, only apparent limitations in the basis on which it was made.

In order to assess the possible contribution of external flooding to the core-melt frequency, a probabilistic analysis of the occurrence of flooding should be performed. A site-specific flood analysis should consider the following causes:

- River flooding
- Upstream dam failure (includes all secondary causes such as earthquakes, overtopping, antecedent dam failures)
- Failure of dikes and levees
- Tsunamis
- Surges
- Seiches
- Wind Waves
- Precipitation (including hurricanes and sequences of storms)
- Snow melt

The possible occurrence of flooding at the Zion site may result from the occurrence of a combination of events such as precipitation, wind wave action, antecedent conditions,

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*This review was conducted by Jack R. Benjamin & Associates, Inc., under contract to Sandia National Laboratories.

and river flooding. A probabilistic model should allow for such combinations, particularly in evaluating the frequency of extreme events. We suggest that potential combinations of events be explicitly identified and reviewed. Those events considered likely to occur should be incorporated in a probabilistic model.

We feel that the deterministic methodology that has been employed is inappropriate because the uncertainties in the flood assessment have not been considered. The large uncertainties in evaluating flood frequencies and structure and component fragilities may be important at the Zion site for determining the frequency of core melt due to external flooding.

In addition to evaluating the likelihood of occurrence of events and event combinations, consideration should also be given to possible dependencies that may exist. Examples include time dependence of meteorological events and spatial dependencies of flooding due to different sources (Reference 2-27).

Because of the conclusion reached concerning the occurrence of flooding, no consideration has been given to the likelihood of leakage into the plant and the development of structure and equipment fragilities. In the event that a probabilistic analysis warrants consideration of flooding at the site, fragility curves should be developed for the pertinent structures and safety-related equipment.

2.7.3.2 Review of 7.4.2, Internal Flood Sources

This section was reviewed with respect to the adequacy of the analysis as presented in the ZPSS. During the Indian Point study review, a summary was provided of the procedure used to identify sources of internal flooding and to determine their effect. Three steps were followed:

1. Identify sources of flooding.

2. Identify locations vulnerable to floods from those sources determined in 1.

3. Simulate initiating events and evaluate the impact.

We generally agree that the above steps are required to conduct an internal flood analysis. We suggest that the internal flood analysis should be conducted in a more systematic manner, possibly including the development of flood analysis fault trees. This would ensure that a thorough, systematic analysis of critical events and event sequences are considered. We suspect that existing fault trees have been used to some degree in the analysis.

1. 7.4.2.1 Auxiliary Building

For the first potential internal flood source, tanks in the Auxiliary Building are identified. The statement is made that these tanks are not pressure vessels, and therefore catastrophic failure is an extremely unlikely event. The basis for this statement is not known. Releasing the contents of the tanks may lead to adverse consequences.

For a rupture in the RHR system piping during cooldown and heatup conditions, a LOCA will occur. However, this event is not treated as a core-melt initiator because of the leak detection and isolation provisions which exist at the plant. These provisions are apparently described in the Zion FSAR (Chapter 6) which was not available for review. No basis is provided in the Zion PRA report to support an assumption of 100 percent reliability of the detection and isolation system, thus allowing this event to be ignored.

For the fourth flood source, a large leak or rupture in the component cooling water (CCW) piping, a failure rate of 1.8×10^{-4} per year is assumed. The basis of the 10^{-3} to 10^{-4} probability of leak going unnoticed and of the 0.01 chance of a failure resulting in a rupture is not given. An assumption is also made here that alarms and instrumentation are reliable. The basis for this assumption is not supported.

We express similar concerns about the basis for the frequency of service water pipe failure, namely the basis for determining the rate of failure of 1.5×10^{-8} per year.

The flood drains and stairwells are assumed to have a capacity to discharge flood waters to elevation 542'. The basis for assuming that these channels of flow are perfectly reliable is not given.

The uncertainty in the frequency estimates given in this subsection should be documented.

2. 7.4.2.2 Containment Building

It is not clear whether all possible failure mechanisms have been evaluated. What does the frequency of failure of the service water pipes correspond to? For example, is it the probability of failure?

3. 7.4.2.3 Turbine Building

Our comments for subsection 7.4.2.1 regarding the reliability of floor drains and stairwells and the capacity to discharge flood waters also apply here.

2.7.4 Tornadoes, Tornado Missiles, Aircraft Accidents and Turbine Missiles.

The following sections were reviewed: II.7.4, 7.5, (Tornadoes and Tornado Missiles); II.7.5, 7.6, (Aircraft Accidents); and II.7.7, 7.8 (Turbine Missiles).

The most notable feature of the subject sections is their reliance on older references and the absence of reference to recent work. The next most noticeable feature is the manner in which most conclusions are stated. There is a vagueness in the stating of assumptions.

In Section II.7.4 (which summarizes Section 7.5) the latest reference is from 1978, but neither of two programs sponsored by EPRI is mentioned. In Section 7.6 (summarized by II.7.5) three items seem worthy of note: No reference is made to military aircraft, though the Table 7.6-1 listing airports and aircraft does include NAS Glenview, and the aircraft in the table are mostly obsolete, i.e., 720, DC-3, Beach 18, DC-6 or 7. The use of the air carrier accident rate for business jet aircraft is stated, but not justified. It does not appear that any appreciable contribution to risk is available in this area and, if all three items were considered in detail, no change in conclusions would be expected.

In Section 7.8 (summarized by II.7.7) the only reference is out of date. Another article entitled "A Reassessment of Turbine Generator Failure Probability" has been published by the same author in 1978. Since that time much work has been done. The conclusion is likely correct, but the presentation does not instill confidence. The statement regarding further pursuit of the problem seems out of context in a one-time study. The assessment of risk associated with turbine-generated missiles does not discuss the targets struck by such missiles. Early work concentrated on the containment structure with less emphasis on control rooms, cable rooms, etc. It is not clear whether this analysis considered all these items. If it did not, then the results could change significantly.

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3. Accident Sequence Analysis

3.1 Introduction

In this section, we discuss the dominant accident sequences. These include the sequences which, by our estimates, dominate core-melt frequency or plant damage state frequency. We identified 14 such sequences (Figure 3.1-1). Of these, five are on the list of dominant accident sequences (ZPSS Table 8.10-1) presented in the ZPSS. The remaining nine are sequences which did not appear on the ZPSS list either because they were not interpreted as leading to core melt or because the frequency calculated in the ZPSS was not high enough to consider them dominant. The plant damage state used in the tables is: S or A denote small or large LOCA and T denotes transient, E or L denote early or late core melt, F and C denote fans and spray working respectively. For each of those sequences not addressed in the ZPSS, we describe the sequence and the frequency calculations. For sequences which appeared in the ZPSS, we review the analysis.

First we compare the ZPSS point estimates, their posterior means, to alternative estimates based, where possible, on the reported Zion data or on alternative data sources or assumptions. For the most part, our alternative estimates are obtained by modifying a few terms in the ZPSS models--eg, β-factors and human error probabilities--so the resulting point estimates are a mixture of ZPSS Bayesian results and our own less formal (but not necessarily less realistic) point estimates. We regard our results as "working values"-- reasonable estimates to be used in subsequent calculations. Because any point estimate, no matter how derived, can convey an unwarranted aura of precision, we have also carried out a statistical uncertainity analysis for internal event accident sequences. The limited data available to us was not sufficient to do a similar statistical analysis for external events.

To the extent possible, we have identified data pertaining to the parameters of interest, then combined them to obtain statistical confidence limits. The methodology used is that given in a report by Maximus[3-1] extended to include the estimation of failure rates as well as failure probabilities. This methodology consists of a collection of reduction rules whereby the data pertaining to "components" are reduced to effective "system" data by ways that account for the series parallel structure of the system and for the possible repeated use of the same component data. In our analysis we have generally simplified the model so that only the dominant cut sets, in terms of both the estimated occurrence frequency and the imprecision of these estimates, are

TABLE 3.1-1

Revised Zion Dominant Accident Sequences

Rank With Respect to Core Melt	Sequence	Plant State	Annual Frequency
1	CCW Failure (causing failure of all charging and SI pumps, seal LOCA)	SEFC	~2(-4)
2	Loss of off-site power: failure of component cooling water: failure to recover off-site power in 4 hours (recovery in 8 hours)	SEFC	4.6(-5)
3	Loss of off-site power: failure of component cooling water: failure to recover off-site power in 1 hour (recovery in 4 hours)	SEFC	4.0(-5)
4	Loss of off-site power: failure of component cooling water, failure to recover off-site power in 8 hours, failure of containment fans	SEC	1.8(-5)
5*	Small LOCA : failure of recirculation cooling	SLF	1.6(-5)
6	Loss of off-site power: failure of component cooling water, failure to recover off-site power in 8 hours	SEFC	7.9(-6)
7	Failure of DC Bus 112: (causing failure of 1 PORV and loss of AC Bus 149), failure of auxiliary feedwater	TEFC	~7(-6)
8*	Seismic: loss of all AC power	SE	5.6(-6)
9*	Large LOCA: failure of recirculation cooling	ALF	4.9(-6)
10*	Medium LOCA: failure of recirculation cooling	ALF	4.9(-6)
11	Loss of off-site power: failure of component cooling water, failure to recover off-site power in 8 hours, failure of containment sprays and fan coolers	SE	4.7(-6)
12*	Large LOCA: failure of low pressure injection	AEFC	1.4(-6)
13	Loss of off-site power: failure of auxiliary feed-water, failure of feed and bleed, failure to restore off-site power in 4 hours (recovery in 8 hours)	TEFC	1.1(-6)
14	Loss of off-site power: failure of auxiliary feed-water, failure of feed and bleed, failure to restore power in 1 hour (recovery in 4 hours)	TEFC	1.0(-6)
	Interfacing System LOCA**	V	1.1(-7)

*Sequences identified by the ZPSS to be dominant.

**Included here because of its potential impact on consequence analysis, not one of the dominant core-melt sequences.

ZPSS TABLE 8.10-1

Comparison of Core Melt and Release Frequency Contributions --
Impact of Containment and Engineered Safety Systems

Rank With Respect to Core Melt	Sequence	Mean Annual Frequency (Contribution to Core Melt)	Containment Split Fraction to Serious Release	Mean Annual Frequence of Serious Release	Relative Rank With Respect to Serious Release Frequency
1	Small LOCA: failure of recirculation cooling	1.62-5	1-4	1.62-9	4
2	Seismic: loss of all AC power	5.60-6	1-0	5.60-6	1
3	Large LOCA: failure of recirculation cooling	4.89-6	1-4	4.89-10	5
4	Medium LOCA: failure of recirculation	4.89-6	1-4	4.89-10	6
5	Loss of main feedwater: ATWS, failure to control pressure rise (i.e., failure of augmented auxiliary feedwater or primary pressure relief)	3.89-6	1-4	3.89-10	8
6	Turbine trip: ATWS, failure to control pressure rise (i.e., failure of augmented auxiliary feedwater or primary pressure relief)	2.76-6	1-4	2.76-10	7
7	Spurious safety injection: failure to control the SI, recirculation cooling	1.64-6	1-4	1.64-10	9
8	Spurious safety injection: loss of off-site power, loss of ESF Buses 148 and 149	1.43-6	1-4	1.43-10	10
9	Large LOCA: failure of low pressure injection	1.32-6	1-4	1.32-10	11
10	Medium LOCA: failure of low pressure injection	4.36-7	1-4	4.36-11	14

ZPSS TABLE 8.10-1 (Continued)

Rank With Respect to Core Melt	Sequence	Mean Annual Frequency (Contribution to Core Melt)	Containment Split Fraction to Serious Release	Mean Annual Frequence of Serious Release	Relative Rank With Respect to Serious Release Frequency
11	Loss of main feedwater: loss of off-site power, loss of ESF Buses 148 and 149, failure of auxiliary feedwater	2.91-7	2-4	5.82-11	12
12	Reactor trip: loss of off-site power loss of ESF Buses 148 and 149 failure of auxiliary feedwater	2.23-7	2-4	4.46-11	13
13	Turbine trip: loss of off-site power loss of ESF Buses 148 and 149, failure of auxiliary feedwater	2.14-7	2-4	4.28-11	15
14	Turbine trip due to loss of off-site power: loss of all AC power, failure of auxiliary feedwater	2.00-7	1.0	2.00-7	2
15	Loss of main feedwater: failure of auxiliary feedwater, failure of bleed and feed cooling	1.33-7	1-4	1.33-11	16
16	Interfacing system LOCA (RHR inlet valves)	1.05-7	1.0	1.05-7	3

considered. Also, where only subjective estimates are available, we have translated them to effective data. The resulting bounds don't have the status of statistical confidence limits, but they are the best we can do until data can be obtained.

An intermediate step in obtaining statistical confidence limits by the Maximus approach is the calculation of the "maximum likelihood estimate" of the accident sequence rate. These estimates also are plausible working values. Since we are dealing with reduced models, however, and since the maximum likelihood estimate of the probability of an event that hasn't happened is zero, we believe it more prudent to use the point estimates obtained from modifying the ZPSS estimates as working values in subsequent calculations. One has a great deal of leeway in choosing point estimates and one purpose of calculating statistical confidence limits is to show a range within which one could choose a point estimate and not be inconsistent with the data.

Correct interpretation of statistical confidence limits is important. Stating that a statistical upper 95 percent confidence limit on λ, the occurrence rate of a particular accident sequence, is 1.5 (-5)/yr., for example means the following: If λ were actually greater than 1.5(-5), then the chance of observing data as favorable as those observed, or more so, is less than .05. That is, values of λ greater than 1.5(-5) are inconsistent with the data, to the extent indicated; the observed data would be fairly unlikely. Values of λ less than 1.5(-5) are more consistent with the data since in that case the chance of such favorable data is greater than .05. Note that the chance, or probabilistic, basis of statistical confidence limits is the random variation of possible data, for a fixed λ. The parameter λ is not a random variable (nor is it in the ZPSS analysis--rather one's presumed state of knowledge about that unknown constant is expressed as a probability distribution), so a 50 percent confidence limit, for example, is not the median of some distribution of λ. There is no distribution of λ available for which a median can be calculated. Neither do statistical confidence limits represent, necessarily, our feelings about λ. They are simply statements about values of λ that are consistent with the available data, given certain models that link the data to λ. We think it is important to get a clear picture of what information can be extracted from the data.

This review also tested the readability and reproducibility of the ZPSS. In several respects, the report was found wanting. The sources of numbers used in the event tree calculations (Section 1.3) were difficult to trace because of:

- Incorrect references, e.g., a referenced section sometimes would not contain the information claimed to be there.

- Incomplete references, e.g., a reference to 1.5.2 would actually be to 1.5.2.3.4.6.2.1.6

- Unclear descriptions of events and the ZPSS modeling of them.

Specific instances will be cited in the following sections. To aid the reader, pertinent page copies from the ZPSS are included where appropriate.

3.2 Zion Dominant Accident Sequence Review

We must stress that the accident sequences discussed below which involve the loss of component cooling water are based on a system success criterion of two pumps operating. Such sequences also implicitly assume a service water system success criterion of two pumps operating. We have been given information by Commonwealth Edison (see Appendix E) that suggests that one CCW pump is sufficient but that three SW pumps are required. In Section 4.9, we consider these criteria as a sensitivity issue, and the overall values computed there do not significantly differ from those presented below. Futhermore, as this report goes to press, we understand that Zion personnel are reexamining the success criteria for specific situations (e.g., the short-term service water system pumping requirements following a loss of off-site power). Therefore, the reader should fully realize, as we do, that the accident sequence frequencies presented here are potentially subject to change.

3.2.1 Failure of Component Cooling Water (CCW), SEFC

Point Estimation

The complete loss of component cooling water frequency is given in the ZPSS as 9.4(-4) per reactor year (see I.E.13b in Table 1.5.1-50). This frequency was derived by a two-stage Bayesian analysis. To date, no such events have occurred at Zion or any other plants considered in the ZPSS data base. No further information is provided by the ZPSS.

The ZPSS analysts assumed, based on information available, that failure of component cooling water did not lead directly to core melt, without additional system failures. Specifically, it was believed that loss of component cooling water did not cause failure of the safety injection pumps and charging pumps during the injection phase. Consequently, it was concluded that loss of component cooling water as an initiating event would result in core melt only if it were combined with independent failure of these pumps (or associated hardware).

Subsequent information from Commonwealth Edison is that both charging pumps and safety injection pumps will fail "in a short period of time," given loss of component cooling water. On this basis the following sequence is applicable.

1. Component cooling water is lost with consequent loss of cooling to the reactor coolant pump seal thermal barriers.

2. The two centrifugal charging pumps fail. We estimate that each pump would fail in about 5 minutes based on information received from Consolidated Edison for similar pumps during our Indian Point Safety Study Review (Reference 3-2). Since the pumps would be operated in succession, seal cooling would be lost approximately 10 minutes after CCW failure.

3. All four reactor coolant pump seals fail in about 30 minutes after loss of seal cooling with maximum loss of coolant through each seal of 300 gallons per minute (total 1200 gallons per minute).

4. Both safety injection pumps are actuated by low reactor coolant pressure and fail due to loss of cooling in about 5 minutes.

5. With loss of makeup capability through either the charging or safety injection pumps, core uncovery will ensue. A core-melt accident will be assured unless cooling to the safety injection pumps is restored in about 45 minutes.

As noted above, the causes and corrective actions for the postulated component cooling water system failure are not given in the ZPSS. However, CCW system failure due to pipe rupture is discussed in the fault tree section of the report. Here the pipe break frequency is given as 8.60(-10) per hour per pipe section (ZPSS page 1.5-609). Thirty pipe sections are identified which would lead to system failure, resulting in an hourly failure rate for the system, due to pipe rupture of 8.6(-10) times 30, or 2.6(-8) per hour. To compute an annual frequency for this cause we multiply this hourly rate by 8760 hours (per year) and arrive at 2.3(-4). Zion operating procedure AOP-8 provides the indications which would be present for this event and the operator actions to be taken. Based on this information and the configuration and location of the CCW system components, we assume a 0.5 recovery rate. Since system capacity is approximately 25,000 gallons, a major pipe rupture could empty the system to the point of failure in less than 5 minutes, and a pipe rupture would have to be isolated and the system refilled (at 800 gallons per minute makeup capacity). On this basis we assume that 2.3(-4) of the 9.4(-4) CCW system failure is attributable to pipe rupture, and that half of this pipe rupture figure, 1.2(-4) results in core melt.

The remaining CCW losses

$$9.4(-4) - 2.3(-4) = 7.1(-4)$$

would be due to causes unknown.

For recovery, we assume that system failure is the result of failure of the three running component cooling pumps and that the recovery action would be to start the two standby pumps. Failure to recover in this case is equated to the failure to start these pumps. This could be caused by

Pump failure to start	7.2(-4)	ZPSS page 1.5-612
Pump out for maintenance	.032	ZPSS page 1.5-606
	.033	

and, since unavailability of either pump would fail the system, the probability is 2 times .033, or .066. On this basis, the failure to recover for CCW failures not caused by pipe rupture is

$$7.1(-4) \cdot .066 = 4.7(-5) \quad .$$

Note that we have neglected human error, which we believe would not contribute significantly to failure to start. Based on these calculations, the overall core-melt frequency due to loss of CCW is

$$4.7(-5) + 1.2(-4) = 1.7(-4) \quad .$$

Due to the conjectural nature of the analysis, we believe that the accuracy implied by two significant figures is not appropriate, so we state the core-melt frequency for this event as ~ 2(-4).

The following potential conservatisms and unconservatisms are present in this sequence analysis:

Conservatism

1. The two-pump criterion for CCW system success in preventing a seal LOCA may be conservative since one pump appears to be sufficient if some loads (such as the spent fuel pit heat exchanger) were isolated.

2. Although we have no basis for adopting another pipe rupture frequency, we believe that the one used is probably conservative.

Unconservatism

1. In the nonpipe break case the failure to recognize the problem and start the standby pumps is not quantified.

2. The ZPSS gives two frequencies for loss of CCW, 9.4(-4) in Table 1.5.1-50 and 5.37(-3) on page 1.3-314. We assume the lower frequency is correct.

Confidence Limits

Because of the absence of data pertaining to the probability estimates used to obtain our sequence estimate of 2(-4) occurrences per year, statistical confidence limits on the sequence rate cannot be obtained. However, for the purpose of combining this sequence with others, this estimate will be regarded as an upper 50 percent statistical confidence limit based on zero occurrences. Thus, the "effective data" assumed are zero occurrences in 3500 reactor-years. Given these "data," an upper 95 percent confidence limit on the sequence rate would be 1(-3) occurrences/year.

3.2.2 Failure of DC Bus 112, Failure of Auxiliary Feedwater, TEFC

Failure of DC Bus 112 would cause loss of main feedwater and reactor trip. It would also remove DC power from one of the two PORV's. Since Bus 112 provides control power for AC Bus 149, the auxiliary feedwater system would lose the availability of one motor-driven pump.

The sequence of interest in this case is failure of DC Bus 112, loss of main feedwater, reactor trip, loss of auxiliary feedwater, and failure of feed and bleed capability due to loss of one PORV. The sequence leads to core melt. The frequency of occurrence of the events in this sequence is discussed below.

Point Estimation

The loss of DC Bus 112 frequency is given in the ZPSS as 0.28 based on an analysis of switching errors associated with the Unit 1 DC Bus cross-tie procedure. This procedure calls for the monthly cross-tie of each Bus to an equivalent Bus in Unit 2 so that batteries and battery chargers can be isolated and a battery equalizing charge can be performed. We feel the value of .28 is a reasonable estimate because Zion data also suggests a similar value. (See confidence limits discussion.)

As noted above, loss of main feedwater and reactor trip would occur with a probability of 1.0.

Since control power has been lost to AC Bus 149, AC power has been lost to one of the auxiliary feedwater pumps. The frequency of auxiliary feedwater system failure, given the unavailability of AC Bus 149, is 2.3(-4). (See Section 2.4.1.12 for analysis.)

Power would be lost to one of the two power-operated relief valves. Since both PORV's are required for successful feed and bleed, the probability of failure of this core cooling operation is 1.0.

Based on the above, the frequency of the sequence is

Q_{cm} = probability of DC 112 failure · probability of AFW failure

= 0.28 · 2.3(-4) = 6.4(-5)·

Events with probability 1.0 are not included in the calculation.

This estimate, however, does not include possible recovery actions by the operators. Zion data suggests (see confidence limits discussion) that all Zion loss of DC events were quickly recovered. Assuming 6 or 8 recovery events yields a nonrecovery probability of ~.1 at 50 percent confidence, the sequence frequency is thus estimated as ~7(-6).

The loss of DC Bus 112 and AC Bus 149 does not prevent the function of containment fans and sprays. Therefore, this sequence category is TEFC.

Conservatisms

Information received from the utility indicates that it is questionable whether failure of a single PORV to open will fail feed and bleed core cooling (Reference 3-4).

Confidence Limits

The occurrence rate for this sequence is given by

$$\lambda = \Phi_{112} \cdot Q_{AFW} \quad .$$

where Φ_{112} denotes the failure rate of DC Bus 112 and Q_{AFW} denotes the failure probability of the auxiliary feedwater system. The effective data pertaining to the parameters are as follows:

Φ_{112}: A review of the Zion reactor trip data (pp. 1.5-146,154) shows six events which were DC Bus failures, two which may have been. Conservatively counting all of these yields eight failures in 33 Bus-years since there are three Buses per unit and a total of 11 unit-years Zion experience. These data yield an estimated Bus failure rate of 8/33 = .24 failures/yr. and an estimated variance of this estimate of .007. By way of contrast, the ZPSS estimate based on an estimate of the probability of a switching error is a posterior distribution which has a mean of .28 failures/yr. and a variance of .11, which amounts to effective data of one occurrence in three years.

Further review of the DC Bus data, however, indicates that all the failures were quickly recoverable. Hence, for the purpose of obtaining statistical confidence limits, the data used will be 0/33 Bus-years, i.e., (8/33)(0/8) = (0/33).

Q_{AFW}: AFW fails if no motor-driven pump train and one turbine-driven pump train are both unavailable due to either failures or maintenance, with the exception that both pumps cannot be simultaneously out for maintenance. The basic events that contribute to Q_{AFW} and their corresponding data are as follows:

Q_T: Failure of the turbine-driven pump to start and run 8 hours. The ZPSS reported data are 6/231, but it is assumed that half of such failures are recoverable so 3/231 will be assumed. Failure-to-run data are 0/1900 hrs., which are relatively negligible.

Q_{TM}: Maintenance unavailability of the turbine-driven pump. The ZPSS data showed an increasing trend (p.1.5-92) in unavailability and so in our initial review a conservative (apparently) estimate was used. However, subsequent data provided by

Commonwealth Edison and transmitted to us by S. Newberry, U.S. NRC, showed the following:

Year	Unavailability
1979	.100
1980	.007
1981	.025
1982	.040

(Note: This 1980 figure differs from that in the ZPSS because the ZPSS authors apparently counted some outages during a cold shutdown period.) These data show the trend has not continued at the pre-1980 level. Treating the year-to-year differences as random leads to an estimated unavailability of .026. The estimated variance of this estimate is 1.4(-4) and these results lead to effective data for Q_{TM} of 5/185.

Q_M: Failure of the motor-driven pump to start and run for 8 hours. The failure-to-start data given in the ZPSS (entry 36 in Table 1.5.1-5) are 4/462 and the fail-to-run data are 1/3800 hrs. Combining these for an 8-hour mission yields effective data of 5/462.

Q_{MM}: Maintenance unavailability of a motor-driven pump. The ZPSS data (p.1.5-91) yield effective binomial data of 12 failures in 930 demands.

The model for Q_{AFW} is

$$Q_{AFW} = Q_T \cdot Q_M + Q_{TM} \cdot Q_M + Q_{MM} \cdot Q_T$$

Applying the Maximus methodology and using the Zion data pertaining to these parameters yields effective data pertaining to Q_{AFW} of three failures in 5000 demands. Combining these with the effective data for Φ_{112} yields effective sequence data of 0/41,000 reactor-years. The resulting upper 95 percent confidence limit is 7(-5)/yr. The upper 50 percent confidence limit is 2(-5)/yr.

3.2.3 Loss of Off-site Power: Loss of Component Cooling Water: Failure to Restore Power in 4 Hours, SEFC

In this sequence the initiating event, loss of off-site power, is followed by loss of component cooling water with failure to restore power in 4 hours. Given loss of component cooling water, a series of events leading to a seal LOCA with loss of makeup capability and thus to core melt will occur, as described in Section 3.2.1. The frequency of the events in this sequence is described below.

Point Estimation

The loss of off-site power frequency as an initiating event is given in the ZPSS as 5.7(-2). In addition we consider the following events and frequencies from the ZPSS: turbine trip followed by loss of off-site power, 1.3(-3); loss of main feedwater followed by loss of off-site power, 1.8(-3); and reactor trip followed by loss of off-site power, 1.3(-3). Each of these events could lead to the accident sequence in question. The sum of these frequencies is .061. Our analysis assumes loss of off-site power to both Zion units, which is consistent with the ZPSS analysis. (See discussion of seal LOCA in ZPSS Section 1.3.3.8.) The ZPSS analyst has indicated that their study of loss of off-site power leads to the conclusion that it will nearly always affect both units.

The failure to restore off-site power in 4 hours is, in effect, the failure to restore power in 30 minutes, 60 minutes, and 4 hours and, by implication, the success in restoring power by 8 hours. The frequency of these events is assigned on the basis of plant historical data developed by the Electric Power Research Institute[3-3] as indicated below. These frequencies are different from the ones used in the ZPSS, which were based on an analysis of the Zion electric power distribution system. See Section 2.5 for comments.

EP30 (Failure to restore power in 30 minutes)	0.52
EP60 (Failure to restore power in 60 minutes)	0.38
EP 4 (Failure to restore power in 4 hours)	0.25
EP 8 (Failure to restore power in 8 hours)	0.10

Since we apply these frequencies sequentially in our evaluation, the following calculations are necessary.

$$\text{EP60(Given EP30)} = \frac{0.38}{0.52} = 0.73$$

$$\text{EP4(Given EP30 and EP60)} = \frac{0.25}{(0.52)(0.73)} = 0.66$$

$$\text{EP8(Given EP30, EP60, and EP4)} = \frac{0.10}{(0.52)(0.73)(0.66)} = 0.4$$

$$\overline{\text{EP8}} \text{ (Given EP30, EP60, and EP4)} = 0.6$$

The probability that component cooling water (CCW) fails, given loss of off-site power, is dependent on the state of the vital AC power Buses. The calculation of CCW failure probability for each degraded power state is detailed in Section 2.4.1.10. For convenience, the results are repeated here.

Power on Buses	CCW Failure Frequency
All	4.3(-5)
147 and 148	2.5(-3)
147 and 149	2.5(-3)
148 and 149	2.5(-3)
147	3.2(-2)
148	3.2(-2)
149	3.2(-2)
None	0.17

Based on the above, we consider the sequence

$$EP30 \cdot CCW' \cdot EP60 \cdot EP4 \cdot \overline{EP8} \quad .$$

The calculations for each degraded power state are tabulated as follows:

Power on Buses	(.15)(CCW') (Conditional probability of power state, given no ES signal)	
All	(.15)(4.3-5)(.38)	= 2.4(-6)
147 and 148	(.15)(2.5-3)(3.6-2)	= 1.4(-5)
147 and 149	(.15)(2.5-3)(3.6-2)	= 1.4(-5)
148 and 149	(.15)(2.5-3)(.45)	= 1.7(-4)
147	(.15)(3.2-2)(3.2-3)	= 1.5(-5)
148	(.15)(3.2-2)(4.5-2)	= 2.2(-4)
149	(.15)(3.2-2)(4.5-2)	= 2.2(-4)
None	(.15)(.17)(3.9-3)	= 9.9(-5)
	Total	7.5(-4)

where .15 is the product of EP30, EP60, EP4, and $\overline{EP8}$, and the last term (e.g., 0.38) is the conditional probability of the given electric power state (see Section 2.4.1.1). The total 7.5(-4) is the frequency of the sequence, given loss of off-site power. To complete the calculation we must multiply by the frequency of the initiating events.

$$7.5(-4) \cdot .061 = 4.6(-5) \quad .$$

Since this sequence involves a seal LOCA and off-site power is restored by 8 hours, the containment sprays and fans would be available. Consequently this is an SEFC sequence.

The following conservatisms were identified based on the ZPSS report and this analysis:

Conservatism

1. The two-pump success criterion for the CCW system in preventing a seal LOCA is potentially conservative since one pump appears to be sufficient if nonessential loads (such as the spent fuel pit heat exchanger) were isolated.

2. The recovery-of-off-site-power failure probabilities used in the analysis may be high because historical data primarily reflects LOP situations in which emergency diesel generators were available. Given the loss of diesel generators factored into the analysis, a more vigorous effort to recover off-site power might be made, with consequent shorter recovery times.

Statistical Confidence Limits

The model for this sequence is

$$\lambda = \Phi_{11b} \cdot Q_{CCW} \cdot Q_{4,8},$$

where

Φ_{11b} = occurrence rate of loss of off-site power, which is initiating event 11b,

Q_{CCW} = conditional probability of component cooling water failure, given loss of off-site power,

$Q_{4,8}$ = conditional probability of off-site power recovery between 4 and 8 hours.

In order to obtain statistical confidence limits for λ, the sequence occurrence rate, we need to obtain data pertaining to each event in the sequence. We do this as follows:

Φ_{11b}: In response to questions concerning the ZPSS estimate of .06 occurrences per year for loss of off-site power, which is lower than "conventional" estimates, Pickard, Lowe, and Garrick collected new data and repeated their two-stage Bayesian analysis.[3-4] The new data show 34 events in 229 reactor-years; 0/9 at Zion. Examining the data shows that with the exception of Turkey Point, the industry-wide data are quite consistent across plants. With this exception, the data show 26 occurrences in 220 reactor-years. The assumed availability of each Zion unit is .71. Thus, we will use data of .71(26) = 18 occurrences in 220 reactor-years to estimate Φ_{11b}.

Q_{CCW}: There are a variety of diesel generator and component cooling water pump failures and maintenance unavailabilities that contribute to Q_{CCW}. These events and their corresponding data are as follows:

DGFS: Diesel generator fail-to-start. This event has three subevents: H, failure of the diesel to start; J, failure of a Bus feed breaker to close; and K, failure of a Bus feed breaker to open. The Zion data for these are:

H: 30 failures; 1693 demands
J: 5/3120
K: 1/3120

These are series events so using the Maximus methodology to combine them leads to effective data of 33/1693 for DGFS.

DGFM: Diesel generator maintenance unavailability. The average observed annual unavailability is .03, with a squared standard error of 2.3(-15), which, for the purpose of combining these data with others, corresponds to effective data of 40 occurrences in 1300 demands.

DG: DG = DGFS + DGFM. The effective data are 65/1300.

CCWP: Component Cooling Water Pump fails-to-start. The Zion reported data (entry 11 in Table 1.5.1-5) are 3/3138.

CCWM: Component Cooling Water pump maintenance unavailability. The Zion data (Table 1.5.1-20) show an average annual unavailability of .06 with a squared standard error of 1.0(-3). These results correspond to effective data of 3.4/56. The ZPSS estimated unavailability is .032 (p. 1.5-606).

CCW2M: Simultaneous unavailability of two CCW pumps. The ZPSS, p.1.5-606, says, "of the actions reviewed, five maintenance actions occurred on a CCW pump with one pump already out for service." Though the number of actions reviewed is not given, the ratio of the Zion values of CCW2M to CCWM suggests data of 5/21 for this factor, which we will denote by Q_{M2}.

For the purpose of obtaining effective data for Q_{CCW}, the expressions used to obtain point estimates were simplified. In particular, only four power states were considered. Those power states and the simplified models for those probabilities and the conditional probabilities of CCW failure are:

Power on Buses	Prob (Power State)	(CCW/Power State)
148,149	.5	6 DG CCW2M
148	.5 DG	4DG · CCWM + 3 CCW2M
149	.5 DG	4DG · CCWM + 3 CCW2M
None	.5 DG^2	2 (DG + CCWM)

The factor of .5 in the power state probabilities arises because Bus 147 is connected to the swing diesel. A conservatism in the above expressions is that the possibility of simultaneous maintenance on two diesels is not subtracted out. This will compensate somewhat for the nonconservatism of not including all power states.

Multiplying and summing the terms in the above table yields

$$Q_{CCW} = 6\ DG \cdot CCWM \cdot Q_{M2} + 5\ DG^2 \cdot CCWM + DG^3$$

Dropping the last term, which is relatively negligible, and applying the Maximus methodology to the others yields effective data of two failures in 412 demands.

$Q_{4,8}$: The recovery data in EPRI-NP-2301 show six recoveries of off-site power in the period of 4-8 hours out of 42 instances of loss of power.

Multiplying the data-based estimates of the terms in the model for λ yields an estimate of

$$\lambda^* = \frac{18}{220} \cdot \frac{2}{412} \cdot \frac{6}{42} = 5.7(-5)/yr.$$

which is not importantly different from our point estimate above of 4.6(-5) derived primarily from the ZPSS posterior means. The Maximus methodology yields effective sequence data of one occurrence in 17,000 reactor-years, which leads to an upper 95 percent statistical confidence limit of 3(-4)/yr. and a lower 95 percent limit of 3(-6)/yr.

3.2.4 Loss of Off-site Power: Loss of Component Cooling Water: Failure to Restore Power in 1 Hour, SEFC

Note: This sequence is the same as the sequences discussed in Sections 3.2.2 and 3.2.5, except for the time at which off-site power is restored. All three sequences contribute to plant damage state SEFC.

In this sequence the initiating event, loss of off-site power, is followed by loss of component cooling water, with failure to restore power in 1 hour. Given loss of component cooling water, a series of events leading to a seal LOCA with loss of makeup capability, and, thus, to core melt will occur, as described in Section 3.2.1. The frequency of the events in this sequence is described below.

Point Estimation

The loss of off-site power frequency as an initiating event is given in the ZPSS as 5.7(-2). In addition we consider the following events and frequencies from the ZPSS: turbine trip followed by loss of off-site power, 1.3(-3); loss of main feedwater followed by loss of off-site power, 1.8(-3); and reactor trip followed by loss of off-site power, 1.3(-3). Each of these events could lead to the accident sequence in question. The sum of these frequencies is 0.061. Our analysis assumes loss of off-site power to both units, which is consistent with the ZPSS analysis (see ZPSS Section 1.3.3.8). The ZPSS analyst has indicated that their study of loss of off-site power leads to the conclusion that it will nearly always affect both units.

The failure to restore off-site power in 1 hour is, in effect, the failure to restore power in 30 minutes and 1 hour, and by implication, the success in restoring power in 4 hours. The frequency of power recovery events is assigned on the basis of plant historical data developed by the Electric Power Research Institute as follows:

EP30 (Failure to restore power in 30 minutes)	0.52
EP60 (Failure to restore power in 60 minutes)	0.38
EP4 (Failure to restore power in 4 hours)	0.25

$$\text{EP60(Given EP30)} = \frac{0.38}{0.52} = 0.73$$

$$\text{EP4(Given EP30 and EP60)} = \frac{0.25}{(0.52)(0.73)} = 0.66$$

$$\overline{\text{EP4}}\ \text{(Given EP30 and EP60)} = .34$$

The probability that component cooling water (CCW) fails, given loss of off-site power, is dependent on the state of the vital AC power Buses. The calculation of CCW failure probability for each degraded power state is detailed in Section 2.4.1.10. For convenience, the results are repeated here.

Power on Buses	CCW Failure Frequency
All	4.3(-5)
147 and 148	2.5(-3)
147 and 149	2.5(-3)
148 and 149	2.5(-3)
147	3.2(-2)
148	3.2(-2)
149	3.2(-2)
None	0.17

Based on the above, we consider the sequence

$$EP30 \cdot CCW' \cdot EP60 \cdot \overline{EP4}$$

The calculations for each degraded power state are tabulated as follows:

Power on Buses	(.13)(CCW') (Conditional probability of power state, given no ES signal)	
All	(.13)(4.3-5)(.38)	= 2.1(-6)
147 and 148	(.13)(2.5-3)(3.6-2)	= 1.2(-5)
147 and 149	(.13)(2.5-3)(3.6-2)	= 1.2(-5)
148 and 149	(.13)(2.5-3)(.45)	= 1.5(-4)
147	(.13)(3.2-2)(3.2-3)	= 1.3(-5)
148	(.13)(3.2-2)(4.5-2)	= 1.9(-4)
149	(.13)(3.2-2)(4.5-2)	= 1.9(-4)
None	(.13)(.17)(3.9-3)	= 8.5(-5)
	Total	6.5(-4)

where .13 is the product of EP30, EP60, $\overline{EP4}$.

The total 6.5(-4) is the frequency of the sequence, given loss of off-site power. To complete the calculation we must multiply by the frequency of the initiating events.

$$6.5(-4) \cdot .061 = 4.0(-5) \quad .$$

Since this sequence involves a seal LOCA and off-site power is restored by 8 hours, the containment sprays and fans would be available. Consequently this is an SEFC sequence.

The following conservatisms were identified, based on the ZPSS and this analysis:

Conservatism

1. The two-pump success criterion for the CCW system in preventing a seal LOCA is potentially conservative since one pump would be sufficient if non-essential loads (such as the spent fuel pit heat exchanger) were isolated.

2. The recovery-of-off-site-power failure probabilities used in the analysis may be high because historical data primarily reflects LOP situations in which emergency diesel generators were available. Given the loss of diesel generators factored into the analysis, a more vigorous effort to recover off-site power might be made, with consequent shorter recovery times.

Statistical Confidence Limits

This sequence differs from that considered in Section 3.2.2 only in that power is recovered in the period of 1-4 hours rather than 4-8 hours. The EPRI-NP-2301 recovery data show six recoveries out of 42 in the 1-4 hour range, the same as in the 4-8 hour range, so the same numerical results are obtained:

λ_{U95} = 3(-4)/yr., λ_{L95} = 3(-6)/yr.

3.2.5 Loss of Off-site Power: Failure of Component Cooling Water: Failure to Recover Off-site Power in 8 Hours, Failure of Containment Fans, SEC

Point Estimation

This sequence is the same as that described in Section 3.2.7 except that it also includes containment fan system failure. Since the success criterion for the containment fan system is three of five fan coolers operating, the dominant cause of fan system failure in this sequence is loss of power from two of the three Unit 1 AC Buses. For this, we continue the calculation in Section 3.2.7:

Power on Buses	(.10)(CCW') (Conditional probability of power state, given no ES signal)
147	(.10)(3.2-2)(3.2-3) = 1.0(-5)
148	(.10)(3.2-2)(4.5-2) = 1.4(-4)
149	(.10)(3.2-2)(4.5.2) = 1.4(-4)
Total	2.9(-4)

The total 2.9(-4) is the frequency of the sequence, given loss of off-site power. To complete the calculation, we multiply by the frequency of the initiating events

$$2.9(-4) \cdot 0.061 = 1.8(-5) \quad .$$

Since this sequence involves a seal LOCA with loss of containment fan coolers, the plant damage state is SEC.

The following conservatisms were identified, based on the ZPSS and this analysis:

Conservatism

1. The two-pump success criterion for the CCW system in preventing a seal LOCA is potentially conservative since one pump may be sufficient if non-essential loads (such as the spent fuel pit heat exchanger) were isolated.

2. The recovery-of-off-site-power failure probabilities used in the analysis may be high because historical data primarily reflects LOP situations in which emergency diesel generators were available. Given the loss of diesel generators factored into the analysis, a more vigorous effort to recover off-site power might be made, with consequent shorter recovery times.

3. The "three-of-five fan" success criterion is potentially conservative. Two fans may be sufficient for cooling in the small LOCA case.

3.2.6 Small LOCA: Failure of Recirculation Cooling, SLF

Point Estimation

By Zion's estimates (see page 8.10-7), this accident is the most probable cause of core melt. Zion's dominant sequence (p. 1.3-128) occurs when AC power is available at all three Buses and recirculation cooling fails (R-2). Zion's mean value for the probability of R-2 is given as 4.55(-4). Multiplying by the mean Small LOCA rate (3.54(-2) per year) and by the probability of power at all AC Buses (~1) yields 1.6(-5) per year as the estimated core-melt rate by this sequence.

As discussed elsewhere, the initiating event estimates (given as posterior means and variances) are reasonably consistent with the data presented. Consider now the estimate of R-2. The referenced supporting sections are Section 1.3.3 and 1.5.2. In the former (p. 1.3-35) it is said that the split fraction used will be calculated (conservatively) assuming fan coolers are not available. A search of Section 1.5.2 turns up a calculation of recirculation system unreliability when the fan coolers are unavailable and all electric Buses are available (Section 1.5.2.3.4.4.2.1.6), which would seem to be appropriate. However, there the mean value given is 3.9(-4), not seriously different from the above 4.55(-4), but enough to be a puzzle. (From subsequent conversations with the authors we learned that late changes in the fault tree estimates were not fed through the event tree models if their impact was deemed unimportant.)

The calculations on p. 1.5-463 also show that the authors' DPD arithmetic can have a considerable effect. The expression for this event (denoted $Q_{HI\text{-}HEAD}$, rather than Q_{R-2}) is

$$Q_{HI\text{-}HEAD} = 1.25Q_{H1} + Q_{SUMP} + \text{(doubles)}$$

where H1 denotes the human error of failure to initiate the switchover to the recirculation phase (approximately 40 percent of total) and SUMP denotes blockage of the containment sump. The variances for Q_{H1} and Q_{SUMP} are given as 4.54(-7) and 6.4(-6), respectively. Thus the variance of $Q_{HI\text{-}HEAD}$ should exceed $(1.25)^2(4.54(-7)) + 6.4(-6) = 7.1(-6)$. But the variance given is 1.7(-7), less by a factor of 40 than this value. Either a mistake has been made or (more likely), when the assumed lognormal distributions for Q_{H1} and Q_{SUMP} were discretized, chopping off the tails and choosing the discrete values and their probabilities reduced the variance considerably. In effect, the DPD methodology increased the (apparent) information by a factor of 40. The authors should have given the means and variances of the actual distributions used.

The Zion estimates for $Q_{H1\text{-}HEAD}$ are dominated by the estimates of Q_{H1} and Q_{SUMP}. Both are wholly subjective--unmodified by any data. We have no data on which to base an evaluation of these estimates. Component failures of interest are primarily motor-operated valve failures. The Zion data alone, 14/11,310 (comp. 6 in Table 1.5-1.5) lead to an estimated failure probability of 1.24(-3) with an estimated variance of 1.09(-7). The Zion posterior mean and variance are 1.55(-3) and 6.3(-8). The differences are not enough, relative to the dominating Q_{H1}, to be concerned with. Hence we accept Zion's estimated rate of occurrence of core melt via this accident sequence.

<u>Statistical Confidence Limits</u>

The model for this sequence is

$$\lambda = \Phi_3 \cdot Q_{R-2} \quad .$$

where ϕ_3 denotes the occurrence rate of small LOCAs. The ZPSS-reported data for this event are three occurrences in 131 reactor-years and there is no strong evidence of plant-to-plant differences. Thus these data will be used in obtaining confidence limits.

The dominant hardware contributors to Q_{R-2} are the simultaneous failures of two pumps or of any of four pairs of MOVs. The Zion estimates of the probabilities of these dual failures are based on their universal β-factor of .014. Alternatively, suppose the testing procedure is such that if a pump or a valve fails a test, then the redundant pump or valve is immediately tested. If so, then each test is effectively a test of a redundant pair. The ZPSS lists 14 MOV failures, two on the same day for the same reason (p.1.5-17), in 11,310 tests. Thus, we will estimate the probability of dual valve failure from data of one occurrence in 11,310 demands. The pump data show no dual failures in 3,138 tests, so data of 0/3138 will be used for the probability of a dual pump failure.

If we ignore the human error (which seems quite conservatively estimated), then the effective Q_{R-2} data become 1/2828. This ratio which equals 3.5(-4) is not appreciably different from the above point estimate of 3.9(-4) so we will use this to obtain sequence confidence limits.

Combining the data for the two parameters in the sequence model yields effective sequence data of .5/60,000 reactor-yrs. which lead to statistical confidence limits of:

λ_{u95} = 6(-5)/yr., λ_{L95} = 3(-8)/yr.

3.2.7 Loss of Off-site Power: Loss of Component Cooling Water: Failure to Restore Power in 8 Hours, SEFC

Note: This sequence is similar to the sequences discussed in Sections 3.2.3 and 3.2.4, except for the time at which off-site power is restored. All three sequences contribute to plant damage state SEFC.

In this sequence, the initiating event, loss of off-site power, is followed by loss of component cooling water, with failure to restore power in 8 hours. Given loss of component cooling water, a series of events leading to a seal LOCA with loss of makeup capability and thus a core melt will occur, as described in Section 3.2.1. The estimated frequency of the events in this sequence are described below.

Point Estimation

The loss of off-site power frequency as an initiating event is given in the ZPSS as 5.7(-2). In addition we consider the following events and frequencies from the ZPSS: turbine trip followed by loss of off-site power, 1.3(-3); loss of main feedwater followed by loss of off-site power, 1.8(-3); and reactor trip followed by loss of off-site power 1.3(-3). Each of these events could lead to the accident sequence in question. The sum of these frequencies is 0.061. Our analysis assumes loss of off-site power to both units, which is consistent with the ZPSS analysis (see ZPSS Section 1.3.3.8). The ZPSS analysts have indicated that their study of loss of off-site power leads to the conclusion that it will nearly always affect both units.

The failure to restore power in 8 hours is, in effect, the failure to restore power in 30 minutes, in 60 minutes, in 4 hours and in 8 hours. The frequency of power recovery events is assigned on the basis of plant historical data developed by the Electric Power Research Institute as follows:

EP30 (Failure to restore power in 30 minutes)	0.52
EP60 (Failure to restore power in 60 minutes)	0.38
EP4 (Failure to restore power in 4 hours)	0.25
EP8 (Failure to restore power in 8 hours)	0.10

Since we apply these frequencies sequentially in our evaluation, the following calculations are necessary.

$$\text{EP60 (Given EP30)} = \frac{0.38}{0.52} = 0.73$$

$$\text{EP4(Given EP30 and EP60)} = \frac{0.25}{(0.52)(0.73)} = 0.66$$

$$\text{EP8(Given EP30, EP60, and EP4)} = \frac{0.10}{(0.52)(0.73)(0.66)} = 0.40$$

The probability that component cooling water (CCW) fails, given loss-of-off-site power, is dependent on the state of the vital AC power Buses. The calculation of CCW failure probability for each degraded power state is detailed in Section 2.4.1.10. For convenience, the results are repeated here.

Power on Buses	CCW Failure Frequency
All	4.3(-5)
147 and 148	2.5(-3)
147 and 149	2.5(-3)
148 and 149	2.5(-3)
147	3.2(-2)
148	3.2(-2)
149	3.2(-2)
None	0.17

Based on the above, we consider the sequence

EP30 · CCW' · EP60 · EP4 · EP8

The calculations for each degraded power state are tabulated as follows:

Power on Buses	(.10)(CCW') (Conditioned power state,	probability of given no ES signal)
All	(.10)(4.3-5)(.38)	= 1.6(-6)
147 and 148	(.10)(2.5-3)(3.6-2)	= 9.0(-6)
147 and 149	(.10)(2.5-3)(3.6-2)	= 9.0(-6)
148 and 149	(.10)(2.5-3)(.45)	= 1.1(-4)
	Total	1.3(-4)

where 0.10 is the failure to restore power in 8 hours. Only those degraded power states which would result in plant damage state SEFC are included in the above calculation. The degraded power states which would result in a SEC plant damage state are discussed in Section 3.2.5. The total 1.3(-4) is the frequency of the sequence given loss of off-site power. To complete the calculation, we multiply by the frequency of the initiating events.

$$1.3(-4) \times 0.061 = 7.9(-6)$$

Since this sequence involves a seal LOCA and, for the power states included, containment sprays and fans would be available, the result is plant damage state SEFC.

The following conservatisms were identified, based on the ZPSS and this analysis:

<u>Conservatism</u>

1. The two-pump success criterion for the CCW system in preventing a seal LOCA is potentially conservative

since one pump may be sufficient if non-essential loads (such as the spent fuel pit heat exchanger) were isolated.

2. The recovery-of-off-site-power failure probabilities used in the analysis may be high because historical data primarily reflects LOP situations in which emergency diesel generators were available. Given the loss of diesel generators factored into the analysis, a more vigorous effort to recover off-site power might be made, with consequent shorter recovery times.

Statistical Confidence Limits

The model for this sequence is

$$\lambda = \Phi_{11b} \cdot Q_{CCW} \cdot Q_{8'}$$

where $Q_{8'}$ denotes the probability of off-site power recovery after 8 hours and the other terms are defined in Section. 3.2.2.

The dominant power state for this sequence is power at Buses 148 and 149, which we estimate to occur with a conditional probability of .5, given loss of off-site power (see Section 3.2.2). The simplified expression for CCW failure, given this power state, from Section 3.2.2, is 6DG · CCW2M. Thus, Q_{CCW} = 3DG · CCW2M. The data pertaining to these parameters are given in Section 3.2.2. Combining them yields effective data corresponding to Q_{CCW} of 2.1/962, which when combined with the other event data yield effective sequence data of one occurrence in 59,000 reactor-years. The resulting statistical confidence limits on λ are λ_{u95} = 8(-5)/yr. and λ_{L95} = 9(-7)/yr.

3.2.8 Seismic: Loss of All AC Power, SE

Jack R. Benjamin & Associates, Inc., reviewed ZPSS Sections 7.2.4 to 7.2.6 which develop a model for core melt due to seismically induced failures. Additional comments are given in Appendix A and in Section 2.7.1 of this report.

In this sequence, a seismic event large enough to fail off-site power and the service water pumps occurs. Failure of the service water pumps causes subsequent failure of the diesel generators, due to lack of cooling. A loss of all AC power results followed by failure of RCP seal cooling and a RCP LOCA. Since safety injection and containment systems require AC power, a core melt ensues that results in damage state SE. The ZPSS frequency estimate for this sequence is 5.6×10^{-6} per year. We conclude that this estimate is "reasonable." In the following subsections we discuss the rationale behind this conclusion.

3.2.8.1 Review of "Plant Logic"

We agree that it is reasonable to assume that off-site power will be lost due to the failure of transformer ceramic insulators as a result of any earthquake large enough to contribute to the frequency of failure of the plant. Since the contribution to the mean frequency of core melt is significant for ground accelerations greater than 0.3g, it is reasonable to assume that off-site transformer ceramic insulators have failed (note that the median capacity of the ceramic insulators is 0.2g).

Based on data we obtained from Lawrence Livermore National Laboratory we confirmed the median acceleration capacity values for recoverable interruptions of electrical components and that nonrecoverable failure is several times the recoverable modes such as relay chatter and breaker trip. However, we question whether it is reasonable to, a priori, eliminate components 2, 3, 5, 6, 7, and 13 listed in Table 7.2-3 from further consideration. One viewpoint is that there are many individual components involved in the electrical equipment. It is possible that these components are in series such that a failure of only one of them may cause severe consequences? Also as mentioned in Chapter 2, Appendix A, we have some concerns about sequences of events and the inability of a component to absorb energy (i.e., brittleness). Also, we question whether relays will also trip at large acceleration levels. It is implicitly assumed in the analysis that this is a recoverable event. This may be a problem for electrical components.

In regard to component 16 (fan cooler duct work), we cannot judge whether the fan coolers are mechanically capable of adequately mixing the containment gases without the duct work. If this is true, this is sufficient reason to eliminate item 16 from further consideration. The argument that it is improbable that all the duct risers would fail from the same earthquake may be weak. If these components are identically constructed and attached to the same portion of the building, their capacities and seismic responses may be highly correlated. If so, then the failure of one would imply the failures of others. We did not investigate the details of construction for the fan cooler duct work.

As stated in Chapter 1, Appendix A, we did not review the fault trees (Figures 7.2-3a through 7.2-3g) for completeness or functional relationships. We did note that the following components were eliminated since it is claimed that their failure is not induced by the range of possible seismic events:

- Power relief valves
- Charging pumps
- Auxiliary feedwater pumps
- RHR pumps
- Pressurizer
- Pressurizer piping (see discussion below)
- Fuel supply to diesel generators
- Service water supply
- Diesel generator (direct failure)
- Switchgear failure
- AC power cables (direct failure)
- Cable trays
- Control building
- Direct failure of piping between Auxiliary building and Containment building (note that piping failures due to soil failure were considered)
- Other failures

We could not find the fragility parameter values for the piping component in Figure 7.2-3d which was eliminated from further consideration. We assume this piping is associated with the pressurizer. In regard to "other failures" which were eliminated from several branches on the fault trees, we question whether the possibility was considered that "other" structures, equipment, or components could fail, fall, and impinge on critical safety-related structures or equipment.

In Figure 7.2-3d, the pressurizer failure was considered to be "nonapplicable." Since the collapse of the pressurizer enclosure roof will occur at a much lower acceleration, we question whether the pressurizer capacity should be replaced by the capacity of this structure. In either case, the effect on the final results is judged to be small.

In the case of piping, all pipe segments are connected in series; thus the frequency of failures for a piping system is not conservatively represented by the frequency of the weakest component, unless the capacities and responses of all segments are individually (i.e., capacity with capacity and response with response) perfectly correlated. Because piping extends a relatively long distance and is supported at many places in a structure, piping response will not be perfectly correlated. Also, because different components may come from different manufacturers or material runs, capacity also is not perfectly correlated. The Licensee's response as discussed in Appendix A suggests that this effect was considered in the analysis and the selection of fragility parameter values for piping systems; however it does not appear that the capacity of a single piping run was reduced for the effects of a series system. A similar problem also exists for electric cables supported by cable trays.

The fragility parameters for the rest of the components listed above were reviewed, and we agree that they do not significantly contribute to the frequency of core melt.

The final Boolean expression for core-melt failure given on page 7.2-7 interacts with the following 10 structures or equipment items:

4 Service Water Pumps

8 Auxiliary Building -- Failure of Concrete Shear Wall

9 Refueling Water Storage Tank

10 Interconnecting Piping/Soil Failure Beneath Reactor Building

12 Condensate Storage Tank

14 Crib House Collapse of Pump Enclosure Roof

17 125 VDC Batteries and Racks

21 Service Water System Buried Pipe 48"

22 CST Piping 20"

26 Collapse of Pressurizer Enclosure Roof

The review concentrated primarily on these items followed by items which were eliminated from the fault trees due to their high seismic capacity.

Figures 3.1-1 and 3.1-2 show the relative contribution of the above-listed 10 items to the total frequency of core melt. The failure fraction versus damage effective ground acceleration is shown for linear and logarithmic scales in these two figures. As shown, the curves become progressively higher (i.e., larger failure fraction) as more and more items are included in the Boolean equation for core-melt failure. Note that in developing these curves the randomness and uncertainty were combined.

Several observations which give perspective to the importance of each structure or equipment item can be seen from these curves. The most important contributor to core melt is the service water pumps, which are the weakest components. The next most important item is the Auxiliary Building shear wall. As explained previously, the integration of the hazard and fragility curves depends primarily on acceleration values above 0.3g. For ground accelerations above 0.3g, neglecting the contribution (i.e., based on the

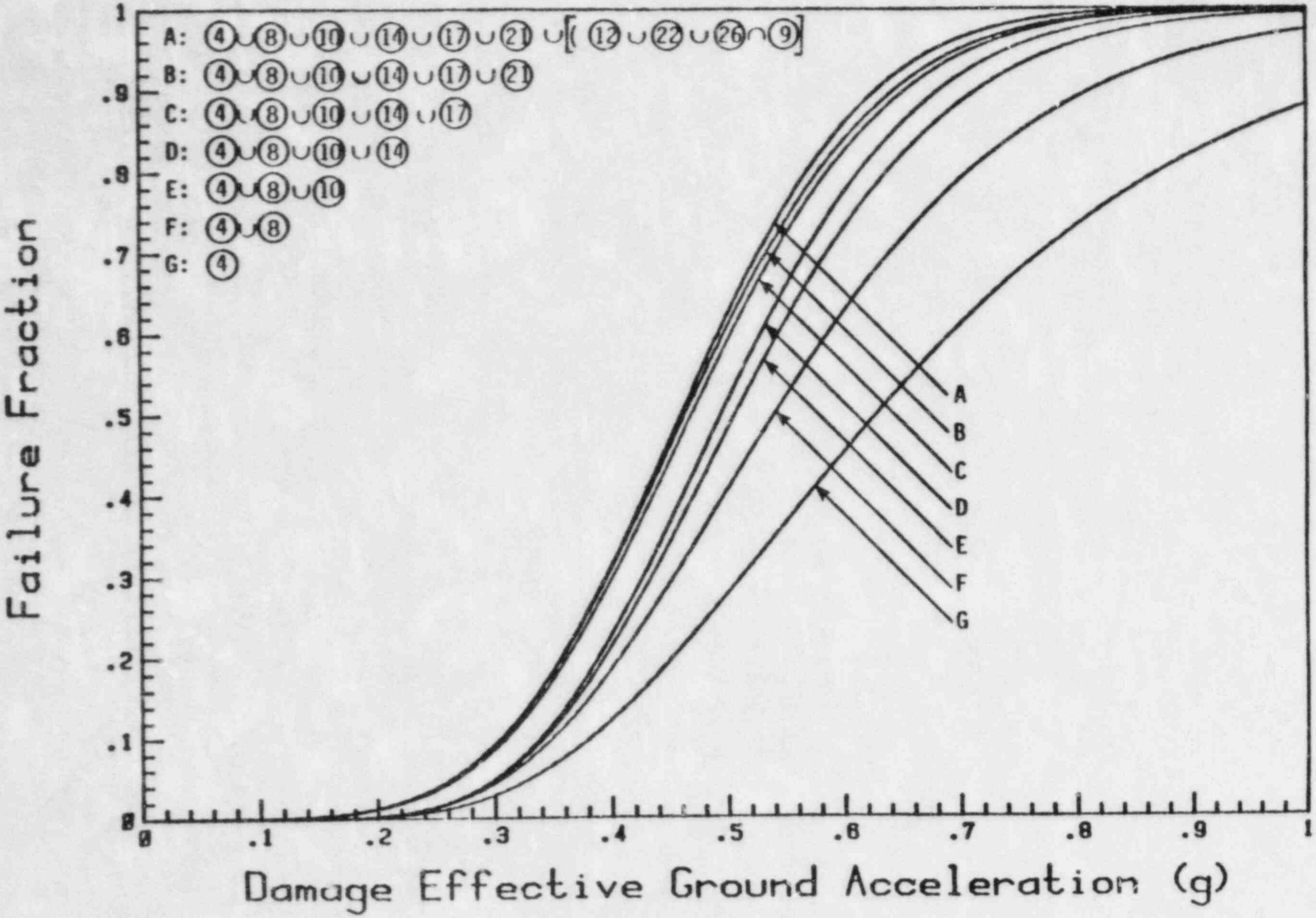

Figure 3.1-1 Partial Fragility Curves - Linear Scale

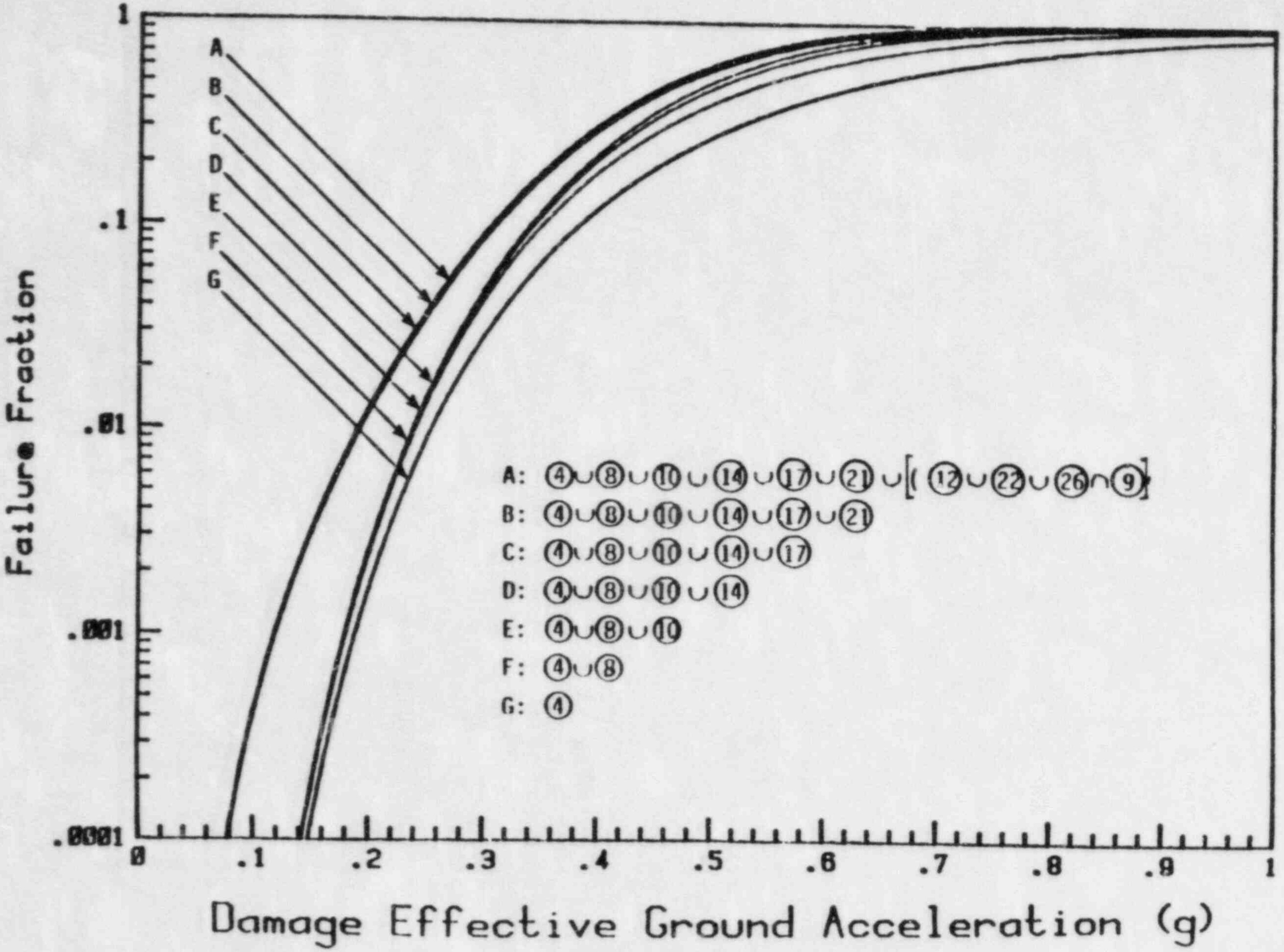

Figure 3.1-2 Partial Fragility Curves - Logarithmic Scale

fragility parameter given in the ZPSS) from all other structures or equipment, would result in underestimating the failure fraction by a factor of only three (a moderate effect). In terms of the probability distribution of the frequency of core melt (i.e., Figure 7.2-5), the mean value would also follow this same relationship. However, the lower tail of the density function would be affected more severely by the elimination of other items. Thus, in order for a major change to occur in the mean frequency of core melt due to fragility effects, the strengths of a number of the structure or equipment items would have to drop down to or below the strength of the service water pumps (or the strength of the service water pumps would have to decrease).

A jump in the probability of failure occurs when item 17, which is the 125 VDC batteries and racks, is included in the Boolean expression (see Figure 3.1-2). This is due to the relatively high uncertainity value (i.e., β_u = 0.63) for this equipment.

Because items 9, 12, 22, and 26 are embedded in an "and" subexpression, they do not contribute significantly to coremelt frequency.

There is one potentially important type of dependency that is not discussed in the ZPSS. This involves the correlation of the response of two or more structures or equipment due to the motion of a common supporting building. In simple terms, if two components are located in the same building, close to each other, the response input to one could be nearly the same as for the other. Thus, if one exhibited a high response, the other would likely also have a high response. Because of the correlation of input (hence response), the failure frequencies for the two components would be correlated. The potential for this type of interaction is present for the following two combinations of components:

1. <u>Crib House</u>

    4 Service Water pumps
    14 Crib House Roof

2. <u>Auxiliary Building</u>

    8 Auxiliary building shear wall
    17 125 VDC batteries and racks

Potentially, the most important contributions due to dependencies would come first from the correlation between 4 and 14, followed by the correlation between 8 and 17.

In the context of the probability of frequency format used in the Zion PRA report, correlation of response affects the results in two different ways. Based on some simple examples, it was found that correlation through the uncertainty factor, β_u, causes greater uncertainty and hence "spreads out" the fragility curves (see Figure 7.2-4). On the other hand, correlation through the randomness factor, β_r, causes the fragility curve for combined components to decrease for components in series and increase for components in parallel. The extreme randomness case is perfect correlation where the frequency of failure would be based on the weakest component in a series configuration or the strongest component in a parallel configuration. For the Boolean expression for core-melt failure, the significant components are in series; thus the effect of building response correlation is to increase the probability spread for uncertainty and decrease the frequency of failure for randomness. In examining the β_r and β_u terms, roughly half of the variability comes from the building response contribution (as compared to the contribution from the particular equipment or structure item). It is our judgment that if the correlation due to building response were included in the probabilistic analysis of the Boolean expression for core-melt failure, the effects on the mean frequency of core melt would be small. If the Boolean expression had been dominated by "and" symbols, this effect could have been more significant. In general, the effect of building response on the correlation of frequency failure between components should be considered.

3.2.8.2 Review of "Seismic Core-Melt Frequencies"

As discussed in our review of Section 7.2.2 (see Appendix A) we believe that the value of 5.6×10^{-6} per year for the mean frequency of core melt is reasonable. We have less belief that the lower-bound value for the 90 percent confidence interval given in the ZPSS (i.e., 3.0×10^{-8} per year) is correct. As discussed above, many factors influence this value (e.g., tails on probability and frequency distributions and dependencies). We judge that there could be a major difference in this value. Also, it is not clear which 90 percent confidence interval is being cited. Is it the one where there is 5 percent probability remaining in each tail?

We question whether the five curves shown in ZPSS Figure 7.2-4 and tabulated in Table 7.2-4 are median values, mean values, or other. It is not clear how they are located in each 20 percent probability slice.

3.2.8.3 Review of "Seismic Plant Damage State Frequencies"

Seismic sequences only contribute significantly to plant damage state SE. The Boolean expression for this state is similar to the expression for the core-melt frequency which was reviewed in the previous section. The same relationship exists between the components which contribute significantly to core melt and to the plant damage state. Thus, no additional comments are made for this section.

3.2.8.4 Potential Conservatisms and Unconservatisms

Conservatism:

1. Capacity of the service water pumps is conservative.

2. The assumption that failure of the Crib House roof will fail all six service water pumps is very conservative.

3. The capacity of the connecting piping between the Auxiliary building and the Reactor building appears to be on the conservative side.

4. The capacity of the refueling water storage tank is conservative.

Unconservatism

1. Neglecting design and construction errors and aging effects is unconservative.

2. The decision to eliminate components 2, 3, 5, 6, 7, and 13, Table 7.2-3, from further consideration may be unconservative.

3.2.9 Large LOCA: Failure of Recirculation Cooling, ALF

Point Estimation

For internally initiated accidents, this sequence is the second leading contributor to core melt. Zion's posterior distribution for the Large LOCA rate has a mean of 9.4(-4) per year and a variance of 5.74(-6), which seems consistent with the available data. Recirculation failure (R-1) pertains to low pressure recirculation and, as in the case of Small LOCA: Recirculation Failure, the assumption made in calculating the split fraction is that fan coolers are not available. Also, in the dominant sequence (p. 1.3-88), all three AC Buses are available. Zion's mean value for (R-1) of 5.19(-3) is derived on pp. 1.5-475 and 476. (Actually, p. 1.5-476 shows a mean value of 5.16 (-3). The system unreliability, $Q_{LO-HEAD}$, is dominated by 1.2 Q_{H1}, where H1 is the human error of failing to initiate switchover. Assuming Zion's stated value for the variance of Q_{H1}, namely β^2 = 1.15(-3) implies the variance of $Q_{LO-HEAD}$ should exceed 1.7(-3). The variance given, however, is 1.55(-4), so again the DPD analysis contributed to the apparent precision of Zion's estimates. As a point estimate, Zion's mean Q_{H1} of 4.03(-3) seems reasonable and so we concur with their estimated rate for this sequence.

Also noteworthy in Zion's Large LOCA analysis is the exclusion of "catastrophic reactor vessel ruptures that are beyond the capability of the ECCS" (p. 1.3-71). Large LOCA followed by recirculation failure has an estimated occurrence rate of 4.9(-6) per year. Zion adopts the WASH-1400 assumptions with respect to reactor vessel ruptures--5th and 95th percentiles of 10^{-8} and 10^{-6} per year--and concludes that adding this term in is unnecessary. If, however, the WASH-1400 values were taken as 20th and 80th percentiles, the mean rate of vessel rupture would be 4.3(-6) per year, which is not so negligible.

The following potential conservatism was identified in our review of this sequence:

Conservatism:

1. The medium LOCA initiating event frequency is conservative relative to that used in other PRAs.

3.2.10 Medium LOCA: Failure of Recirculation Cooling, ALF

The ZPSS analysis and results for this sequence are identical to their treatment of the Large LOCA: Failure of Recirculation Sequence. Our comments are the same.

3.2.11 Loss of Off-site Power: Failure of Component Cooling Water; Failure to Recover Off-site Power in Eight Hours; Failure of Containment Sprays and Fan Coolers, SE

Point Estimation

This sequence is similar to that described in Section 3.2.5 except that it also includes containment sprays and containment fan system failure. The dominant cause of failure of both fans and sprays in this sequence is loss of power from AC Buses in Unit 1. To evaluate this frequency we continue the calculation in Sections 3.2.5 and 3.2.6, adding the spray and fan failure frequencies:

Power on Buses	(.10)(CCW')	(Conditioned probability of power state, given no ES signal)(CS')(F')	
147	(.10)(3.2-2)(3.2-3)(6.8-3)(1.0)		= 7.0(-8)
148	(.10)(3.2-2)(4.5-2)(6.8-3)(1.0)		= 9.8(-7)
149	(.10)(3.2-2)(4.5-2)(6.8-2)(1.0)		= 9.8(-6)
None	(.10)(.17)(3.86-3)(1.0)(1.0)		= 6.6(-5)
	Total		7.7(-5)

Note that in the case of no power on emergency Buses the containment sprays fail with probability 1.0. This is because normally-closed MOV CS-0006 in the outlet of the diesel-driven containment spray pump requires power from an emergency Bus in order to open.

The total is the frequency of the sequence, given loss of off-site power. (Other possible AC Bus states are not shown, since they do not contribute significantly to the results.) To complete the calculation, we multiply by the frequency of the initiating events.

$$7.7(-5) \cdot 0.0614 = 4.7(-6) \quad .$$

Since this sequence involves a seal LOCA with loss of fans and sprays, the plant damage state is SE.

The following conservatisms were identified, based on the ZPSS and this analysis:

Conservatism

1. The two-pump success criterion for the CCW system in preventing a seal LOCA is potentially conservative since one pump would be sufficient if non-essential loads (such as the spent fuel pit heat exchanger) were isolated.

2. The recovery-of-off-site-power failure probabilities frequencies used in the analysis may be high because historical data primarily reflects LOP situations in which emergency diesel generators were available. Given the loss of diesel generators factored into the analysis, a more vigorous effort to recover off-site power might be made, with consequent shorter recovery times.

Statistical Confidence Limits

The model for this sequence is

$$\lambda = \phi_{11b} \cdot Q_{CCW,CS} \cdot Q_{8+}.$$

where Φ_{11b} denotes the occurrence rate of loss of off-site power, $Q_{ccw,cs}$ denotes the joint failure probability of CCW and containment spray, and Q_{8+} denotes the probability that recovery is after 8 hours. The first and last terms and their corresponding data have been discussed in previous sections. The performance of both the CCW and CS systems depends on the electric power state, so it is necessary to consider them together. (The CF system fails with probability 1.0 for each of the power states considered.)

For this sequence there are two dominant power states: Power at Bus 149 only and No Power. The following table gives simplified expressions for the power states and the conditional probabilities of CCW and CS failure.

Power on Buses	Prob. (Power State)	Q(CCW/Power State)	Q(CS/Power State)
149	.5 DG	4DG · CCWM + 3CCW2M	CS
None	$.5DG^2$	2(DG + CCWM)	1.0

All of these terms except CS have been defined in Section 3.2.2. The conditional probability of containment spray is modeled as follows:

$$CS = Q_{MOV,cs} + Q_{DE} + \lambda_{DE}$$

These terms are defined and the ZPSS-reported data are given, as follows:

$Q_{MOV,cs}$: Failure of an MOV in the CS system to operate on demand. Entry 34 in Table 1.5.1-5 shows 10 failures in 1647 demands.

Q_{DE}: Failure of a diesel-driven CS pump to start. The ZPSS data (entry 37 in Table 1.5.1-5) shows 1/183.

λ_{DE}: Hourly failure rate of the diesel-driven CS pump. Entry 38 shows two failures in 33 hours.

Combining these data via the Maximus methodology yields an assessment of CS based on effective data of 2.2 failures in 16.5 demands.

The data for loss of off-site power, discussed in Section 3.2.2, are 18 occurrences in 220 reactor-years and the Q8' hour recovery probability data are 10/42. Combining these with the CS data yields effective sequence data of .5 occurrences in 134,000 reactor-years which lead to statistical confidence limits of λ_{u95} = 3(-5)/yr., λ_{L95} = 1(-8)/yr.

3.2.12 Large LOCA: Failure of Low Pressure Injection, AEFC

Point Estimation

The large LOCA initiating event estimates have been previously discussed. Failure of the low pressure injection system is given a mean value of 1.39(-3), p. 1.3-88, while the supporting calculations (p.1.5-410) result in a mean of 4.65(-4)--another case of nonmatching because late changes in the fault tree estimates were not fed through the fault tree models if their impact was deemed unimportant. The dominant cause of this failure, by Zion's estimates, is a human error: MOV left closed after testing and not discovered in the control room. Their mean probability for this event is 3.64(-4). The hardware and maintenance terms are relatively small, using Zion's estimates and using estimates based strictly on the Zion data, so we will just consider the human error.

The error of concern is failure to reopen both MOV 8812A and B after testing. Zion's personal probability distribution for the probability of this error has a mean of 2.2(-3). This is based on using the HE Handbook for errors of omission, nonpassive tasks, short checklists, checkoffs used in which case the point estimate given is .001. Treating this as a lognormal distribution median and taking an error factor of five yields the above mean. Zion's analysis assumes the checkoff is used. We feel it is more reasonable to assume a 50 percent chance of correct use of the checkoff in which case the median becomes .5(.003) + .5(.001) = .002 which leads to a mean of 4.4(-3).

A second error is failure to correct the previous error of omission. This is assigned a mean probability of .083 based on a nonexistent histogram, so we cannot evaluate it. Note, though, that if this estimate is doubled, then the resulting probability of both errors would be estimated by 1.06(-3), which is still less than the mean value Zion used in their event tree calculations. Hence we accept this value of 1.39(-3).

Confidence Limits

The model for this sequence is

$$\lambda = \phi_1 \cdot Q_{LP-1}$$

where ϕ_1 is the Large LOCA rate and Q_{LP-1} denotes the failure probability of low pressure injection. This failure probability is dominated (in the Zion estimates) by human error. As noted above, we have no disagreements with the point estimate of 1.4(-3) for this probability. For the purpose of calculative confidence limits we will treat this estimate as being based on data of 1/700. This assessment is slightly more conservative than the ZPSS posterior distribution for this event probability.

As discussed earlier, we assume data of 0/500 for the large LOCA rate. Combining these with the Q_{LP-1} effective data yields effective system data of 0/1.8(5) yrs. The resulting confidence limits are λ_{u95} = 2(-5)/yr., λ_{L95} = 0.

The following potential conservatism and unconservatism were identified in our review of this sequence:

Conservatism

1. The Large LOCA initiating event frequency is conservative relative to that used in other PRAs.

Unconservatism

1. The fan coolers were assumed to be available to mitigate the core-melt accident (see Section 4).

3.2.13 Loss of Off-site Power: Failure of Auxiliary Feedwater; Failure of Feed and Bleed; Failure to Restore Off-site Power in 4 Hours, TEFC

In this sequence, the initiating event, loss of off-site power, is followed by loss of auxiliary feedwater and loss of feed and bleed capability, with failure to restore power in 4 hours. The loss of auxiliary feedwater eliminates the capability for secondary cooling, since without off-site power the main feedwater pumps have tripped and cannot be restored. The loss of feed and bleed capability removes the remaining option for core cooling. The frequency of the events in this sequence is described below.

Point Estimation

The loss of off-site power frequency as an initiating event is given in the ZPSS as 5.7(-2). In addition we consider the following events and frequencies from the ZPSS: turbine trip followed by the loss of off-site power, 1.3(-3), loss of main feedwater followed by loss of off-site power, 1.8(-3); and reactor trip followed by loss of off-site power, 1.3(-3). Each of these events could lead to the accident sequence in question. The sum of these frequencies is 0.061. Our analysis assumes loss of off-site power to both Zion units, which is consistent with the ZPSS analysis. (See discussion of seal LOCA in ZPSS Section 1.3.3.8.) The ZPSS analysts have indicated that their study of off-site power loss leads to the conclusion that it will nearly always affect both units.

The failure to restore off-site power in 4 hours is, in effect, the failure to restore power in 30 minutes, 60 minutes and 4 hours, and, by implication, the success in restoring power by 8 hours. The frequency of these events is assigned on the basis of plant historical data developed by the Electric Power Research Institute as follows:

EP30	(Failure to restore power in 30 minutes)	- 0.52
EP60	(Failure to restore power in 60 minutes)	- 0.38
EP4	(Failure to restore power in 4 hours)	- 0.25
EP8	(Failure to restore power in 8 hours)	- 0.10

Since we apply these frequencies sequentially in our evaluation, the following calculations are necessary.

$$\text{EP60 (Given EP30)} = \frac{0.38}{0.52} = 0.73$$

$$\text{EP4 (Given EP30 and EP60)} = \frac{0.25}{(0.52)(0.73)} = 0.66$$

$$\text{EP8 (Given EP30, EP60, and EP4)} = \frac{0.10}{(0.52)(0.73)(0.66)} = 0.60$$

$$\overline{\text{EP8}} = \text{(Given EP30, EP60, and EP4)} = 0.40$$

The probability that auxiliary feedwater (AFW) and feed and bleed (F&B) fail, given loss of off-site power, is dependent on the state of the vital AC power Buses. The calculation of AFW and F&B failure probability for each degraded power state is detailed in Sections 2.4.1.12 and 2.4.1.4. For convenience, the results of those calculations are repeated here.

Power on Buses	AFW Unavailability	F&B Unavailability
All	3.4(-5)	3.0(-3)
147 and 148	2.3(-4)	3.0(-3)
147 and 149	2.3(-4)	3.0(-3)
148 and 149	3.4(-5)	3.0(-3)
147	0.039	3.0(-3)
148	2.3(-4)	5.1(-3)
149	2.3(-4)	8.6(-3)
None	0.039	1.0

Based on the above, we consider the sequence

$$\text{EP30} \cdot \overline{\text{CCW}'} \cdot \text{EP60} \cdot \text{AFW} \cdot \text{F\&B} \cdot \text{EP4} \cdot \overline{\text{EP8}} \cdot$$

The calculation for each degraded power state is as follows:

Power on Buses	(.15)(CCW')(AFW')(F&B')	(Conditional Probability of Power state, given no ES signal
All	(.15)(1)(3.4-5)(3.0-3)(.38)	= 5.8(-9)
147 and 148	(.15)(1)(2.3-4)(3.0-3)(3.6-2)	= 3.7(-9)
147 and 149	(.15)(1)(2.3-4)(3.0-3)(3.6-2)	= 3.7(-9)
148 and 149	(.15)(1)(3.4-5)(3.0-3)(.45)	= 6.9(-9)
147	(.15)(.97)(.039)(3.0-3)(3.2-3)	= 5.4(-8)
148	(.15)(.97)(2.3-4)(5.1-3)(4.5-2)	= 7.7(-9)
149	(.15)(.97)(2.3-4)(8.6-3)(4.5-2)	= 1.3(-9)
None	(.15)(.83)(.039)(1)(3.9-3)	= 1.8(-5)
	Total	1.8(-5)

where .15 is the product of EP30, EP60, EP4, and $\overline{EP8}$. The total 1.8(-5) is the frequency of the sequence, given loss of off-site power. To complete the calculation, we multiply by the frequency of the initiating events.

$$1.8(-5) \cdot 0.061 = 1.1(-6) \quad .$$

Since this sequence is a transient and off-site power is restored in 8 hours, the containment fans and sprays would be available. Consequently, this is a TEFC sequence.

The following conservatism was identified, based on the ZPSS and this analysis:

Conservatism

1. The recovery-of-off-site-power failure probabilities used in the analysis may be high because historical data primarily reflects LOP situations in which emergency diesel generators were available. Given the loss of diesel generators factored into the analysis, a more vigorous effort to recover off-site power might be made, with consequent shorter recovery times.

Statistical Confidence Limits

The model for the dominant terms in this sequence is

$$\lambda = \phi_{11b} \cdot Q_{4,8} \cdot Q_O \cdot Q_{AFW} \quad ,$$

where the first two terms refer to the loss of off-site power for 4 to 8 hours, Q_O denotes the conditional probability that power is not available to any of the bases, and Q_{AFW} denotes the unavailability of the turbine-driven auxiliary feedwater pump. The data pertaining to ϕ_{11b}

are 18/220 reactor-years and to $Q_{4,8}$ are 6/42. The power state simplified probability expression is .5 DG^2, where the effective DG data (see Section 3.2.2) are 65/1300. The turbine-driven AFW unavailability, which includes both pump failure and maintenance unavailability, was estimated as .0494. This estimate correponds to effective data of 7.4/185.

Combining these data using the Maximus methodology yields effective sequence data of two occurrences in 3.5 reactor-years and statistical confidence limits of

$$\lambda_{u95} = 2(-6)/yr., \lambda_{L95} = 1(-7)/yr \quad .$$

3.2.14 Loss of Off-site Power: Failure of Auxiliary Feedwater: Failure of Feed and Bleed: Failure to Restore Off-site Power in 1 Hour, TEFC

In this sequence, the initiating evert, loss of off-site power, is followed by loss of auxiliary feedwater and loss of feed and bleed capability, with failure to restore power in one hour. The loss of auxiliary feedwater eliminates the capability for secondary cooling, since without off-site power, the main feedwater pumps have tripped and cannot be restored. The loss of feed and bleed capability removes the remaining option for core cooling. Therefore core cooling will not occur. The frequency of the events in this sequence are described below.

The loss of off-site power frequency as an initiating event is given in the ZPSS as 5.7(-2). In addition we consider the following events and frequencies from the ZPSS: turbine trip followed by the loss of off-site power, 1.3(-3); loss of main feedwater followed by loss of off-site power, 1.8(-3); and reactor trip followed by loss of off-site power, 1.3(-3). Each of these events could lead to the accident sequence in question. The sum of these frequencies is 0.061. Our analysis assumes loss of off-site power to both Zion units, which is consistent with the ZPSS analysis. (See discussion of seal LOCA in ZPSS Section 1.3.3.8.) The ZPSS analysts have indicated that their study of off-site power loss leads to the conclusion that it will nearly always affect both units.

The failure to restore off-site power in 1 hour is, in effect, the failure to restore power in 30 minutes and 60 minutes. The frequency of these events is assigned on the basis of plant historical data developed by the Electric Power Research Institute (Reference 3-3) as follows:

EP30 (Failure to restore power in 30 minutes) - 0.5
EP60 (Failure to restore power in 60 minutes) - 0.38
EP4 (Failure to restore power in 4 hours) - 0.25

Since we apply these frequencies sequentially in our evaluation, the following calculations are necessary.

$$\text{EP60 (Given EP30)} = \frac{0.38}{0.52} = 0.73$$

$$\text{EP4 (Given EP30 and EP60)} = \frac{0.25}{(0.52)(0.73)} = 0.66$$

$$\overline{\text{EP4}}\text{ (Given EP30, EP60)} = 0.34$$

The probability that auxiliary feedwater (AFW) and feed and bleed (F&B) fail, given loss of off-site power, is dependent on the state of the vital AC power Buses. The calculation of AFW and F&B failure probability for each degraded power state is detailed in Sections 2.4.1.12 and 2.4.1.4. For convenience, the results of those calculations are repeated here.

Power on Buses	AFW Failure Probability	F&B Failure Probability
All	3.4(-5)	3.0(-3)
147 and 148	2.3(-4)	3.0(-3)
147 and 149	2.3(-4)	3.0(-3)
148 and 149	3.4(-5)	3.0(-3)
147	0.039	3.0(-3)
148	2.3(-4)	5.1(-3)
149	2.3(-4)	8.6(-3)
None	0.039	1.0

Based on the above, we consider the sequence

$$\text{EP30} \cdot \overline{\text{CCW}'} \cdot \text{EP60} \cdot \text{AFW} \cdot \text{F\&B} \cdot \overline{\text{EP4}} \cdot$$

The calculation for each degraded power state is as follows:

Power on Buses	(.13)($\overline{CCW}$')(AFW')(F&B')	Conditional Probability of Power state, given no ES signal
All	(.13)(1)(3.4-5)(3.0-3)(.38)	= 5.0(-9)
147 and 148	(.13)(1)(2.3-4)(3.0-3)(3.6-2)	= 3.2(-9)
147 and 149	(.13)(1)(2.3-4)(3.0-3)(3.6-2)	= 3.2(-9)
148 and 149	(.13)(1)(3.4-5)(3.0-3)(.45)	= 6.0(-9)
147	(.13)(.97)(.039)(3.0-3)(3.2-3)	= 4.7(-8)
148	(.13)(.97)(2.3-4)(5.1-3)(4.5-2)	= 6.7(-9)
149	(.13)(.97)(2.3-4)(8.6-3)(4.5-2)	= 1.1(-9)
None	(.13)(.83)(.039)(1)(3.86-3)	= 1.6(-5)
	Total	1.6(-5)

where .13 is the product of EP30, EP60, EP4. The total 1.6(-5) is the frequency of the sequence, given loss of off-site power. To complete the calculation, we multiply by the frequency of the initiating events:

$$1.63(-5) \cdot 0.061 = 1.0(-6) \quad .$$

Since this sequence is a transient and off-site power is restored in one hour, the containment fans and sprays would be available. Consequently, this is a TEFC sequence.

The following conservatism was identified, based on the ZPSS and this analysis:

Conservatism

1. The recovery-of-off-site-power failure frequencies used in the analysis may be high because historical data primarily reflects LOP situations in which emergency diesel generators were available. Given the loss of diesel generators factored into the analysis, a more vigorous effort to recover off-site power might be made, with consequent shorter recovery times.

Statistical Confidence Limits

This sequence differs from that in Section 3.2.13 only in that recovery occurs between 1 and 4 hours, rather than 4 to 8 hours. In both cases the available recovery data are 6/42 so the same confidence limits result:

$$\lambda_{u95} = 2(-6)/yr., \ \lambda_{L95} = 1(-7)/yr \quad .$$

3.2.15 Event V: The Interfacing LOCA, V

Event V leads to release category 2 which, by Zion's risk estimates, is one of the dominating releases. The dominant V sequence is the joint failure of two motor-operated valves in the RHR suction path. A brief discussion of this event and the resulting estimates are given on pages 1.3-76 and 1.3-77. Conversations with the authors indicate that this discussion is inaccurate.

The situation, based on these conversations, is apparently this: After a refueling outage both valves are closed with probability 1.0, this assumption being made because the valves are interlocked and repressurization cannot occur if they are open. Failure can then occur in two ways during the subsequent year:

1. The downstream valve transfers open then the upstream valve ruptures.

2. The upstream valve ruptures, then the downstream valve ruptures.

The events (both valves transfer open) and (upstream transfers open, downstream ruptures) are excluded because it is assumed that the valve subjected to pressure cannot transfer open.

The Zion report says that the former event, (open-open), was excluded because if both valves had failed open, RCS startup would reveal this and the situation would be remedied. But the actual concern is failure during the period after successful startup, so the report statement is irrelevant except to support the assumption that both valves will initially be closed. The report also says that one of the two cutsets involving disc open and disc rupture, that is, either (open-rupture) or (rupture-open), can be excluded because at least one of the MOVs must close to allow start-up. Again this is irrelevant with respect to failure during the period after startup, but it suggests that the assumption that both valves are initially closed with probability 1.0 is suspect. Zion's discussion deals with startup failure, while the event of interest is failure during the time period (1 year assumed) following successful startup. The confusion is further confounded by the calculation of estimates for this event.

<u>Point Estimation</u>

Let λ_1 denote the rate at which "valve transfers open" occurs and let λ_2 denote the rate at which "valve rupture" occurs. Zion gives the following posterior distribution moments.

	λ_1 (transfer open)	λ_2 (rupture)
Mean (failures/hr.)	3.14 (-8)	2.66 (-8)
Variance	1.56 (-15)	4.32 (-15)

No source is given for these values on p. 1.3-76, but by looking back in the section and at Table 1.5.1-5 it is possible to infer the sources. For λ_1, the source is entry 7 in Table 1.5.1-5 (see attached excerpt). Note though that the exponent on the variance should be -14, not -15. The correct value appears to have been used in the calculations. Note also that the source for their prior distribution is N-1363 (EG&G) and WASH-1400 estimates pertaining to external leakage, a different failure mode from transfer to open. The posterior is said to pertain to transfer open/excessive leakage. We were told external leakage would be (a) detected and (b) not large enough to cause a LOCA and so was not a concern.

The λ_2 estimates do not appear in Table 1.5.1-5. However, on p. 1.3-73 the same numbers are given for check valve disc rupture with the notation: "from WASH-1400." Going to WASH-1400 one finds (Table III 2-1) a range of 10^{-9} to 10^{-7} per hour for the rate of occurrence of "External leak-rupture" in motor-operated (and all other) valves. Equating these limits to the 5th and 95th percentiles of a lognormal distribution leads to a distribution with a mean and variance equal to the above values for λ_2. Conversations with the authors confirm that this is indeed the basis for their estimates.

Consider the "rupture-rupture" event. Zion's model for this (apparently) is

$$q^2 = (\lambda_2 \tau/2)^2 \quad ,$$

where τ is the time between refuelings and is taken as one year (8760 hr.). (This model is wrong, but we'll get to that later.) That this is the Zion model can be inferred by calculating the mean and variance of $q = \lambda_2\tau/2$, given the assumed mean and variance of λ_2. We get mean (q) = 1.17(-4), var(q) = 8.29(-8), which corresponds to Zion's results on p. 1.3-77. Deriving the mean of q^2 yields

$$\text{mean}(q^2) = \text{mean}^2(q) + \text{var}(q) = (1.17 \times 10^{-4} + 8.29 \times 10^{-8}$$
$$= 9.7 \times 10^{-8}$$

(Adding the mean value of $\lambda_1\lambda_2 t^2/4$ (= 1.6(-8)) yields a total of 1.13(-7), which corresponds closely enough to Zion's 1.05(-7) p. 1.3-77).

TABLE 1.5.1-5 (continued)

SPECIALIZED COMPONENT HARDWARE FAILURE DATA

Component Description and Failure Mode	Plant-Specific				Generic		
	No. of Failures	Service Hours or Demands	Updated		Mean [$\lambda 20$]	$\lambda 80/\lambda 20$ [$\lambda 80$]	Comments - Data Sources
			Mean	Variance			
7) System: All Component Type: Motor Operated Valves Failure Mode: Transfer Open/Excessive Leakage Through Valve	0	6.95(5) hours	3.14(-8) /hour	1.56(-14) /(hour)^2	1(-7) /hour	100	N-1363 PWR Remote + MOV - No command fault. External leakage $\overline{X}$ = 1(-7)/hr W-1400 MOV's External leakage/ rupture $\lambda 5$ = 1(-9)/hr $\lambda 95$ = 1(-7)/hr
8) System: All Component Type: Air Operated Valves Failure Mode: Failure to Operate on Demand	3	1540 demands	1.44(-3) /demand	7.93(-7) /(demand)^2	9(-4) /demand	10	N-1363 PWR AOV Fail to operate. No commmand faults. $\overline{X}$ = 9(-4)/dem W-1400 AOV. Fails to operate $\lambda 5$ = 1(-4)/dem $\lambda 95$ = 1(-3)/dem
9) System: All Component Type: Air Operated Valves Failure Mode: Transfer Closed, Plugged	0	2.13(6) hours	1.12(-7) /hour	1.84(-14) /(hour)^2	[2.8(-8)] /hour	[2.8(-7)] /hour	W-1400 AOV. Plug. Failure to remain open. Used 1 dem 45 days to convert to 1 hour. $\lambda 5$ = 3(-5) = 2.8(-8)/hr $\lambda 95$ = 3(-4) = 2.8(-7)/hr
10) System: All Component Type: Air Operated Valves Failure Mode: Transfer Open, External Leakage			(NO DATA FOUND)		1(-7) /hour	100	N-1363 AOV. External leakage $\overline{X}$ = 1(-7)/hr W-1400 AOV. External leakage/ rupture $\lambda 5$ = 1(-9)/hr $\lambda 95$ = 1(-7)/hr
11) System: All, Except Auxiliary Feedwater Component Type: Pumps-Motor Driven Failure Mode: Failure to Start on Demand	3	3.138(3) demands	7.21(-4) /demand	1.91(-7) /(demand)^2	5(-4) /demand	10	N-1205 Standby system. Does not start. No command faults $\overline{X}$ = 5(-4)/dem W-1400 Electric motor. Failure to start. $\lambda 5$ = 1(-4)/dem $\lambda 95$ = 1(-3)/dem
12) System: Auxiliary Feedwater Component Type: Turbine Driven Auxiliary Feedwater Pump Failure Mode: Failure to Start on Demand	6	2.31(2) demands	2.29(-2) /demand	7.60(-5) /(demand)^2	4(-3) /demand	100	N-1205 Standby System. No command faults. Does not start. $\overline{X}$ = 4(-3)/dem $\lambda 80/\lambda 20$ based on engineering judgment.

NOTE: 1.23(4) indicates 1.23×10^4
W-1400: WASH-1400, Table III 2-1.
N-1363: NUREG/CR-1363, Table 23, page 63.
N-1205: NUREG/CR-1205, Table 14, page 35.
1362: NUREG 1362, Table page 51.

Now suppose we take 10^{-9} and 10^{-7} to be the 20th and 80th percentiles of λ_2. Then,

$$\text{mean}(\lambda_2) = 4.28(-7) \qquad \text{var}(\lambda_2) = 3.36(-10)$$
$$\text{mean}(q) = 1.87(-3) \qquad \text{var}(q) = 6.45(-3)$$
$$\text{mean}(q^2) = 6.45(-3)$$

so an apparently minor change in assumed percentiles leads to a five orders of magnitude increase in the estimated probability of this event. Part of the increase in estimating q^2 is due to the error in approximating $1 - \exp(-\lambda^2\tau/2)$ by $\lambda_2\tau/2$, but a substantial portion of the conservatism is due to using the posterior mean as an estimate of q^2. We did a 5000-run Monte Carlo and obtained 3.9(-4) as an estimate of the mean of $(1 - \exp(-\lambda_2\tau/2)^2)$ which is what Zion is approximating by $(\lambda_2\tau/2)^2$, so there are still four orders of magnitude attributable to the seemingly innocent change from 5/95 bounds to 20/80. This change results from the change in var (λ) and the way in which var (λ) contributes to Zion's mean value, their point estimate (see our Section 2.6).

Let μ and σ denote the mean and standard deviation of $\ln(\lambda)$. For Zion's assumptions $\sigma = 1.40$. Using the 20/80 assumptions yields $\sigma = 2.74$. In both cases $\mu = \ln(10^{-8}) = -18.42$. The mean of the lognormal distribution is $\exp(\mu + \sigma^2/2)$. By increasing σ from 1.40 to 2.74, the mean is increased by a factor of 16. The variance of a lognormal distribution is $\exp(2_\mu + \sigma^2)(\exp(\sigma^2) - 1)$. Here the increase in α results in increasing the variance by a factor of 77,719. In the above calculations for mean (q^2), var (λ) overshadows $\text{mean}^2(\lambda)$ and thus the large difference is obtained. It is disturbing that important results are so sensitive to the assumptions made.

Figure 3.2.15-1 shows the two lognormal distributions just discussed, drawn on a linear scale. Innocent-looking normal distributions on a log-scale transform to greatly skewed distributions on a linear scale. Whether they accurately depict anybody's state of knowledge is open to question. From this figure, the large increase in the variance of λ using the 20/80 assumptions, rather than 5/95, is not at all apparent. However, the 20 percent of the distribution beyond $\lambda = 10^{-7}$ in the former case versus the 5 percent in the latter exerts considerable leverage.

Now consider Zion's model. For the (rupture-rupture) event the downstream valve is not at risk until the upstream valve has ruptured. If it is assumed that the rupture rate of the downstream valve is the same as that of the upstream (as Zion implicitly did), then the probability of both valves rupturing in a period of 1 year is approximately

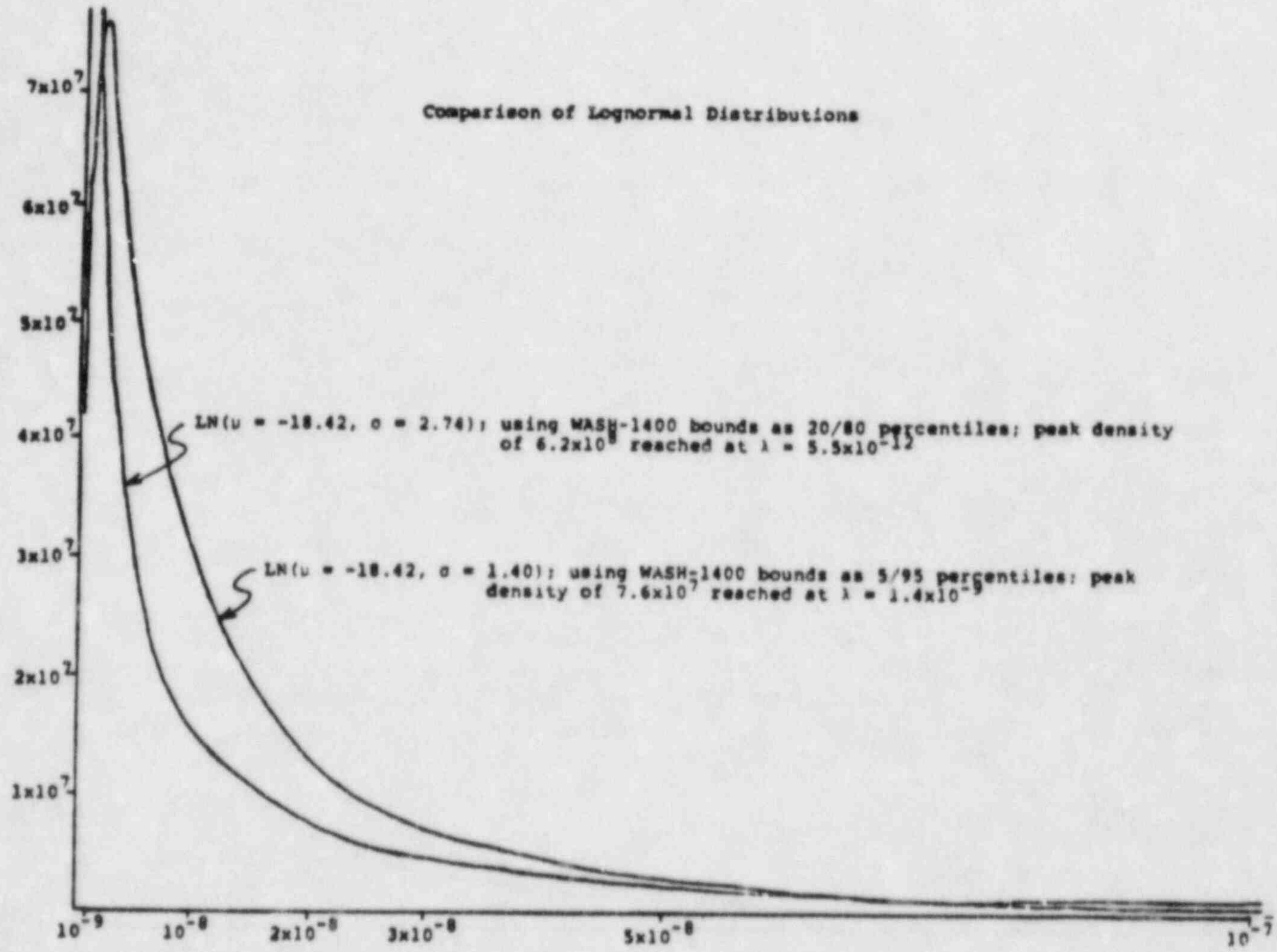

Figure 3.2.15-1. Comparison of Lognormal Distribution

$(\lambda_2\tau)^2/2$, where τ = 8760 hrs. This is a factor of two larger than Zion's assumed model. Similarly, for (open-rupture) the downstream valve must transfer open before the upstream valve ruptures, so this event occurs with probability $\lambda_1\lambda_2\tau^2/2$, approximately (p. V-43 of WASH-1400 gives a misprinted derivation of this result; no doubt there are other references). Thus the model should be

$$V = \frac{(8760)^2}{2} \quad \lambda_1\lambda_2 + \lambda_2^2$$

and Zion's mean value should be doubled and its variance quadrupled, even if their estimated rates are accepted.

There are thus several concerns about the Zion analysis:

- The textual description does not match the model.
- The model is not given explicitly.
- The model is off by a factor of two (or maybe more; no consideration is given to non-independent, or "other" failures).
- The estimation procedure used does not conform to their stated methodology.

This does not mean we can say their final estimate is high or low with respect to reality. Our concern is that a dominant source of the Zion plant risk, by Zion's estimate, is so capriciously analyzed and explained.

Now, what of alternative estimates? In response to similar criticisms of the Indian Point analysis, PLG developed a new model and analysis. These would seem to apply also to Zion, but no such analysis has been supplied. However, it is possible to at least approximate that analysis.

The new model is

$$P(V) = [1 - e^{-\lambda T}(1 + \lambda T)] + 2P(1 - e^{-\lambda T})$$
$$\approx (\lambda T)^2/2 + 2P\lambda T$$

where P = probability of valve failure to close on demand in an undetected manner
λ = valve rupture failure rate (hr^{-1})
T = time between refuelings (hr) (assumed to be 8,760 hrs = 12 mos)

Note that the first term, for rupture-rupture, is double that in the ZPSS model and that both combinations of left-open/rupture are now considered. Also, valve-left-open is now modeled as a demand failure.

In the revised IPPSS analysis, λ has a posterior mean of 1.2(-8) occurrences/hour and a posterior variance of 3.7(-15). The posterior mean of P is 5.8(-5) for IP-2 and 3.8(-5) for IP-3. These latter results were obtained, essentially, by multiplying the estimated MOV failure-to-operate-on-demand probabilities for the two units by 19/781, which was the fraction of industry-wide valve failures deemed undetectable. The Zion posterior mean for valve failures is 1.6(-3), which would lead by the same multiplication to a Zion posterior mean for P of 3.9(-5). Then a revised ZPSS analysis, by their methodology, would yield a posterior mean for P(V) of 1.6(-7), as opposed to the ZPSS result of 1.05(-7), obtained by an incorrect model and inappropriate data.

Statistical Confidence Limits

The available data permit a statistical analysis of the V sequence. The data pertaining to λ, the valve rupture rate, industry-wide, are zero occurrences in 7.0(7) hrs. The Zion MOV failure data are 14/11,310 and the data pertaining to the probability a failed valve is undetectable are 19/781. Combining these data via the Maximus methodology and the above model yields an assessment based on effective data of zero occurrences in 2.1(7) demands. In particular an upper 95 percent statistical confidence limit on P(V) is $P(V)_{95}$ = 1.4(-7). Even though V does not occur at a constant rate, under the above model, this probability can also be regarded as the annual rate of occurrence for the sake of comparability to other accident sequences. These results suggest that the ZPSS value of 1.0(-7) is a reasonable point estimate to use in subsequent calculations.

3.2.16 Other ZPSS Dominant Sequences

A number of accident sequences which appeared in the ZPSS dominant accident sequence list (ZPSS Table 8.10-1, which was repeated earlier in Section 3) have been omitted from the foregoing discussion because their importance to plant damage state frequencies has diminished. This is primarily due to the fact that they have been supplemented by sequences which, in our analysis, have higher frequencies of occurrence. Included in this group are the sequences which the ZPSS numbered 5, 6, 7, 8, 10, 11, 12, 13, 14, and 15. Our review of these sequences resulted in frequency estimates which were substantially lower for sequences 7 and 8. In the assessment of these spurious safety injection sequences, it appears that the ZPSS assumed that the pressurizer safety valves would open and remain open. Subsequent to the ZPSS analysis, the PORV block valves were changed from normally closed to normally open. Under these conditions, the PORVs rather than the safety valves would open. If the PORVs stuck open, the plant operators could respond by closing the PORV block valves. Our estimate of these revised sequences results in a frequency which is lower by at least an order of magnitude. Sequences 11 through 13 of Table 8.10-1 involve loss of off-site power, loss of emergency AC Buses 148 and 149, failure of auxiliary feedwater, and failure of feed and bleed. Our reanalysis of these sequences resulted in somewhat lower values, because of the change in PORV block valve status mentioned above and because recent plant maintenance data leads to a lower value for auxilary feedwater system availability. Because of the decreased importance of the sequences listed, in our estimation, a detailed discussion is not warranted.

REFERENCES

3-1. Handbook for the Calculation of Lower Statistical Confidence Bounds on System Reliability, Maximum, Inc., McLean, VA, February 1980.

3-2. Review and Evaluation of the Indian Point Probabilistic Safety Study, NUREG/CR-2934, December 1982.

3-3. Loss of Off-site Power at Nuclear Power Plants: Data and Analysis, EPRI NP-2301, Interim Report, March 1982.

3-4. Summary of NRC Staff and Consultants Questions on the Zion Probabilistic Safety Study - Commonwealth Edison Response, 1982.

4.0 Special Issues

4.1 Core-Melt Interactions with Containment Systems

As mentioned in Section 2.2.1, the Zion event trees imply that the containment spray system and fan cooler system may be utilized to protect the containment from overpressure during a core-melt accident. The fault tree analysis of these systems also assumes that the system reliability will not be degraded due to the adverse environment within containment following a core melt. In this section we will investigate the effect that not giving credit for these systems has on the ZPSS plant damage state estimates.

Following a core meltdown, the fan cooler system may possibly fail by one or a combination of the following mechanisms:

1. Cable or instrumentation failure due to containment hydrogen burns.
2. Cable or instrumentation failure due to radiation exposure.
3. Plugging of fan cooler filters or cooling coils due to aerosol generation.

The ZPSS analysis team does not feel aerosol plugging is a likely failure mechanism because the amount of aerosols reaching the coolers should be insignificant since most small aerosols (2 to 4 micron) will be scrubbed out in the water in the reactor cavity and larger aerosols (100 micron --1 mm) will fall out due to gravity before reaching the fans.[4-1] Also, they believe that temperatures resulting from hydrogen burns would not be high enough to fail fan cooler components due to the short duration of the burn and the thermal inertia of the components.[4-1] (The ZPSS team did not formally respond to the radiation concern.)

Though the preceding seem like good reasons, we did not attempt to resolve this issue due to the limited time available to perform this review, and the fact that the issue is currently being addressed in several NRC and Sandia equipment qualification research programs. Rather, a sensitivity analysis was performed which investigated the effect that assuming fan cooler failure has on the plant damage states.

If it is assumed the fan coolers will fail during a core melt and the containment spray injection system is not available because the RWST cannot be refilled (see discussion in Section 2.2.1.), the following changes to the ZPSS damage states are made:

1. SEFC becomes SEC,
2. SEF becomes SE,
3. SLFC becomes SL,
4. SLF becomes SL,
5. SLC becomes SL,
6. TEFC becomes TEC,
7. TEF becomes TE,
8. AEFC becomes AEC,
9. AEF becomes AE,
10. ALFC becomes AL,
11. ALF becomes AL, and
12. ALC becomes AL.

As discussed in Section 2, these plant damage states can be combined into plant damage state groups as follows:

1. Early core melt with containment cooling
2. Early core melt without containment cooling
3. Late core melt with containment cooling
4. Late core melt without containment cooling
5. Containment bypass before core melt.

These groupings have been adopted for the comparisons shown in Table 4.1-1.

The frequencies listed in the first column of Table 4.1-1 assume the fan coolers are capable of operating in a post core-melt environment and represent our best estimate frequencies, as discussed in Chapter 5. The frequencies listed in the second column were calculated with the assumption assuming the fans fail following a core melt.

We have recently received preliminary information that the Zion plant intends to develop a procedure for refilling the RWST to allow for continued spray operation in the recirculation phase (see discussion in Section 2.2.1). If complete credit is given for RWST refill, the plant damage state frequencies listed in Table 4.1-2 are applicable.

4.2 Feed and Bleed Capability

The ZPSS gave credit for post shutdown decay heat removal via feed and bleed (FB) core cooling. FB would be utilized during small LOCAs and transients if the auxiliary feedwater system (i.e., the normal decay heat removal system) was unavailable. Initiation of FB at Zion requires the operator to:

1 Recognize that auxiliary feedwater and secondary heat removal has failed.

2. Start a safety injection pump (if pressure is low enough).

TABLE 4.1-1

Comparison of Zion Damage States With and Without the Availability of the Containment Fan Coolers Following a Core-Melt Accident (Events/Reactor Yr.) (No RWST Refill Credit Given)

NRC Defined Plant Damage States	Fans Potentially Available (Taken from Table 5.2-4 of this Report)	Fans Not Available
Early core melt with containment cooling	3.3(-4)	3.3(-4)
Early core melt without containment cooling	1.1(-5)	1.1(-5)
Late core melt with containment cooling	2.7(-5)	0
Late core melt without containment cooling	1.0(-7)	2.7(-5)

TABLE 4.1-2

Comparison of Zion Damage States With and Without the Availability of the Containment Fan Coolers Following a Core-Melt Accident (Events/Reactor Yr.) (Complete RWST Refill Credit Given)

NRC Defined Plant Damage States	Fans Potentially Available	Fans Not Available
Early core melt with containment cooling	3.3(-4)	3.3(-4)
Early core melt without containment cooling	1.1(-5)	1.1(-5)
Late core melt with containment cooling	2.7(-5)	2.7(-5)
Late core melt without containment cooling	1.3(-8)	<1(-7)

3. Open both pressurizer power operated relief valves.

4. Verify that adequate heat removal is taking place.

FB is currently not a fully accepted core cooling method at the NRC. We have been asked to assess the effect that giving credit for FB has on the core-melt frequency and on the risk calculated in the ZPSS. Before presenting the quantitative results, it should be noted that Zion operators have received FB simulator training, and that a FB procedure (EOP-11) exists.

If it is assumed that feed and bleed cooling is not possible, one replaces the revised ZPSS probabilities for event tree events OP-1 and OP-2 with 1.0. This was done for the dominant accident sequences for each event tree and includes the affect of other significant findings of this report. The "no-feed and bleed" dominant accident sequences are summarized in Table 4.2-1. As can be seen from the table, assuming feed and bleed is not possible primarily affects plant damage state TEFC.

It should be noted that we feel that feed and bleed core cooling should be given credit. Recent TRAC computer runs at Los Alamos suggest that it is a viable core cooling option for Zion.[4-2]

4.3 Reactor Coolant Pump Seal LOCA

Several of our revised dominant internal and external accident sequences involve reactor coolant pump (RCP) seal failure. Seal failure is assumed to occur following failure of the redundant means of providing seal cooling (i.e., charging system and component cooling system) and is predicted in the ZPSS to lead to a 1200 gpm LOCA at 30 minutes. The reason that seal LOCAs appear in so many dominant sequences is because failure of the component cooling water system or AC power causes common mode failure of the seal cooling systems and the emergency core cooling safety injection pumps. If, however, a seal LOCA did not occur following loss of seal cooling, the reactor coolant system would not lose inventory and the safety injection pumps would not be required. With an intact reactor coolant system, decay heat could be removed with the AC independent turbine driven auxiliary feedwater pump via the steam generators. In this section, we assume that a seal LOCA will not occur following a loss of seal cooling and requantify the Zion dominant accident sequences.

We suspect that the seal LOCA may not occur for two reasons. One, the ZPSS 1200 gpm assumption was based on a very simplistic bounding analysis.[4-1] Two, an experiment

TABLE 4.2-1

Zion Sequences Predominantly Affected by The No-Feed and Bleed Assumption

Accident Sequence	With Viable Feed and Bleed	Without Feed and Bleed	Plant Damage State
Loss of main feedwater and loss of auxiliary feedwater (ET7, Sequence 9)	5.3(-7)	1.8(-4)	TEFC
Loss of off-site power and loss of auxiliary feedwater (LOP event tree 11b Sequences 9 and 10)	2.0(-8)	6.6(-6)	TEFC

Note: In our analysis of sequences not initiated by loss of main feedwater or loss of off-site power, credit was given for main feedwater recovery with a probability of failure ≈0.01. As a result, the feed and bleed issue did not have significant impact on sequences initiated by turbine trip, reactor trip or MSIV closure.

performed on a Byron Jackson RCP showed that significant leakage did not occur for 56 hours following interruption of seal cooling to a static RCP seal.[4-3] We recognize that Byron Jackson RCP seals are not identical to Westinghouse RCP seals. However, similarities do exist which might indicate that Westinghouse seals would not leak significantly.

The following discussion describes the impact of the assumption (that loss of RCP seal cooling does not cause a seal LOCA) on the dominant accident sequence (see Figure 3.1-1):

<u>Sequence 1</u> - The ~2(-4) unavailability of component cooling water is multiplied by the failure rate for the auxiliary feedwater system 3.4(-5), since auxiliary feedwater failure would have to occur for this sequence to result in core melt. The resulting frequency is 6.8(-9). Note also that the plant damage state would change from SEFC to TEFC.

<u>Sequence 3</u> - The frequency is multiplied by 3.3(-4) for auxiliary feedwater system failure (summed over all power states for all LOP). The resulting frequency is 1.5(-8) and the plant damage state changes from SEFC to TEFC.

<u>Sequence 4</u> - The frequency is multiplied by 3.3(-4) for auxiliary feedwater system failure. The resulting frequency is 1.3(-8) and the plant damage state changes from SEFC to TEFC.

<u>Sequence 5</u> - The frequency is multiplied by 3.3(-4) for auxiliary feedwater system failure. The resulting frequency is 5.9(-9) and the plant damage state changes from SEC to TEC.

<u>Sequence 7</u> - The frequency is multiplied by 3.3(-4). The resulting frequency is 2.6(-9) and the plant damage state changes from SEFC to TEFC.

<u>Sequence 11</u> - The frequency is multiplied by 3.3(-4). The resulting frequency is 1.4(-9) and the plant damage state changes from SE to TE.

<u>Sequence 8</u> - The frequency is multiplied by .039. The resulting frequency is 3.1(-7) and the sequence changes from SE to TE.

Table 4.3-1 compares the estimated plant damage state frequencies assuming that component cooling water failure results in a reactor coolant pump seal LOCA with frequencies assuming that a seal LOCA would not occur. Since inadequate evidence is available to support that a seal LOCA will not occur, we conservatively assume that the seal LOCA will result from loss of AC or CCW and therefore the frequencies in the first column represent our best estimates.

In Section 5.2, we summarize the effect that the assumption of the RCP seal LOCA has on our revised plant damage states.

4.4 Testing of the Room Cooling System

At Zion, pump room cooling is provided by supplying service water to cooling units and circulating air through these units with motor-operated fans. The motor circuitry is arranged so that the cooling fans in a pump room are energized whenever a pump in the room is operating. The need for room cooling arises, in part, from the fact that many of the pumps at Zion are installed in small rooms where the opportunity for natural heat dissipation is limited.

At the time of our review, it was noted that the Zion procedures did not include inspection or testing of room cooling system function. We were advised by plant personnel, however, that a room cooling test was in the process of being incorporated. On this basis, our analysis assumed testing of the room cooling system, concurrent with the established monthly pump tests.

For the purpose of evaluating the consequences of room cooling failure, we assessed the impact on the residual heat removal pumps. These pumps were selected for two reasons. First, we assume that room cooling is not essential to the pump rooms during the injection phase, following a LOCA, since cold water will be supplied to the pump suctions. Second, the residual heat removal pumps provide suction supply in the recirculation phase to the safety injection pumps, the charging pumps and the containment spray headers. We conclude that, lacking room cooling, the residual heat removal pumps would fail in the recirculation phase, sometime after the suction supply was switched from cold to hot water. On this basis, the consequence of room cooling failure would be late core melt for any loss of coolant accident initiator.

In the absence of periodic testing for room cooling, the potential for system failure is high. We assess this potential failure rate by integrating an hourly failure rate for motor-driven fans of 1.0(-5)[4-4], over a 40-year period of plant operation and averaging the result as follows:

TABLE 4.3-1

Comparison of Zion Plant Damage State Frequencies With and Without the Assumption that CCW Failure Results in Reactor Coolant Pump Seal LOCA (Events/Reactor Yr.)

Plant Damage States	If AC or CCW Failure Causes Seal LOCA	If AC or CCW Failure Does Not Cause Seal LOCA
AEFC	1.9(-6)	N.C.
AEF	1.9(-10)	N.C.
AEC	8.2(-9)	N.C.
AE	1.1(-11)	N.C.
ALFC	--	N.C.
ALF	9.8(-6)	N.C.
ALC	--	N.C.
AL	4.0(-10)	N.C.
SEFC	3.0(-4)	<1.0(-6)
SEF	5.5(-9)	N.C.
SEC	1.9(-5)	<1.0(-7)
SE	1.0(-5)	<1.0(-7)
SLFC	--	N.C.
SLF	1.7(-5)	1.6(-5)
SLC	--	N.C.
SL	1.0(-7)	N.C.
TEFC	1.0(-5)	1.1(-5)
TEF	1.6(-9)	N.C.
TEC	9.3(-7)	1.2(-6)
TE	7.7(-7)	<1.5(-6)
V	1.1(-7)	N.C.
	3.7(-4)	4.3(-5)

N.C. = No Change

Failure of two residual heat removal pump room coolers

Probability of failure at time t = $(1-e^{-\pi t})^2$

where failure rate π is assumed constant,

$$\text{Average probability in time } T = \frac{\int_0^T (1-e^{-\Pi t})^2 dt}{T}$$

$$= 1 - \frac{2}{\pi T}(1 - e^{-T\pi}) + \frac{1}{2\pi T}(1 - e^{-2T\pi}) \quad .$$

For π = 1.0(-5) per hour, T = 40 years (350,400 hours) the average probability of failure is 0.59.

Using the above result, we evaluate the accident sequences as follows:

Plant Damage State SLF

Small LOCA (.0354) x Failure of recirc. (.59) = 2.1(-2)

Plant Damage State ALF

Medium LOCA (9.4 E-4) x Failure of recirc. (.59)

= 5.6(-4)

Large LOCA (9.4 E-4) x Failure of recirc (.59)

= 5.6(-4)

Plant Damage State SL

Small LOCA (.0354) x recirc. failure (.59) x Fan Failure

(3.1-6) = 6.5(-8)

Plant Damage State AL

Large or Medium LOCA (1.88 E-3) x recirc. failure (.59)

x Fan Failure (3._-6) = 3.4(-9)

These results are tabulated below.

Plant Damage States	With Monthly Test of Room Cooling	With Room Cooling Untested
SLF	1.7(-5)	2.1(-2)
SL	1.0(-7)	1.7(-7)
ALF	9.8(-6)	1.1(-3)
AL	4.0(-10)	3.8(-9)

Changes to other damage state frequencies are not significant.

4.5 Concurrent Sequences

In Section 3 of this report, we discuss several dominant accident sequences which involve loss of component cooling water, either as an initiating event or following loss of off-site power. Loss of component cooling water is important at Zion because it leads to a reactor coolant pump seal LOCA, and causes failure of the RCS makeup capability. (See discussion in Section 3.2.1.) The sequences involving loss of component cooling water become even more significant because this system is common to both Unit 1 and Unit 2 at Zion. Thus, failure of the component cooling water system would result in core melt at both units.

It is important to note, however, that the plant damage states and frequencies arising from these sequences are not necessarily the same at Unit 1 and Unit 2. This fact could have a significant impact on the timing of containment failure (if any) between the two units, and therefore, the overall risk. For this reason, further examination of the dominant sequences is in order. The examination of the sequences involving CCW failure has two elements:

1. Determining what percentage of a given plant damage state at Unit 1 is represented by each plant damage state at Unit 2.

2. Applying these percentages to the applicable sequences and summing the results across the plant damage states.

4.5.1 Determining Plant Damage State Percentages

The evaluation of component cooling water system failure, given loss of off-site power is detailed in Section 2.4.1.10. This evaluation addresses four cases.

Case 1: No power is available at the Unit 1 AC buses
Case 2: Power is available at one AC bus of Unit 1
Case 3: Power is available at two AC buses of Unit 1
Case 4: Power is available at all AC buses of Unit 1

For these cases, the plant damage state at Unit 2 may be SEFC, SEF, SEC, or SE, depending on the availability of the Unit 2 emergency AC buses which would provide power to the containment spray and fan cooler systems (and the probability of spray and fan failure, given power). The analysis in Section 2.4.1.10 assessed the probability of CCW failure for each of the four cases. The equations which were used to examine these cases in Section 2.4.1.10 are reexamined here for their implications with regard to the status of Unit 2. Note that in the following calculations the failure probabilities of containment fans and sprays in degraded power states are applied. In addition, some of the four cases must be subdivided because of the swing diesel generator. Thus, for example, if only one bus is available at Unit 1, we must ask whether it is Bus 147 or Bus 148 (or 149). If it is 147, Bus 247 cannot be energized, but if it is 148 (or 149), then there is a 90 percent probability that 247 has power (see Section 2.4.1.1).

Case 1: No Power at Unit 1

The equation for CCW failure is:

$$Q_{LS} = 2(DGFS + DGM) + 2(CCWP + CCWM) = 0.17$$

where Q_{LS} is the probability of loss of RCP seal cooling (i.e., loss of CCW).

The values contributed by the terms in the equation are
first term = 0.104
second term = 0.065

First, let us consider the situation in which Bus 247 is not available. Note that in Case 1 we assume that three CCW pumps are unavailable due to lack of AC power at Unit 1. This means that two pumps are potentially available, given power on Buses 248 and 249, at Unit 2. If either of these two pumps is unavailable (due to pump maintenance or failure to start or to lack of power on its associated AC bus) the CCW system will fail. The first term in the Case 1 equation, 2(DGFS + DGM), addresses the probability that one or the other of the two AC buses will lack power because of diesel generator unavailability. If this occurred, power would be available at only one AC bus of Unit 2 which feeds a CCW pump. Therefore, the Unit 2 containment fans would fail with probability 1.0, and the probability of Unit 2 containment spray system failure is 6.8(-2) if power is available on AC Bus 249 or 6.8(-3) if power is available on AC Bus 248 (see Table 2.4-1). We assume that the probability of power on Bus 249 is the same as the probability of power on Bus 248, i.e., 0.5. With this in mind, we find, for the first term of the equation, the following damage state probabilities:

$$P_{SE} = 0.1 \left[\frac{.0502}{.17}(6.8\text{-}2) + \frac{.0502}{.17}(6.8\text{-}3)\right] = 2.2(-3)$$

and

$$P_{SEC} = 0.1 \left[\frac{.0502}{.17}(1\text{-}6.8\text{-}2) + \frac{.0502}{.17}(1\text{-}6.8\text{-}3)\right] = 0.0569$$

where .0502 is half of the 0.104 failure frequency contributed by the first term of the equation, and 0.1 is that fraction of the time when Bus 147 is unavailable and Bus 247 is unavailable as well.

The second term of the Case 1 equation addresses the probability that one CCW pump will fail and, therefore, implies that power is available at Buses 248 and 249 of Unit 2. The failure frequency contributed by this term is 0.065. With these buses available at Unit 2, fan coolers are potentially available (failure frequency = 9.2-5) and sprays are potentially available (failure frequency = 4.6-4).* Under these conditions the calculation of damage state probabilities applicable to the second term of the equation is:

$$P_{SEFC} = 0.1 \left[\frac{.065}{.17}\right]\left[1\text{-}9.2(-5) \cdot 1\text{-}4.6(-4)\right] = 0.0382$$

$$P_{SEF} = 0.1 \left[\frac{.065}{.17}\right]\left[1\text{-}9.2(-5) \cdot 4.6(-4)\right] = 1.8(-5)$$

$$P_{SEC} = 0.1 \left[\frac{.065}{.17}\right]\left[9.2(-5) \cdot 1\text{-}4.6(-4)\right] = 3.5(-6)$$

$$P_{SE} = 0.1 \left[\frac{.065}{.17}\right]\left[4.6(-4) \cdot 9.2(-5)\right] = 1.6(-9)$$

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* We assume that system failure probabilities given power on Buses 248 and 249 are the same as system failure probabilities given power on Buses 148 and 149 (see Table 2.4-1).

and summing the plant damage state proportions for the two terms of the equation for Case 1, we have for the situation with Bus 247 unavailable

$$P_{SE} = 2.2(-3)$$

$$P_{SEC} = 0.0569$$

$$P_{SEF} = 1.8(-5)$$

$$P_{SEFC} = 3.82(-2)$$

That is, 5.7 percent of the component cooling water failures arising from Case 1 would result in an SEC damage state, 3.8 percent in an SEFC. Note that those probabilities sum to approximately 0.1 which is that fraction of the time that Bus 247 is unavailable, given that Bus 147 is unavailable.

To ascertain the total plant damage state probabilities at Unit 2 for Case 1, we must also consider the situation where Bus 247 is available, which occurs 90 percent of the time when 147 is not. This availability does not affect the CCW failure probabilities because Bus 247 powers no CCW pump, but it does affect the availability of the fan and spray systems at Unit 2.

Once again, we separate the Case 1 CCW failure equation into the same two terms in which for one, either Bus 248 or 249 is available (as well as 247 in this situation), and for the other term, both Buses 248 and 249 are available (as well as Bus 247). The first term yields probability of Unit 2 plant damage states of

$$P_{SE} = (0.9)\left[\frac{.0502}{.17}\right]\left[(1.2\text{-}4)(1.1\text{-}2) + (4.6\text{-}4)(1.1\text{-}2)\right] = 1.7(-6)$$

$$P_{SEC} = (0.9)\left[\frac{.0502}{.17}\right]\left[(1.1\text{-}2)(1\text{-}1.2(-4)) + (1.1\text{-}2)(1\text{-}4.6(-4))\right]$$

$$= 5.8(-3)$$

$$P_{SEF} = (0.9)\left[\frac{.0502}{.17}\right]\left[(1.2\text{-}4)(1\text{-}1.1(-2)) + (4.6\text{-}4)(1\text{-}1.1(-2))\right]$$

$$= 1.5(-4)$$

$$P_{SEFC} = (0.9)\left[\frac{.0502}{.17}\right]\left[(1-1.2(-4))(1-1.1(-2)) + (1-4.6(-4))(1-1.1(-2))\right] = 0.526$$

where 0.17 is the CCW failure probability for Case 1, .0502 is half the contribution of the first term in the CCW failure equation (half because 248 or 249 availability is equally likely), 0.9 is the probability that Bus 247 is available, given 147 is not, and the fan and spray failure probabilities are taken from Table 2.4-1.

For the second term of the Case 1 CCW failure equation, all three Unit 2 buses are available, and hence it is highly probable that the fans and sprays will be too (see Table 2.4-1). Therefore, plant damage state SEFC is most likely at Unit 2, and

$$P_{SEFC} = (0.9)\ \frac{.065}{.17} = 0.344 \quad .$$

Thus, for Case 1 and the situation where Bus 247 is available, the plant damage state conditional probabilities for Unit 2 are

$P_{SE} = 1.7(-6)$

$P_{SEC} = 5.8(-3)$

$P_{SEF} = 1.5(-4)$

$P_{SEFC} = 0.87$.

The final Unit 2 plant damage states arising from Case 1 are simply the sum of the two situations:

$P_{SE} = 2.2(-3)$

$P_{SEC} = 6.3(-2)$

$P_{SEF} = 1.7(-4)$

$P_{SEFC} = 0.91$.

Case 2: Power Available at One AC Bus of Unit 1

The equation for CCW failure is

$$Q_{LS} = (DGFS^2 + 2DGFS \cdot DGM) + \left[4(DGFS + DGM) \cdot (CCWP + CCWM)\right]$$

$$+ 3(CCWP^2 + 2\ CCWP \cdot CCWM + CCW2M) = 0.032$$

and the values contributed by the terms of the equation are

first term = 1.5(-3)
second term = 6.3(-3)
third term = 2.3(-2) .

In Case 2, we have two entirely different situations: the bus available at Unit 1 is Bus 147, and it is Bus 148 or Bus 149. The reasons these must be considered separately is that the conditional probabilities of these situations is different (see Section 2.4.1.1), and that in the first situation, 147 is available, Bus 247 cannot be, but in the latter situation, 148 or 149 is available, Bus 247 may or may not be available.

Bus 147 Available

Let us first consider the situation in which the bus available at Unit 1 is in fact 147. The first term of the Case 2 equation ($DGFS^2 + 2DGFS \cdot DGM$) addresses the probability that both Buses (248 and 249) at Unit 2 would fail, and, therefore no fans or sprays would be available. For this reason all of the contribution of the first term is to damage state SE

$$P_{SE} = \frac{1.5-3}{3.2-2} = .049 \quad .$$

The second term of Case 2 implies power available at either 248 or 249 in Unit 2. For this term the SE state would require failure of sprays, so

$$P_{SE} = \frac{3.15(-3)}{.032}(6.8-3) + \frac{3.15(-3)}{.032}(6.8-2) = 7.4(-3)$$

and

$$P_{SEC} = \frac{3.15(-3)}{.032}(1-6.8-3) + \frac{3.15(-3)}{.032}(6.8-2) = 0.19$$

where 3.15(-3) is half of the 6.3(-3) contribution of the second term.

The third term implies power available at both the 248 and 249 buses of Unit 2. For this term, the SE state would require the failure of both fans and sprays:

$$P_{SE} = \frac{.023}{.032}\left[9.2(-5) \cdot 4.6(-4)\right] = 3.0(-8) \quad .$$

$$P_{SEF} = \frac{.023}{.032}\left[1\text{-}9.2(-5) \cdot 4.6(-4)\right] = 3.3(-4) \quad .$$

$$P_{SEC} = \frac{.023}{.032}\left[1\text{-}4.6(-4) \cdot 9.2(-5)\right] = 6.6(-5) \quad .$$

$$P_{SEFC} = \frac{.023}{.032}\left[1\text{-}4.6(-4) \cdot 1\text{-}9.2(-5)\right] = 0.718 \quad .$$

Summing the probabilities for the three terms for this situation of Case 2, we have

$$P_{SE} = 0.06$$

$$P_{SEC} = 0.19$$

$$P_{SEF} = 3.3(-4)$$

$$P_{SEFC} = 0.72 \quad .$$

Thus, for example, 72 percent of the time in which CCW failure occurs given power available for Unit 1 only at Bus 147, we have plant damage state SEFC at Unit 2.

Bus 148 or 149 Available

Now let us consider the situation in which the bus available at Unit 1 is either 148 or 149 (both equally likely). In this circumstance, the CCW failure equation still identifies the probabilities of Bus 248 and 249 availabilities. Also, 10 percent of the time Bus 247 will not be available, and 90 percent of the time, it will. When Bus 247 is not available, the Unit 2 plant damage state probabilities are simply those for the situation when the available bus at Unit 1 was 147, but multiplied by 0.1 to reflect the 247 unavailability. Thus,

$$P_{SE} = 6.0(-3)$$

$$P_{SEC} = 1.9(-2)$$

$$P_{SEF} = 3.3(-5)$$

$$P_{SEFC} = 7.2(-2) \quad .$$

When Bus 247 is available, the first term of the Case 2 CCW failure equation has power available at neither Bus 248 nor 249. Hence, the fan failure probability is 1.0 (see Table 2.4-1), and the probabilities of Unit 2 damage states SEF and SEFC are 0.0 in this circumstance. For the other two Unit 2 damage states we have

$$P_{SE} = (0.9)\ \frac{1.5-3}{3.2-2}\ (6.8-3) = 2.9(-4)$$

$$P_{SEC} = (0.9)\ \frac{1.5-3}{3.2-2}\ (1-6.8(-3)) = 4.2(-2) \quad .$$

For the second term of the Case 2 CCW failure equation, we have power at either Bus 248 or 249 as well as 247.

$$P_{SE} = (0.9)\ \frac{3.15-3}{.032}\ \Big[(1.1-2)(1.2-4) + (1.1-2)(4.6-4)\Big]$$
$$= 5.7(-7) \quad .$$

$$P_{SEC} = (0.9)\ \frac{3.15-3}{.032}\ \Big[(1.1-2)(1-1.2(-4)) + (1.1-2)(1-4.6(-4))\Big]$$
$$= 1.9(-3) \quad .$$

$$P_{SEF} = (0.9)\ \frac{3.15-3}{.032}\ \Big[(1.2-4)(1-1.1(-2)) + (4.6-4)(1-1.1(-2))\Big]$$
$$= 5.1(-5) \quad .$$

$$P_{SEFC} = (0.9)\ \frac{3.15-3}{.032}\ \Big[(1-1.2(-4))(1-1.1(-2))$$
$$+ (1-4.6(-4))(1-1.1(-2))\Big] = 0.18 \quad .$$

For the third term of the Case 2 CCW failure equation, both Bus 248 and Bus 249 have power, as well as Bus 247. Therefore, because the spray and fan failure probabilities are small with power at all three buses, only damage state SEFC need be considered

$$P_{SEFC} = (0.9)\ \frac{2.3-2}{.032} = 0.65 \quad .$$

With either Bus 148 or 149 available at Unit 1 and the Case 2 CCW failure, the plant damage state probabilities for Unit 2 are

$$P_{SE} = 6.3(-3)$$

$$P_{SEC} = 6.3(-2)$$

$$P_{SEF} = 8.4(-5)$$

$$P_{SEFC} = 0.90 \quad .$$

<u>Case 3: Power Available at Two Buses of Unit 1</u>

The equation for CCW failure is

$$\begin{aligned} Q_{LS} = & \left[2(CCWP + CCWM)\ (DGFS^2 + 2\ DGFS \cdot DGM)\right] \\ & + \left[6(DGFS + DGM)\ (CCWP^2 + CCWP \cdot CCWM + CCW2M)\right] \\ & + \left[4(CCWP^3 + 3\ CCWP^2 \cdot CCWM + 3\ CCWP \cdot CCW2M)\right] = 2.5\text{-}3 \end{aligned}$$

and the values contributed by the terms are

first term = 1.0(-4)
second term = 2.4(-3)
third term = 6.7(-5)

As in Case 2, we have in Case 3 two entirely different situations: the two buses available at Unit 1 are 147 and 148 or 147 and 149, and the two buses available at Unit 1 are 148 and 149.

Buses 147 and 148 or 147 and 149 are Available

In this circumstance, Bus 247 cannot be available at Unit 2. The first term of the Case 3 equation addressed the probability that one Unit 1 pump would fail and both the 248 and 249 buses at Unit 2, so no fans or sprays would be available. Therefore, all of the contribution of the first term is to the SE damage state

$$P_{SE} = \frac{1.0(-4)}{2.5(-3)} = 0.04 \quad .$$

The second term implies power at either Bus 248 or 249 of Unit 2. For this term the SE state would require failure of the sprays, so

$$P_{SE} = \frac{1.2(-3)}{2.5(-3)}\,(6.8-2) + \frac{1.2(-3)}{2.5(-3)}\,(6.8-3) = 0.036 \quad .$$

$$P_{SEC} = \frac{1.2(-3)}{2.5(-3)}\,(1-6.8-2) + \frac{1.2(-3)}{2.5(-3)}\,(1-6.8-3) = 0.924 \quad .$$

The third term implies power at both the 248 and 249 buses of Unit 2. For this term, the SE state would require the failure of both fans and sprays:

$$P_{SE} = \frac{6.7(-5)}{2.5(-3)}\left[9.2(-5)\cdot 4.6(-4)\right] = 1.1(-9) \quad .$$

$$P_{SEC} = \frac{6.7(-5)}{2.5(-3)}\left[9.2(-5)\cdot 1-4.6(-4)\right] = 2.5(-6) \quad .$$

$$P_{SEF} = \frac{6.7(-5)}{2.5(-3)}\left[1-9.2(-5)\cdot 4.6(-4)\right] = 1.2(-5) \quad .$$

$$P_{SEFC} = \frac{6.7(-5)}{2.5(-3)}\left[1-9.2(-5)\cdot 1-4.6(-4)\right] = 2.7(-2) \quad .$$

Summing the probabilities for the three terms of Case 3, we have

$P_{SE} = 7.6(-2)$

$P_{SEC} = 0.92$

$P_{SEF} = 1.2(-5)$

$P_{SEFC} = 2.7(-2)$.

Buses 148 and 149 are Available

In this circumstance, Bus 247 may or may not be available. If it is not, then the Unit 2 plant damage state probabilities for this power state at Unit 1 and CCW failure are simply those calculated above (for Buses 147 and 148 or 147 and 149 available), but multiplied by 10 percent probability that 247 is unavailable, given 147 is unavailable.

$P_{SE} = 7.6(-3)$

$P_{SEC} = 9.2(-2)$

$P_{SEF} = 1.2(-6)$

$P_{SEFC} = 2.7(-3)$

When Bus 247 is available, the first term of the Case 3 CCW failure equation has power available at neither Bus 248 nor 249. Hence, the fan failure probability is 1.0. The Unit 2 damage state probabilities for 247 available, this Unit 1 power state, and CCW failures are

$$P_{SE} = (0.9)\,\frac{1.0(-4)}{2.5(-3)}\,(6.8-3) = 2.4(-4) \quad .$$

$$P_{SEC} = (0.9)\,\frac{1.0(-4)}{2.5(-3)}\,(1-6.8(-3)) = 3.6(-2) \quad .$$

$$P_{SEF} = 0.0 \quad .$$

$$P_{SEFC} = 0.0 \quad .$$

For the second term of the Case 3 CCW failure equation, we have power available at either Bus 248 or 249 as well as 247.

$$P_{SE} = (0.9)\ \frac{1.2-3}{2.5-3}\left[(1.1-2)(1.2-4) + (1.1-2)(4.6-4)\right]$$

$$= 2.8(-6) \quad .$$

$$P_{SEC} = (0.9)\ \frac{1.2-3}{2.5-3}\left[(1.1-2)(1-1.2(-4)) + (1.1-2)(1-4.6(-4))\right]$$

$$= 9.5(-3) \quad .$$

$$P_{SEF} = (0.9)\ \frac{1.2-3}{2.5-3}\left[(1-1.1(-2))(1.2-4) + (1-1.1(-2))(4.6-4)\right]$$

$$= 2.5(-4) \quad .$$

$$P_{SEFC} = (0.9)\ \frac{1.2-3}{2.5-3}\Big[(1-1.1(-2))(1-1.2(-4))$$

$$+ (1-1.1(-2))(1-4.6(-4))\Big] = 0.85 \quad .$$

For the third term of the Case 3 CCW failure equation, we have both the 248 and 249 buses available as well as Bus 247. Therefore, we need only determine the probability of the SEFC damage state at Unit 2.

$$P_{SEFC} = (0.9)\ \frac{6.7-5}{2.5-3} = 2.4(-2) \quad .$$

With Buses 148 and 149 available at Unit 1 and the Case 3 CCW failure, the plant damage state probabilities at Unit 2 are

$P_{SE} = 7.8(-3)$

$P_{SEC} = 1.3(-1)$

$P_{SEF} = 2.5(-4)$

$P_{SEFC} = 0.87$

Case 4: Power Available at Three Buses of Unit 1

The CCW failure equation is

$$Q_{LS} = \left[3(DGFS^2 + 2\ DGFS \cdot DGM)\ (CCWP^2 + 2\ CCWP \cdot CCWM + CCW2M) \right]$$

$$+ \left[8(DGFS + DGM)\ (CCWP^3 + 3\ CCWP^2 \cdot CCWM + 3\ CCWP \cdot CCW2M) \right]$$

$$+ \left[5(CCWP^4 + 4\ CCWP^3 \cdot CCWM + 6\ CCWP^2 \cdot CCW2M) \right] = 4.3\text{-}5 \quad ,$$

and the values contributed by the terms of the equation are

first term = 3.6(-5)
second term = 6.9(-6)
third term = 1.2(-7)

Note that in this case, Bus 247 cannot be available at Unit 2 because Bus 147 is available at Unit 1. The first term of Case 4 implies no power at Unit 2, so no fans or sprays. Therefore, all of the contribution of the first term is to the SE damage state

$$P_{SE} = \frac{3.6(-5)}{4.3(-5)} = 0.837 \quad .$$

The second term implies power at one Unit 2 bus. For this term the SE state would require failure of the sprays, so

$$P_{SE} = \frac{3.45(-6)}{4.3(-5)}\ (6.8\text{-}2) + \frac{3.45(-6)}{4.3(-5)}\ (6.8\text{-}3) = 6.0(-3) \quad .$$

$$P_{SEC} = \frac{3.45(-6)}{4.3(-5)}\ \ (1\text{-}6.8\text{-}2) + (1\text{-}6.8\text{-}3) \ = 0.154 \quad .$$

The third term of Case 4 implies power at two Unit 2 buses

$$P_{SE} = \frac{1.2(-7)}{4.3(-5)} \left[(4.6\text{-}4) \cdot (9.2\text{-}5) \right] = 1.2\ (-10) \quad .$$

$$P_{SEC} = \frac{1.2(-7)}{4.3(-5)} \left[1-4.6(-4) \cdot 9.2(-5)\right] = 2.6(-7) \quad .$$

$$P_{SEF} = \frac{1.2(-7)}{4.3(-5)} \left[4.6(-4) \cdot 1-9.2(-5)\right] = 1.3(-6) \quad .$$

$$P_{SEFC} = \frac{1.2(-7)}{4.3(-5)} \left[1-4.6(-4) \cdot 1-9.2(-5)\right] = 2.8(-3) \quad .$$

Summing the probabilities for the three terms of Case 4, we have

$P_{SE} = 0.84$

$P_{SEC} = 0.15$

$P_{SEF} = 1.3(-6)$

$P_{SEFC} = 2.8(-3)$

Based on these calculations, we have the following distribution of damage states in Unit 2, given a core melt in Unit 1 which is initiated by loss of off-site power.

Unit 1 Power State	Unit 2 Damage State Probability
All buses available	SE = 0.84 SEC = 0.15 SEF = 1.3(-6) SEFC = 2.8(-3)
Buses 147 and 148 or 147 and 149 available	SE = 7.6(-2) SEC = 0.92 SEF = 1.2(-5) SEFC = 2.7(-2)
Buses 148 and 149 available	SE = 7.8(-3) SEC = 0.13 SEF = 2.5(-4) SEFC = 0.87
Bus 147 available	SE = 6.0(-2) SEC = 0.19 SEF = 3.3(-4) SEFC = 0.72

Bus 148 or 149 available	SE = 6.3(-3) SEC = 6.3(-2) SEF = 8.4(-5) SEFC = 0.90
No buses available	SE = 2.2(-3) SEC = 6.3(-2) SEF = 1.7(-4) SEFC = 0.91

To summarize, the figures in the above table indicate the distribution of plant damage states in Unit 2 given core melts in Unit 1 which would be initiated by loss of off-site power followed by loss of component cooling water. For example, if the losses of component cooling water occurred even though all AC buses at Unit 1 were available, 84 percent (0.84) of the time the Unit 2 the plant damage state would be SE, 15 percent of the time the Unit 2 damage state would be SEC, etc. These distributions are next applied to the Unit 1 core-melt sequence probabilities.

4.5.2 Application of Percentages to Unit 1 Core-Melt Sequences

To continue our evaluation, we address the dominant LOP sequences, as listed in Table 3.1-1.

Sequences 3 and 4 have frequencies of 4.6(-5) and 4.0(-5) respectively. These sequences result in a Unit 1 plant damage state of SEFC. Because off-site power is restored in these sequences, the predominant result for Unit 2 will be SEFC also. Therefore, we estimate the same frequencies and damage states for both units from these sequences.

Sequence 7 results in Unit 1 damage state SEFC. No power is recovered in 8 hours. This sequence could occur if either all Unit 1 buses or two of the three Unit 1 buses have power available. Using the proportions developed above we calculate

With all Unit 1 buses available	Unit 1 frequency	Unit 2 frequency
	SEFC = 9.1(-8)	SE = 7.7(-8) SEC = 1.4(-8) SEF = ε SEFC = ε

With Buses 147 and 148 available	Unit 1 frequency	Unit 2 frequency
	SEFC = 5.5(-7)	SE = 4.1(-8) SEC = 5.0(-7) SEF = ε SEFC = 1.5(-8)

With Buses 147 and 149 available	Unit 1 frequency	Unit 2 frequency
	SEFC = 5.5(-7)	SE = 4.1(-8) SEC = 5.0(-7) SEF = ε SEFC = 1.5(-8)

With Buses 148 and 149 available	Unit 1 frequency	Unit 2 frequency
	SEFC = 6.8(-6)	SE = 5.3(-8) SEC = 8.8(-7) SEF = 1.8(-9) SEFC = 5.9(-6)

In addition to these loss of off-site power sequences we have the sequence initiated by loss of component cooling water, leading to an SEFC damage state, for both units of ~2(-4). Thus, the combined results for the Unit 1 SEFC state are

Unit 1	Unit 2
SEFC = ~3.0(-4)	SE = 2.2(-7) SEC = 1.9(-6) SEF = 1.8(-9) SEFC = 3.0(-4)

The SEF damage state frequency at Unit 1 is dominated by a sequence which includes LOP, loss of component cooling water and loss of containment sprays in the degraded power state. This sequence could occur if either all Unit 1 or two of the three Unit 1 buses have power available. Using the developed proportions we find

With all Unit 1 buses available	Unit 1 frequency	Unit 2 frequency
	SEF = 5.7(-12)	SE = ε
		SEC = ε
		SEF = ε
		SEFC = ε

With Buses 147 and 148 available	Unit 1 frequency	Unit 2 frequency
	SEF = 6.1(-10)	SE = 4.6(-11)
		SEC = 5.6(-10)
		SEF = ε
		SEFC = 1.6(-11)

With Buses 147 and 149 available	Unit 1 frequency	Unit 2 frequency
	SEF = 2.6(-10)	SE = 2.0(-11)
		SEC = 2.4(-10)
		SEF = ε
		SEFC = 7.0(-12)

With Buses 148 and 149 available	Unit 1 frequency	Unit 2 frequency
	SEF = 3.4(-9)	SE = 2.7(-11)
		SEC = 4.4(-10)
		SEF = ε
		SEFC = 3.0(-9)

The combined results for the Unit 1 SEF damage state are

Unit 1	Unit 2
SEF = 4.2(-9)	SE = 9.3(-11)
	SEC = 1.2(-9)
	SEF = ε
	SEFC = 3.0(-9)

The SEC damage state frequency at Unit 1 is dominated by sequence 5. This sequence could occur with all buses, two buses, or one bus available at Unit 1.

With all Unit 1 buses available	Unit 1 frequency	Unit 2 frequency
	SEC = ε	SE = ε SEC = ε SEF = ε SEFC = ε

With Buses 147 and 148 available	Unit 1 frequency	Unit 2 frequency
	SEC = 6.1(-9)	SE = 4.6(-10) SEC = 5.6(-9) SEF = ε SEFC = 1.6(-10)

With Buses 147 and 149 available	Unit 1 frequency	Unit 2 frequency
	SEC = 6.1(-9)	SE = 4.6(-10) SEC = 5.6(-9) SEF = ε SEFC = 1.6(-10)

With Buses 148 and 149 available	Unit 1 frequency	Unit 2 frequency
	SEC = 8.6(-10)	SE = ε SEC = 1.1(-10) SEF = ε SEFC = 7.5(-10)

With Bus 147 available	Unit 1 frequency	Unit 2 frequency
	SEC = 6.1(-7)	SE = 3.7(-8) SEC = 1.2(-7) SEF = 2.0(-10) SEFC = 4.4(-7)

With Bus 148 available	Unit 1 frequency	Unit 2 frequency
	SEC = 9.2(-6)	SE = 5.8(-8) SEC = 5.8(-7) SEF = 7.7(-10) SEFC = 8.3(-6)

With Bus 149 available	Unit 1 frequency	Unit 2 frequency
	SEC = 9.2(-6)	SE = 5.8(-8)
		SEC = 5.8(-7)
		SEF = 7.7(-10)
		SEFC = 8.3(-6)

The combined results for the Unit 1 SEC damage state are

Unit 1	Unit 2
SEC = 1.8(-5)	SE = 1.5(-7)
	SEC = 1.3(-6)
	SEF = 1.7(-9)
	SEFC = 1.7(-5)

The SE damage state frequency at Unit 1 is dominated by sequences 8 and 11. The seismic initiated sequence would lead to an SE damage state at both units with estimated frequency of 5.6(-6). Sequence 11 occurs most probably if either one AC bus or no AC buses are available.

With Bus 147 available	Unit 1 frequency	Unit 2 frequency
	SE = 4.2(-9)	SE = 2.5(-10)
		SEC = 8.0(-10)
		SEF = ε
		SEFC = 3.0(-9)

With Bus 148 available	Unit 1 frequency	Unit 2 frequency
	SE = 6.1(-8)	SE = 3.8(-10)
		SEC = 3.8(-9)
		SEF = ε
		SEFC = 5.5(-8)

With Bus 149 available	Unit 1 frequency	Unit 2 frequency
	SE = 6.1(-7)	SE = 3.8(-9)
		SEC = 3.8(-8)
		SEF = ε
		SEFC = 5.5(-7)

With no Unit 1 buses available	Unit 1 frequency	Unit 2 frequency
	SE = 4.1(-6)	SE = 9.0(-9)
		SEC = 2.6(-7)
		SEF = 7.0(-10)
		SEFC = 3.7(-6)

Adding the sequence frequencies, the SE results are

Unit 1	Unit 2
SE = 1.0(-5)	SE = 5.6(-6)
	SEC = 3.0(-7)
	SEF = 7.0(-10)
	SEFC = 4.3(-6)

The overall results of the double core-melt analysis are shown in Table 4.5-1.

4.6 ZPSS Fire Analysis

The Zion fire analysis appears to lack the depth found in the Indian Point fire PSS. In particular, the following was found:

1. The Zion fire analysis only analyzed two plant areas--the auxiliary electrical equipment room and the cable spreading room. Other important plant areas were either qualitatively assessed (e.g., Auxiliary Building Zone 11-3.0) or not addressed at all in the Zion PRA (e.g., component cooling water (CCW) pump area).

2. The Zion fire analysis did not address seal LOCA events caused by the loss of CCW.

3. The Zion fire analysis did not consider that power to both electrical AFW pumps and to the steam regulating valve of the steam-driven AFW pump all run through the same cable spreading room.

4. The Zion fire analysis did not consider the loss of service water or component cooling water by fire in combination with an unavailability of redundant components due to maintenance.

5. The Zion fire analysis assumed correct operator actions with a mean probability of 2.5×10^{-2} even under high-stress fire conditions.

In addition to these problems, the analyses that were reported in the ZPSS have not considered equipment or cable damage by hot gas layers or failure of cabling at temperatures below autoignition temperatures. These failure mechanisms are discussed in Section 2.7.4 of NUREG/CR-2934, "Review and Evaluation of the Indian Point Probabilistic Safety Study." Finally, the ZPSS has not thoroughly documented, unlike the Indian Point PSS, each of the numerical factors used to estimate the frequency of the two core-melt fire scenarios which were analyzed. Each of these factors, along with the numerous plant modifications planned by Zion

TABLE 4.5-1

Zion Double Core-Melt Frequency

Total Damage State at Unit 1	Corresponding Damage States at Unit 2 From Common Mode Core Melt	**
1. SEFC = 3.0(-4)*	1. SEFC = ~3.0-4*	1.8(-4)
	SEF = 1.8(-9)	1.2(-9)
	SEC = 1.9(-6)	1.2(-6)
	SE = 2.2(-7)	1.4(-7)
2. SEF = 5.5(-9)	2. SEFC = 3.0(-9)	1.9(-9)
	SEF = ε	ε
	SEC = 1.2(-9)	7.7(-10)
	SE = 9.3(-11)	6.0(-11)
3. SEC = 1.9(-5)	3. SEFC = 1.7(-5)	1.1(-5)
	SEF = 1.7(-9)	1.1(-9)
	SEC = 1.3(-6)	8.3(-7)
	SE = 1.5(-7)	9.6(-8)
4. SE = 1.0(-5)[t]	4. SEFC = 4.3(-6)	2.8(-6)
	SEF = 7.0(-10)	4.5(-10)
	SEC = 3.0(-7)	1.9(-7)
	SE = 5.6(-6)[t]	3.6(-6)

*predominantly CCW loss as initiating event

t 5.6-6 is seismic event

**Values adjusted to reflect the probability of .64 that both units will be in operation at the time of an initiating event.

by Zion to comply with 10 CFR50 Appendix R requirements for fire protection, complicated Sandia review efforts.

To help supplement missing information, we conducted a review of docketed correspondence between NRC and Commonwealth Edison on fire protection issues considered from July 1982 through March 1983. In addition, R. Furguson and R. Eberly (both NRC) were contacted to discuss the status of Appendix R reviews of Zion and to confirm what types of fire protection improvements have been committed to by the utility. Based on this information and the ZPSS documentation, it appears that nearly all plant areas in which a fire may cause a probabilistically significant accident sequence have been addressed by plant improvements committed to by the utility. The exceptions to this involve the cable spreading rooms and auxiliary electrical equipment room which were analyzed in the ZPSS. For these areas, one or more of the factors listed above involving component cooling water failure, auxiliary feedwater failure, hot gas layer effects, or operator response probability can be interpreted as causing new accident scenarios or increasing the estimated frequency of those scenarios already analyzed. Unfortunately, without a detailed reanalysis of the auxiliary electrical equipment room and the cable spreading rooms, credible estimates of the effects of these factors or core-melt frequency cannot be made. Because of this, it was decided to treat these issues in terms of a sensitivity analysis of the ZPSS.

Auxiliary Electrical Equipment (AEE) Room

In the ZPSS, the major contributor to the estimated core-melt frequency for the AEE room is a fire which damages cabinets to an extent that operators receive incorrect diagnostic information and effect incorrect recovery actions involving auxiliary feedwater or high-pressure injection. Under these circumstances, the ZPSS assumed the probability for incorrect operator actions to be 2.5×10^{-2}. We believe that this figure may be optimistic and that a more defendable number of 10^{-1} should be used for the high-stress situation which would occur during a fire. Using this revised estimate, the mean frequency for a core melt from a fire in the AEE room increases from 2.8×10^{-6}/yr. to around 10^{-5}/yr. for damage state TEFC.

Cable Spreading Room

In the ZPSS, the major contribution to the estimated core-melt frequency for the cable spreading room is a fire in which the motor-driven auxiliary feedwater pump power cables fail, the turbine-driven auxiliary feedwater pump fails randomly, and operators fail to start feed and bleed decay heat removal. The estimated mean frequency of this scenario in the ZPSS is 1.8×10^{-6}/yr. for damage state TEC.

However, from the ZPSS, it appears as though the Zion Unit 1 cable spreading room contains the following cabling:

- power feeds for three component cooling water pumps and three service water pumps
- power feeds for two charging pumps
- power feeds for two auxiliary feedwater pumps
- control cabling for five fan coolers
- control cabling for at least two containment spray pumps

Docketed information from Commonwealth Edison indicates that the cable spreading room also contains power cables for the turbine-driven auxiliary feedwater (AFW) pump steam supply valves separated by a minimum distance of 20 feet from the motor-driven auxiliary feedwater pump power cables. Information on the location of safety injection pump cabling and the third containment spray pump is not provided in the ZPSS. Furthermore, the exact cable routing location of the CCW and service water power cables and the steam-driven AFW valve power cable is not provided.

Because of this information (or lack thereof), four concerns are identified:

1. Hot gas layer fire failure mechanisms can be postulated which cause the Zion TEC damage state to become a TE damage state due to loss of containment sprays.
2. Hot gas layer fire failure mechanisms can be postulated which cause the Zion TEC damage state to increase in frequency due to the ability of smaller fires than those postulated in the Zion PRA to cause damage.
3. The mean probability for incorrect operator actions of 2.5×10^{-2} for the cable spreading room TEC damage state may be optimistic and should be more like 10^{-1}.
4. Cable spreading room fires may damage three component cooling water pump power cables. This, together with a maintenance outage of one of the two unaffected CCW pumps ($\sim 6 \times 10^{-2}$/demand), could result in a seal LOCA and loss of all high-pressure injection. The

result would be an SE, SEF, SEC, or SEFC damage state depending on whether containment spray pumps or fan cooler cabling is affected by the same fire.

Without a detailed analysis, which is outside the scope of Sandia's review, the numerical significance of these concerns cannot be determined. However, an estimate of their potential magnitude can be made by combining insights from the Zion fire PSS with analyses reported in the Indian Point fire PSS.

With regard to transient induced core-melt sequences, it is not unreasonable for the Zion TEC sequence to be replaced by a TE sequence in which the fire damages the motor-driven auxiliary feedwater pump power cabling, the turbine-driven AFW steam valve cabling, the containment spray and fan cooler cabling, and all high-pressure injection power cabling. If this occurs, local operator control of the turbine-driven auxiliary feedwater pump may prevent core melt. Assuming a 10^{-1} chance of doing this, then the TE frequency can be estimated as:

$$7.2(-3)/\text{yr.} \times 0.05 \times 0.1 \cong 4(-5)/\text{yr. for TE}$$

cable spreading room fire frequency for Indian Point	X	fires large enough and located so as to cause problems	X	operator local control of turbine AFW pump

With regard to LOCA induced core-melt sequences, the SE frequency can be estimated as:

$$7.2(-3)/\text{yr.} \times 0.05 \times 0.06 \cong 2(-5)/\text{yr. for SE}$$

as above	chance of at least one CCW pump being out for maintenance

Clearly, these numbers represent gross estimates of what are most likely worst-case conditions. For example, no credit for suppression activities has been given. From the Indian Point PSS this credit could represent a factor of at least 0.5. Nevertheless, the estimates for TE and SE appear to indicate the need for a further analysis of whether these types of fire scenarios can truly be ignored at Zion.

Based on the above, the following changes would apply to plant damage state frequencies resulting from internal and external events

Plant Damage State	Revised Damage State	Estimated Frequency Based on Worst-Case Conditions
TE	7.7(-7)	~4(-5)
SE	1(-5)	~2(-5)

4.7 ATWS Initiated by Loss of Main Feedwater

This sequence is initiated by loss of main feedwater followed by the failure of the reactor protective system to trip the reactor. If the reactor fails to trip, an increase in primary system pressure occurs. The rise in pressure will be limited to less than 3200 psi by sufficient secondary cooling and primary pressure relief. Failing these moderating events, pressure will rise above 3200 psi. In this case, it is assumed that a small LOCA will occur. It is also assumed that damage to the check valves between the primary system and the high-pressure injection system will render them inoperable, preventing makeup of fluid to the reactor coolant system. The result of this sequence will be core melt.

Of particular interest for this sequence is the status of the turbine generator. If the turbine is running and fails to trip, its demand on the steam generator will cause them to dry out rapidly, leading to loss of secondary cooling and hence to rapid, uncontrolled rise in primary system pressure. The probability of turbine trip failure is high if the trip signal is generated solely by the same circuitry which causes reactor trip. We are advised by Commonwealth Edison personnel that this is not the case at Zion. The information received is that an independent turbine trip signal is also generated by the main feedwater pump trip circuitry, arranged so that the trip of both main feedwater pumps will generate a turbine trip signal.

Based on the above information, we calculate the loss of main feedwater ATWS sequence as follows:

$$Q_{CM} = LOMF \cdot RPS \cdot L2 \cdot PL \cdot HH2$$

$$= 5.2 \cdot 1.8(-4) \cdot 5.9(-4) \cdot 0.5 \cdot 1.0 = 2.8(-7)$$

where:

Loss of main feedwater (LOMF)	= 5.2
Failure of reactor to trip (RPS)	= 1.8(-4)
Failure of augmented aux. feedwater (L2)	= 5.9(-4)
Probability that power level is above 80% (PL)	= 0.5
Probability of high-head injection failure (HH2)	= 1.0

If turbine trips were initiated only by the reactor trip circuitry, the loss of main feedwater/ATWS sequence would be calculated as follows:

$$Q_{CM} = \text{LOMF} \cdot \text{RPS} \cdot \text{TT} \cdot \text{PL} \cdot \text{HH2}$$

$$= 5.2 \cdot 1.8(-4) \cdot 1 \cdot 0.5 \cdot 1 = 4.7(-4)$$

where values for loss of main feedwater, reactor trip, failure of high-head injection and the probability that power level is above 80 percent are the same as above; and turbine trip and the augmented auxiliary feedwater are assumed to be ineffective.

Since the LOMF/ATWS sequence would result in plant damage state SEFC, the following comparison can be made:

SEFC (assuming independent turbine trip circuit)	3.3(-4)
SEFC (assuming no independent turbine trip circuit)	7.7(-4)

We feel that the two sequence calculations presented above bound ATWS for the Zion units. Because of the nature of this review, we are not able to analyze the situation in full detail; the examination of central wiring diagrams of the RPS, AFWS, and turbine trip instrumentation and logic circuitries as well as common manufacturer questions is beyond the scope of this review.

In Section 2.4.1.2, however, we note that the Zion trip breakers do not have the shunt coils in their design. In light of the current ATWS rule-making process, we consider the possible benefits gained by the addition of shunt coils. Information presented in Reference 4-5 suggests that the RPS circuit breaker failure probability could be decreased by an order of magnitude if the shunt coils were added. Hence, if we assume the worst case, that the Zion turbine trip is not independent of the RPS, then the upper-bound sequence becomes

$$Q_{CM} = \text{LOMF} \cdot \text{RPS} \cdot \text{TT} \cdot \text{PL} \cdot \text{HH2}$$

$$= 5.2 \cdot 1.8(-5) \cdot 1 \cdot 0.5 \cdot 1 = 4.7(-5)$$

where all the values are the same as before with the exception of the RPS failure probability being lowered by a factor of 10. (In Zion, it is dominated by the breaker failure.)

4.8 Completeness

One of the major sources of uncertainty in any PRA is completeness. These types of uncertainties arise from the inability of the PRA analysts to completely identify all possible accident sequences and system failure modes. Our review identified several accident sequences and system failure modes which were apparently omitted in the ZPSS. The more important omissions are summarized below:

- Pressurized thermal shock--discussed in Section 2.1 and not evaluated in this review.
- Several fire analysis omissions--discussed in Section 4.6.
- Safety System failure caused by core meltdown phenomena--discussed and evaluated in Section 4.1.
- An initiating event caused by a pipe break in the component cooling water system--discussed and evaluated in Section 3.2.1.
- Failure of safety injection and RHR pumps due to failure of room cooling systems--discussed and evaluated in Section 4.4.
- Component cooling water, service water, auxiliary feedwater, containment spray and containment spray system β factors were omitted--discussed and evaluated in Section 2.4.
- Reactor coolant pump seal ruptures were not included in the small LOCA initiating event data base--discussed and evaluated in Section 2.1.
- Steam generator overfill scenarios were not considered--not discussed or evaluated in this report.
- Cold shutdown events--discussed in Section 2.1 and not evaluated in this review.
- Failure of safety injection and charging pumps due to failure of component cooling water system --discussed and evaluated in Section 3.2.1.
- Design and construction errors and the effects of aging were not considered.

The effect of the post core-melt environment on electrical terminal blocks inside containment, which could lead to incorrect plant status indications in the control room--discussed in Section 5.

4.9 Special Sensitivity Issue (Success Criteria)

In the calculation of system failure probabilities for the component cooling water and service water systems, an important issue is the system success criteria. The calculation of these probabilities (Sections 2.4.1.10 and 2.4.1.11) was based on the assumption that two component cooling water pumps are required for component cooling water system success and two service water pumps for service water system success. We recognize, however, that there is substantial uncertainty associated with these assumptions. There are some arguments which suggest that the correct success criteria are one operating pump for the component cooling water system and three operating pumps for the service water system. (See Appendix E for Commonwealth Edison comments on this issue.) Because the selection of success criteria can have a significant impact on plant damage state frequencies, we have evaluated system failure probabilities based on alternative success criteria. This evaluation is described below.

Component Cooling Water System

The effect of assuming that only one component cooling water pump is required for system success has a negligible impact on accident sequences initiated by events other than loss of off-site power, since none of these sequences contributes significantly to plant damage state frequencies. There are, however, important sequences which involve CCW failure following loss of off-site power. Consequently, we calculated the probability of system failure for various degraded electric power states in Section 2.4.1.10. These calculations are repeated here using a one-pump system success criterion.

Case 1: The diesel generators at Buses 147, 148, and 149 have failed.

Two buses are potentially available at Unit 2. If two of two buses fail, or if one of two buses and one pump fail, or two of two pumps fail, event LS will occur. For this probability we calculate

Q_1 = 2 of 2 diesel generators failing, or 1 diesel generator and 1 pump failing, or 2 of 2 pumps failing

$$= DGFS^2 + 2(DGFS)(DGM) + 2(DGFS + DGM)(CCWP + CCWM)$$

$$+ CCWP^2 + 2(CCWP)(CCWM) + CCW2M = .016 \quad .$$

<u>Case 2: One diesel generator (at AC Bus 147, 148, or 149) has succeeded and the other two have failed.</u>

We again have the possibility of pumps available on Buses 248 and 249. If two of two diesel generators and one pump fail, or if one of two diesel generators and two of two pumps fail, or if three of three pumps fail, event LS will occur.

$$Q_2 = (DGFS^2)(CCWP + CCWM) + 2(DGFS)(DGM)(CCWP + CCWM)$$

$$+ 2(DGFS + DGM)(CCWP^2) + 2(DGFS + DGM)(CCWP)(CCWM)$$

$$+ 2(DGFS + DGM)(CCW2M) + CCWP^3 + 3(CCWP^2)(CCWM)$$

$$+ 3(CCWP)(CCW2M) = 8.7(-4) \quad .$$

<u>Case 3: Two diesel generators are available at Buses 147, 148, or 149 and the other diesel generator has failed.</u>

We have the possibility of pumps available on four buses. If two of two diesel generators and two of two pumps fail, or if one of two diesel generators and three of three pumps fail, or if four of four pumps fail, event LS will occur.

$$Q_3 = (DGFS^2)(CCWP^2 + CCW2M) + 2(DGFS^2)(CCWP)(CCWM)$$

$$+ 2(DGFS)(DGM)(CCWP^2 + CCW2M) + 4(DGFS)(DGM)(CCWP)(CCWM)$$

$$+ 2(DGFS + DGM)(CCWP^3) + 6(DGFS + DGM)(CCWP^2)(CCWM)$$

$$+ 6(DGFS + DGM)(CCWP)(CCW2M) + CCWP^4 + 4(CCWP^3)(CCWM)$$

$$+ 6(CCWP^2)(CCW2M) = 1.4(-5) \quad .$$

Case 4: Three diesel generators are available at Buses 147, 148, and 149.

We have the possibility of CCW pumps available on all five buses. If two of two Unit 2 buses fail and three of three pumps fail, or if one of two buses fail and four of four pumps fail, or if all five pumps fail, event LS will occur.

$$Q_4 = (DGFS^2)(CCWP^3) + 3(DGFS^2)(CCWP^2)(CCWM)$$

$$+ 3(DGFS^2)(CCWP)(CCW2M) + 2(DGFS)(DGM)(CCWP^3)$$

$$+ 6(DGFS)(DGM)(CCWP^2)(CCWM) + 6(DGFS)(DGM)(CCWP)(CCW2M)$$

$$+ 2(DGFS)(CCWP^4) + 8(DGFS)(CCWP^3)(CCWM)$$

$$+ 12(DGFS)(CCWP^2)(CCW2M) + CCWP^5 + 5(CCWP^4)(CCWM)$$

$$+ 10(CCWP^3)(CCW2M) = 2.9(-8)$$

The four cases discussed above correspond to the headings of Table 2.4-1. That is, Case 1 applies to the heading "No Buses"; Case 2 to headings "Bus 147," "Bus 148," and "Bus 149"; Case 3 to headings "Buses 147-148," "Buses 147-149" and "Buses 148-149"; Case 4 to "All Buses." It is apparent that the CCW system failure probabilities calculated in this section (based on a one-pump success criterion) are much lower than those which appear in Table 2.4-1 (based on a two-pump criterion.)

Service Water System

The service water system is not an important contributor to plant damage state frequencies for initiating events other than seismic or loss of off-site power. Consequently, we evaluate here only the probability of service water system failure for the degraded power states, assuming that three operating pumps are required for system success. The following calculations are similar to those in Section 2.4.1.11.

Case 1: No emergency buses are available at Unit 1.

Case 1A: Only two buses are potentially available at Unit 2, therefore service water fails with probability of 1.0.

Case 1B: One bus is available and two more buses are potentially available at Unit 2. The system fails if either one of two buses at Unit 2 is unavailable or one of three service water pumps fail.

Q_{1B} = 2 (DGFS + DGM) + 3(SWP + SWM) = .11

The total for case one is the probabilistic combination of cases 1A and 1B.

Q_1 = (.10)(Case 1) + .90 (Case 2) = .20

Case 2: Power available at Bus 147 and two buses of Unit 2.

A total of three buses are potentially available. The system fails if either one of two buses at Unit 2 fails or one of three service water pumps fail. This is the same as Case 1B above, i.e., .11.

Case 3: Power available at Bus 148 or 149

Case 3A: Only two buses are potentially available at Unit 2. The system fails if either one of two buses at Unit 2 fails or one of three SW pumps fail.

Q_{3A} = 2(DGFS + DGM) + 3(SWP + SWM) = .113

Case 3B: One bus is available and two buses are potentially available at Unit 2. The system fails if two of two Unit 2 buses fail, or if one of two buses and one of three pumps fail, or if two of four pumps fail.

$$Q_{3B} = DGFS^2 + 2(DGFS)(DGM) + 2(DGFS + DGM)(3)$$
$$(SWP + SWM) + 6\ SWP^2 + (4)(SWP)(3)(SWM) = 2.5(-3)$$

Total for Case 3

$$Q_3 = (.1)(.113) + (.9)(2.5-3) = 1.4(-2) \quad .$$

NOTE: In "B" cases, such as 3B above, the swing diesel is assumed to be on Bus 247. Since the diesel must be running before it is connected to a bus and since failure to continue running is not considered in the analysis, Bus 247 is counted as "available" rather than "potentially available."

Case 4: Power at 147 and 148 or 147 and 149 (two buses at Unit 2).

Two buses are potentially available at Unit 2. The system fails if both buses at Unit 2 fail, or if one of two buses and one of three service water pumps fail, or if two of four service water pumps fail.

$$Q_4 = DGFS^2 + 2(DGFS)(DGM) + 2(DGFS + DGM)(3)(SWP + SWM)$$
$$+ 6SWP^2 + 12(SWP)(SWM) = 2.7(-3) \quad .$$

Case 5: Power at 148 and 149.

Case 5A: Only two buses are potentially available at Unit 2. The system fails if both buses at Unit 2 fail, or if one of two buses and one of three service water pumps fail, or if two of four service water pumps fail. This is the same as Case 4 above - 2.7(-3).

Case 5B: One bus is available and two buses are potentially available at Unit 2. In this case the system fails if two of two buses and one of three service water pumps fail or if one of two buses and two of four service water pumps fail, or if three of five service water pumps fail.

$$Q_{5B} = (DGFS^2)(3)(SWP + SWM) + 2(DGFS)(DGM)(3)(SWP + SWM)$$
$$+ 2(DGFS + DGM)(SWP^2) + 2(DGFS + DGM)(12)(SWP)(SWM)$$
$$+ 10\ SWP^3 + 10(SWP^2)(SWM) = 1.6(-5) \quad .$$

Total for Case 5

$$Q_5 = (.1)(2.7\text{-}3) + (.9)(1.6\text{-}5) = 2.8(-4) \quad .$$

Case 6: Power available at all buses of Unit 1.

In this case power is potentially available at two buses of Unit 2. The system fails if two of two buses at Unit 2 and one of three service water pumps fail, or if one of two buses and two of four service water pumps fail, or if three of five service water pumps fail.

$$Q_6 = [(DGFS^2 + 2(DGFS)(DGM)] \, [3(SWP + SWM)]$$

$$+ 2(DGFS + DGM)(6\ SWP^2) + 2(DGFS + DGM)(12)(SWP)(SWM)$$

$$+ 10\ SWP^3 + 10(SWP^2)(SWM) = 1.6(-5) \quad .$$

The six cases discussed above correspond to the headings of Table 2.4-1. That is, Case 1 corresponds to Table 2.4-1 heading "No Buses"; Case 2 corresponds to heading "147"; Case 3 corresponds to headings "148" and "149"; Case 4 corresponds to headings "147-148" and "147-149"; Case 5 corresponds to heading "148-149"; and Case 6 corresponds to the heading "All." The results of the above calculations for service water system failure probability (based on a three-pump success criterion) are significantly higher than those given in Table 2.4-1 (based on a two-pump success criterion.)

If the results of these calculations are applied to the accident sequences, the following changes occur:

- The frequency of sequences initiated by loss of off-site power and followed by loss of component cooling water is greatly reduced. Consequently, these sequences (sequences 3, 4, 5, 7, and 11 of Table 3.1-1) would be eliminated from the dominant accident sequence list.

- The frequency of sequences initiated by loss of off-site power and followed by loss of service water is greatly increased. As a result, these sequences would appear in the dominant accident sequence list.

To illustrate the impact of these changes, Table 4.9-1 compares the important loss of off-site power sequences involving loss of service water (three-pump success criterion) with the loss of off-site power/loss of component cooling water sequences of Table 3.1-1.

Table 4.9-1

Comparison of LOP/CCW and LOP/SW Sequence Frequencies

Table 3.1-1 Sequence	Frequency of LOP/CCW Sequence (CCW-2-pump, SW2-pump criteria)	Frequency of Corresponding LOP/SW Sequence (CCW1-pump, SW3-pump criteria)
3(SEFC)	4.6(-5)	2.7(-5)
4(SEFC)	4.0(-5)	2.3(-5)
5(SEC)	1.8(-5)	1.5(-5)
7(SEFC)	7.9(-6)	2.0(-6)
11(SE)	4.7(-6)	5.5(-6)

Note: In the calculation of the above LOP/SW frequencies, it was assumed that three pumps were necessary to prevent a reactor coolant pump seal LOCA, but that one pump was sufficient for cooling of the diesel-driven containment spray pump.

REFERENCES

4-1. Summary of NRC Staff and Consultants Questions on the ZPSS - Commonwealth Edison Response, 1982.

4-2. N. S. DeMuth, et. al., "Loss of Feedwater Transients for the Zion 1 PWR," Nuclear Safety, Vol. 24, No. 1, January- February 1983.

4-3. Memorandum from J. Zudans, NRC, to Z. Rosytoczy, NRC, Subject: St. Lucie 2; Reactor Coolant Pump Seal Hot Standby Test, September 19, 1980.

4-4. Interim Reliability Evaluation Program Procedures Guide, NUREG/CR-2728, January 1983.

4-5. Appendix A "Scram Reliability" of Enclosure D "Recommendations of the ATWS Task Force" to Final Amendment of 10 CFR 50.62, presented by NRC staff to the Commission on September 30, 1983.

5. Summary and Conclusions

Over the past several months, we have reviewed the Zion probabilistic Safety Study. Our review was limited to the treatment of the plant systems and external events. This section summarizes some of our more substantive findings.

Section 5.1 lists several of the more important findings in Sandia's review of the Zion Probabilistic Safety Study (ZPSS). Section 5.2 presents our recommended estimate of plant damage state frequencies for use in the containment and consequence analysis. This estimate reflects, to the degree possible given the limited scope of our review, our best judgment of these frequencies. Included in these estimates are the significant quantitative conclusions presented in the text. Section 5.2.1 summarizes our findings for the internal events, and Section 5.2.2 summarizes our findings for the external events. Section 5.2.3 combines these, and Section 5.2.4 highlights the sensitivity issues investigated.

In general, we found the systems analysis portion of the study to be consistent in scope and detail with ongoing probabilistic risk assessments. The treatment of external events represents an advancement over what has been done in the past. We commend the ZPSS analysis team for their utilization of plant-specific data in their analysis.

We found the documentation for the report, though voluminous, often lacking. This made review difficult and, at times, raised questions. Many of these questions, however, were resolved through the cooperation of those who performed the study.

Our principal findings are summarized in the following section. By the very nature of the review process, we concentrate on negative findings and impressions with respect to the ZPSS. We have tried, however, to place these in perspective with respect to their impact on the frequency of core melt and risk. In some instances, we note where the Zion treatment appears reasonable to us.

5.1 Important Findings

Among the important findings of our review are the following, grouped by topic:

<u>Initiating Events</u>

- The initiating events covered in the ZPSS seem to be relatively complete compared to those addressed in previous PRAs, and their estimates of initiating event frequencies appear reasonable.

- Though the initiating events considered were relatively complete, the ZPSS failed to recognize several dependencies between initiating events and responding safety systems. Examples of these were component cooling water, DC power, and service water initiating events.

- The initiating event frequencies for each plant are based on the operating history of each plant.

Event Trees

- The treatment of the containment spray system (CSS) is questionable. The ZPSS assumes that the CSS can be used throughout an accident in the injection mode rather than having to draw from the sump. They assume that the operator will act to refill the refueling water storage tank (RWST) if depleted. We question this assumption and do not give credit for refilling the RWST.

- The ZPSS did not give credit for the main feedwater system following reactor trip. We feel this is an overly conservative assumption.

- Core melt caused by overpressure failure of containment (e.g., S_2C type accidents in WASH-1400) were not considered. However, this would have negligible effects based on our review.

Success Criteria

- Success criteria used in the analysis in general appears to be reasonable and consistent with those used in PRAs of similar plants.

- An exception is the success criterion for CCW to prevent a seal LOCA. The ZPSS assumed that one CCW pump was sufficient. We believe it is more reasonable to assume that two pumps are required.

Fault Trees

- In general, the fault trees presented in the ZPSS are an accurate representation of the Zion systems.

- The Zion plant test procedures do not require staggered tests of safety systems. This practice tends to increase system failure probabilities due to common mode test errors. We feel this practice provides a basis for our use of common mode β factors.

- The analyses are inconsistent in the application of common cause failure possibilities. The ZPSS, however, should be commended for its examination of common cause failures, although we would recommend more reliance on historic data. In some instances, we modified ZPSS common cause failure estimates via examination of generic historic data.

- In the degraded power states, the ZPSS ignored random failure and maintenance unavailability for the pumps which could still receive power in the component cooling water and service water systems.

- The ZPSS failed to provide fault trees for several room cooling systems and thus failed to identify that room coolers were not being completely tested at Zion.

Human Reliability Analysis

- The human reliability analysis reflected a diligent and sincere effort to use accepted human reliability analysis methods. A complete evaluation by us was not possible due to a lack of documentation.

  - We judged that undue optimism existed in the assessment of credit for human redundancy.

  - We recommend less optimistic assessments of human performance under stress, especially for the case of multiple problems.

  - We believe that estimates of operator performance should be based on simple measurements rather than personal estimates.

  - We found inadequate documentation on the use of expert opinion.

  - We judged that optimistic assessments of dependence among tasks done by the same person were used.

  - We found apparent nonconsideration of some possibilities for common cause failures from human errors.

  - We found possible insufficient consideration of errors in restoring safety components after test, maintenance, or calibration.

- In general, we found the ZPSS human error probability estimates to be reasonable.

Estimation Methods

- The ZPSS estimates of maintenance unavailabilities appear to be consistent with Zion data.

- The treatment of uncertainty associated with estimates from existing data sources is inconsistent. Generally, five and 95 percent bounds from WASH-1400 were used as 20 percent and 80 percent limits in the ZPSS. Notable exceptions to this were the treatment of interfacing system LOCAs, pressure vessel rupture, and pipe ruptures. In all three cases, substantially higher estimates would have been obtained had their general rule been followed. The results are highly sensitive to this assumption.

- The Bayesian methodology used to estimate accident sequence rates was evaluated. We did not find the chosen prior distributions to be particularly meaningful or compelling. However, where Zion data exist and are used to modify ZPSS's prior probability distributions, the effect of the prior distributions is generally unimportant with respect to the estimated accident sequence rates. Where Zion data are not available or used, the estimates are quite sensitive to the assumed prior distribution. This sensitivity is not acknowledged or accounted for in the ZPSS.

<u>External Events</u>

- The seismic analysis was, in general, difficult to review due to lack of clear documentation. The following observations are made:

  - Sequences considered included only those events caused by the seismic event; that is, combinations of seismic and non-seismic events leading to core melt were not considered.

  - The overall methodology used seems, in general, to be appropriate.

  - The treatment of the choice of the boundaries of the seismogenic zones and the rate of seismic activity are questionable.

  - The summary of seismicity in the main body of the Zion report does not faithfully restate the conclusions and reproduce the results contained in the appendices (Zion report, Module 6, Section 7.9.1).

  - The imposition of an upper bound of effective peak acceleration is unusual and, if relaxed, would probably lead to moderate increases in final mean seismic risk estimates.

  - The definition of damage effective ground acceleration used in the analysis may not be appropriate for electrical and mechanical equipment, buried pipe, and equipment which depends on functional operation as opposed to ductile strength capacity.

- Design and construction errors and aging were not considered.

- The assumption of lognormal distribution for all variables needs further justification.

- We believe that the mean frequency of core melt due to seismic events given in the ZPSS is on the conservative side. We would be surprised if the "true" value was more than a factor of 10 different; however, because of the newness of these type of analyses, a factor of 2 to 3 is possible. However, changes to the tails of the probability density function for core melt are expected to be substantial.

- Because of the need fo subjective input for probabilistic studies of seismic hazard, and the resulting large uncertainties, the results of these studies are more appropriate for making relative comparisons of seismic hazard than determining absolute values.

- The ZPSS fire analysis

  - Only analyzed the auxiliary equipment room and the cable spreading room. Other important areas such as the Auxiliary Building Zone 11.3-0 and the component cooling water pump room were either assessed qualitatively or not addressed at all.

  - Did not address seal LOCA events caused by loss of component cooling water.

  - Did not consider that power to both motor-driven auxiliary feedwater pumps and to the turbine-driven pump regulating valve all run through the same cable spreading room.

  - Did not consider the loss of service water or component cooling water components by fire in conjunction with loss of redundant components due to maintenance.

- Based on the above and several other omissions in the fire analysis, a complete review of the fire analysis was found to be beyond the scope and resources of this study.

- The treatment of floods does not appear to be appropriate for the purposes of a probabilistic risk assessment.

Accident Sequence Analysis

- In general, the ZPSS accident sequence analysis was difficult to follow because of

  - Incorrect and/or incomplete references.

  - Nonmatching numerical results.

  - Unclear or incomplete description of events or the modeling of them.

- The ZPSS inappropriately modeled accident sequences initiated by a loss of component cooling water or a DC bus as well as accidents initiated by a loss of off-site power followed by failure of component cooling water. It became necessary for us to remodel and requantify these accident sequences. These sequences had a significant impact on the results.

- The presence of open terminal blocks inside containment is a concern not addressed in the ZPSS. Recent tests indicate that condensation on the blocks in a post-LOCA environment can cause current leakage in instrumentation circuits, resulting in severe distortion of instrument readings in the control room. The consequences of incorrect operator actions due to false indications of plant status should be analyzed.

5.2 Estimated Plant Damage State/Release Category Frequencies and Sensitivity Issues

5.2.1 Internal Events

Table 5.2-1 summarizes the effect that the findings discussed in the previous sections have on the Zion internal event plant damage states and release category frequencies.

The first column is a listing of 21 plant damage states defined in the ZPSS. The nomenclature is: S or A denotes small or large LOCA, T denotes transient, V denotes interfacing systems LOCA, E or L denotes early or late core melt, F and C denote fans and sprays working, respectively. Also appearing in column one are the mean frequencies of those damage states as calculated in the ZPSS.

The second column represents the revised estimates of the ZPSS plant damage states, based on the significant findings in Sections 2 through 4. It can be noted that a dash appears instead of a frequency estimate in several places. A dash denotes that we did not attempt to recalculate a frequency because these damage states were found to have a small impact on risk as calculated in the ZPSS.

TABLE 5.2-1

Zion Internal Event Results
(Events/Reactor/Yr.)

ZPSS Plant Damage States		Revised Plant Damage States				Revised NRC Defined Plant Damage States			
	Mean		Point Estimate	L95	U95		Point Estimate	L95	U95
SEFC	7.4(-6)	SEFC	~3.0(-4)	4(-6)	1(-3)				
AEFC	1.7(-6	AEFC	1.9(-6)	0	3(-6)				
SEC	1.8(-8)	SEC	1.9(-5)	8(-7)	7(-5)	Early Core Melt With Containment Cooling	~3.3(-4)	2(-5)	2(-3)
AEC	8.2(-9)	AEC	--	--	--				
TEFC	8.3(-7)	TEFC	~1(-5)	8(-7)	2(-3)				
TEC	9.3(-7)	TEC	--	--	--				
SEF	1.3(-9)	SEF	--	--	--				
AEF	1.9(-10)	AEF	--	--	--				
TEF	1.6(-9)	TEF	--	--	--				
SE	6.5(-10)	SE	4.7(-6)	1(-7)	1(-5)				
TE	2.3(-7)	TE	7.7(-7)	1(-7)	2(-6)	Early Core Melt Without Containment Cooling	5.5(-6)	1(-8)	3(-5)
AE	1.1(-11)	AE	--	--	--				
SLFC	1.9(-5)	SLFC	0						
ALFC	9.8(-6)	ALFC	0						
SLC	1.9(-6)	SLC	0			Late Core Melt With Containment Cooling	2.6(-5)	3(-8)	3(-5)
ALC	4.0(-10)	ALC	0						
SLF	4.7(-9)	SLF	1.6(-5)	3(-8)	7(-5)				
ALF	7.3(-10)	ALF	9.8(-6)	0	3(-5)				
SL	1.3(-8)	SL	1.0(-7)						
AL	2.5(-13)	AL	--	--	--	Late Core Melt Without Containment Cooling	1.0(-7)	--	--
V	1.1(-7)	V	1(-7)	0	1(-7)	Bypass	1(-7)	0	1(-7)

The third column represents the revised "NRC defined" plant damage states. The "NRC defined" states consist of the sum of ZPSS damage states listed to the left.

Also listed in columns two and three are the upper and lower 95 percent confidence limits for the damage states. These were obtained by estimating the sum of the accident sequence rates for those dominant sequences that make up each damage state, using the Maximus methodology.

Via comparison, it can be noted that 14 of the 21 ZPSS damage state frequencies have been revised. These revisions are summarized below.

- SEFC - The value ~3.0(-4) is primarily the summation of 4 numbers. They are:

    1. ~2(-4) = loss of component cooling water event discussed in Section 3.2.1,

    2. 4.6(-5) = loss of off-site power event discussed in Section 3.2.3,

    3. 4.0(-5) = loss of off-site power event discussed in Section 3.2.4,

    4. 7.9(-6) = loss of off-site power event discussed in Section 3.2.7.

- AEFC - The value 1.9(-6) is primarily the summation of two numbers. They are:

    1. 1.4(-6) = the large LOCA event discussed in Section 3.2.12,

    2. 4.36(-7) = a medium LOCA and failure of low pressure injection (Sequence 10 on ZPSS Table 8.10-1),

- SEC - The value 1.9(-5) is primarily a loss of off-site power event discussed in Section 3.2.5,

- SLF - The value 1.6(-5) was calculated in Section 3.2.6 and represents a small LOCA event.

- ALF - The value 9.8(-6) is the summation of two numbers. They are:

    1. 4.9(-6) = the large LOCA event discussed in Section 3.2.9,

    2. 4.9(-6) = the medium LOCA event discussed in Section 3.2.10.

- TEFC - The value ~1(-5) is primarily the summation of three numbers. They are:

  1. ~7(-6) = loss of DC bus event discussed in Section 3.2.2.

  2. 1.1(-6) = the loss of off-site power event discussed in Section 3.2.13.

  3. 1.0(-6) = the loss of off-site power event discussed in Section 3.2.14.

- SE - The value 4.7(-6) is a loss of off-site power event discussed in Section 3.2.11.

- V - The value 1(-7) is the interfacing systems LOCA event described in Section 3.2.15.

- SLFC, ALFC, SLC, ALC - The value zero reflects our finding in Section 2.2.1 that these damage states are not possible.

- SL - The value 1.0(-7) is our assessment of ZPSS damage states SL and SLC. In Section 2.2.1, we found that damage state SLC should be SL.

5.2.2 External Events

Table 5.2-2 summarizes the effect that the findings discussed in the previous sections have on the Zion external event plant damage states. (The ZPSS did not report the external event plant damage state frequencies. They were deduced by comparing ZPSS, Tables 8-2 and 8.10-1 and Figure 8.10-1 presented in ZPSS, Section 8 for external events.)

The first column is a listing of the ZPSS external event plant damage states. The nomenclature is: S or A denote small or large LOCA, T denotes transient, E or L denote early or late core melt, F and C denote fans and sprays working, respectively. Also appearing in column one are the deduced mean frequencies of those damage states as calculated in the ZPSS.

The second column represents the revised estimates of the ZPSS plant damage states based on our significant findings in Sections 2 through 4. It can be noted that a dash appears instead of a frequency estimate in several places. A dash denotes that we did not attempt to recalculate a frequency because these damage states were found to have a small impact on risk as calculated in the ZPSS.

TABLE 5.2-2

Zion External Event Results (Events/Reactor Yr.)
(Excluding Fire)

ZPSS Plant Damage States	(Mean)	Revised Plant Damage States*	(Point Estimate)	Revised NRC Defined Plant Damage States*	(Point Estimate)
AEFC	<1(-7)	AEFC	--		
AEF	<1(-7)	AEF	--		
AEC	<1(-7)	AEC	--		
SEFC	<1(-7)	SEFC		Early Core Melt With Containment Cooling	~1(-7)
SEC	<1(-7)	SEC	--		
TEFC	--	TEFC	--		
TEF	--	TEF	--		
TEC	--	TEC	--		
AE	<1(-7)	AE	--		
SE	5.6(-6)	SE	5.6(-6)	Early Core Melt Without Containment Cooling	5.6(-6)
TE		TE	--		
SLF	<1(-7)	SLF	<1(-7)	Late Core Melt Without Containment Cooling	--

*Reflect seismic contribution only. See Section 4.6 for discussion of fire analysis.

The third column represents the revised "NRC defined" plant damage states. The NRC defined states consist of the sum of ZPSS damage states listed to the left. Because review of the seismic analysis resulted in general agreement with the ZPSS numerical results and because we were unable to reach firm quantitative results from review of the fire analysis (see discussion in Section 4.6), our revised plant damage state frequencies, as shown in Table 5.2-2 do not differ from those of the ZPSS.

5.2.3 Combined Internal and External Events

Table 5.2-3 lists the revised dominant core-melt internal and external accident sequences. Table 5.2-4 summarizes the effect that the internal and external event findings have on the "NRC defined" plant damage state frequencies. The frequencies listed in Table 5.2-4 were obtained by summing the frequencies listed in Tables 5.2-1 and 5.2-2.

As can be seen, the revised damage state frequency estimates are within a factor of two of the ZPSS estimate except for "Early Core Melt With Containment Cooling." In the field of PRA, factors of two are usually not considered a significant disagreement.

The difference in the "Early Core Melt With Containment Cooling" category is due primarily to the inclusion of sequences involving loss of component cooling and a DC power initiated sequence in our revised frequency estimate. The ZPSS did not identify such sequences. (See Section 3.)

In closing, it can be noted that we did not attempt to place statistical confidence limits on our final combined internal and external event plant damage state frequencies. Although we estimated confidence limits for internal events, we did not feel comfortable estimating external event uncertainties because of the paucity of data and immaturity of the methodology. We commend the ZPSS authors for attacking this difficult problem, a problem which the majority of PRAs did not address. However, the ZPSS external event data and the mathematical models, as well as the alternative data and models used in our review, are somewhat simplistic.

Rank With Respect to Core Melt	Sequence	Major Plant State/ Release Category	Mean Annual Frequency
1	CCW Failure (causing failure of all charging and SI pumps, seal LOCA)	SEFC	(2-4)
2	Loss of off-site power: failure of component cooling water: failure to recover off-site power in 4 hours	SEFC	4.6(-5)
3	Loss of off-site power: failure of component cooling water: failure to recover off-site power in 1 hour	SEFC	4.0(-5)
4	Loss of off-site power, failure of component cooling water, failure to recover off-site power in 8 hours, failure of containment fans	SEC	1.9(-5)
5	Small LOCA, failure of recirculation cooling	SLF	1.6(-5)
6	Loss of off-site power, failure of component cooling water, failure to recover off-site power in 8 hours.	SEFC	7.9(-6)
7	Failure of DC Bus 111 (causing failure of 1 PORV and loss of AC bus 149), failure of auxiliary feedwater	TEFC	~7(-6)
8	Seismic: Loss of all AC power	SE	5.6(-6)
9	Large LOCA: Failure of recirculation cooling	ALF	4.89(-6)
10	Medium LOCA: Failure of recirculation cooling	ALF	4.89(-6)
11	Loss of off-site power, failure of component cooling water, failure to recover off-site power in 8 hours, failure of containment sprays and fan coolers	SE	4.7(-6)

Figure 5.2-3. Dominant Core Melt Internal and External Accident Sequences (Excluding Fire)

Rank With Respect to Core Melt	Sequence	Major Plant State/ Release Category	Mean Annual Frequency
12	Large LOCA: Failure of low pressure injection	AEFC	1.4(-6)
13	Loss of off-site power: Failure of auxiliary feedwater: failure of feed and bleed: failure to restore off-site power in four hours	TEFC	1.1(-6)
14	Loss of off-site power: Failure of auxiliary feedwater: Failure of feed and bleed: Failure to restore power in four hours	TEFC	1.0(-6)
	Interfacing system LOCA	V	1.1(-7)

Figure 5.2-3. Dominant Core Melt Internal and External Accident Sequences (Excluding Fire) (cont'd.)

TABLE 5.2-4

Revised Zion Combined Internal and External Event Results

NRC Defined Damage State	ZPSS Frequency (Mean)	Revised Frequency (Point Estimate)
Early Core Melt With Containment Cooling	1.5(-5)	3.3(-4)
Early Core Melt Without Containment Cooling	5.8(-6)	1.0(-5)
Late Core Melt With Containment Cooling	3.1(-5)	2.7(-5)
Containment Bypass Prior to Core Melt	1.1(-7)	1.1(-7)

5.2.4 Sensitivity Issues

Presented below is a summary of the results of sensitivity analyses for selected issues.

Issue	Results
Core Melt/System Interaction (Section 4.1)	"Late core melt with containment cooling" sequences become "Late core melt <u>without</u> containment cooling" sequences. The frequency of the latter increases from 1.0(-7) to 2.7(-5) per year.
No Feed and Bleed (Section 4.2)	"Early core melt with containment cooling" plant damage state increased by 48 percent.
Reactor Coolant Pump Seal LOCA (Section 4.3)	Assuming a reactor coolant pump seal LOCA may occur yields: a) 3.3(-4)--"Early core melt with containment cooling." b) 1.0(-5)--"Early core melt without containment cooling." c) 2.7(-5)--"Late core melt with containment cooling." Assuming a reactor coolant pump seal LOCA does not occur yields: a) 1.5(-5)--"Early core melt with containment cooling." b) 1.2(-6)--"Early core melt without containment cooling". c) 2.6(-5)--"Late core melt with containment cooling."
Testing of the Room Cooling System (Section 4.4)	Assuming that room cooling systems are tested monthly yields: a) 2.7(-5)--"Late core melt with containment cooling."

Issue	Results
	b) 1.7(-7)--"Late core melt without containment cooling."
	Assuming that room cooling system is not tested yields:
	a) 2.1(-2)--"Late core melt with containment cooling."
	b) 1.7(-7)--"Late core melt without containment cooling."
ATWS (Section 4.6)	Assuming that the turbine can be tripped by loss of main feedwater yields: 3.3(-4)--"Early core melt with containment cooling."
	Assuming that the turbine cannot be tripped by loss of main feedwater yields:
	7.1(-4)--"Early core melt with containment cooling."
Fire Analysis (Section 4.6)	Assuming that the results of the ZPSS fire analysis are correct yields:
	a) 1(-5) for "Early core melt without containment cooling."
	Assuming that gross estimates of the effects of sequences omitted or incorrectly evaluated in the ZPSS are correct yields:
	a) 8(-5) For "Early core melt without containment cooling."
Success Criteria (Section 4.6)	Assuming that two CCW pumps are required for CCW system success and two SW pumps are required for system success yields:

Issue	Results
	a) ~3.3(-4)--"Early core melt with containment cooling."
	b) 1.0(-5)--"Early core melt without containment cooling."
	Assuming that one CCW pump is required for system success and three SW pumps are required for system success yields:
	a) ~3.4(-4)--"Early core melt with containment cooling."
	b) ~1.1(-5)--"Early core melt without containment cooling."

Appendix A

REVIEW
OF THE
ZION PROBABILISTIC SAFETY STUDY
SEISMIC FRAGILITY
AND FLOODING

by

John W. Reed

Martin W. McCann, Jr.

prepared for

Sandia National Laboratories

Albuquerque, New Mexico

May 18, 1983

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1. INTRODUCTION

Jack R. Benjamin and Associates, Inc. (JBA) was retained by Sandia National Laboratories (Sandia), Albuquerque, New Mexico to perform an in-depth critical review of the following sections of the Zion Probabilistic Safety Study (referred to as the Zion PRA report); NRC Docket Numbers 50-295 and 50-304, not dated:

Summary Report

II.2.2, paragraph 1	Major Contributions to Risk - Major Seismic Event
II.7.1	External Events - Seismic
II.7.3	External Events - Flooding

Main Report

7.2	Seismic Events
7.9.2	Conditional Probabilities of Seismic-Induced Failures For Structures and Components For the Zion Nuclear Generating Station
7.9.3	Comments on Effective Ground Acceleration Estimates
8.8.1	Determination of Risk From External Initiating Events - Seismic Risk

Except for Section II.7.3, these Zion PRA report sections give the results of the probabilistic analysis of the Zion plant for the effects of earthquakes and in particular the structural fragility calculations. In addition, Sandia requested JBA to make comments on other parts of the Zion PRA report which pertained to the seismic hazard curves and the systems analysis.

Paragraph 1, Section II.2.2 and Section II.7.1 are sections from the summary report, which repeat in a condensed form information given in Section 7 of the main report (in particular, Section 7.2). Rather than offering comments for the material in the summary report, which is repeated in the main report, all comments are given for the main report sections. Comments for the seismic hazard curves and the systems analysis are integrated with our comments for the main report section 7.2.

Section II.7.3 summarizes the flooding analysis which was performed for the Zion plant. Since more information is given in the main report, Section 7.4, our comments are given for this section rather than Section II.7.3 from the summary report.

The review of the Zion PRA report was directed by Dr. John W. Reed. Dr. Martin W. McCann, Jr. assisted Dr. Reed in review of the report sections, concentrating primarily on the seismic hazard analysis and the flooding analysis. Dr. Jack R. Benjamin participated in the review of the concrete capacity analyses reported in Section 7.9.2.

The remaining chapters in this report discuss the review of the overall methodology, provide review of specific Zion PRA report subsections, discuss the review of the fragility calculations, and end with the final conclusions of the review and recommendations. These chapters are entitled:

Overall Methodology
Report Sections
Calculations
Conclusions and Recommendations

The remaining sections of this chapter describe the approach used to review the Zion PRA report and present the results of a sensitivity study which was conducted to gain insight into the stability of the integration of the seismic hazard and fragility curves.

In order to avoid confusion in reading this report, the chapter sections are not numbered. The figures, tables, pages and references are each numbered consecutively. In contrast, sections, figures, tables and pages of the Zion PRA report have a decimal (or sometimes dashed) numbering system. By organizing the review report in this manner, references to the locations of material in the Zion PRA report and in this report are more apparent.

The major part of the review was conducted during January and February of 1982. A draft report was issued 22 February 1982, which listed findings to date, including questions which we were unable to asnwer. Subsequent to the draft report, seventeen of the most important questions were formally transmitted to the Licensee. These questions and the Licensee's responses are contained in Appendix A of this report. In April 1983, Dr. Reed visited the Zion site and inspected the plant facilities. In addition the reviewers, during the period between the draft and final reports, participated in the review of the Indian Point Probabilistic Study (Ref. 21) which included similar analyses for both seismic and flood external events. Based on the Licensee's responses to the seventeen questions, the Zion plant inspection, and additional information obtained during the review of the Indian Point study, the draft report was revised. This report presents the final results of the review of the Zion PRA report.

REVIEW APPROACH

A dual approach was used to review the Zion PRA report. One part consisted of systematically reading, reviewing, and commenting on the sections and subsections of the report. In the second part, the review consisted of a continuous search for the parameters, assumptions, etc. which controlled or contributed significantly to the results of the analysis. As part of this effort, a sensitivity study was conducted to determine how the mean frequency of core melt changes as the relationship between the hazard and fragility curves is varied. Using this procedure, structures and equipment which contributed significantly to the frequency of core melt were identified. The review effort concentrated more heavily on the major contributors.

The Seismic Safety Margins Research Program (SSMRP) being conducted by the Lawrence Livermore National Laboratory (LLNL) for the Nuclear Regulatory Commission (USNRC) is currently developing a procedure for estimating the risk of an earthquake-caused radioactive release from commercial nuclear power plants. Zion Nuclear Generation Station has been used as a model facility for the development of the SSMRP methodology. We have utilized the results which have been published prior to February 1982 for the SSMRP in our review of the Zion PRA report. It should be noted that the engineers who contributed to the development of fragility data for the SSMRP are the same professionals who performed the fragility analyses for the Zion PRA report. In this sense, the results of the SSMRP studies are not an independent comparison of the PRA results. However, numerous detailed analyses of the structures and probabilistic sensitivity studies have been performed in the SSMRP, which provide an independent indication of the appropriateness of some of the assumptions made in the Zion PRA study.

We have used selected results from the SSMRP in our review of the Zion PRA report. Because the published results from the SSMRP at this time are not complete, we have not attempted to use the SSMRP to make general overall comparisons. We feel that when the SSMRP studies of Zion are completed, it would be worthwhile to perform a detailed comparison between the two approaches.

In our review, we have attempted to make comments on both minor and major issues, looking for both conservative and unconservative assumptions. In order to help the reader and to maintain perspective ourselves, we have tried to indicate, where possible, the ultimate impact of the issues which we have raised. As an aid in doing this we have selected the mean frequency of core melt as the basis for comparison. We have adopted the following scale to quantify our comments in reviewing the Zion PRA report:

Comment	Effect on Mean Frequency of Core Melt
Small	Factor $\leq$ 2
Moderate	2 < Factor $\leq$ 10
Large	Factor > 10

We have indicated in our report in several places where effects of changes in parameters will have a greater effect on the tails of the frequency of the core melt density function. In general, we expect a greater impact on the tails as compared to the mean frequency; however, we feel that the mean frequency is a more important parameter to the Zion PRA study. In general, we do not feel that the tails of the probability density function on core melt are particularly meaningful, except in a relative sense (i.e., the density function is very broad).

SENSITIVITY ANALYSIS

As part of our attempt to understand how changes in the analysis parameters might affect the mean frequency of core melt, we integrated the hazard and fragility curves using the same discrete probability distribution procedure applied in the Zion PRA report. The value given in the report is 5.6×10^{-6} per year, which was used as the value for comparison.

We first integrated the hazard curves using the discrete values given in Table 8.8-1* with the fragility data tabulated in Table 7.2-4 (note that in all our analyses we used the same fragility values--

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* The tables and figures with decimal system notation are from the Zion PRA report.

specifically the Table 7.2-4 values). It was our understanding based on a discussion with Pickard, Lowe, and Garrick (PLG) that the hazard values in Table 8.8-1 are equal to the Dames and Moore report results from Section 7.9.1 where the acceleration axis had been shifted by a factor of 1.23 and truncated for the assumption of maximum acceleration cut-off. Our understanding is that the Table 8.8-1 values were used in the final PRA analysis to obtain the frequency of core melt distribution. Our analysis produced a value of 4.2×10^{-6} per year which is about 25 percent smaller (a small effect) than the value reported in the Zion PRA report. In trying to understand the reason for the difference, we plotted the Table 8.8-1 curves along with the curves obtained from Section 7.9.1 which were shifted by the factor of 1.23 and truncated, as described in the Zion PRA report. The curves we plotted are shown in Figures 8, 9, and 10, which are found at the end of Chapter 3. As can be seen from these figures, the Table 8.8-1 curves exceed the Section 7.9.1 curves (which have been shifted and truncated) above the acceleration value of roughly 0.3g. It was found by studying the contributions from different acceleration regions that most of the mean frequency of core melt value comes from contributions from the integration process above 0.3g. Based on the Licensee's response given in Appendix A (Question 1), we believe that the hazard values given in Table 8.8-1 are incorrect. The comparison of the hazard curves in Figures 8, 9, and 10 leads us to believe that the 25 percent difference which was found may be due to the apparent error in the Table 8.8-1 values.

We ran several additional runs to quantify the sensitivity of the input. Below is a tabulation of the runs and the result.

INTEGRATION PROCESS

Hazard Curves	Fragility Curves	Mean Annual Frequency of Core Melt
Section 7.9.1 - all 27 curves	Table 7.2.4	5.9×10^{-6}
Section 7.9.1 - 9 curves each analysis		
only 0.5g cut-off curves	Table 7.2.4	3.8×10^{-6}
only 0.8g cut-off curves	Table 7.2.4	6.8×10^{-6}
No cut-off curves	Table 7.2.4	6.8×10^{-6}
Table 8.8-1 shifted by 1.25	Table 7.2.4	7.5×10^{-6}
Table 8.8-1	Table 7.2.4	4.2×10^{-6}

Note that the 27 hazard curves in Section 7.9.1 consisted of three sets of nine curves with corresponding maximum acceleration cutoff values of 0.5g, 0.8g and no cutoff. The curves were assigned probability values which summed to unity. Several conclusions can be made based on these results:

1. Shifting the hazard or fragility curves by a factor of 1.25 causes less than 100 percent change in the mean frequency of core melt (i.e., 4.2×10^{-6} compared to 7.5×10^{-6}--a small effect).

2. Truncating the hazard curves changes the results by less than 15 percent (i.e., 5.9×10^{-6} compared to 6.8×10^{-6}--a small effect).

3. Truncation of the acceleration values at 0.8g or larger is the same as no truncation.

The experience we gained in these analyses was used in estimating the effects of potential changes of individual parameters of the safety-related structures and components.

2. OVERALL METHODOLOGY

The methodology used in the Zion PRA report for seismic effects is appropriate and adequate to obtain a rational measure of the probability distribution of the frequency of core melt. The procedure is based on a simple probabilistic model which uses some data, but currently relies heavily on engineering judgment. In the application of the methodology, we offer the following comments.

The notion of separating variability into randomness and uncertainty components is an appropriate concept. Randomness by definition is irreducible while uncertainty in the parameters and models can be eliminated by analysis, testing, research, or contributions of these techniques. However, it is our experience that in practice these definitions become blurred. What is randomness today may be uncertainty tomorrow. In other words, as the state-of-the-art advances, new techniques are developed which can be used to solve problems which yesterday were unsolvable. Even the classic example of the randomness of compressive stresses obtained from testing concrete cylinder samples may someday fall prey to an advanced analysis technique. Hence, knowing for certain the values of some obscure set of parameters (e.g., aggregate shape and location, cement properties, etc.), the compressive stress may be predicted almost perfectly. In reality, this may never occur, because today we have remaining such a small randomness component that there may not be sufficient incentive to pursue the development of a more refined theory.

In the methodology used in the Zion PRA report, the median capacity value is the only uncertain parameter. It should be kept in mind that there are other uncertainties associated with the methodology (e.g., randomness β_r, the logarithmic model, and even β_u itself). What is implicitly assumed is that the variability in these other parts of the methodology, is relatively small so that their uncertainty can be neglected. Also, there is some evidence that variability may be constant with response level (Ref. 1).

There are some who believe that all variability is uncertainty and the frequency of failure (fragility) curve for a component is equal to 0 up to some uncertain acceleration value and equal to 1 for higher values (i.e., the "cookie-cutter" fragility curve). Others choose to think of variability as being all randomness (e.g., the procedure used in the SSMRP). The Zion PRA report has taken a middle road and considers both types to be present. The implication of how dependencies are affected by these two types of variability is discussed later in this report. We personally feel that generally it is more rational to have more uncertainty and less randomness for structural components subject to seismic and other forces.

It is important that the industry adopt a consistent approach to be applied to PRA analyses. In this manner, results between PRA's can be compared (e.g., "apples with apples"). It is naive to think that the answers we produce are absolute truth. The best we can do today is to be rationally consistent and to communicate to others exactly how our analyses are performed, so that the results can be compared in a relative sense.

After reviewing the procedures used to produce the fragility data we have a general impression which bears on the issue of consistency. We feel that the uncertainty of the parameters in the Zion PRA report has probably been understated. There are various levels of sophistication which have been used to develop the fragility parameter values, but we do not sense that enough uncertainty has been assigned to components where parameter values are based on more distant information. Although in fairness to the Zion PRA report, the values for β_u are generally larger for generic components as compared to plant specific components. We indicate in the discussion for the various Zion PRA report sections where the uncertainty assignment may be low.

On the other hand, we also believe that the median capacity values are probably low. Structural and mechanical engineers have an inbred tradition to be conservative, and our guess is that this tendency has persisted in developing median capacity values. It is useful to remember that the median value is the value in which there is a 50 percent chance that the "true" value is larger. We suspect that over-conservatively stating the median values and understating the uncertainties is sufficiently self-compensating such that reasonable final results are still obtained.

Several obvious elements of uncertainty have been left out of the seismic fragility analysis. First, design and construction errors (e.g., the problem of piping supports at Diablo Canyon) and aging effects are not generally included in the seismic fragility or fault tree analysis. These become extremely important for series systems such as piping systems and cables (i.e., cable trays). One failure and the system may be lost. We noted for several subsections which we reviewed that the authors did not check the calculations which formed the basis for the fragility parameters which were developed. Thus, errors in the calculations <u>could not</u> be discovered. The Licensee's response given in Appendix A (Question 11) addresses the issue of design and construction errors and aging. Our conclusion is that the issue is only partially addressed and that in general, design and construction errors and aging are not systematically included. This is a complex issue and considerable research is needed before it will be resolved.

In an approximate way the lower tail on the lognormal distribution for capacity accounts for possible errors. This is true since the capacity tail goes to zero which is not supported by reality. However, the frequency distribution for design and construction errors certainly varies from component to component. Since the lognormal tail is a function of only the capacity parameter, it may or may not properly account for these types of errors. Our conclusion is that design and construction errors are not specifically accounted for in the analysis.

Another uncertainty (and bias in the median value) is created by the fact that structural components are not built to produce the maximum allowable stress. Construction practices dictate that components generally are stronger than needed. It is tempting, but incorrect, to say that design and construction errors can be balanced by over-construction such that these effects in total can be neglected. We feel these considerations individually should be taken into account in the systems analysis.

In the Zion PRA report the weakest part of a structure or equipment was used to develop fragility data. In general, this approach is satisfactory. It should be pointed out that it is possible for a slightly stronger part to produce a greater frequency of failure. This occurs if the variability of the stronger component is large enough to overcompensate for the weaker but less variable part. Thus, it is not always sufficient to consider just the weakest part. Slightly stronger parts should also be reviewed and discarded if their variability is found to be relatively small.

One approach used to develop fragility curves was based on analysis of generic data. Rather than working with the analysis of a plant specific component, failure and/or response data from similar components in similar environments are used as the basis to develop a fragility curve for the particular plant component being considered. We feel this procedure is appropriate under certain circumstances. If after determining the fragility of a particular plant component using generic data it is found that the capacity is sufficiently high so that the component does not influence the frequency of core melt analysis, then we feel the analysis is appropriate. On the other hand, if the component is found to have a low capacity such that it influences (or could if changed by a small amount) the frequency of core melt analysis, then a more detailed analysis for that component should be conducted. Examples of components that fall in this category are:

Component	Median Ground Acceleration Capacity (g)
Ductwork and dampers	0.97
Batteries and racks	1.01
Relief tank	1.19
Transformer	1.39

We feel that because the batteries and racks could affect the core melt analysis, a more detailed analysis should be conducted to verify that the generic-based capacity is appropriate. In regards to the other three components, the Licensee's response given in Appendix A (Question 6) convinces us that detailed analyses are not necessary.

It is important that median parameter values be selected to give frequency of behavior (i.e., failure, capacity, response, etc.) at acceleration values which are significant to the frequency of core melt analysis. For example, in the integration of the hazard and fragility curves, most of the mean frequency of core melt came from the range of acceleration values between 0.3 and 0.6g. Thus, in developing the median factor for damping, the stress level in a structure for this range of accelerations should be taken into account in selecting a damping value. If the stress level is less than yield, then 3 percent may be appropriate, or if yield level is reached, 10 percent would be more representative. This is particularly important for equipment items which have natural frequencies close to a fundamental building frequency. Other examples can be cited. In our review, we kept this concept in mind.

One assumption implicit in the methodology is that everything occurs at once, and no phasing of events is considered. Structures and components either fail or do not fail at the same instant in time. For ductile structures, the loading sequence is less critical compared to the maximum load or number of cycles of large motion. For brittle elements, the loading sequence is more important. There is a dependence between the loading and response in reality, because structures fail sequentially leading to many possible failure histories.

We wonder how this process might be applied to electrical control functions and the interaction of electrical equipment functional failures with failures of structural elements. The elimination of the electrical components from the analysis because their capacities are apparently high should be reconsidered in light of this idea.

As reviewers, there is one area which is missing from the Zion PRA report which should be part of all public documentations of PRA studies. Results of sensitivity calculations should be performed to provide the reader with an understanding of what elements control the results of the analysis. For example, how sensitive is the frequency of core melt to the upper-bound earthquake magnitude cut-off? What would happen to the mean frequency of core melt if the median acceleration capacity of the service water pumps was one-half of the computed value? As discussed in the introduction chapter, we have attempted to do this to a small degree to assist us in our review. We feel that the results of sensitivity studies should be provided as part of all basic PRA documentation.

In our review of the Zion PRA report, we spot-checked calculations which could easily be done as we read the report. We also performed sensitivity studies of the hazard and fragility curve integration (see Chapter 1). As documented in Chapter 4, Dr. Reed met with Structural Mechanics Associates (SMA) and reviewed the fragility calculations for seven structure/components. We did not review in any detail the original structural calculations for the Zion Plant. We did review one report (Ref. 2).

As a result of our tour of the Zion plant, we question whether the Zion PRA report has considered all possible failures of non safety-related structures or equipment, which could impact on safety-related items. We believe that a systematic study should be conducted to identify and quantify the effects of all possible secondary failures throughout the entire plant which could affect safety-related structures and equipment.

One area which we have not commented on concerns the adequacy of the fault and event trees, except we question the absence of consideration for a moderate size earthquake occurring during a time when some safety-related components may not be available due to maintenance procedures, etc.

Thus, for the purposes of our review, we accept the fault trees given in Section 7.2. In addition, Sandia has reviewed these trees and has determined that the safety-related components which are included are complete. Based on the fault trees presented in this subsection, we checked the Boolean algebra and determined that the final expression for M_S is correct. We noted in Section II.7.1.3 (summary section) that the event: "Failure of Safety Inspection Pumps" is included in the Boolean equation. Based on the fault trees, failure of the safety injection pumps cannot cause core melt since the charging pumps will be available. However, in Section 7.2.4 (main report section), this event is absent from the Boolean equation for core melt, which is correct.

We accept the methodology used in the Zion PRA report. Our comments on specific aspects are offered in the context that the methodology is appropriate for a PRA study.

3. REPORT SECTIONS

Comments given in this chapter are directed to specific sections of the Zion PRA report. The comments are organized by the following main report sections:

7.2 External Events - Seismic

7.4 External Events - Flooding

7.9.2 Conditional Probabilities of Seismic-Induced Failures For Structures and Components For the Zion Nuclear Generating Station

7.9.3 Comments on Effective Ground Acceleration Estimates

8.8.1 Determination of Risk From External Initiating Events -- Seismic Risk

7.2 EXTERNAL EVENTS - SEISMIC

No comments for introductory paragraph.

7.2.1 Methodology

We agree that a seismic safety analysis consists of the five main steps which are listed in this subsection.

7.2.2 Seismicity

We agree that the overall seismic hazard methodology utilized by Dames and Moore and described in Section 7.9.1 is appropriate. However, we feel that a number of assumptions in the application of the methodology need further clarification as to their appropriateness.

In the seismic hazard analysis, the variability in ground motion attenuation has been accounted for by a lognormal distribution with a standard deviation, $\sigma_{\ell n\ a}$, of 0.60, a typical value used in ground motion attenuation models. The basis for this value is the scatter observed in acceleration strong motion data. Recent studies suggest that $\sigma_{\ell n\ a}$ is a composite parameter whose components include travel path, building, and local geologic effects (Refs. 3 and 4). In fact, the variability due to buildings has been identified as a function of the depth of embedment. In the seismic risk analysis, soil-structure interaction effects and variability in response are considered. Since free-field accelerations are specified, it may be more appropriate to account for the part of variability in the attenuation equation due to building effects in the soil-structure interaction factor. The standard deviation corresponding to embedment effects, β , was found to be approximately 0.16, corresponding to a factor of 1.17 for data from the 1971 San Fernando earthquake (Ref. 3). However, we feel that removal of this component from the randomness in attenuation will have a small effect on the frequency of ground shaking levels.

In Section 7.9.1, the lognormal distribution about the attenuation curve has been truncated to reflect the possibility that the effective accelerations are limited. To account for this possibility, three levels of a_{max} were assumed in the analysis. These a_{max} values are independent of magnitude and distance. We agree that upper limits on a_{max} should be considered; however, the manner in which the analysis was conducted is inconsistent with the arguments presented in this section for limiting accelerations on the basis of a dependence on maximum magnitudes (and hence intensities). The approach used in the Zion PRA report requires that the lognormal distribution be normalized (Ref. 5) for each value of M_b and distance (see Figure 1). An alternative approach would have been to truncate the distribution at a level equivalent to a constant number of standard deviations (i.e., $2\sigma_{\ell n\ a}$, $3\sigma_{\ell n\ a}$, etc.), or an a_{max} which is dependent on M_b and distance. However, we judge that the overall effect on the annual

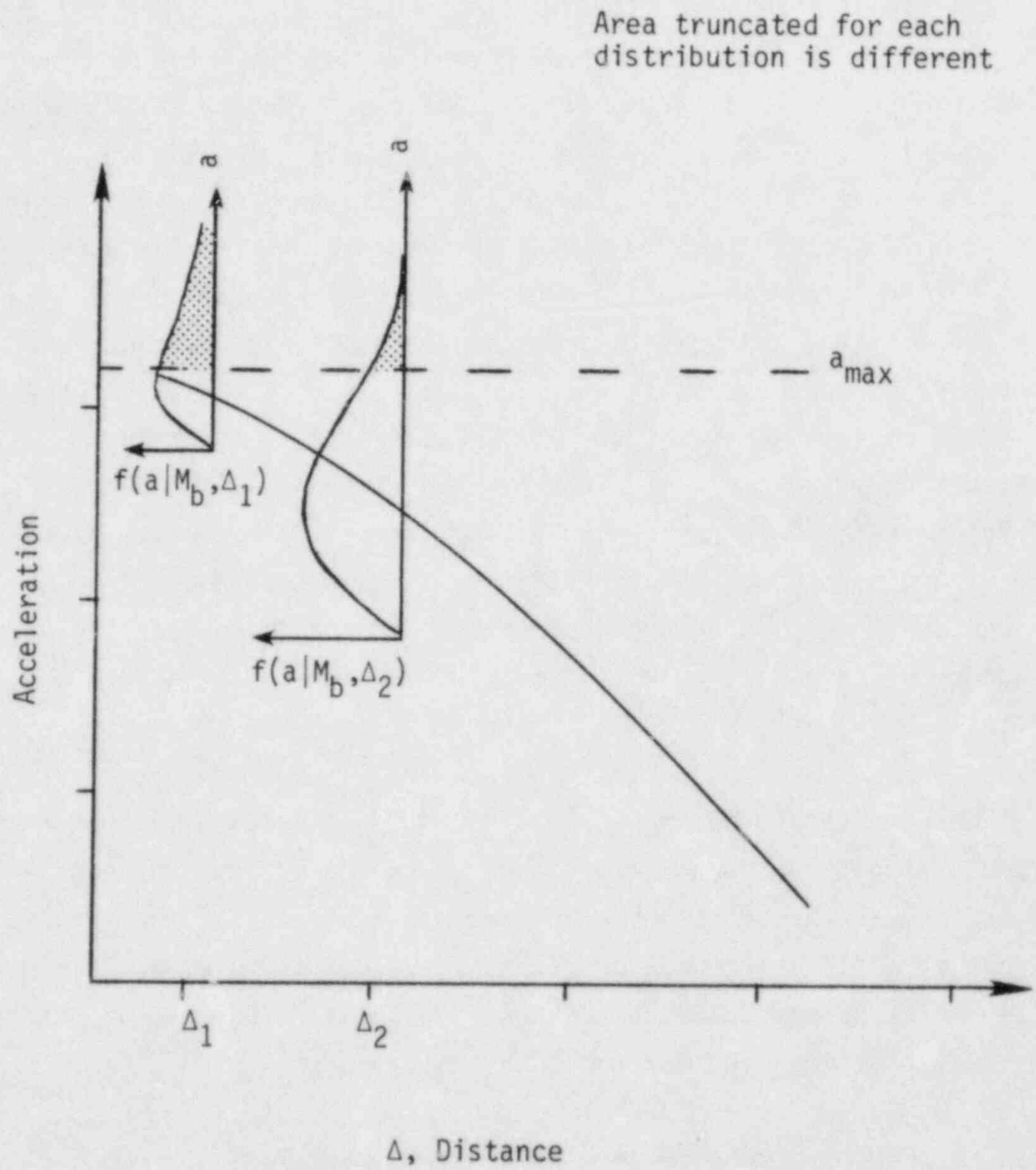


Figure 1 Demonstration of Truncation of the Attenuation Curves

frequency of exceedance curves, and the frequency of core melt will be small if the distributions were in fact not normalized.

The seismic hazard curves presented in Section 7.9.1 have been modified in two ways before being integrated with the seismic fragility curves. First, peak acceleration values have been shifted by a factor of 1.23, and second, hazard curves have been truncated to reflect the belief that there is a maximum ground shaking intensity which can occur.

In regards to the definition of ground motion, we believe that it may be more appropriate to use 1.25 a_s (where a_s is the sustained level of acceleration corresponding to the third highest peak in the acceleration time history) as the damage-effective ground acceleration value for <u>some</u> components in the Zion analysis. This is in contrast to the value of a_s which was used as the damage-effective ground acceleration value for <u>all</u> structures and components. The basis for our viewpoint is given in our review of Section 7.9.3, "Comments on Effective Ground Acceleration Estimates". However, adopting this definition for all structures and components would increase the mean frequency of core melt by approximately 80 percent, which is a small effect.

The results of the ground motion hazard analysis have been modified by truncating the hazard curves at acceleration values corresponding to the maximum intensity values considered appropriate for the Zion site. The basis for limiting peak ground acceleration given a specific value of intensity is discussed in Section 7.9.3. We offer comments here in regards to limiting the maximum intensity value.

The results of the seismic hazard analysis, Table 3 from Section 7.9.1, have been truncated by using the following relationship between magnitude, M_b and intensity, I_o:

$$M_b = 0.5\ (I_o + 3.5)$$

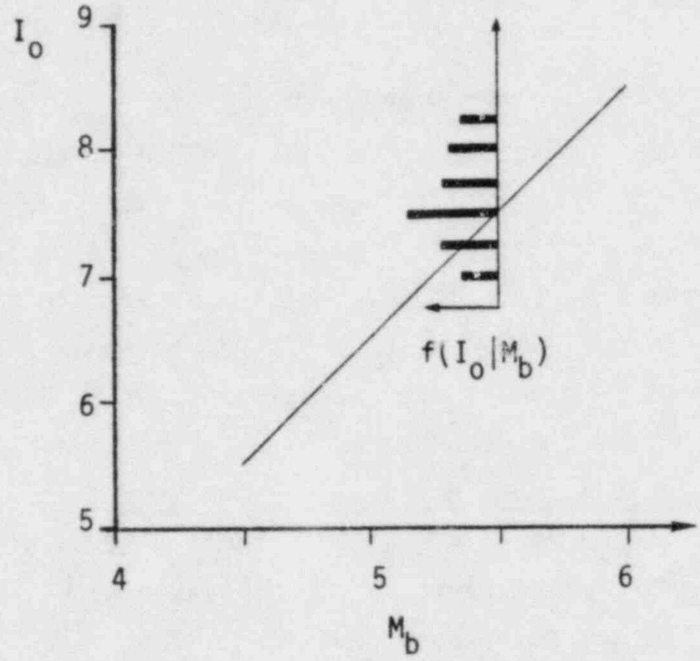
Maximum M_b values were assumed and thus using the above equation, upper-bound intensity values were the basis to limit acceleration values. However, since this equation is uncertain (see Figure 2a), this leads necessarily to a distribution on maximum estimates of ground acceleration (Figure 2b). The maximum magnitudes used in the seismic hazard analysis were 5.6, 5.8, and 6.0. The corresponding maximum accelerations in g units are approximately 0.43, 0.55, and 0.64. It is not evident from the text how these upper-bound accelerations were used to modify Table 3 from Section 7.9.1 to give Figure 7.2-1 of the Zion PRA report (repeated in Figure 3). These maximum accelerations apparently correspond to values interpolated on the basis of the implied I_{mm} - M_b - EPA_{max} relationship given in Section 7.9.3.

In the seismic hazard analysis, predictions of sustained acceleration at the Zion site were made by the following path,

$$I_o \rightarrow M_b \rightarrow a_s$$

where an arrow refers to an empirical relationship. Using the same sequence of steps, although with a different set of arguments, limits on sustained acceleration were obtained. With respect to ground motion attenuation, the variability in the $I_o \rightarrow M_b$ step is not included in the estimate of a_s. Further, in establishing limits on a_s, the variability in the path $I_o \rightarrow M_b \rightarrow a_{smax}$ is also not taken into account (see Figure 2) in the final family of seismicity curves. We note that in Section 7.9.1, variability in the maximum acceleration was considered. However, these results were not used in the seismic risk calculations. The logarithmic standard deviation value used in the seismicity study was 0.60, which is a value typical of the variability in magnitude - distance regressions on peak ground acceleration. The effect of increased values of $\sigma_{\ell n\ a}$ (increased due to uncertainty in the path taken to estimate a_s) on seismic risk is judged too small.

(a)



(b)

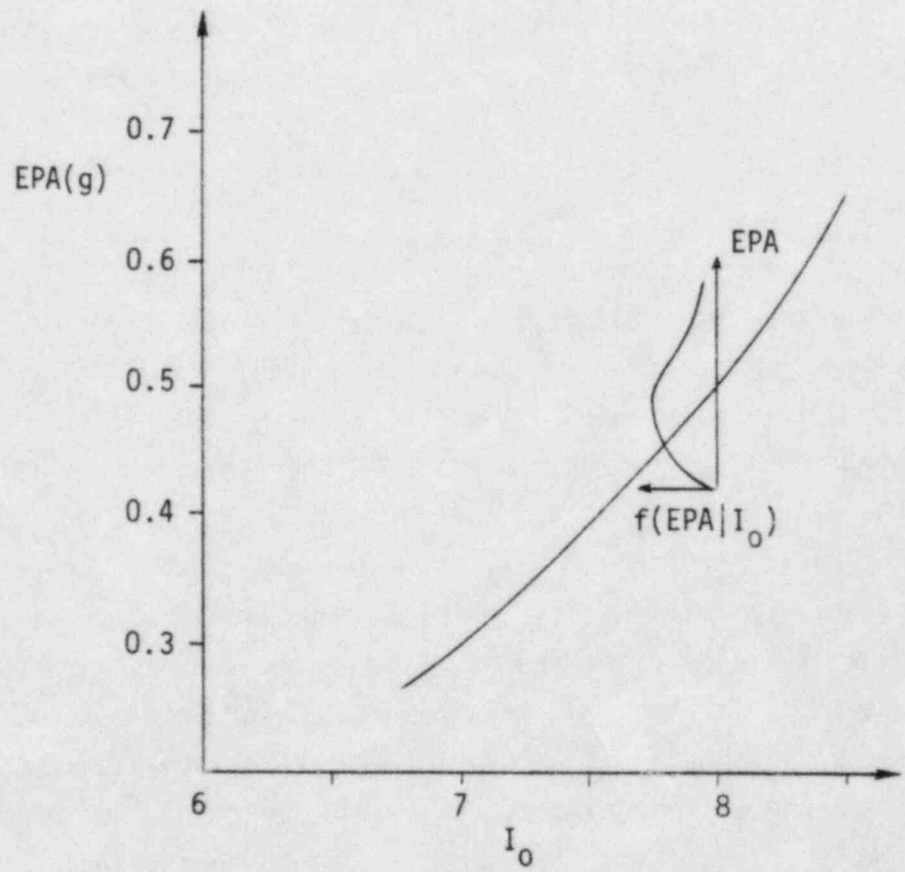


Figure 2 (a) M_b - I_o Relationship;

(b) I_o - Upper Bound Acceleration Relation

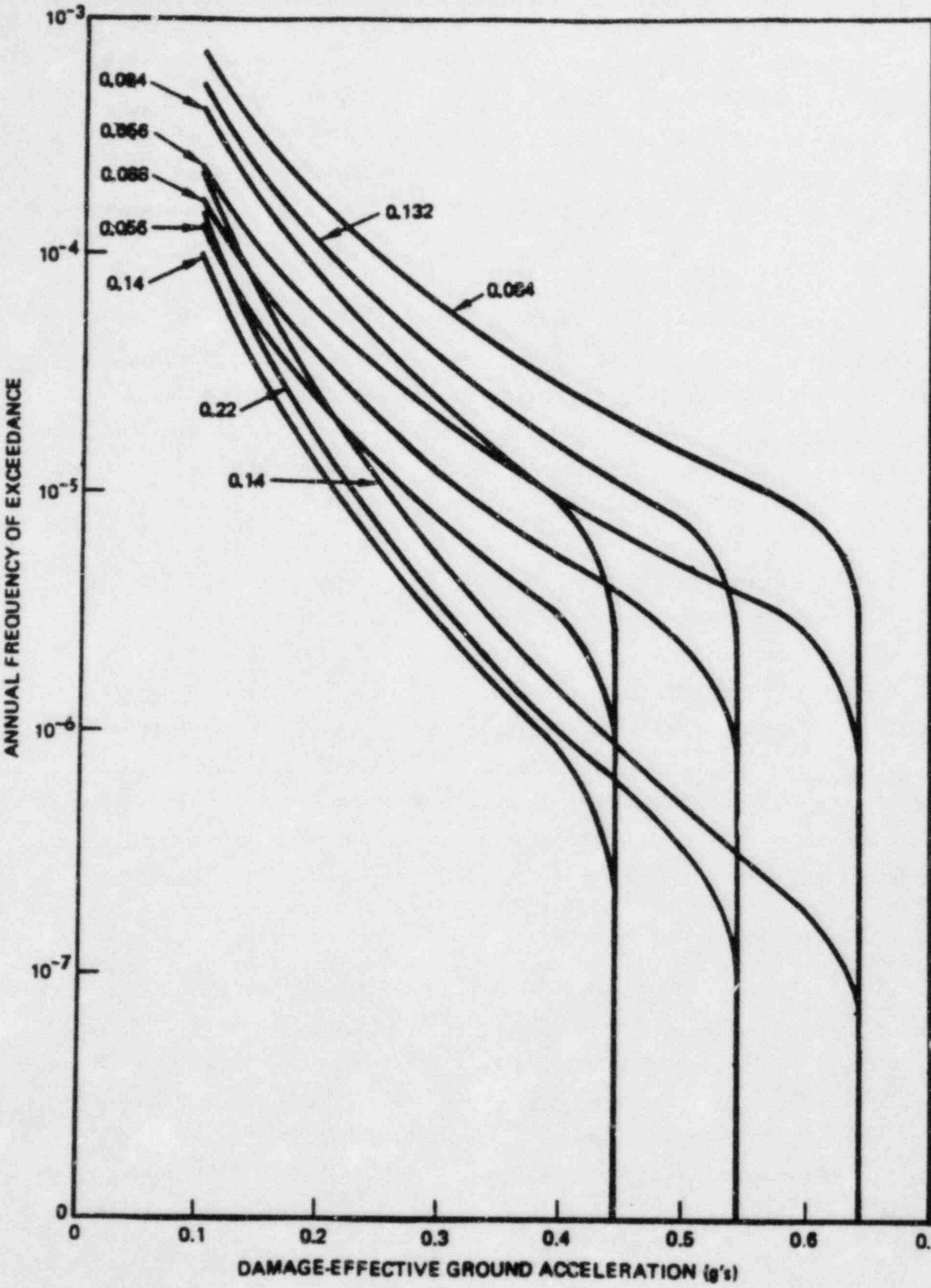


Figure 3 The Seismicity Family Curve From the Zion PRA Report (Figure 7.2-1)

We agree that the possibility of upper-bound accelerations should be considered. However, based on the concerns presented above and in our review comments for Section 7.9.3, we believe this effect could have been incorporated in a more realistic manner in the risk analysis. Note that one truncation of acceleration values was made in the Section 7.9.1 analysis and again in subsection 7.2. We believe that it is more appropriate not to truncate the hazard curves, but to reflect a limit on damageability in development of the fragility curves. The mechanism to handle this effect is currently not an element of the fragility analysis. A new factor or a redefinition of an existing factor is required to treat the frequency dependent effect. By virtue of calculations we have made on the frequency of core melt, we feel that only small changes in the hazard curves and the mean frequencies of core melt would result in making any of these modifications.

We have made a comparison of the mean Zion report hazard curve with results from the SSMRP study. The SSMRP study provides information based on expert opinion surveys, a probabilistic seismic hazard analysis, and the estimated historical record. The resulting curves are shown in Figure 4. The largest discrepancies between the Zion report and the SSMRP curve occur at accelerations below 0.2g. Above this level, the median results are quite similar. We conclude from this and our previous comments that the Zion hazard curves are in general appropriate, and that any modifications to the analysis will result in only small changes.

As explained in the first chapter and in the discussion above, it is not clear exactly how the hazard curves presented in Section 7.9.2 were modified before integration with the fragility curves. In Section 7.2.2 it is claimed that the acceleration values were decreased by a factor of 1.23. Based on personal correspondence with PLG it was verified that the values from Table 8.8-1 (from Section 8.8, Determination of Risk from External Initiating Events) were used in the final computer run; however, the Licensee's response given in Appendix A

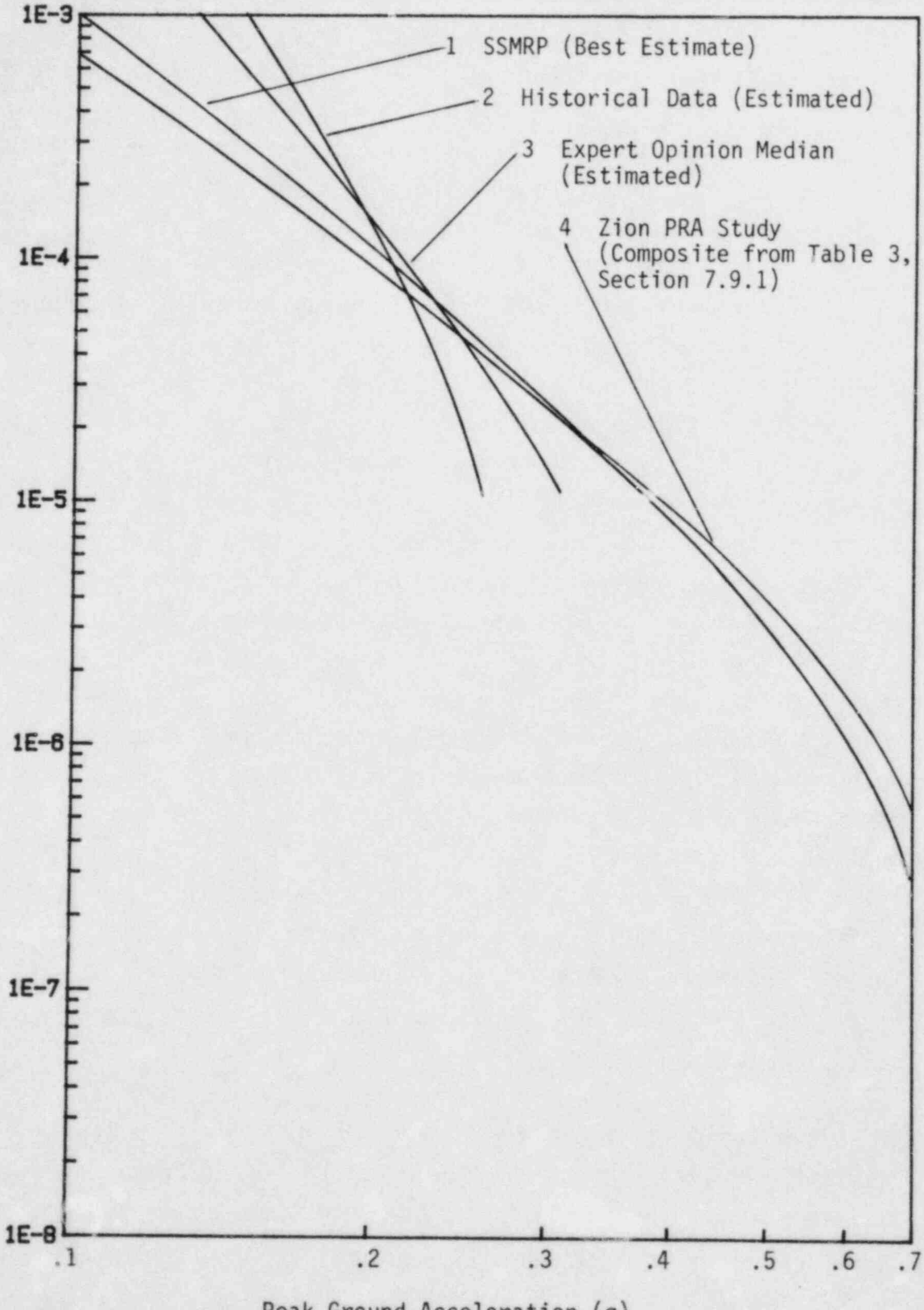


Figure 4 A Comparison of the Median Zion Hazard Curve With Results of the SSMRP Study

(Question 1) leads us to believe the Table 8.8-1 values are incorrect. An attempt by us to duplicate the analysis produced a mean value of core melt frequency equal to 4.2×10^{-6} per year, which is 25 percent lower than the value quoted in the Zion PRA report (i.e., 5.6×10^{-6} per year). Other computer runs were made to determine the influence of shifting the acceleration axis by a factor of 1.25 to increase the frequency of core melt to investigate the influence of using 1.25 a_s (rather than a_s, see discussion above and for Section 7.9.3) as the damage effective ground acceleration value. From our analysis, the calculated mean value was found to be 7.5×10^{-6} per year. This was surprising since this value exceeded the 5.9×10^{-6} per year we computed using the hazard values directly from Section 7.9.1 (see Chapter 1). In resolving the differences, Figures 8, 9, and 10 (which are given at the end of this chapter) were prepared which show the hazard curves from Table 8.8-1 and the Section 7.9.1 curves with the acceleration axis shifted by a factor of 1.23 and truncated for the effects of maximum acceleration cut-off. It is seen that the two sets of curves are not the same, which they should be according to the Zion PRA report.

There may be some error in the above comparison due to the differences in the integration schemes which we and PLG used. We estimate the differences to be less than 30 percent. In summary, we believe that the value of 5.6×10^{-6} per year given in the Zion PRA report for mean core melt frequency is reasonable. Our judgment is that shifts in the hazard curves for the types of factors discussed above result only in small changes to the final results.

7.2.3 Fragility

This subsection discussed, in general terms, the methodology used to develop the fragility curves for structures and equipment. We agree that this methodology is appropriate for the Zion plant. The basis for accepting the methodology and specific comments concerning application of this methodology to the Zion study are given in Chapter 2 of this report.

We noted the statement that the factor of safety is equal to the resistance capacity divided by the response associated with the DBE. In the probabilistic analysis, dividing median values for capacity and response implicitly assumes that these parameters are independent. Due to the effects of load combinations and failure sequences, this may not always be true.

7.2.3.1 Definition of Failure

Structural failure is defined as ". . .The onset of significant structural damage, not necessarily corresponding to structure collapse." This definition may be conservative in some cases and will tend to produce higher frequency of failure estimates compared to a definition based on collapse where functional failure is not an issue. It would be more appropriate to use a realistic definition for the component being considered and add uncertainty for the definition.

We agree that it is appropriate to define failure as either rupture/collapse or loss of function, whichever occurs first.

7.3.2.2 Fragility Curve Formulation

We agree with separating variability of seismic response and structural capacity into randomness and uncertainty components.

Use of the lognormal distribution is appropriate as long as the extreme tails of the density function do not significantly influence the results of the analysis. It was found in performing the integration of the hazard and fragility curves that most of the contribution (i.e., greater than 90 percent) to the mean frequency of core melt was not significantly influenced by the frequency of failure values for accelerations below 0.3g. This value corresponds to less than three standard deviations from the median value for the structure/equipment components which contribute significantly to the mean frequency of core melt. We feel that the use of the lognormal distribution should be justified for each PRA analysis in which it is used. There may be cases where it is not an appropriate distribution, being either too conservative or possibly unconservative.

It should be noted that the lower tail of the probability density function for frequency of core melt (see Figure 7.2-5) is influenced by the lower tails of the lognormal density functions. Thus, high confidence values for frequency of core melt (e.g., 99 or 95 percent) are influenced by the tails and must be used with caution. However, as explained above, the central results (e.g., mean value) are not significantly affected by the lower tail and the use of the lognormal distribution is reasonable for the Zion study.

The shape of the upper tail of the lognormal density function does not significantly affect the results since the cumulated probability of failure is close to 1.0, and variations in tail shape do not significantly affect the integration process and the final frequency of core melt values.

The results of the fragility analysis are given in Tables 7.2-1 and 7.2-2 (also, key Zion structures and equipment are listed in Table 7.2-3). As noted in Chapter 1, the review concentrated on those structures and equipment which contributed significantly to the frequency of core melt. As discussed in the next subsection (i.e., 7.2.4, Plant Logic), the basis for the fragility for ten structures and equipment were reviewed in detail. Other components in Tables 7.2-1 and 7.2-2 were reviewed generally (i.e., do the fragility parameter values look reasonable, and are they consistent relative to the main contributing items?). For the non-key components, the possibility that they may be much weaker than calculated in the fragility analyses was considered. Specific comments on the fragility parameters for the structures and equipment are given in review of Section 7.9.2, "Conditional Probabilities of Seismic Induced Failure For Structures and

Components For the Zion Nuclear Generating Station". Some general comments on Tables 7.2-1, 7.2-2, and 7.2-3 are included in the discussion below.

7.2.4 Plant Logic

We agree that it is reasonable to assume that offsite power will be lost due to failure of transformer ceramic insulators as a result of any earthquake large enough to contribute to the frequency of failure of the plant. Since the contribution to mean frequency of core melt is significant for ground accelerations greater than 0.3g, it is reasonable to assume that offsite transformer ceramic insulators have failed (note that the median capacity of the ceramic insulators is 0.2g).

Based on data we obtained from LLNL we confirmed the median acceleration capacity values for recoverable interruptions of electrical components, such as relay chatter and breaker trip, and that non-recoverable failure is several times that for recoverable modes. However, we question whether it is reasonable to a priori eliminate components 2, 3, 5, 6, 7, and 13 listed in Table 7.2-3 from further consideration. One viewpoint is that there are many individual components involved in electrical equipment. It is possible that these components are in series such that a failure of only one of them may cause severe consequences. Also as mentioned in Chapter 2, we have some concerns about sequences of events and the inability of a component to absorb energy (i.e., brittle). Also, we question whether relays will also trip at large acceleration levels. It is implicitly assumed in the analysis that this is a recoverable event. This may be a problem for electrical components.

In regard to component 16 (fan cooler ductwork), we cannot judge whether the fan coolers are mechanically capable of adequately mixing the containment gases without the ductwork. If this is true, this is sufficient reason to eliminate component 16 from further consideration. The argument that it is improbable that all the duct risers would fail from the same earthquake may be weak. If these components are identically constructed and attached to the same portion of the building, their capacities and seismic responses may be highly correlated. If so, then the failure of one would imply the failures of others. We did not investigate the details of construction for the fan cooler ductwork.

As stated in Chapter 1, we did not review the fault trees (Figures 7.2-3a through 7.2-3g) for completeness or functional relationships. We did note that the following components were eliminated since it is claimed that their failure is not induced by the range of possible seismic events:

- Power relief valves
- Charging pumps
- Auxiliary feedwater pumps
- RHR pumps
- Pressurizer
- Pressurizer piping (see discussion below)
- Fuel supply to diesel generators
- Service water supply
- Diesel generator (direct failure)
- Switchgear failure
- AC power cables (direct failure)
- Cable trays
- Control building
- Direct failure of piping between Auxiliary building and Containment building (note that piping failure due to soil failure was considered)
- Other failures

We could not find the fragility parameter values for the piping component in Figure 7.2-3d which was eliminated from further consideration. We assume this piping is associated with the pressurizer. In regard to "other failures" which were eliminated from several branches on the fault trees, we question whether the possibility was considered that "other" structures, equipment, or components could fail, fall, and impinge on critical safety-related structures or equipment.

In Figure 7.2-3d, the pressurizer failure was considered to be "non-applicable." Since the collapse of the pressurizer enclosure roof will occur at a much lower acceleration, we question whether the pressurizer capacity should be replaced by the capacity of this structure. In either case, the effect on the final results is judged to be small.

In the case of piping, all pipe segments are connected in series; thus, the frequency of failures for a piping system may not be conservatively represented by the frequency of the weakest component, unless the capacities and responses of all segments are individually (i.e., capacity with capacity and response with response) perfectly correlated. Because piping extends a relatively long distance and is supported at many places in a structure, piping response will not be perfectly correlated. Also, because different components may come from different manufacturers or material runs, capacity also is not perfectly correlated. The Licensee's response given in Appendix A (Question 5) suggests that this effect was considered in the analysis and the selection of fragility parameter values for piping systems; however, it does not appear that the capacity of a single piping run was reduced for the effects of a series system. A similar problem also exists for electric cables supported by cable trays.

The fragility parameters for the rest of the components listed above were reviewed and we agree that they do not significantly contribute to the frequency of core melt.

The final Boolean expression for core melt failure given on page 7.2-7 interacts the following ten structures or equipment items:

(4) Service water pumps
(8) Auxiliary building--failure of concrete shear wall
(9) Refueling water storage tank
(10) Interconnecting piping/soil failure beneath reactor building
(12) Condensate storage tank
(14) Crib house collapse of pump enclosure roof
(17) 125 VDC batteries and racks
(21) Service water system buried pipe 48"
(22) CST piping 20"
(26) Collapse of pressurizer enclosure roof

Our review concentrated primarily on these items followed by items which were eliminated from the fault trees due to their high seismic capacity.

Figures 5 and 6 show the relative contribution of the above-listed ten items to the total frequency of core melt. The failure fraction versus damage effective ground acceleration is shown for linear and logarithmic scales in these two figures. As shown, the curves become progressively higher (i.e., larger failure fraction) as more and more items are included in the Boolean equation for core melt failure. Note that in developing these curves the randomness and uncertainty was combined.

Several observations which give perspective to the importance of each structure or equipment item can be seen from these curves. The most important contributor to core melt is the service water pumps, which are the weakest components. The next most important item is the Auxiliary Building shear wall. As explained previously, the integration

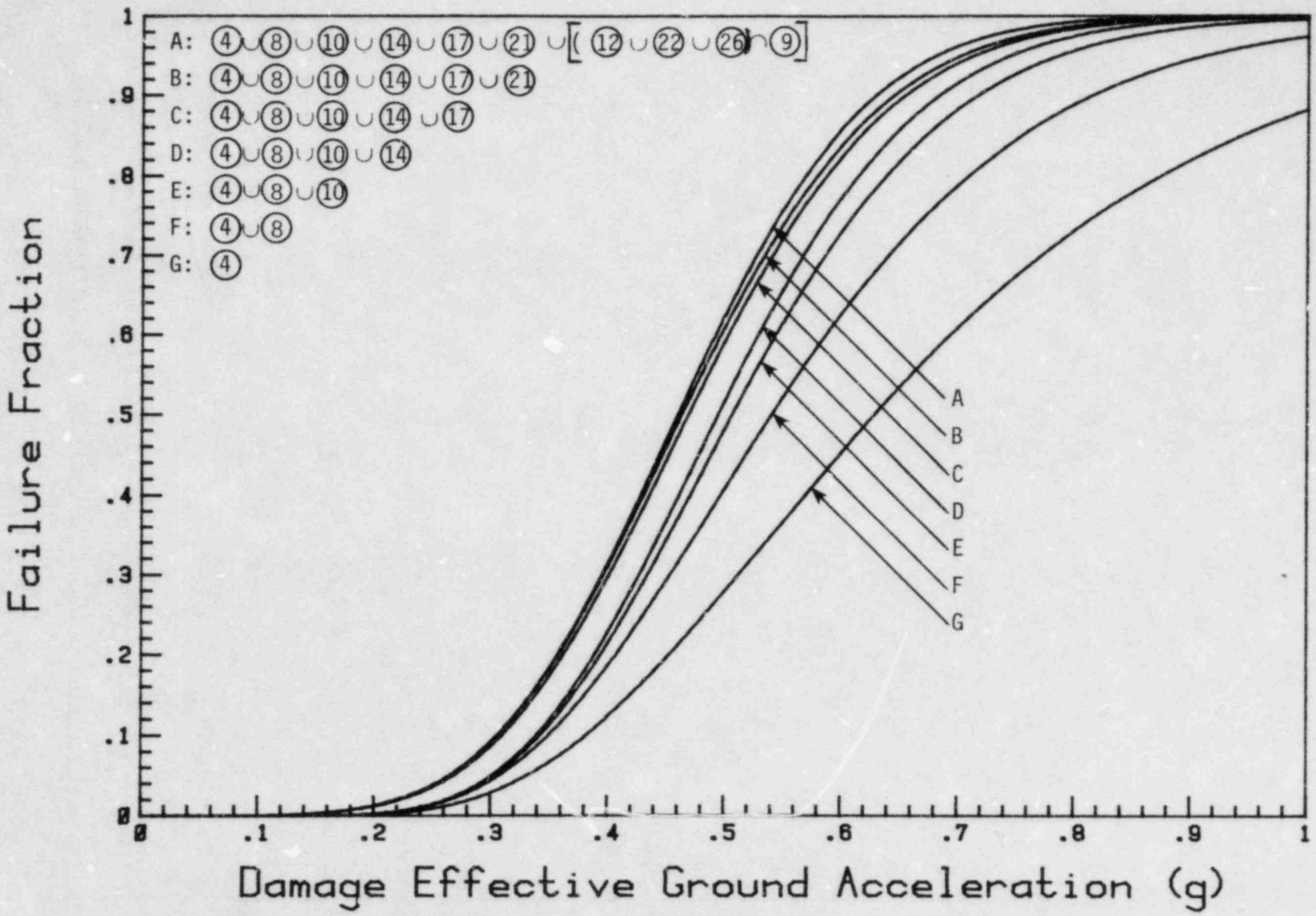


Figure 5 Partial Fragility Curves - Linear Scale

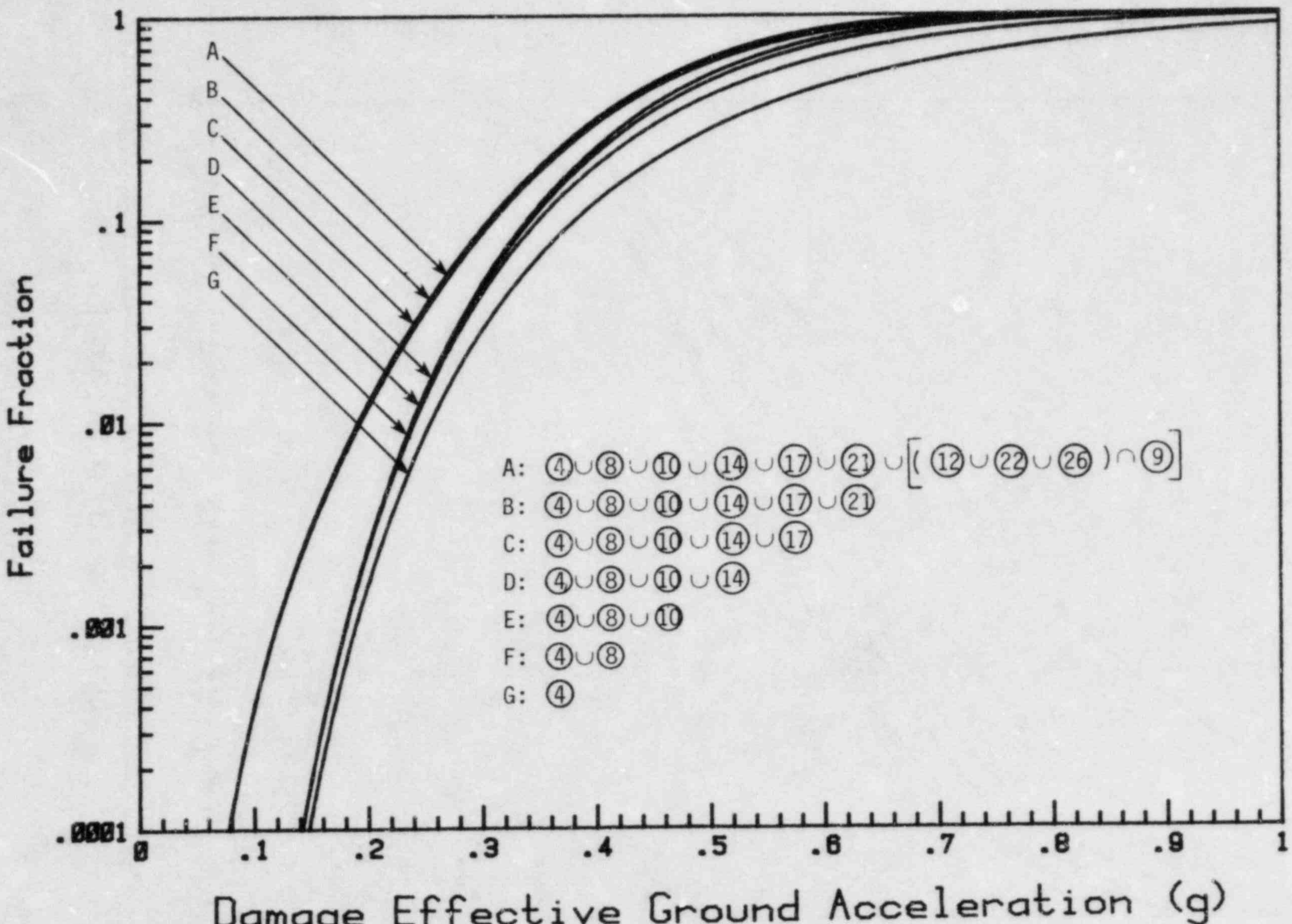


Figure 6 Partial Fragility Curves - Logarithmic Scale

of the hazard and fragility curves depends primarily on acceleration values above 0.3g. For ground accelerations above 0.3g, neglecting the contribution (i.e., based on the fragility parameter given in the Zion PRA report) from all other structures or equipment, would result in underestimating the failure fraction by a factor of only three (a moderate effect). In terms of the probability distribution of the frequency of core melt (i.e., Figure 7.2-5), the mean value would also follow this same relationship. However, the lower tail of the density function would be more severely affected by the elimination of other items. Thus, in order for a major change to occur in the mean frequency of core melt due to fragility effects, the strengths of a number of the structure or equipment items would have to drop down to or below the strength of the service water pumps (or the strength of the service water pumps would have to decrease).

A jump in the probability of failure occurs when item 17, which is the 125 VDC batteries and racks, is included in the Boolean expression (see Figure 6). This is due to the relatively high uncertainty value (i.e., $\beta_u = 0.63$) for this equipment.

Because items 9, 12, 22, and 26 are embedded in an "and" sub-expression they do not contribute significantly to core melt frequency.

There is one potentially important type of dependancy that is not discussed in the Zion PRA report. This involves the correlation of the response of two or more structures or equipment due to the motion of a common supporting building. In simple terms, if two components are located in the same building, close to each other, the response input to one could be nearly the same as for the other. Thus, if one exhibited a high response, the other would likely also have a high response. Because of the correlation of input (hence response), the failure frequencies for the two components would be correlated. The potential for this type of interaction is present for the following two combinations of components:

1. Crib House

   (4) Service water pumps

   (14) Crib house roof

2. Auxiliary Building

   (8) Auxiliary building shear wall

   (17) 125 VDC batteries and racks

Potentially, the most important contributions due to dependencies would come first from the correlation between (4) and (14), followed by the correlation between (8) and (17).

In the context of the probability of frequency format used in the Zion PRA report, correlation of response affects the results in two different ways. Based on some simple examples it was found that correlation through the uncertainty factor, β_u causes greater uncertainty and hence "spreads out" the fragility curves (see Figure 7.2-4). On the other hand, correlation through the randomness factor, β_r, causes the fragility curve for combined components to decrease for components in series and increase for components in parallel. The extreme randomness case is perfect correlation where the frequency of failure would be based on the weakest component in a series configuration or the strongest component in a parallel configuration. For the Boolean expression for core melt failure, the significant components are in series; thus the effect of building response correlation is to increase the probability spread for uncertainty and decrease the frequency of failure for randomness. In examining the β_r and β_u terms, roughly half of the variability comes from the building response contribution (as compared to the contribution from the particular equipment or structure item). It is our judgment that if the correlation due to building response were included in probabilistic analysis of the Boolean expression for core melt failure, the effects on the mean frequency of core melt would be small. If the Boolean expression had been dominated by "and" symbols this effect could have been more significant. In general, the effect of building response on the correlation of frequency failure between components should be considered.

7.2.5 Seismic Core Melt Frequencies

As discussed in our review of Section 7.2.2 we believe that the value of 5.6×10^{-6} per year for the mean frequency of core melt is reasonable. We have less belief that the lower-bound value for the 90 percent confidence interval given in the Zion PRA report (i.e., 3.0×10^{-8} per year) is correct. As discussed above, many factors influence this value (e.g., tails on probability and frequency distributions and dependencies). We judge that there could be a major difference in this value. Also, it is not clear which 90 percent confidence interval is being cited. Is it the one where there is 5 percent probability remaining in each tail?

We question whether the five curves shown in Figure 7.2-4 and tabulated in Table 7.2-4 are median values, mean values, or other. It is not clear how they are located in each 20 percent probability slice.

7.2.6 Initial Assembly Leading to Containment Tree Entry States

Only the plant state SE contributes to the plant matrix M^S. The Boolean expression for this state is similar to the expression for core melt frequency which was reviewed in the previous section. The same relationship exists between the components which contribute significantly to core melt and to the plant state. Thus, no additional comments are made for this section.

7.2.7 Seismic Effect on Containment

Comments on fragility values for impact between the reactor building and auxiliary building and for other Containment building failure modes are given in our review of Section 7.9.2.

7.2.8 Conclusions

No new comments are made for this section.

7.4 EXTERNAL EVENTS - FLOODING

7.4.1 External Flood Sources

It was concluded in the Zion PRA report that the contribution to the frequency of core melt due to external flood sources is insignificant. This conclusion was based on a deterministic analysis of assessing flood levels. The basis for the conclusion, including a description of the methodology and pertinent data, was not provided in the report. We feel that the approach presented is not appropriate nor adequate for purposes of a probabilistic risk assessment. We discuss below pertinent issues that should be addressed. Note, at this time comments cannot be made concerning the conclusion presented, only the inappropriateness of the basis on which it was made.

In order to assess the possible contribution of external flooding to the core melt frequency, a probabilistic analysis of the ocurrence of flooding should be performed. A site-specific flood analysis should consider the following causes:

- River flooding
- Upstream dam failure (includes all secondary causes such as earthquakes, overtopping, antecendent dam failures)
- Failure of dikes and levees
- Tsunamis
- Surges
- Seiches
- Wind waves

- Precipitation (including hurricanes and sequences of storms)
- Snow melt

A screening of these causes may indicate that certain of them may not exist at a site. For example, in the absence of an upstream dam, flood due to a dam failure can be eliminated as a possible cause and excluded from the probabilistic analysis. We recommend that all sources of flooding including, but not necessarily limited to, the above list be explicitly identified and reviewed as to their potential occurrence at the Zion site. The Zion PRA report has not discussed the events that might cause flooding.

The possible occurrence of flooding at the Zion site may result due to the occurrence of a combination of events such as precipitation, wind wave action, antecendent conditions, and stream flooding. A probabilistic model should allow for such combinations, particularly in evaluating the frequency of extreme events. We suggest that potential combinations of events be explicitly identified and reviewed. Those events considered likely to occur, should be incorporated in a probabilistic model.

We feel that the deterministic methodology that has been employed is inappropriate because the uncertainties in the flood assessment have not been considered. The large uncertainties in evaluating flood frequencies and structure and component fragilities may be important at the Zion site for determining the frequency of core melt due to external flooding.

In addition to evaluating the likelihood of occurrence of events and event combinations, consideration should also be given to possible dependencies that may exist. Examples include time dependence of meteorological events, and spatial dependencies of flooding due to different sources (Ref. 6).

Based on the conclusion that external flood is not a significant risk contributor, no consideration was given to the likelihood of leakage into the plant and the development of structure and equipment fragilities. In the event that a probabilistic analysis warrants consideration of flooding at the site, fragility curves and systems analysis should be developed for the affected structures and safety-related equipment.

7.4.2 Internal Flood Sources

This section was reviewed with respect to the adequacy of the analysis as presented in the Zion PRA report. During the Indian Point study review, a summary was provided of the procedure used to identify sources of internal flooding and to determine their effect. Three steps were followed:

1. Identify sources of flooding.

2. Identify locations vulnerable to floods from those sources determined in 1.

3. Simulate initiating events and evaluate the impact.

We generally agree that the above steps are required to conduct an internal flood analysis. We suggest that the internal flood analysis should be conducted in a more systematic manner, possibly including the development of flood analysis fault trees. This would ensure that a thorough, systematic analysis of critical events and event sequences are considered. We suspect that existing fault trees have been used to some degree in the analysis.

7.4.2.1 Auxiliary Building

For the first potential internal flood source, tanks in the Auxiliary building are identified. The statement is made that these tanks are not pressure vessels, and therefore catastrophic failure is an extremely unlikely event. The basis for this statement is not known. If rupture were to occur, releasing the contents of the tanks may cause adverse consequences.

For a rupture in the RHR system piping during cooldown and heatup conditions, a LOCA will occur. However, this event is ignored as a core melt initiator because of the leak detection and isolation provisions which exist at the plant. These provisions are apparently described in the Zion FSAR (Chapter 6) which was not available for review. No basis is provided in the Zion PRA report to support an assumption of 100% reliability of the detection and isolation system, thus allowing this event to be ignored.

For a large leak or rupture in the component cooling water (CCW) piping, a failure rate of 1.8×10^{-4} per year is assumed. The basis of the 10^{-3} to 10^{-4} probability of leak going unnoticed and of the 0.01 chance of a failure resulting in a rupture is not given. An assumption is also made here that alarms and instrumentation are reliable. The basis for this assumption is not supported.

We express similar concerns about the basis for the frequency of service water pipe failure, namely the basis for determining the rate of failure of 1.5×10^{-8} per year.

The flood drains and stair wells are assumed to have a capacity to discharge flood waters to elevation 542'. The basis for assuming that these channels of flow are perfectly reliable and available is not given.

The uncertainty in the frequency estimates given in this subsection should be documented.

7.4.2.2 Containment Building

It is not clear whether all possible failure mechanisms have been evaluated. What does the frequency of failure of the service water pipes correspond to? For example, is it the mean probability of failure?

7.4.2.3 Turbine Building

Our comments for subsection 7.4.2.1 regarding the reliability of floor drains and stairwells and capacity to discharge flood waters also apply here.

7.9.2 CONDITIONAL PROBABILITIES OF SEISMIC INDUCED FAILURES FOR STRUCTURES AND COMPONENTS FOR THE ZION NUCLEAR GENERATING STATION

1. Introduction

No comments.

2. General Criteria for Development of Median Seismic Safety Factors

2.1 Definition of Failure

We accept the definition of failure used in this study.

2.2 Basis for Safety Factors Derived in Study

It is important to note the comment given at the bottom of page 2-3: "There was a general lack of detailed information available for this study on seismic fragility of structures and equipment." This comment was kept in mind during the review of uncertainty values assumed for the various parameters.

2.2.1 Structural Response and Capacity

The dynamic analyses of the Reactor building, the Turbine/Auxiliary building, and Crib house were not reviewed. We did not check the design calculations. We assumed that the mechanics of performing the structural analysis was done correctly. It is stated in the second paragraph of this subsection that Structural Mechanics Associates (SMA) also did not review the design calculations. As discussed in Chapter 2, we do not know whether the range of uncertainty included consideration for possible design and construction error.

2.2.2 Seismic Class I, Piping and Equipment Response and Capacity

We agree with the material in this subsection.

2.3 Formulation Used for Fragility Curves

We believe that the mathematical presentation in this section tends to confuse the casual reader. Because of the inherent simplicity of the method we offer the following explanation of how it works.

It is assumed in the analysis that the capacity of a structure or equipment, in terms of ground acceleration, is lognormally distributed. Thus, the frequency of failure is a function of three parameters: (1) the median capacity value, $\check{A}$; (2) the logarithmic standard deviation for capacity, β_r, and (3) the ground motion input acceleration value. Note that any randomness in the ground motion value or building or equipment response is included in the β_r value. Figure 7(a) shows the capacity density function which is determined by $\check{A}$ and β_r. If the ground motion value is A_g, then failure occurs for all values of A less than A_g. Thus, the frequency of failure is just the area under the density function between A equal to 0 and A_g. We could stop at this point and just use this procedure to obtain various values of frequency of failure (for different A_g values) and plot the fragility curve as shown in Figure 7(a).

The problem is that $\check{A}$ is not known with certainty. (It is assumed that the logarithmic model and β_r value are known in a relatively certain sense). Thus, a second lognormal distribution for $\check{A}$ is used to quantify the uncertainty for this parameter. It is determined by two parameters: the median value, $\check{\check{A}}$, and the logarithmic standard deviation for uncertainty in the median value, β_u. The probability density function for $\check{A}$ is shown in Figure 7(b).

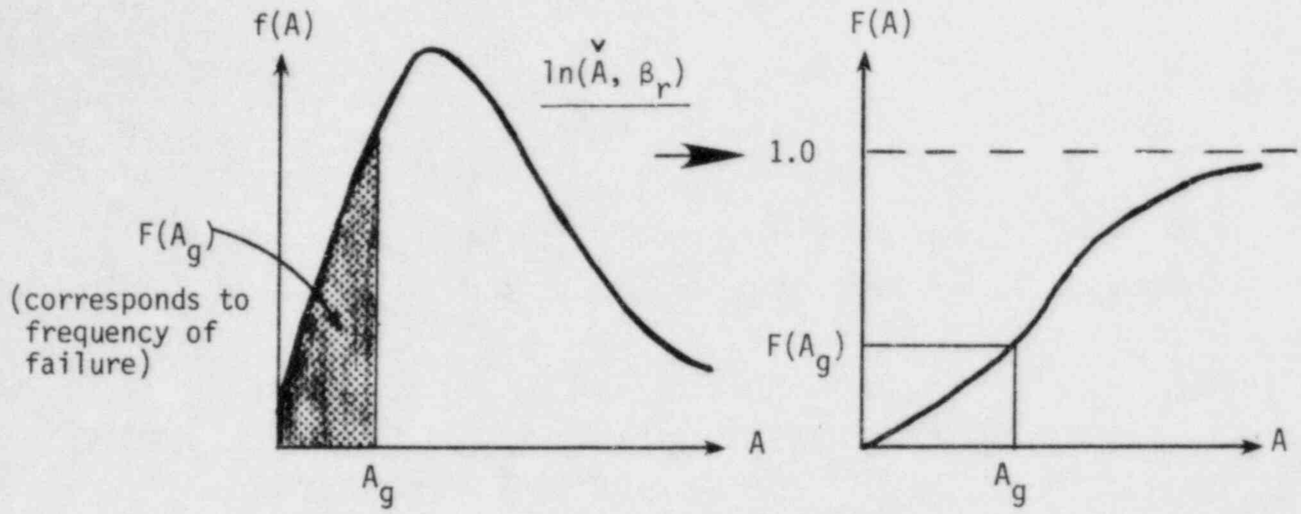
Now depending on what value of $\check{A}$ is picked from the distribution on $\check{A}$ (see Figure 7(b)), a corresponding fragility curve can be calculated (see Figure 7(a)). For example, if the 95% probability fragility curve was desired, then $\check{A}$ would be selected such that there is a 0.95 probability that a larger median value would occur. If $\check{\check{A}}$ is 0.77g and β_u = 0.39 then for the 0.95 probability level $\check{A}$ = .4g. This value comes from the following equation, which is the mathematical representation of the solution shown in Figure 7(b):

$$\check{A} = \check{\check{A}} \exp\left[\beta_u \Phi^{-1}(1 - p)\right]$$

where $\Phi(\cdot)$ = Standard cumulative normal distribution and Φ^{-1} is the inverse function

p = Probability value (e.g., 0.95)

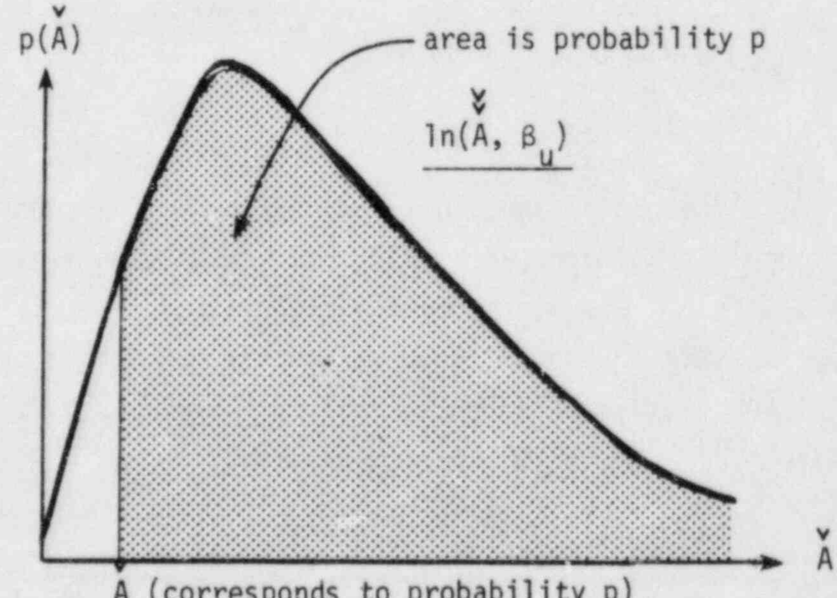
Now, if the fragility frequency of failure value, assuming β_r is 0.36 is desired corresponding to a ground acceleration A_g equal to 0.4g, the answer can be found from the lognormal distribution with median value equal to 0.4g (see Figure 7(a) and β_r equal to 0.36. The answer



Density Function

Fragility Curve

(a) Capacity Frequency Density Function and Fragility Curve



(b) Probability Density Function for $\check{A}$

Figure 7 Probability and Frequency Functions for Fragility Analysis is 0.50 and is found from the following equation:

$$F(A_g) = \Phi\left[\frac{\ell n\ A/\check{A}}{\beta_r}\right]$$

Thus, there is only a 5 percent probability (which corresponds to the 95 percent confidence level) that the frequency of failure exceeds 0.5 for a ground acceleration of 0.4g.

3. Differences Between Current Methods and Criteria Used for Zion for Seismic Qualification of Structures and Equipment

3.1 Earthquake Level Specified for Design

No comment.

3.2 Free Field Structure Response Spectrum Anchored to Peak Ground Acceleration

It is not clear that a_s (sustained acceleration or damage effective) can be applied to anchor the ground response spectral shapes from Regulatory Guide (RG) 1.60 which is based on peak response acceleration. It would be more appropriate to redo the statistical analysis using recomputed spectral shapes for the earthquake time histories normalized to the a_s response level (as opposed to peak ground acceleration as done in the original study for RG 1.60)

In the text, two methods for defining design spectra are recognized: specifying site-dependent spectra, or using broad-banded spectra such as in Regulatory Guide 1.60. The Zion PRA risk analysis used a broad-banded spectra. By this selection a source of modeling error is created in the analysis. Comparisons of the Regulatory Guide 1.60 average spectra and the response spectra from the SSMRP study suggests an uncertainty component corresponding to a factor of approximately 1.5 should be used.

In the Zion PRA report, there is no uncertainty component for variability in the response spectra at all, only randomness. If this were true, then there would be no motivation to ever conduct site studies to develop site-specific spectra. Remember that randomness is irreducible and the Zion PRA report broad-banded ground response spectra have no uncertainty. We believe that the Licensee's response in Appendix A (Question 9) does not adequately address our concern. We believe that a larger uncertainty for the response spectra should be included; however, we doubt that the additional uncertainty will have a large effect on the mean frequency of core melt.

3.3 Damping

The damping values given in Table 3-1 are reasonable values for structures and equipment items when the applied stress is near yield. These values are the same as values recommended by Newmark and Hall (Ref. 7). A study of the sensitivity of response of the Zion Auxiliary building for different effects was conducted for the SSMRP program (Ref. 8). As part of this study, the effect of damping on structure response was investigated. It was found that structure response for a particular earthquake time history (or set of time histories) is weakly affected by damping in the range of 3 to 10 percent. Variation of the median response value was less than 25 percent in this range.

For in-structure response spectra (which affects equipment response) the damping of the structure had a minor effect except near the fundamental frequency of the structure where the difference was approximately a factor of 2 between the response for 3 and 10 percent damping. This last result indicates that the fragility curves for equipment or substructures with natural frequencies near the fundamental frequency of a supporting structure should reflect the expected structural damping.

Reference 8 (a report for the SSMRP) also presents a table of free-field acceleration values corresponding to various levels of response (one-half yield to yield) for the Zion Reactor building, Auxiliary building, and Crib house. Response levels corresponding to yield and one-half yield from Reference 8 are listed in Table 1 along with the median capacity values for the critical structure and equipment items contained in each building from the Zion PRA report. Table 1 indicates that except possibly for the Reactor building, it is reasonable that the yield level value of 10 percent damping be assumed for the structure damping in developing the fragility values for enclosed structure/equipment items. In regard to the pressurizer roof located in the Reactor building, even if 3 percent damping was used in the Reactor building analysis, the effect on the mean core melt frequency due to the contribution from the pressurizer would be small.

Table 1 Free-Field Ground Acceleration Values for Buildings and Supported Structure/Equipment Items

Structure/Equipment	Response Level: 1/2 Yield (3% Damping)	Response Level: Yield (10% Damping)	Median Capacity
Reactor building internals	2.7g	5.5g	
Pressurizer enclosure roof			1.8g
Auxiliary building	0.35	0.70	
Shear wall			0.73
Refueling storage tank			0.73
Safety injector pump			0.90
Batteries and racks			1.01
Crib house	0.35	0.70	
Service water pumps			0.63
Crib house roof			0.86

3.4 Location at Which Free Ground Response Spectra are Specified

No comment.

3.5 Soil-Structure Interaction

No comment.

3.6 Combination of Responses for Earthquake Direction Components

We agree that the alternate method, consisting of combining 40 percent of the response in two orthogonal directions of motion with 100 percent of the response in the principal direction, is appropriate to use as a median centered method.

3.7 Specification of Seismic Input For Piping and Equipment

It is not clear from the description what differences were found between the algebraic summation procedure and the SRSS procedure for combining modal responses. We assume that these differences, if any, were incorporated into the development of the piping fragility parameters.

3.8 Load Combinations

The possibility of a severe event which causes a LOCA, followed by an aftershock should be considered. Pressurization of the Containment building may fail the prestress tendons which would weaken the capacity of the building. If this situation occurs, an aftershock could cause additional damage and possibly failure. We reviewed the Licensee's response given in Appendix A (Question 12). We still believe that this issue chould be addressed quantitatively, although we doubt that this type of occurrence will contribute significantly to the frequency of core melt.

3.9 Stress Criteria for Seismic Design of Critical Structures and Containment

Since in the Zion Reactor buildings analysis the reinforcing steel was held to the yield value rather than allowing the ductile capacity to be developed, the Zion design criteria appear to be generally more conservative than the USNRC Standard Review Plan criteria.

3.10 Allowable Stress Criteria for Seismic Design of Piping and Mechanical Equipment

Since above-ground piping were found not to be critical components, this section was not reviewed in detail. Thus, no specific comments are made for this subsection. However, comments concerning piping as a series system are made for subsection 5.2.3.1.

3.11 Seismic Class I Electrical and Instrumentation

No specific comments are made for this section. Comments concerning develoment of fragility values for electrical components and instrumentation are made later in this chapter.

4. Structures

4.1 Safety Factors, Logarithmic Standard Deviations, and Coefficients of Variation

No comment on the introductory paragraphs.

4.1.1. Structure Capacity

No comment on the introductory paragraph.

4.1.1.1. Concrete Compressive Strength

Based on the measured strength values given in Table 4-1 of the Zion PRA report, we concur with the median strength factors for concrete test cylinders. The logarithmic standard deviations are supported by data summarized for the SSMRP given in Appendix A of Reference 9.

It is implied in this section that the strength of test cylinders is similar to the strength of in-place concrete. However, it is stated in this subsection that some increase in strength is present in in-place concrete. We believe that the variability between test cylinders and in-place concrete is larger than the variability factors for concrete cylinders. Thickness of concrete members and the availability of moisture contribute to actual concrete strength in members. Our estimate is that a logarithmic standard deviation of at least 0.2 would be appropriate.

What is more relevant to the question of capacity is the properties of in-place reinforced concrete strength which includes factors such as construction joints, boundary condition, shrinkage and creep properties, etc. which can be more important than the $\sqrt{f'_c}$ value for concrete material.

4.1.1.2. Reinforcing Steel Yield Strength

values used were compared with similar values given in Appendix A of Reference 9 and were found to be in agreement.

We feel that it is inappropriate to lump No. 3 through No. 11 bars in the same category. No. 3 bars are stronger per unit area than No. 11 bars. However, larger bars comprise the reinforcement generally found in reinforced concrete members in nuclear power plants. This may create a slightly unconservative bias. However, we judge that the effect of this bias is small.

4.1.1.3. Shear Strength of Concrete Walls

The basis for equation 4-3 given in this section of the Zion PRA report was reviewed and we agree that this equation is an acceptable prediction of the ultimate strength of shear walls bounded on four sides by concrete members. We feel that the contribution of reinforcement steel given by equation 4-5 is questionable. This equation implies that for an aspect ratio (height/width) of 1.0 the vertical steel has no effect on the strength. We find this hard to believe. Based on a recent study, we would shift the values of coefficents used in equation 4-5 so that the horizontal and vertical steel contribute equally at an aspect ratio of 1.0 (Ref. 22).

We feel that it is more appropriate to base the strength of reinforced concrete shear walls on test data reflecting boundary conditions which most closely represent the in-place configuration. The strength of shear walls depends to a large degree on the boundary conditions which are present. In the case of the Auxiliary building shear walls, the concrete sections are enclosed by steel members which are connected by shear studs to the concrete. Concrete shrinkage may cause small cracks between the steel and enclosed concrete. Items which contribute to strength include the strength of shear studs, anchorage of wall reinforcing bars in the floor slabs, vertical load, and confining strength of the steel frame. A large capacity due to shear friction may be developed if the proper boundary conditions are present. We suspect that the median capacity may be larger than indicated by equation 4-3; however, we did not review the boundary conditions in detail or perform any structural analysis. We also believe that the variability in strength is larger than a logarithmic standard deviation value of 0.15 which was used. A value of 0.20 was obtained based on an analysis of 54 wall sections from 5 different sources (Ref. 22).

The strength factor for shear walls is a major contributor to fragility of this type of component. We assume as a minimum that the anchorage of the reinforcement bars was reviewed by SMA, since proper embedment is required to develop the type of strength predicted by the equations cited in this subsection.

4.1.1.4 Strength of Shear Walls in Flexure Under In-Plane Forces

We did not review this sub-section in detail. We have no comments.

4.1.1.5 Strength of Shear Studs

The strength of shear studs depends on many factors. Most data for which we are aware for stud tests has come from testing composite beams with the concrete slab in compression, which confines the studs, and the steel beam in tension. Studs can perform in a ductile or brittle manner depending on the details of confinement, concrete shrinkage cracks, concrete stress state (i.e., compression or tension), and location of reinforcing bars close to the studs. Based on these considerations, we feel that the variability values cited for shear studs is low.

4.1.2. Structure Ductility

In regards to the ductility value of 4 which was calculated in the Zion PRA report from Reference 10, it is not clear whether this value was used in the analysis of all three Zion buildings. We noted that in a SSMRP report (Ref. 11) a system ductility ratio of 2 was estimated for the Auxiliary building.

Figure 4-3 in the Zion PRA report shows the relationship between the ductility value and the deamplification factor used to increase the median capacity of shear walls for inelastic energy absorption. It should be noted that the results shown in this figure are based on single-degree-of-freedom (SDOF) elasto-plastic systems. At a workshop held in December 1981, sponsored by the USNRC, SMA presented the results of a research project directed to the development of a basis for selecting design response spectra based on free-field motion. The results of the analytical studies support the deamplification curves given in Figure 4-3. It was found for one example comparison that the difference between Figure 4-3 (Zion PRA Report) and the methodology developed by SMA when applied to a broad-banded spectrum was less than 10 percent. The study done for the NRC is based on a different approach than taken by Riddell and Newmark (Ref. 12) which is the basis for Figure 4-3 and thus is a good check.

Both the SMA and the Riddell and Newmark studies were based on SDOF models. As noted in Reference 11, considerable uncertainty exists in the application of these techniques to multi-degree-of-freedom (MDOF) systems. No accepted methods currently exist for applying the deamplification factor for SDOF models to MDOF systems. This problem is particularly complex when localized ductilities contribute significantly to the overall strength of a building.

In addition to the variability in the ductility model (it appears from the calculations that a value of 0.10 was used), an uncertainty measure should also be included for the inaccuracy of using a SDOF model to predict behavior of a MDOF System. A nonlinear MDOF analysis of the Auxiliary building was conducted for the SSMRP (Refs. 9 and 13). As part of this study, five input time histories were applied to the model until a ductility value of 4 was reached in the weakest element. The ratio of the peak ground acceleration value at failure (defined at a ductility value of 4) to the corresponding value at yield was found to range between 1.33 and 1.60 with a median value of 1.43. In comparison, the method used in the Zion PRA report to account for inelastic behavior (Figure 4-3) gives a deamplification factor of 0.43 for 10 percent damping. The inverse of this value is 2.35, which is much larger than the more rational median value of 1.43.

This comparison can be interpreted two ways. As a minimum, it demonstrates the magnitude of the uncertainty which exists between the SDOF model and the MDOF response. Another interpretation is that since the analysis described in References 9 and 13 is more rational, the median inelastic energy absorption factor based on the MDOF analysis should be used; although, as noted in the Licensee's response given in Appendix A (Question 4), the model used in the SSMRP analyses did not include the Turbine building which is attached to the Auxiliary building. Subsection 4.3.1 describes the analysis for the Auxiliary building shear wall failure. Several issues were raised in regards to this subsection, and it is not clear whether the result of the above example comparison should be applied. One interpretation is that the inelastic energy absorption factor should be reduced by the ratio 1.43/2.35 (or 0.6). This level of change may have a moderate effect on the mean core melt frequency value.

We reviewed the Licensee's response given in Appendix A (Question 3). We still feel that the uncertainty value of 0.10 assigned to the use of the equation for the ductility factor (based on the review of the calculations performed for the Auxiliary building shear wall) is too low. Also, we are not convinced that the SDOF model always results in a conservative estimate. The example given above indicates one case where the SDOF model produces unconservative results.

4.1.2.1 Structure Response

We noted that the numbering system for this subsection is misleading. In actuality, structure response is a general category which includes contributions from the following four primary categories:

- Modal response
- Combination of modes
- Combination of earthquake components
- Soil-structure interaction effects

These are discussed in the subsections which follow.

4.1.2.2 Modal Response

This category includes the effects of:

- Input ground spectra
- Damping
- Frequency
- Mode shape

We generally agree with the approach used in this subsection except for the following areas.

As discussed above (see comments for subsection 3.2), a larger uncertainty value should be included for the response spectrum input to reflect the potential error between site-specific spectra and the broad-banded site-independent spectra which were used in the analysis.

We have also compared the randomness in the input ground spectra from the Zion PRA study to the results of the SSMRP study (Ref. 14). We note that β values varied from 0.25 to 0.33 in the SSMRP study, accounting for a β of 0.12 due to the combination of earthquake components. This compares to a value for β_r of 0.18, a value typically used in the Zion PRA study, which appears to be slightly low.

There is in general a coupling (dependency) between damping and frequency effects. The logarithmic standard deviation values would be different if a combined value were calculated rather than computing the contributions from frequency and damping separately. We judge that this consideration would have a small effect on the Zion PRA results.

In regards to the frequencies of the structures which were obtained from the original design and used in the analysis, it is not known how these values compare to the results of the SSMRP for the Reactor building and the Auxiliary building. If there are differences, the implications of these differences for the fragility curves should be incorporated.

The logarithmic standard deviation for frequency was estimated to be 0.3. The results of a study conducted for the SSMRP, where four mathematical models were developed for the same structure, support using this value (Ref. 9).

For the effect of mode shape, logarithmic standard deviation values of 0.15 and 0.10 were used for the Reactor building and Auxiliary building/Crib house, respectively. We agree that these are reasonable values as long as the model has sufficient detail to predict the response of interest. For example, if a flexible floor slab is lumped at a column line in a finite element model, the uncertainty in predicting vertical response at the center of the floor is much larger as compared to results obtained from a model where the floor slab details are included. Based on the Licensee's response given in Appendix A (Question 14), they believe that the models have sufficient detail to produce accurate response in the elements of interest.

4.1.2.3 Modal Combination

The values used for this consideration appear to be reasonable based on the data provided in Figure 4-4.

4.1.2.4 Combination of Earthquake Components

The 100%, 40%, 40% method is discussed in Reference 15 where it is stated that it is more conservative than the SRSS method. However, we feel that either of the two methods can be used to predict median response.

The logarithmic standard deviation value for this consideration is assumed to be 0.17. The basis for this value is not obvious. (It appears that the -30% to +40% change is assumed to be ± 2 standard deviation range).

4.1.2.5 Soil-Structure Interaction Effects

An indication of deconvolution is given in an SSMRP report (Ref. 1) where cumulative distributions of free-field and Reactor building foundation peak accelerations are shown. The variability in the free-field peak accelerations is due to the range of the peak acceleration values for the time histories used in the analysis. However, the ratio of the median values (i.e., Reactor building foundation/free-field) is a good measure of deconvolution. Values of 0.66, 0.61, and 0.58 are found for the X direction, Y direction, and Z direction, respectively. This is much smaller than the value of 0.9 which was used in the Zion PRA report. Thus, the Zion analysis appears to have a conservative bias for the effects of deconvolution. Comments concerning bias and variability in the soil-structure interaction (SSI) component of the analysis are given in Section 7.2.

It is not clear what the variability should be for SSI effects. Some limited studies were conducted for the SSMRP where only the soil modulus and damping were varied (Ref. 15). The variability in these parameters did not significantly increase the variability of structure response. However, the effect of different models (e.g., soil spring versus finite element) was not considered. This is an area where knowledge is lacking. Our intuition is that if you engaged several competent independent engineers to perform SSI analyses of the Auxiliary building, the size of the uncertainty logarithmic standard deviation parameter would be larger than 0.15 (the value used in the Zion PRA report). On the other hand, it could be argued that part of the frequency uncertainty value of 0.30 is due to SSI effects. This is reasonable since the four SSMRP models for the Auxiliary building were fixed-base models and produced logarithmic standard deviation values in the range of approximately 0.1 to 0.3 (Ref. 9).

However, from another viewpoint, the uncertainty due to SSI may not get much larger than 0.15. We noted in our comments for subsection 7.2 that variability in response in basements of buildings, for effects of buildings themselves, corresponds to a β value of 0.17 (Refs. 3 and 4). This contribution is due to SSI.

4.1.2.6 Response Factor Estimate

It appears that the contribution of modal response to the combined parameter values for the Reactor building was left out. A summary of values and the percent of difference is given in Table 2.

The effect of the difference given in Table 2 on the mean core melt frequency value is judged to be small.

Individual failure modes for the Reactor building, Auxiliary building, and Crib house are described in general terms in two SSMRP reports (Refs. 16 and 17), which were prepared by the engineers who also participated in preparation of the Zion PRA fragility data. These two reports give descriptions of the modes of failure which could occur. We noted that fragility data was developed in the Zion PRA report for many of these failure modes. We believe that the significant failure modes for these building have been considered.

Table 2 Response Parameters for Reactor Building

Response Factor	F_r	β_c	Location in Zion PRA Report
Modal response	1.1	0.27	p. 4-15
Mode combination	1.0	0.17	p. 4-15
Earthquake components	1.0	0.12	p. 4-38, 39
Soil-structure interaction	1.3	0.25	p. 4-19
Combined values	1.43	0.42	
Values p. 4-19 (Zion PRA report)	1.3	0.34	
% Difference	10%	24%	

The basis for dividing the variability into randomness and uncertainty components is not provided in the Zion PRA report. This information should be given in the report.

In addition, no uncertainty was assumed for spectral shape. See comments concerning this issue in the comments for subsection 7.2. No randomness was assumed for modeling which includes both frequency and mode shape considerations Based on random variations due to mass and stiffness a value of β_r equal to 0.1 would be appropriate (Ref. 8).

4.2 Reactor Building

Tables 4-3 through 4-10 give the fragility parameter values for the significant modes of failure for the Reactor building. The factors in these tables were not checked in detail. They were reviewed for reasonableness and compared in some cases with results obtained from the SSMRP studies. Specific comments for each of the failure modes is given below.

4.2.1 Soil Failure

This failure mode is important to the piping fragility evaluation discussed later. The fragility data for soil failure beneath the Reactor building is based primarily on an analysis performed by Sargent and Lundy (Reference 66 to this section of the Zion PRA report--our Reference 2). A curve developed from this report and other data are given in Figure 4-7 which give median and $\pm 1\beta$ Reactor building displacement versus peak ground acceleration curves. Since there is no inelastic energy absorption consideration for this failure mode, we question how the peak ground acceleration value used in this study relates to the damage effective peak acceleration value (i.e., a_s) adopted for the Zion analysis. We noted that displacement values for the three curves in Figure 4-7 appear to be unbounded; however, based on the Licensee's response given in Appendix A (Question 8), we agree that overturning due to an earthquake motion is bounded.

Reference 2 is cited as the basis for determining the Reactor building capacity for soil failure, structure-to-structure impact (subsection 4.2.2) and interconnecting piping failure (subsection 5.2.3.1.6). A copy of this reference was obtained and reviewed. We found that the strength factor for soil failure was determined by dividing the soil failure acceleration value of 0.67g from Reference 2 by the DBE value of 0.17g (i.e., 0.67/0.17 equals 3.9). We feel that this value is slightly biased on the conservative side since the analysis in Reference 2 was performed using a time history with a base acceleration value of 0.14, which produced a response spectrum which enveloped the broad-shaped spectral amplification curve anchored to 0.17g. So effectively, the median capacities based on using Reference 2 can be increased by the ratio of 0.17 divided by 0.14 (i.e., a factor of 1.21).

We also noted that the authors of Reference 2 were hesitant to support the results of the analysis and recommended that additional modeling and parametric studies be performed before the conclusions of the report are finalized. In developing the fragility curves based on this report, no uncertainty apparently was included to reflect the preliminary nature of the study which was conducted.

4.2.2 Structure-to-Structure Impact

This failure mode does not influence the core melt frequency, but is important to the failure of the containment barrier and potential for release. The fragility data for this failure mode were probably the same as used for the soil failure analysis; thus, the same general comments made for Section 4.2.1 apply to the impact mode.

Putting aside our concerns about the Reactor building soil analysis, we suspect that the fragility data may be biased to the conservative side. If the fragility parameters are based on just closing the gap between the two structures, then damage significant enough to cause a large hole is unlikely. A median-centered value could be determined by calculating the damage in the containment barrier caused by various sizes of interacting displacements. In addition, any impact would tend to break up the sinusoidal motion of the impacting structures causing a decrease in resonance and subsequent build up of motion.

4.2.3 Pressurizer Enclosure Failure

The fragility parameters for this failure mode have a minor effect on the frequency of core melt. We noted that the combined β_r value for strength and inelastic energy absorption is about twice the value given for exactly the same failure mode in a SSMRP report (Ref. 11, page 4-24). We do not understand why there should be a difference between the values for this component from the Zion PRA and the SSMRP reports.

4.2.4 Gross Structural Failure

As discussed in this section, in the event of a LOCA, a substantial amount of prestress capacity would be required for pressure loads and the seismic capacity would be reduced for all gross structure failure modes. Our comments for subsection 3.8 concerning the possibility of a LOCA followed by an aftershock is pertinent to the Reactor building. If it were determined that the probability of this combined event is significant, the median capacity value for many of the Reactor building gross structure failure modes would reduce substantially.

4.3 Auxiliary Building

Tables 4-11 through 4-13 give the fragility parameter values for the significant modes of failure for the Auxiliary building. The factors in these tables were not checked in detail. They were reviewed for reasonableness and compared in some cases with results obtained from the SSMRP studies. Specific comments for each of the failure modes is given below.

This is a particularly complex building which consists of several buildings which have been structurally joined together.

4.3.1 Shear Wall Failure

The shear wall failure mode is the second most important mode of failure which affects the frequency of core melt.

As we stated in our discussion of subsection 4.1.2, we are concerned about using a SDOF model to predict the inelastic energy absorption capacity of a MDOF system. For the particular wall considered, the problem is compounded further by the complexity of the various capacity modes of the wall and the proportion of recoverable and non-recoverable energy which is effective. As noted in the comparison of the SDOF and MDOF models of the Auxiliary building, the inelastic energy absorption median factor may be different by a factor of 0.6.

In addition to our concern with the median inelastic energy absorption factor, we believe that the combined uncertainty factor, β_u , of 0.20 for strength and ductility is low. Additional uncertainty should be included to reflect the uncertainty in the model (i.e., SDOF model used to predict MDOF response).

4.3.2 Other Auxiliary Building Failure Modes

We noted that the values for the median, β_r, and β_u for the strength and inelastic energy absorption factors are not consistent with the values give in an SSMRP report for the same components (Ref. 11). Table 3 summarizes the values from the Zion PRA and SSMRP reports. The ratio values for the median parameter are equal to the median capacity from the SSMRP divided by the Zion PRA report capacity using only the strength and inelastic energy absorption factors. Note that the SSMRP capacity values do not include the effects of building response, but possibly include effects of local response. Although the ratios individually are not meaningful, they should be the same for all three structures since in each report the same point of reference is used (i.e., ground for the Zion PRA and Node 3006 for the SSMRP). The logarithmic standard deviations are the same for the shear wall, but are different for the other two structures. The reasons for these differences are not known.

4.4 Crib House

No comment on the introductory paragraph.

Detailed information for the failure modes and assumptions used in the analysis for the Crib house structures is given in Reference 4. The fragility parameter values agree between the SSMRP and Zion PRA reports. We noted in the tables of fragility parameter values that the randomness due to spectral shape effects is small (β_r is less than 0.09). This is probably because the frequency of the Crib house is high and thus close to the point where the response spectra returns to the ground acceleration value.

Table 3 Comparison of Auxiliary Building Structures Strength/Ductility Parameter Values

Structure	Median Parameter Factor	Ratio	β_r	β_u
Shear wall				
SSMRP*	1.1g	1.44	0.12	0.20
Zion PRA	(2.5)(1.8)(.17g) = .77		0.12	0.20
Roof diaphragm				
SSMRP*	3.0g	2.08	0.07	0.22
Zion PRA	(3.4)(2.5)(.17g) = 1.45		0.15	0.16
Masonry wall				
SSMRP*	1.7g	0.93	0.23	0.24
Zion PRA	(4.7)(2.3)(.17g) = 1.84		0.42	0.16

* Reference 11

In regards to failure modes which were considered, we wondered whether the capacity of either the block walls which support the roof slab, or the connection between the roof slab and block walls is less than the capacity of the roof slab.

4.5 Liquefaction Potential

In regards to the physical properties of the soil and rock, we question the level of earthquake motion for which these properties are applicable.

4.6 Condensate Storage Tank

The structural calculations used as the basis for the fragility curves for the condensate storage tank are not known. We question whether the flexibility of the tank and potential motion amplification were considered in the analysis.

Again, we noted that the randomness value for spectral shape is small ($\beta_r = 0.01$). This may be because the fundamental frequency of the tank is close to 33 Hz. However, a field inspection of the tank indicates that it is probably below 33 Hz.

The welds to the anchor straps appear to be properly constructed.

4.7 Underground Piping

The structural calculation basis for the fragility curves for the service water pipes is not known. It appears that the definition of failure may be conservatively biased since the failure mode may only cause leakage, but not flow blockage. The important issue concerns how much water is needed for the safety-related function. Can significant leakage be tolerated?

5. Equipment Fragility

5.1 General Approach and Information Sources

We noted that no new analyses were conducted for equipment items.

5.1.1 Information Sources for Equipment

We noted that it was stated in the Zion PRA report that many of the design reports for non-NSSS items did not contain summaries and the calculations were difficult to follow. Thus, it was impractical to use all of the information available.

5.1.2 Equipment Categories

No comment for this subsection.

5.1.3 Response Factor Categories

We agree with the categories in the subsection.

5.1.4 Structural Response

As noted for subsection 4.1.2.2, we raised the issue of mode shape ordinate error due to flexibility of a local element or substructure. This is particularly appropriate for development of fragility data for subsystems which are supported by the structure. We question whether the structure models are of sufficient detail to represent the response of locations where equipment items are supported. The Licensee believes that the effect of local response is small (see Appendix A, Question 14); however, it is not clear whether each equipment support was systematically reviewed to establish that in-structure response spectra for local modes are not significant.

One category of structure response which is not included is "modal combination." We assume that the floor response spectra were developed by an approach which did not involve the SRSS combination of modal responses (e.g., direct integration or modal analysis with time phasing). However, if this is not true, then the variability due to the inaccuracy of the modal combination procedure should be included.

5.2 Equipment Capacity Factors

Specific comments are made for each of the subsections in the following text. In order to assist in determining the implication of issues and questions which are raised, the components listed in Table 5-6 of the Zion PRA report were associated with the various report sections. Table 4 lists the Zion PRA report sections, components, and median ground acceleration values. Particular attention was given to key equipment (see Zion PRA report Table 7.2-3).

Table 5 lists components for which we could not determine what basis was used for the capacity values from the Zion PRA report. For example, the service water pumps, which are the most critical components besides the ceramic insulators, are listed as a plant-specific component; however, no report section could be found which discussed the capacity of the service water pumps.

In reviewing many of the fragility parameters, it was not clear what specifically constituted the underlying bases. This question is asked about specific parameters in order to determine which ones are based on data, engineering judgment, or a combination of these sources. One parameter which is common to almost all components is material yield strength. The basis for assuming that the median yield value is 1.25 times the code specified value is not given. Information given in Reference 23 indicated that the ratio may be as low as 1.0. Also, the basis for the variability β_c value of 0.14 and the associated randomness and uncertainty components of β_r equal to 0.1 and β_u equal to 0.1 is not provided. For these cases, we are interested in knowing the bases for completeness.

It was learned at a meeting with PLG during the review of the Indian Point PRA that the separation of variability into its randomness and uncertainty components was primarily based on judgment. We believe that this should be documented in the Zion PRA report. In instances where analysis or data form the basis for selecting parameter values, this also should be documented. We do not object to determining parameter values subjectively, but feel it is imperative that the reader know what was done.

Table 4 Summary of Equipment Capacity Values

Zion PRA Report Subsection	Component	Median Ground Acceleration Capacity (g)
5.2.1.1	Reactor Pressure Vessel	7.31
5.2.1.2	* Reactor Pressure Vessel Internals	1.16
5.2.1.3	Steam Generator	3.29
5.2.1.4	Reactor Coolant Pump	3.29
5.2.1.5	Pressurizer	2.63
5.2.1.6	Control Rod Drive Mechanism	3.33
5.2.1.7	Reactor Coolant Loop Piping	11.08
5.2.1.8	* Safety Injection Pump	0.9
5.2.1.9	Residual Heat Exchanger	7.19
5.2.1.10	Component Cooling Water Heat Exchanger	8.32
5.2.1.11	Accumulator Tanks	37.19
5.2.1.12	Boron Injection Tank	8.66
5.2.1.13	Main Steam Isolation Valve	11.14
5.2.2.1	* Containment Fan Coolers	1.74
5.2.2.2	Residual Heat Removal Pumps	4.22
5.2.2.3	Centrifugal Charging Pump	3.77
5.2.3.1	Piping and Supports	3.35 - 10.45
5.2.3.2	Generic Equipment Structural Mode	
	* Ductwork and Dampers	0.97
	* Transformer	1.39
	* Relief Tank	1.19
	* Batteries and Rack	1.01
5.2.4	Capacities Derived from Tests	2.69
5.2.5	* Generic Capacities From Military Shock Test Data	0.60 to 0.86
5.2.6	Generic Capacities for Valves	4.41 to 6.93
5.2.7	Cable Trays	4.74
5.2.8	* Offsite Power	0.2

* Key Zion Equipment (Zion PRA Report Table 7.2-3)

Table 5 Equipment for Which Fragility Basis Not Known

Component	Median Ground Acceleration Capacity (g)
* Service Water Pumps	0.63
Fuel Oil Transfer Pumps	6.63
Motor Driven Pumps (Containment Spray System)	6.63
Diesel Driven Pump	6.63
Diesel Driven RHR Pump	7.56
Positive Displacement Pump	6.85
Component Cooling Pumps	6.63
Motor Driven Pumps (Auxiliary Feedwater System)	6.85
Turbine Driven Pump	6.85
CDR Mechanisms	5.33
* Refueling Water Storage Tank	0.73

* Key Zion Equipment (Zion PRA Report Table 7.2-3)

5.2.1 Plant-Specific Structural Capacities Derived From Design Reports

It is stated in this subsection that the logarithmic standard deviations for strength, randomness and uncertainty were derived in the same manner as for structures. We noted for subsection 4.1.2.6 that the basis for separating the total variability into randomness and uncertainty components for structures is not provided. Thus we do not know the basis for splitting variability for both structures and equipment. Our understanding from information we obtained during the review of the Indian Point PRA is that this was done primarily using engineering judgment. This should be documented in the Zion PRA report.

The ductility factor used for equipment (Equation 5-5) is different from the approach used for structures, which was based on deamplification factors for elastic-perfectly plastic systems (see Figure 4-3 in Zion PRA report). For structures, the ductility factor is a function of ductility and damping, while the factor for equipment is a function of only ductility. However, the differences between the two approaches are small.

Since both factors (i.e., for structures and equipment) are for single-degree-of-freedom elastic-perfectly plastic systems, there is inherent error in using these models for multidegree-of-freedom equipment (see comments for subsection 4.1.2 for the same problem for structures). In regard to this problem, we question the basis of the logarithmic standard deviation value of 0.2 for uncertainty. We suspect this value is higher and that an additional small value for randomness also should be included.

5.2.1.1 Reactor Pressure Vessel

It is not clear from the description exactly how the median strength and variability was calculated. In particular, we question whether variability was considered for the shape factor. Also, it is not clear what variability was used to determine the two logarithmic standard deviation bounds. Finally, we question how the ultimate strength (i.e., versus the yield strength) was used in determining the upper-bound strength.

Since the variability in strength includes more than just a contribution from material properties, we question the basis for separating β_{c_s} into β_{r_s} equal to 0.15 and β_{u_s} equal to 0.24. Similarly, the basis for splitting β_{μ} into its two components is not obvious.

The variability of equation 5-5 due to variability of only ductility gives β equal to 0.23 based on the combined value being 0.30 and the variability in the equation being equal to 0.20. The value of β equal to 0.23 apparently comes from the following calculation:

$$\beta = \frac{1}{2} \ln \frac{\sqrt{2(3) - 1}}{\sqrt{2(1.5) - 1}}$$

Another way to compute the value is to use a Taylor series expansion approach which gives a median value of F_{c_μ} equal to 2.27 (compared to 2.24) and β equal to 0.21 (compared to 0.23). Thus, the method used in the Zion PRA report gives acceptable values.

5.2.1.2 Reactor Pressure Vessel Internals

In response to our initial review of the reactor pressure vessel internals, the Licensee answered to a question which is documented in Appendix A (Question 16). However, we still are uncertain how the strength factor equal to 1.66 was derived. As indicated on page 5-14 of Section 7.9.2 of the Zion PRA report, a realistic static collapse moment capacity based on experimental data is 1.144 times the service level allowable of 1.8 Sm (or 2.06 Sm). For a typical austenitic steel, Sm is approximately 0.9 Sy, where Sy is the minimum yield strength. Hence, the static collapse moment capacity is approximately 1.86 Sy (this is the value given in the Zion PRA report).

For the Housner spectrum anchored to 0.5g, the maximum stress in the guide tube is reported to be 102.4 percent of the 1.5 Sm value or 1.54 Sm, which is equal to 1.38 Sy. Thus, the margin (or capacity factor) is just 1.86/1.38 or 1.35, not 1.66. The difference is a factor of approximately 1.25, which is coincidentally the value used to relate minimum yield capacity to median yield capacity in other parts of the PRA report; however, we do not know why this factor should be included here.

We are uncertain why our calculations differ with the Zion PRA report. If the more correct value is 1.35 instead of 1.66, then the capacity of the guide tubes would be approximately 0.94g, instead of 1.16g. Examination of the final Boolean equation for core melt shows that the status of the reactor core internals is not important to the frequency of core melt; thus, the difference noted above is not significant for the Zion plant.

5.2.1.3 Steam Generator

The variability values given for the steam generator ductility factor are different from the values given for the reactor pressure vessel on page 5-13. It is not clear why they should be different.

The approach used for this component appears to be reasonable. Any small changes in the parameter values will not affect the frequency of core melt analysis since the median capacity is relatively high.

5.2.1.4 Reactor Coolant Pump

No comment for this component.

5.2.1.5 Pressurizer

In determining the median ductility value for bolting, it was assumed by us that the anchorage detail was investigated to confirm that the bolts will (most likely) yield before they pull out.

The reason is not given for not including additional uncertainty for the capacity of the pressurizer, since its capacity was based on results from a similar plant.

It is our judgment that the issues raised for the pressurizer are unlikely to affect the frequency of core melt analysis since the median capacity is relatively high.

5.2.1.6 Control Rod Drive Mechanisms

Initially, we could not find this component in Table 5-6. The Licensee's response given in Appendix A (Question 17) indicates that the capacity of the Control Rod Drive mechanism is 3.33g. The basis for the capacity value is also given and appears to be reasonable.

5.2.1.7 Reactor Coolant Loop Piping

Basically the capacity of this component is the same as for the reactor pressure vessel except thermal stresses have been removed since they are considered to be self-limiting. Even if the thermal stresses were included, the capacity of the component is very high and thus will not affect the frequency of core melt calculations.

5.2.1.8 Safety Injection Pump

In developing the median strength factor value of 1.64, a shape factor of 1.5 and a yield strength factor of 1.25 are assumed (i.e., 1.64 = (35 x 1.50 x 1.25/40)). The shape factor value should be documented in the report.

We understand, based on a meeting with PLG during the Indian Point PRA review, that the composite logarithmic standard deviation for material equal to 0.14 is based on data. This fact should be documented along with the data or literature source where the analysis of the data can be found.

In developing the uncertainty for the strength factor, uncertainty also should be included for the fact that the pump material is not specified and an assumption that it is carbon steel was made. Although the shaft/bearing interaction median capacity is slightly larger, variability for this failure mode should be computed. A large variability for a slightly weaker mode may produce a larger probability of frequency of failure at acceleration values below the median.

5.2.1.9 Residual Heat Exchanger

In developing the uncertainty for the strength factor, allowance also should be included for the fact that the heat exchanger shell material is not known and an assumption that it is 516-Gr 60 was made. The basis for separating the variability into randomness and uncertainty components is not given.

Since the median capacity for this component is relatively high, the resolution of the issues will not affect the frequency of core melt analysis.

5.2.1.10 Component Cooling Water Heat Exchanger

The capacity for this component is relatively high.

5.2.1.11 Accumulator Tanks

The median ground acceleration capacity for this component is 37.19g; thus, it is unlikely that any rational adjustment to the fragility parameters would affect the frequency of core melt analysis.

5.2.1.12 Boron Injection Tank

The basis of the variability β_c-value of 0.2 assumed for the failure mode is not given. The capacity of this component is relatively high.

5.2.1.13 Main Steam Isolation Valve (MSIV)

The basis for assuming that a ductility of 1.0 is two logarithmic standard deviations below the median value should be given. Also, the basis of the 0.1 value assumed for β_r should be provided.

Since the median capacity for this component is relatively high, the information requested will not affect the frequency of core melt analysis.

5.2.2 Plant-Specific Functional Capacities Derived From Design Reports

We believe that eliminating inelastic energy absorption is conservative; however, it may be more appropriate in some cases to include the effect of ductility. In these cases, the median capacity would be higher, which would be offset to some degree by a higher uncertainty value to reflect the inability to determine when a functional failure occurs.

5.2.2.1 Containment Fan Coolers

The basis for assuming that the manufacturing tolerance stack is equivalent to -3β value on clearance is not given. Also, we questioned what data were used to determine that the median clearance value is 1/8 inch greater than the allowable deflection value of 1/8 inch. The basis for separating the variability into randomness and uncertainty components is not given in the report.

Based on a meeting with PLG during the Indian Point PRA review, we learned that the worst case manufacturing tolerance stackup would occur approximately 2 in 1,000 cases (i.e., approximately -3σ) based on manufacturing experience. We feel that this should be documented in the Zion PRA report along with the data or literature source for the data. The basis for other assumptions in this section should also be documented.

Calculations for the containment fan coolers were investigated as part of the Indian Point PRA review. The calculations show the development of the safety factors and associated logarithmic standard deviations. The development follows the procedure given in the Zion PRA report, and the variabilities are consistent with the general assumptions used throughout the report. The selection of the critical strength factor from three possible failure modes is documented; nowever, the main data are only referenced and not otherwise given. From this information we are unable to conclude about the accuracy of the strength factors. All we can state is that a systematic procedure was used.

The median ground acceleration capacity for the containment fan coolers is 1.74g. It is unlikely that reasonable changes in the assumptions made to determine the fragility parameters would affect the frequency of core melt calculation.

5.2.2.2 Residual Heat Removal Pumps (RHR)

The issues raised for the containment fan cooler fragility analysis also are applicable to the calculations for the residual heat removal pumps. The basis for the assumptions made in this section should be documented.

Since the median capacity for this component is relatively high, the resolution of the issues will not affect the frequency of core melt analysis.

5.2.2.3 Centrifugal Charging Pump

In quantitative terms, we do not know what is meant by: "Bearings can frequently withstand twice the rated load for short durations." The basis that the allowable load is a -3β capacity for short-term duration should be given. Also, the basis for separating the variability into randomness and uncertainty components should be provided.

Since the median capacity for this component is relatively high, the resolution of these issues will not affect the frequency of core melt analysis.

5.2.3 Generic Structural Capacities Derived From Design Criteria

No comment for introductory section.

5.2.3.1 Piping and Supports

We feel that it may be unconservative to base the fragility of the piping system on the single component type most likely to fail. This procedure implicitly assumes that the individual components are perfectly correlated. In reality, a piping system consists of a series of components whose capacities and responses are each partially dependent (Ref. 18). One approach for including this effect would be to determine an equivalent number of independent components, which would be based on the type of elements (e.g., butt welds, their number, location, etc.). Because piping systems can be very long, it is prudent to make a best estimate of the effect of dependency even if it is only based on engineering judgment.

In discussions with PLG during the review of the Indian Point PRA, it was stated that most piping systems have only one or two critical components. The rest of the components are generally understressed. It this is the case, then it does not matter whether or not partial independence is assumed. We believe that it is prudent to look at each safety-related piping system to determine that it is in fact reasonable to assume that only one component controls the capacity.

5.2.3.1.1 Piping Failure Modes

It is not clear in later development of the fragility parameters if the effect of the combination "0.75i" in the stress acceptance equation was incorporated.

5.2.3.1.2 Support Failure Modes

The decision to base the fragility analysis on supports that only carry seismic load implicitly assumes that the total applied stress as a percentage of the design stress is essentially the same whether normal stresses are present or not. This assumption appears to be reasonable.

5.2.3.1.3 Piping Fragility

The basis for splitting the logarithmic standard deviation for shape factor into randomness and uncertainty components should be documented.

In developing the ratio of static collapse load to allowable design load (i.e., P_L/P_D) , if we use a ratio of S_h to yield to be between 0.625 and 0.9 (along with the other factors given in this subsection), we find that P_L/P_D ranges between 1.62 and 2.33. If we then incorporate the various P_N/P_D and P_{OBE}/P_D ratios given in this section into equation 5.4, we obtain a median value of 4.6 (compared to 5.9) and a β_S of 0.40 (compared to 0.27). The basis for not considering a range on the P_L/P_D ratio (i.e., 1.62 to 2.33) should be explained.

The basis for the fragility parameter assumed for the ductility factor is not given. The lower bound system collapse factor (i.e., 1.0) seems conservative. The basis for it being 2β below the median value should be documented.

We believe that these differences would not affect the frequency of offsite consequences.

5.2.3.1.4 Support Fragility Description

It was learned at a meeting with PLG during the Indian Point PRA review that the logarithmic standard deviation value of 0.42 for the strength factor was obtained by establishing a lower-bound factor of safety using a minimum strength (code yield stress of 25 ksi reduced 15 percent for welding or threads, i.e., 21.2 ksi) and a maximum load stress of 1.1 times design stress which is 50/4 x 1.2 x .75 = 11.25 ksi where 1.2 is a short term load factor and 0.75 is also a factor for threaded connections. The lower bound factor of safety is then equal to 21.2/(1.1 x 11.25) or 1.7. Then β is equal to 1/3 (ln 5.9/1.7) or 0.42, where 5.9 is the median factor.

We believe that this is incorrect since the effect of threaded connections appears to be included twice and the code yield stress is not increased by a factor of 1.25 to a median value. A more rational β-value would be 0.28 instead of 0.42. On the other hand, a 3σ range seems high. If a more defendable 2σ range is used, the β-value is back to 0.43. Thus, we concur with the value used.

5.2.3.1.5 Governing Criterion for Piping

Except for the issue of dependence between piping system components, we feel that the issues raised will not affect the frequency of core melt analysis. However, as stated above, since the piping systems can be long with many components (hence potentially many locations for failure), the effect of dependency could lower the effective piping capacity sufficiently such that piping becomes an important component. We are willing to accept the argument, in general, that only one or two components are stressed to allowable values in a piping system; however, we feel that each critical piping system should be reviewed to determine that this assumption is appropriate.

5.2.3.1.6 Piping Subjected to Relative Building Motion

We agree that the equation for $P_f | A_i$ is appropriate for determining the effects for relative building motion. The terms $P_{oc\,\delta i} | A_i$ should reflect the steep uncertainty curves in Figure 4-7 if they are correct. No information is given which explains how values for the term $P_f\ \delta_i$ were developed.

Concern for the soil displacement versus ground acceleration curves in Figure 4-7 was expressed for subsection 4.2.1 in regards to failure of soil beneath the reactor building. One interpretation of the curve in Figure 4-7 is that at minus one standard deviation (0.55g) the displacement is unbounded. Thus, there is a probability of approximately 0.8 that the soil (hence all attached piping) will fail (i.e., frequency of failure equal to 1) for a peak ground acceleration greater than about 0.55g (value scaled from Figure 4-7). Thus, the -1β curve in Figure 5-4 should be to the left of the peak ground acceleration value of 0.55g. The other two curves (i.e., median and +1β) would be shifted to the left also. However, we agree that the reality of the situation is that the displacements are not unbounded, which could be shown to be true via multiple time history nonlinear analyses. Our guess is that without additional analyses, the median capacity is better than the value obtained as discussed in the example above, but that the uncertainty value is greater than β_U equal to 0.33. If the nonlinear analyses were performed, the uncertainty value would decrease.

We note in Table 5-6 which summarizes the fragility parameters that the median capacity for the 34-inch main steam pipe is 3.84g, which is based on the generic structural capacity derived from design criteria (see section 5.2.3.1.4). This value is much larger than the median value associated with Figure 5-4 (i.e., approximately 0.8g). In addition, the median ground acceleration capacity values for the other piping systems listed on page 5-44 also appear to be based on generic capacities derived from design criteria. The capacities for the piping systems connected to the Containment building are not based on the lower capacity displacement considerations. However, we noted in reviewing Section 7.2 that the fragility parameters based on the lower capacity due to relative building motion were used in the analysis.

5.2.3.2 Generic Fragility Descriptions for Other Equipment That Fails in a Structural Mode

It is not clear from the information given in this subsection how a median strength factor of 6.38 and a β_s of 0.51 were obtained. The basis for the derivations of these parameter values should be given. Also, the basis for separating the variability of the strength factor into randomness and uncertainty components is not documented.

It is not clear from the information given in this subsection how the median strength and variability parameters were derived for the ductility factors. The basis for the parameter value given on page 5-48 should be provided.

We note in the summary Table 5-6 that median ground acceleration capacity values for components in the category are as follows:

Ductwork and dampers	0.97g
Batteries and racks	1.01
Relief tank (RCS)	1.19
Transformer	1.39

The batteries and racks are part of the series expression for core melt; thus, a major change in the median value could affect the frequency of core melt estimate. Based on our review of the calculation for the batteries and racks, we recommend that the fragility parameters for this component should be recomputed (see Chapter 4). In regards to the other three components, the Licensee's response given in Appendix A (Question

6) convinces us that detailed analyses are not necessary.

5.2.4 Capacities Derived From Tests for Higher Seismic Zone Criterion

It appears from the discussion in this section that the separation of variability into randomness and uncertainty components was based on engineering judgment.

The capacities for the components that are included in this category are relatively high such that any small changes in the parameter values will not affect the frequency of core melt analysis.

5.2.5 Generic Capacities Derived From Military Shock Test Data Plus Seismic Qualification Reports

No comment for introduction.

5.2.5.1 Electro-Mechanical Equipment

It is not clear from Table 5-6 which component fragility values were developed based on Army Corps of Engineers test data for electrical-mechanical equipment. This should be documented in the Zion PRA report.

Comments concerning capacities determined using the data from the SAFEGUARDS program tests is discussed in the next subsection.

5.2.5.2 Electrical and Control Equipment

Reference 24, which was prepared for the SSMRP by SMA, gives background on reduction of data from the SAFEGUARDS program. This reference does not represent an independent check since both this Zion PRA report section and Reference 24 were prepared by the same authors. We generally concur with the development of fragility curves for relay chatter and breaker trip. However, we are uncomfortable with the general conclusion that failure occurs at a level several times the fragility level for recoverable interruptions.

Our position is based on two points. First, the duration of the input in the SAFEGUARDS tests was only 2 seconds long. During a large seismic event, the duration of motion will be on the order of ten to twenty seconds long. We can conceive of failure at a lower acceleration level due to the effects of duration. Second, we are concerned whether the equipment tested in the SAFEGUARDS program is representative of the specific safety-related equipment at Zion.

We agree that nonrecoverable failure is higher than relay chatter or breaker trip. However, we question whether the strength is a factor of several times higher, or possibly only fifty percent higher in some specific cases. Also, if a relay trips, we question whether it is reasonable to assume that an operator can reset the relay in a timely manner. We recommend that if a particular electrical or control component is a dominant contributor (or potentially a contributor) to offsite consequences, that a specific analysis be performed for that piece of equipment. As a minimum, the particular components (i.e., switches and breakers) should be compared to the units tested in the SAFEGUARDS program. If the units are different, then an independent basis for determining the fragility should be found.

The procedure used to lump together the various test results to determine the median capacity and associated variability (i.e., randomness and uncertainty) is not given. It is mentioned later in the report that the equipment ranged in frequency between 3.5 Hz and 10 Hz. We question the appropriateness of the test data for components with a fundamental frequency below 5 Hz. Also, we wonder what data other than the Army Corps of Engineers data are available to substantiate the assumption that unrecoverable damage would not occur until the median acceleration capacity levels are raised by a factor of at least

2. It also is not clear that it is appropriate to shift the hazard curves by the factor of 1.25 to account for the effects of near-field earthquakes and inelastic response for this equipment. We also feel uneasy that the Licensee does not know the similarity between the components tested in the SAFEGUARDS program and those installed at Zion (see Appendix A, Question 10). Also, see comments for subsection 7.2.4 concerning eliminating these equipment from further consideration.

From Table 5-6, the median ground acceleration capacity values (i.e., recoverable malfunction) based on analysis of the SAFEGUARDS data are as follows:

Distribution panel	0.60g
DC bus work	0.60
Switchgear	0.72
Motor control center	0.72
Diesel generator	0.86

Because these capacities are relatively low, the issues raised above should be addressed.

5.2.6 Generic Capacities for Valves

The acceleration capacity for the sample of values used to develop the fragility parameters ranged from 0.84, to greater than 23g's. Apparently, the capacities were not grouped to correspond to specific types of valves in the plant (i.e., check valves versus motor or air-operated valves).

Based on discussions at a meeting with PLG during the Indian Point PRA review, we concur that the capacities for the safety-related valves at Zion are relatively high.

5.2.7 Cable Trays

Based on the discussion in this subsection, it is assumed that all cable trays important to the safety of the plant are rigid (i.e., frequency greater than 33 Hz). If this is true, the capacity calculation appears to be reasonable. Because of the large number of cable trays, this category of component has the same problem with lack of perfect dependency raised for piping systems.

If the cable trays are not rigid and if many potential tray failures exist, then the effective median ground acceleration capacity may be significantly below 4.74g.

5.2.8 Offsite Power

We agree that the median capacity of ceramic insulators is low and it is reasonable to assume in the systems analysis that they have failed.

5.3 Equipment Response Factors

We have no comment on the introduction to this subsection.

5.3.1 Plant-Specific Equipment Qualified by Dynamic Analysis

The residual heat exchanger was selected as an example to demonstrate the methodology of deriving the response factors. For clarification and use in the review of later sections, the enveloped floor response spectrum used in the design, and the applicable Zion floor response spectra should be provided. No discussion was given as to what method was used to determine the applicable Zion floor spectra. Depending on the method used to develop the applicable Zion floor response spectra, the basis for determining uncertainty due to modeling will be different.

5.3.1.1 Spectral Shape

We agree that there are two factors to account for biases and variability in the floor spectral shape that must be determined. The first concerns the difference between the El Centro ground response spectra and the Zion site spectrum, and the second factor accounts for the difference between the envelope spectrum used in the design and the floor response spectrum based on the Zion site spectrum.

For the example component, the residual heat exchanger, a spectral shape factor is derived based on the difference between the envelope design spectrum for 1% damping and the Zion DBE spectrum for 5% damping at elevation 560'. There is insufficient information to verify a safety factor of 10. Also, no information is provided on the method used to compute the DBE floor spectrum; therefore, this input cannot be reviewed for its appropriateness. However, any change in the median safety factor is not important in lieu of the high capacity of the residual heat exchanger.

The variability in this factor was assumed to be 0.10, corresponding to a multiplicative factor of 1.11. This value appears to be low since at the +4β level there would only be a 1.5 factor increase. The basis for the 0.10 value should be given.

The factor accounting for the difference between the envelope spectrum used in the design, and the floor response spectrum corresponding to the Zion site spectrum has been determined on the assumption that the difference between the two ground input spectra is a constant at all frequencies, which is not the case. This factor varies over a range of 1.4 to 2.0. Not taking this into account prior to determining the second shape factor effect is a source of modeling uncertainty. However, the difference in the variability already determined is likely to be small. Another viewpoint is that the factor should be determined at the frequency of the structure. For which case there is no variability for this effect.

We do not generally agree with the statement given on page 5-60:

> "Note that for rigid equipment, whose fundamental frequency is greater than the frequency at ZPA, that the factor is 1.0 since both spectra are anchored to the same peak ground acceleration."

Since the peak floor acceleration value is a function of the spectral acceleration <u>at the frequency of the structure</u>, the difference in the spectra shown in Figure 5-1 will lead to differences in the peak response of any rigid equipment anchored to the structure. I would be conservative if this statement was in fact followed. The only time that this statement is correct is when the frequency of the structure is equal to or greater than the ZPA value.

Note that our comments for section 7.2 referring to the use of the Regulatory Guide 1.60 spectrum, as opposed to a site-specific spectrum, are also relevant for this subsection.

5.3.1.2 Qualification Method

We agree that the response spectrum method is median centered, with variability set to zero.

5.3.1.3 Damping

We agree that the damping factor has a value of 1. The variability in damping represented by $-\ell n\ (Sa_{\zeta=5\%} / Sa_{\zeta=3\%})$ could not be verified, since the applicable floor spectrum is not available. However, the value of 0.41 for β_C seems very high.

5.3.1.4 Frequency

We agree that the response factor for frequency is 1.0; however, no basis is provided for the statement that the coefficient of variation on frequency equals 0.30. Similarly, that this value corresponds to a 15% change in response for a one logarithmic standard deviation frequency shift is not supported. The degree of variability depends on frequency, and the shape of the floor spectrum and hence is not constant.

5.3.1.5 Mode Shape

We agree that the response factor for mode shape is 1.0. The assumption that the logarithmic standard deviation is 0.15 for multi-degree-of-freedom and 0.1 for single-degree-of-freedom systems is not substantiated in the text or in the referenced report (Ref. 53). Clarification of these values is needed. It is unrealistic to assume that the variability is constant for all equipment.

It has been assumed here and in previous sections that the residual heat exchanger responds predominantly in a single mode. No basis is provided to support this. We anticipate that any change to the mode shape parameter will have a small effect on the frequency of core melt.

5.3.1.6 Mode Combination

We agree with the modal response factor and variability values, with the reservation expressed earlier about the single mode response assumption.

5.3.1.7 Combination of Earthquake Components

We agree with the response factor derived. The basis for the assumed variability of 0.09 should be documented.

5.3.1.8 Combined Response Factor and Variability

We have no additional comments for this subsection.

5.3.2 Plant-Specific Equipment Qualified by Static Analysis

We have no additional comments for this subsection.

5.3.2.1 Flexible Equipment

The applicable floor response spectrum was not available to verify the 0.30g spectral value at the floor elevation, 590'.

We agree with the assumption of single mode response based on the high fundamental frequency of the pressurizer support skirt and flange.

5.3.2.1.1 Spectral Shape

We have no additional comments for this subsection.

5.3.2.1.2 Qualification Method

We agree with the method and results of determining a qualification method factor. The response factor of 3.2 was not checked since the applicable floor response spectrum was not available. The basis for the coefficient of variation of 0.10 for response is not given. It is not clear whether an analysis was conducted to consider the effect of the vertical component.

5.3.2.1.3 Damping

We agree with the approach used to derive the damping factor and variability; however, the values were not verified due to insufficient information.

5.3.2.1.4 Frequency

We agree with the method used to derive frequency response factors and variability; however, we could not verify the results due to insufficient information.

5.3.2.1.5 Mode Shape

We agree with the assumption of single mode response of the pressurizer skirt and flange. Verification is needed regarding the assumption of variability which we feel may not be a constant, even for single mode response.

5.3.2.1.6 Mode Combination

We have no additional comments for this subsection.

5.3.2.1.7 Combination of Earthquake Components

We have no additional comments for this subsection.

5.3.2.1.8 Combined Response Factor and Variability

We have no additional comments for this subsection.

5.3.2.2 Rigid Equipment

We agree that the only response factors to be considered for rigid equipment are the qualification method and earthquake component.

5.3.2.2.1 Qualification Method

The applicable floor response spectrum was not available to verify the qualification method factor of 10.

We agree that there is a small variability associated with the qualification method factor, with the exception that there may be an uncertainty component due to the method of determining the floor response spectrum. This concern was also raised earlier in comments for subsection 5.3.1.

5.3.2.2.2 Earthquake Component Combination

We have no additional comments for this subsection.

5.3.2.2.3 Combined Factors and Variability

We have no additional comments for this subsection.

5.3.3 Plant-Specific Equipment Qualified by Test

We agree that the response factors cited are those which should be considered for equipment qualified by testing.

5.3.3.1 Spectral Shape

We have no additional comments for this subsection.

5.3.3.2 Boundary Conditions

We agree that the test conditions can be assumed to be median centered with respect to the conditions at the plant. We note, however, that different failure mechanisms may exist for the supports in the Zion plant. For example, in the tests, bolt support failure was a possibility while under plant conditions this is not a likely event; however, the report (Ref. 53) does not provide variability for this difference.

5.3.3.3 Damping

We agree that the median response factor due to damping is 1.0; however, insufficient information was provided to verify the derivation of the variability factors.

5.3.3.4 Frequency

Insufficient information was provided to verify the derivation of the response factor and variability. The basis for assuming that the response corresponding to the frequency range 3.5 to 10 Hz is a $\pm 2\beta$ range should be documented. This assumption results in a low logarithmic standard deviation on response.

5.3.3.5 Multi-mode Effects

The basis for assuming the range 1 to 1.5 to be $\pm 2\beta$ above the median should be documented.

5.3.3.6 Earthquake Component Combination

The basis for assuming a $\pm 2\beta$ range for the response range of 0.707 to 0.926 should be documented.

5.3.3.7 Combined Response Factors and Variability

We have no additional comments for this subsection.

5.3.4 Response Factors for Generic Categories of Equipment

The basis for defining various types of equipment as generic, particularly in situations where the systems are complex should be provided. This is often the case for piping systems.

5.3.4.1 Piping 8" in Diameter and Less

We agree that the factors cited are those requiring consideration for the response of generic piping.

5.3.4.1.1 Spectral Shape

We have no additional comments for this subsection.

5.3.4.1.2 Qualification Methods

The basis for assuming a ±2β range on the bounds considered should be documented. It is not clear whether the fact that the piping capacity was determined using the OBE level was taken into account.

5.3.4.1.3 Damping

A simple assumption was used to determine the frequency of all piping systems. Although the estimate appears to be reasonable, there is an additional uncertainty component in the method used to develop the response factor and variability, particularly since the factor is being applied to all piping situations. It is anticipated that only small changes would result if additional uncertainty was added for this effect.

5.3.4.1.4 Frequency

The same comments concerning the frequency response factor discussed for Section 5.3.4.14 apply here as well.

5.3.4.1.5 Mode Shape and Mode Combination

We agree that mode shape and mode combination effects are included in the qualification method factor.

5.3.4.1.6 Combination of Earthquake Components

The basis for a randomness component value of 0.02 due to random phasing should be documented. This value appears low compared to previous estimates for the same effect.

5.3.4.1.7 Total Response Factor and Variability

We have no additional comments for this subsection.

5.3.4.2 Piping 10" in Diameter and Greater

We agree that the factors cited are those to be addressed for this class of piping. The point raised previously (see comments for subsection 5.3.4.1) regarding the use of the OBE level acceleration to determine the capacity also holds true here.

5.3.4.2.1 Spectral Shape

We have no additional comments for this subsection.

5.3.4.2.2 Qualification Method

We have no comments for this subsection.

5.3.4.2.3 Damping

The basis for choosing 10 Hz as the frequency to develop the response factor for damping should be documented. Given the various piping configurations, a single frequency is not appropriate. In addition to the variability associated with the randomness due to material effects, there would also be a component of uncertainty due to the method for selecting pipe frequencies and the variability in frequencies throughout the plant.

5.3.4.2.4 Frequency

We agree that the modal analysis is median centered. The basis for using 10 Hz as the median value should be documented.

5.3.4.2.5 Mode Shape

We agree that the response spectrum analysis is median centered. The basis for the logarithmic standard deviation of 0.15 should be documented. The same comments we made for mode shape for structures (see subsection 4.1.2.2) also apply here.

5.3.4.2.6 Mode Combination

We have no additional comments for this subsection.

5.3.4.2.7 Combination of Earthquake Components

We have no additional comments for this subsection.

5.3.4.3 Valves

We agree that valves can be considered rigid for frequencies above 20 Hz. No reference is provided, however, to support the assumption that all valves have frequencies greater than 20 Hz. We agree that the response acceleration of a rigid valve will be equal to the acceleration of the pipe at the point of attachment. We feel that a similar set of parameters could be developed for valves similar to those developed for piping equal to and less than 8 inches in diameter.

5.3.4.3.1 Spectral Shape

We have no additional comments for this subsection.

5.3.4.3.2 Qualification Method

The basis for using a range equal to the ZPA to 1.5 times the peak spectral acceleration as a $\pm 2\beta$ range should be documented.

5.3.4.3.3 Damping

We agree that the factor is median centered; however, there should also be a component of variability attributable to the valve, in addition to that associated with the piping, albeit this may be small.

5.3.4.3.4 Frequency, Mode Shape and Mode Combination

We have no additional comments for this subsection.

5.3.4.3.5 Combination of Earthquake Components

We agree that this factor is identical to that determined for piping. We have no additional comments for this subsection.

5.3.4.4 Floor and Wall-Mounted Equipment With Generic Capacities

We have no additional comments for this subsection.

5.3.4.5 Cable Trays

We agree with the method for determining the response factor.

5.4 Structural Response Factors

Comments concerning these factors are made for the subsections 4.1.2.1 through 4.1.2.6 from Chapter 4 of Section 7.9.2. However, we noted that many of the values in Tables 5-3, 5-4, and 5-5 are different (and in many cases smaller) than the corresponding values given in

Tables 4-4 through 4-17 of Chapter 4. The reason for the differences between the values in the two chapters is not known.

It appears that the differences between the values in the chapters will cause only a small change in the results of the frequency of core melt analysis.

5.5 Fragility Description

No comment.

7.9.3 COMMENTS ON EFFECTIVE GROUND ACCELERATION ESTIMATES

1. Introduction

We concur with the concept that near-field low magnitude earthquakes are generally less damaging than far-field large magnitude events with the same instrumental peak ground acceleration value. We raise several issues, which are discussed in the next subsection, which question how this concept was applied in the Zion PRA study.

2. Effective Peak Versus Instrumental Peak and Sustained Peak Accelerations

As part of our review for this subsection, we read Reference 19, which explained in more detail the concepts discussed in Section 7.9.3. Reference 19 in turn refers to a report which documents the basis, that for the purpose of predicting elastic response of structure in the 2 to 10 Hz frequency range, median broad-banded amplification spectra (such as used in developing the fragility curves) are more accurately anchored to an acceleration value equal to 1.25 x A_{3F} (Ref. 20). In Reference 20, twelve earthquake response spectra are compared to the mean plus one standard deviation WASH-1255 amplification spectrum anchored to 1.25 x A_{3F} for each time history.

Visually, the comparison between the two types of spectra (actual and broad-banded) in Reference 20 is convincing. In the 2 to 10 Hz frequency region, the comparison appears to be median centered. However, it is difficult to visually determine what the difference would be if the median amplification spectrum (which was used in the Zion PRA report) had been used instead. It would be more comforting if a statistical analysis had been performed to verify that 1.25 is the appropriate factor.

The adjustment of the anchor acceleration value must be done with caution. Near-field low magnitude response spectra tend to be peaked at one (or more) natural frequencies for a particular site. In general, the broad-banded spectrum will be conservative except near the peak of the site-specific spectrum, where it may be just right or even lower. Thus, the correction factor F is appropriate in a median sense; however, there is uncertainty which exists for any specific structure. It makes a difference whether a fundamental building frequency is higher or lower than the frequency corresponding to the peak of a site-specific spectrum, in regards to whether significantly less damage will occur for a near-field low magnitude event.

A rational procedure for determining a value for F for a specific structure would be to determine the relative damageability between the best estimate of the site-specific response spectrum and the broad-banded spectrum used in the Zion PRA analysis at the fundamental frequency of the structures being considered. We noted from the SSMRP work that the ground response spectrum for the Zion site peaked between 2 and 8 Hz (Ref. 1) and that the predominant frequencies of the Containment and Auxiliary buildings are in this range (Ref. 9). Hence, we wonder whether the broad-banded spectrum used in the Zion PRA report is in reality close to being median centered for these structures; thus, no adjustment would be required (i.e., F equal to 1.0).

We are also concerned about applying this concept to equipment located in a building without first confirming that it is appropriate to do so. A structure acts as a filter which smooths the incoming seismic time history to produce a more sinusoidal appearing time trace at equipment support locations. Can the same argument for the factor F be made for equipment housed in a structure as for structures supported on the ground needs to be documented.

The value of F recommended in this section is equal to 1.25. We believe that even if the value were 1.0 that only a small effect would occur to the frequency of core melt analysis. In general, we believe that a value of F equal to 1.25 is on the conservative side for structures. For equipment located in structures, which have a capacity below the capacity of the equipment, this value of F is probably also conservative. The argument given by Structural Mechanics Associates (SMA) at the meeting with PLG during the review of the Indian Point PRA is that the softening of the structure stiffness at high levels of ground motion will decrease the input to the equipment. All safety-related equipment which affects potential offsite consequences falls into this category. This value may not be conservative for certain equipment located on the ground or attached to the base of structures. Equipment, which does not have inelastic energy-absorption capacity or which depends on function capacity, respond more closely to the peak ground acceleration capacity.

3. Upper-Bound Cut-Off On Effective Peak Acceleration

After considerable discussion and thought concerning the use of an upper-bound cut-off on effective peak acceleration, we believe that it is more appropriate not to truncate the hazard curves, but to reflect a limit on damagability in development of the fragility curves. The mechanism to handle this effect is currently not an element of the fragility analysis. A new factor or redefinition of an existing factor is required to treat the frequency dependent effect.

8.8.1 DETERMINATION OF RISK FROM EXTERNAL INITIATING EVENTS - SEISMIC RISK

We feel that the frequency of acceleration values at different probability levels given in Table 8.8-1 are at too coarse a spacing to give stable frequency of core melt values in the tails of the probability density function for frequency of core melt (see Figure 7.2-5). Figures 8, 9, and 10 show plots of these values (with obvious corrections to typographical errors in Table 8.8-1) as compared to the data obtained from Section 7.9.1 which have been shifted by a factor of 1.23 and truncated for maximum acceleration values. We expected the corresponding pairs of curves of curves to coincide. The differences are about 30 percent. The effect on the mean frequency of core melt would be small. However, the effect on the tails of the probability density function would be much larger. Based on the Licensee's response given in Appendix A (Question 1), we believe that the hazard values given in Table 8.8-1 are incorrect.

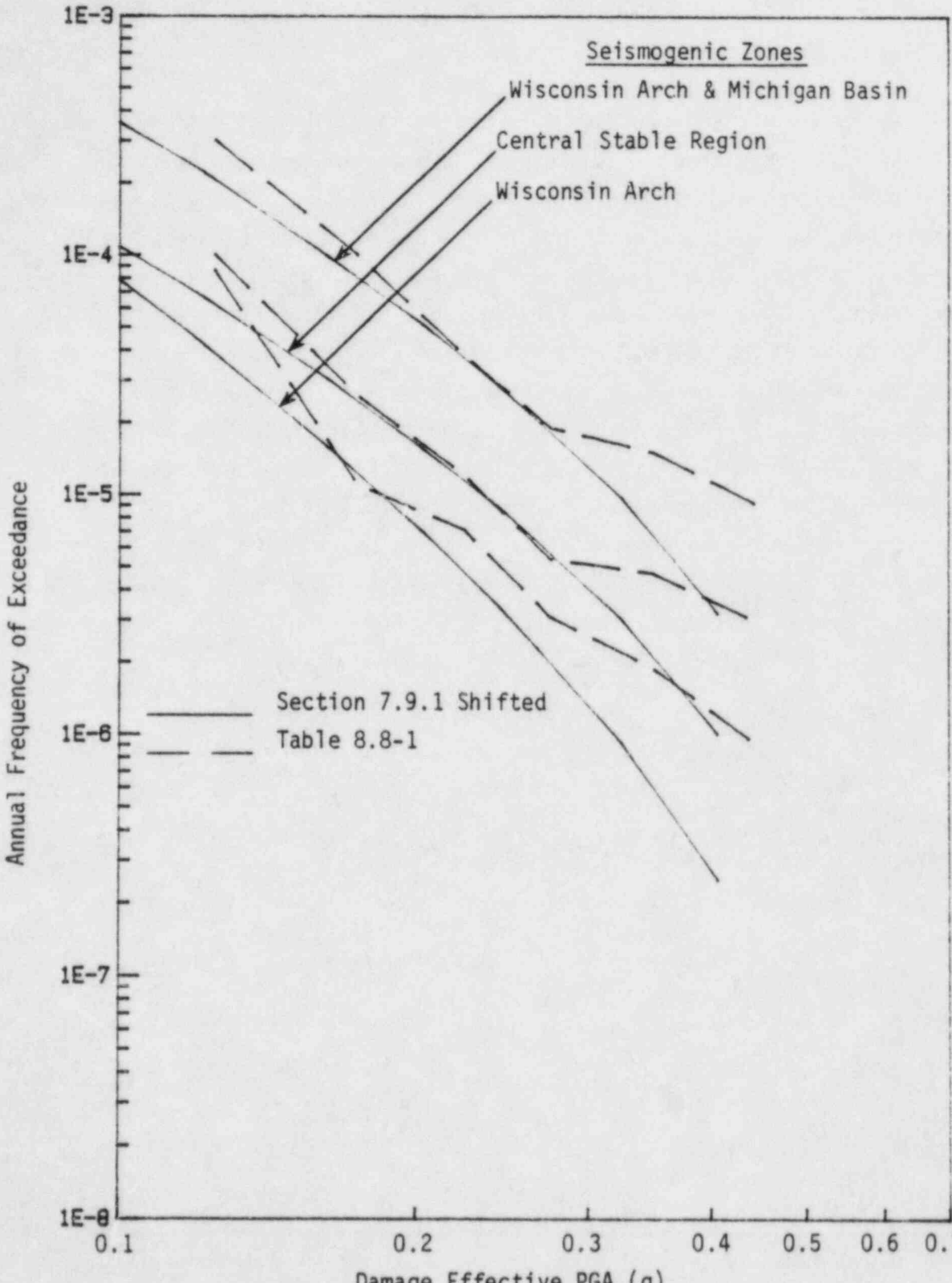


Figure 8 Hazard Curve Comparison - M_b = 5.6 Cutoff

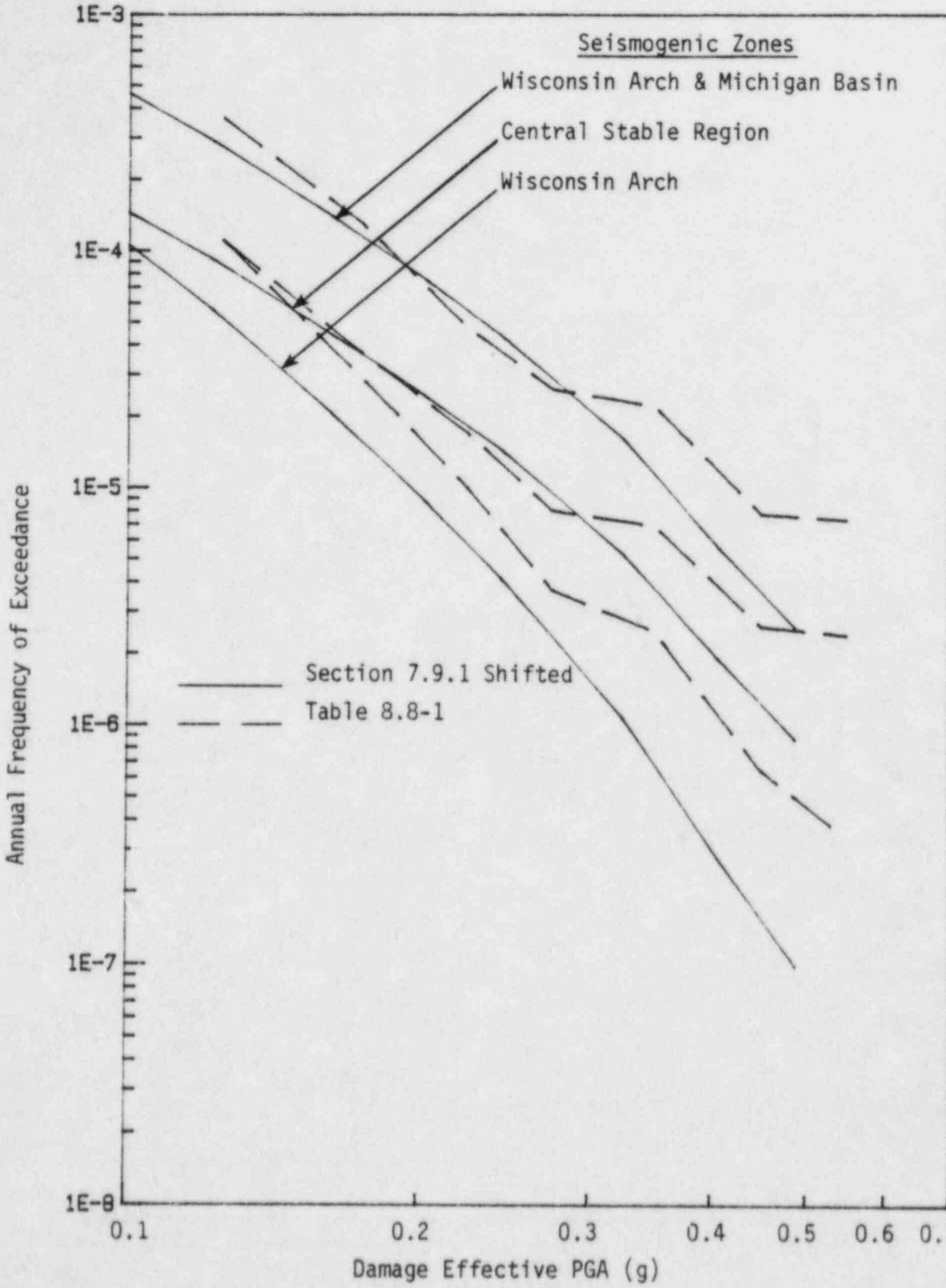


Figure 9 Hazard Curve Comparison - M_b = 5.8 Cutoff

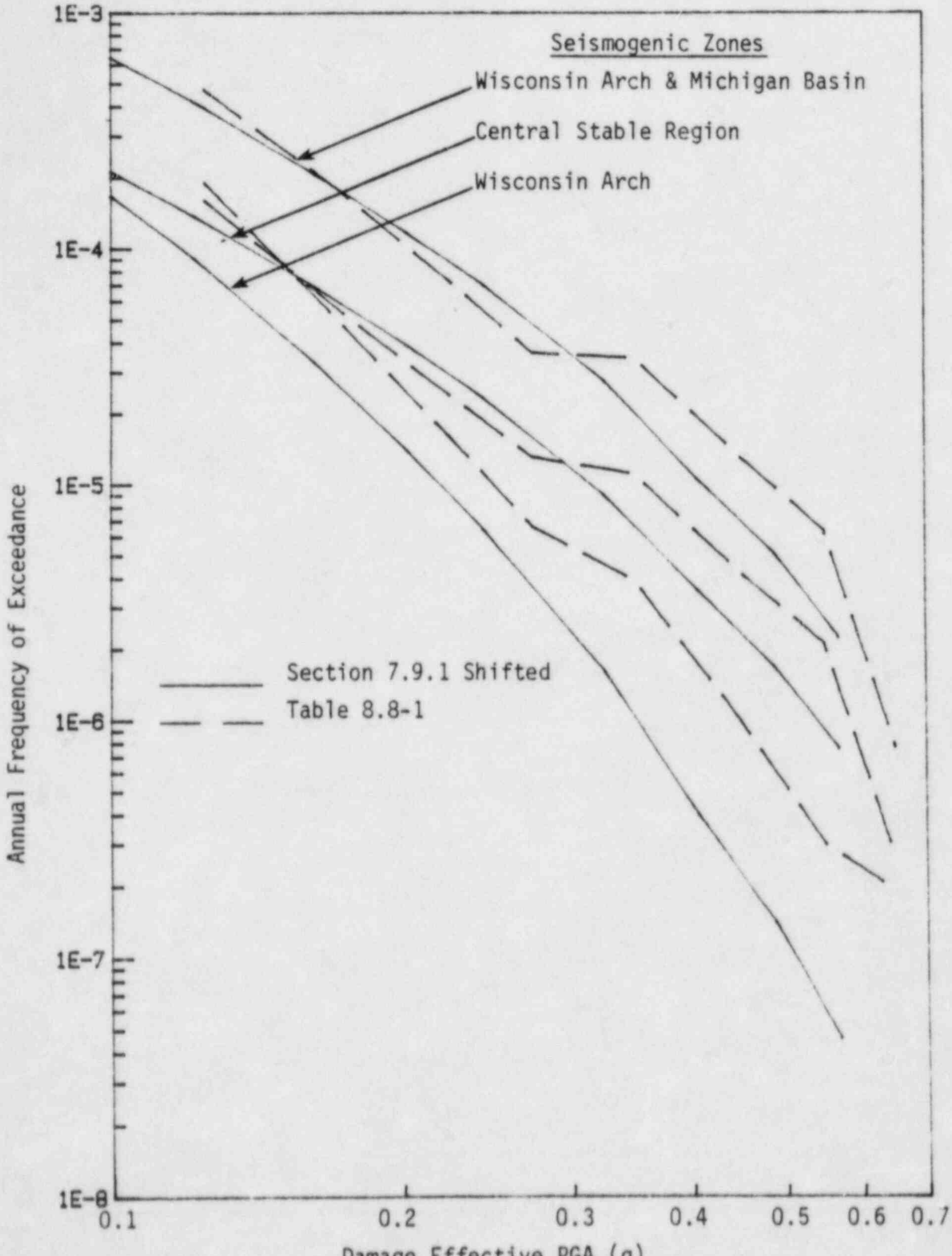


Figure 10 Hazard Curve Comparison - M_b = 6.0 Cutoff

4. CALCULATIONS

A meeting was held with Structural Mechanics Associates (SMA), at the end of the project review for the draft report, to discuss the calculations for seven of the most critical structures/components. Copies of the computations were provided and were reviewed with SMA engineers. A detailed check of the mathematics was not performed. However, the general flow of the calculations and logic was traced, and spot checks of the mathematics were made. The purpose for reviewing the calculations was to gain a better understanding of the assumptions and the analyses that were actually performed to develop the fragility parameters. Several of the issues raised in Chapter 3 were answered based on the discussion with SMA. However, we have left the issues in Chapter 3 since we feel that in many cases their answers should be formally documented in the Zion PRA report.

Comments based on the review of the following structures/components are given below.

- Service water pumps
- Auxiliary building concrete shear wall
- Refueling water storage tank
- Interconnecting piping/soil failure
- Crib house pump enclosure roof
- 125 VDC batteries and racks
- Service water system buried pipe 48"

Service Water Pumps

This component (besides the ceramic insulators) is the largest single contributor to the frequency of core melt. The fragility analysis for this component was based primarily on an incomplete stress analysis report. No detailed drawings for the pumps were available for the analysis. The critical failure mode was assumed to be yielding of the casing and interaction with the pump shaft leading to a functional failure. Thus, no inelastic energy absorption was used.

The analysis followed the procedures presented in Chapter 5 of Section 7.9.2. Two areas were found which indicate that the median capacity value may be overly conservative. First, the mass of the impeller and pump casing, past the last support and possibly the entire casing, was assumed to be be concentrated at the end of the casing. The reduction in frequency due to hydrodynamic mass effects was considered, but the reduction in inertial forces due to the displaced mass of water was not included in the force and stress calculations. Thus, the computed stress reported due to the DBE is higher than would actually occur. The amount of the difference for these considerations on the median capacity value was not estimated.

The second area of conservatism concerns the selection of the median factor for combination of earthquake components. A value of 0.81 was used in the analysis. Because the cross-section of the pump casing is circular and not significantly affected by the vertical earthquake component, a more rational value for the median factor would be closer to 1.0. For this difference alone the median capacity could be increased by a factor close to 1.2.

The total variability β_c value is 0.36. Based on the lack of information (i.e., poor stress report, no drawings or detailed calculations) we feel that this value is low.

We did not verify the calculations for the structural response portion of the analysis.

In conclusion, we judge that the fragility parameters used in the Zion PRA report for the effects of the service water pumps are conservative.

Auxiliary Building Concrete Shear Wall

Relatively detailed calculations were prepared to determine the capacity of the shear wall. Our general impression of the analysis is that more uncertainty should have been provided for the strength and inelastic energy absorption factors (the total uncertainty β-value for these factors is 0.20). A large number of assumptions had to be made leading to the median capacity value. In light of our comment for subsection 4.1.2 concerning the difference between the ductility factor used in the Zion PRA analysis and the value obtained from the SSMRP study, we feel that a larger uncertainty value is more appropriate.

We have made numerous comments throughout this report concerning parameters which affect the median capacity of the Auxiliary building shear wall. Based on the information we have reviewed, we are uncertain whether the median value is high or low.

Refueling Water Storage Tanks

It was assumed in the Zion PRA analysis that the failure of the refueling water storage tanks is controlled by the failure of the Auxiliary building shear wall. This assumption is conservative, but the amount of conservatism is difficult to quantify. However, this assumption implies that these two components are perfectly correlated (i.e., the failure of the wall implies the failure of the storage tanks). Looking at drawings of the Auxiliary building during the inspection of the plant leads us to believe that the capacity of the refueling water storage tanks are much higher than the Auxiliary building shear walls.

Interconnecting Piping Soil Failure

Based on the assumption of the Reactor building displacement versus acceleration curves shown in Figure 4-7, a detailed analysis was performed for the main steam lines which connect between the Auxiliary and Reactor buildings. Because of the steepness of the curves which was conservatively assumed, the fragility parameters are very similar to the fragility parameters for the failure of the soil beneath the Reactor building.

We feel that this analysis is conservative. A more realistic set of curves could have been used in place of the curves in Figure 4-7, which would have increased the capacity of this component. In addition, the results for the main steam line were used for all other pipe lines, which are smaller in size. We believe this also is probably a conservative assumption.

Crib House Pump Enclosure Roof

The median capacity estimate for this component appears to be reasonable. The capacity factor may be larger if a lower natural frequency of vibration was used which corresponds to the 0.86g ground acceleration capacity value (for the critical direction a frequency of 13.5 Hz was used in the analysis). Our judgment is that the median ground acceleration capacity would decrease for this effect. However, this could be offset by a slightly higher allowable capacity value for the critical roof section. Our feeling is that the two effects probably would offset each other. Based on our inspection of the Crib House, we doubt that failure of the Crib House roof will fail all six water pumps. Thus, the analysis for failure of this component is very conservative.

125 VDC Batteries and Racks

Based on our review of the calculations for this component, we feel that a more detailed analysis should be conducted to develop fragility parameters for this component. Based on our inspection of the Zion plant, we still believe that the racks should be analyzed.

The capacity values for this component are based on generic fragility descriptions for equipment which fails in a structural mode (see section 5.2.3.2). The fragility parameters were developed based on requirements of the ASME Code, while the racks were in reality designed to the AISC Code. In addition, a report for a sine dwell test of a battery and rack system actually shows that the racks were not qualified for the design criteria. No drawings of the batteries and racks were available at the time the analysis was performed.

Based on discussion with SMA, they feel that the median capacity value is probably correct based on recent studies they have performed for racks and battery systems. Since this is a critical component, a new analysis should be conducted to document the fragility for this component.

Service Water System Buried Pipe 48"

A consevative soil-wave analysis was performed for this component. Our judgment is that the analysis was conservative, and since the median ground acceleration value is 1.4g, this component will not significantly affect the results of the frequency of core melt analysis.

5. CONCLUSIONS AND RECOMMENDATIONS

We agree that the methodology used in the Zion PRA report for seismic effects is appropriate and adequate to obtain a rational measure of the probability distribution of frequency of core melt. The procedure is based on a simplistic probabilistic model which uses some data, but currently relies heavily on engineering judgment. We offer comments in Chapter 3 in regards to applying the methodology to the Zion plant. In our review of the Zion PRA report sections, we have identified issues which we believe should be addressed. The following unresolved areas of concerns are discussed in Chapters 3 and 4.

- The definition of damage effective ground acceleration used in the analysis may not be appropriate for electrical and mechanical equipment, buried pipe, and equipment which depends on functional operation as opposed to ductile strength capacity.

- Design and construction errors and aging should be considered.

- The possibility of a LOCA followed by an aftershock, or the occurrence of a moderate earthquake, when some safety-related equipment is unavailable, should be considered.

- The affects of variability in SDOF models for MDOF structures for determining the contribution of inelastic behavior should to be included in the analysis.

- The basis for the variability split into randomness and uncertainty components should be documented for critical structures and components.

- Piping systems and cable trays may have less capacity because of the numerous series components present and the potential lack of dependence.

- The fragility curves for the batteries and racks should be recalculated based on more detailed information for this component.

- The coarseness of the data points for the hazard and fragility curves may affect the accuracy of the tails of the probability density function for frequency of core melt, although we do not feel that the tails are particularly meaningful, except in a qualitative sense.

- Information for the basis of the service water pumps capacity should be documented since this is the most critical component.

- The development of the damage effective ground acceleration value in Section 7.9.3 should consider the effect of a best estimate site-specific ground response spectrum relative to the broad-banded spectrum used in the analysis.

- The decision to eliminate the electrical components from further consideration should be reevaluated in light of the comments made in this report.

Other questions and issues cited in the text should also be addressed.

A general impression that we have is that the median values are too conservative, but that the uncertainty is too small.

We recommend that the following should be done:

- Documentation of the bases for assumptions be provided.

- A sensitivity section in the Zion PRA report be included to inform the reader concerning the effect of changes in values of significant parameters on the frequency of core melt analysis.

- When the SSMRP study is completed for the Zion plant, a comparison of the two approaches should be conducted as a check.

- A more detailed probabilistic analysis should be conducted for the effects of internal flooding, and a probabilistic analysis be conducted for external flooding.

Based on our detailed review of the seismic fragility analysis and cursory review of the seismic hazard analysis, we believe that the mean frequency of core melt due to seismic events given in the Zion PRA report (i.e., 5.6×10^{-6} per year) is on the conservative side. Assuming that the systems analysis is correct, we would be surprised if the "true" value was more than a factor of 10 different (a large effect); however, because of newness of these types of analyses, a factor of 2 to 3 is possible.

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APPENDIX A

LICENSEE RESPONSE TO REVIEW QUESTIONS

Seismic Fragility

The following questions were taken from the draft report (dated February 22, 1982): "Review of the Zion Probabilistic Safety Study, Seismic Fragility," prepared for Sandia National Laboratories by Jack R. Benjamin and Associates, Inc., which is Appendix D to March 5, 1982 Sandia letter report. The page numbers below each question refer to the draft report which gives background material for each question. The issues raised represent the most significant concerns for the seismic fragility study which should be addressed and resolved.

1. What seismic hazard curve values were used in the integration with the seismic fragility curves to obtain the frequency of core melt probability distribution? (pp. 5, 6, 21, 23, and 94)

Response

The results of Section 7.9.1 are presented in Table 3 of that section, as the conclusions therein state. However, it was necessary to change the reported parameter from peak sustained-based acceleration a_{ps} to sustained acceleration, a_s, on which the fragility analysis is based (Section 7.9.3). According to Section 7.9.1, page 10, this is accomplished by the use of Equation (3) so that the plotted values of acceleration are the tabulated values divided by 1.23. The second consideration was that there be a correspondence between the predicted maximum site intensity and the maximum damage on which the Modified Mercalli Intensity (MMI) scale and acceleration-related damage are based in the fragility analysis. This leads to a justification for truncation of the seismic hazard curves as discussed in detail in the response to question 15. Then, so that the seismicity and fragility analysis results would be compatible in the assembly process, the exceedance frequencies for the a_{max} infinity cases, Section 7.9.1, Table 3, were truncated at accelerations recommended in Section 7.9.3 and corresponding to each of the three $m_{b,max}$ cases (using $I_o = 2m_b - 3.5$, per Section 7.9.1, page 7) of each seismogenic zone. Accelerations seen in Section 7.9.1, Table 3, were converted to sustained acceleration, the above truncations were applied to the table, and the resultant table was actually used in the calculations.

Seismic Fragility

2. Is the definition of the damage effective ground acceleration appropriate for equipment which depends on functional operation as opposed to ductile strength capacity (e.g., service water pumps)? (pp. 17, 56, 63, and 78)

Response

Equipment response to a seismic event is dependent on the structural response at the equipment location. The damage effective ground acceleration is a measure of expected structural response as opposed to the elastic structural response that would be calculated by anchoring the earthquake spectra to the highest instrumental peak acceleration; thus, for equipment mounted above the basemat, damage effective ground acceleration is appropriate regardless of the failure mode of the equipment. For equipment mounted on the basemat which fails in a structural mode, several cycles of strong motion are required for the equipment to reach peak response, and again, damage effective ground acceleration is a more rational indicator of failure level than instrumental peak.

For equipment mounted on the basemat and which is acceleration sensitive, use of the instrumental peak may be more appropriate. However, most acceleration sensitive devices such as relays are mounted in flexible cabinets or racks which are structural elements requiring several cycles of strong motion to develop peak response. Again, damage effective ground motion is considered appropriate.

Examination of the equipment list for the Zion PRA study reveals that there are no known acceleration sensitive devices that are mounted on the basemat which have rigid links to the basemat. Most of the acceleration sensitive equipment is mounted in the auxiliary building well above the basemat and none of the acceleration sensitive equipment items are mounted in rigid structural elements connected to the basemat.

Seismic Fragility

3. What uncertainty was assigned to the inelastic energy absorption parameter for structures and equipment to account for the variability caused by using single-degree-of-freedom models for multi-degree-of-freedom prototypes? (pp. 50 and 66)

Response

There was an uncertainty value (β_u = 0.2) assigned to the use of the equation for ductility factor. Most of the median equipment ductilities were considered to be 2.0 or less resulting in a ductility factor of 1.41 or less. For a system ductility of 2.0 and considering the properties of the lognormal distribution, the β_u of 0.2 applied to the ductility factor equation would reduce the median ductility factor to 1.0 at 1.71 β's below the median; thus, the model indicates that about 4-1/2 percent of the time the median ductility factor would be less than 1.0. This may be biased on the conservative side, especially for ductilities of less than 2.0.

For very high values of ductility, the uncertainty should be increased above 0.2. However, in the development of fragilities for equipment and piping, median ductilities of no greater than 3.0 were considered and the estimated uncertainty, β_u = 0.2, for use of a SDOF ductility criterion for MDOF systems is considered reasonable.

SMA has conducted some limited studies of MDOF versus SDOF response to mutliple dynamic loads and a general conclusion that can be reached from the studies is that the SDOF models tended to overpredict response to the loads being considered. The studies were conducted for purposes other than the question under consideration and are not conclusive; but based on the trend observed, the use of SDOF response behavior to estimate the beneficial effect of ductility on MDOF systems would, if anything, be biased on the conservative side.

Seismic Fragility

4. Since a more detailed analysis for the effects of inelastic energy absorption was conducted in the SSMRP for the Auxiliary Building, shouldn't the results from this analysis be used to determine the fragility parameters for the concrete shear wall? (pp. 50 and 59)

Response

The analysis conducted for the SSMRP was for a "representative" shear wall structure. This representative structure was based on the Zion auxiliary building, but did not include the turbine building which is integrally connected to the auxiliary building along a common wall. The analysis of this reduced structure consisted only of a two-dimensional analysis for response in the E-W direction. The structure is nearly symmetric about the E-W axis and very little torsional response results from E-W excitation. However, the structure is highly nonsymmetric about the N-S axis and torsion is significant. In comparison to capacity to withstand N-S excitation, the capacity in th E-W direction is considerably higher (by a factor of more than two). The controlling common shear wall capacity results from N-S excitation so that the SSMRP evaluation is not directly applicable. While the uncertainty in the PSS common shear wall fragility could be expected to be reduced by conducting a similar nonlinear analysis for N-S excitation, a three-dimensional model which also includes the turbine building would be required.

Seismic Fragility

5. What effect does the absence of perfect dependence have on the fragility curves for piping systems and cable trays (i.e., cable systems)? (pp. 28, 73, and 79)

Response

The fragility value for piping is intended to apply to a run of pipe from anchor to anchor. In any piping run the number of critically stressed areas is limited to only a few locations. The fragility model conservatively assumes that the most critically stressed element is a butt weld joint, typically at an anchor. This type of element has the least margin against plastic collapse when stressed to code allowables when compared to the margins for other types of elements which typically have higher stress intensification factors, and frequently govern the piping design. Consideration is given to the fact that for a piping system to collapse, more than one plastic hinge must form. It is possible in some piping runs for more than one portion of the run to be a weak link and in these few instances the fragility description may tend to underestimate the probability of failure. This is believed to be approximately accounted for by basing the piping fragility description on the weakest element (butt joint at an anchor) and assuming that this type of element is the highest stressed in the pipe run.

Having established a fragility description for a pipe run from anchor to anchor, the systems analyst then makes an estimate of the number of such runs that might be present in a system.

Cable tray fragilities were based upon test data for various tray and support assemblies and are considered valid for each assembly. The systems analyst then considers the effect of several assemblies in series in developing the fragility curve for a safety system.

Seismic Fragility

6. Since the ductwork and dampers, batteries and racks, relief tank, and transformers have relatively low capacity values based on generic data, shouldn't specific analyses for these components be performed to develop the fragility curves? (pp. 11, 76, 101, and 102)

Response

Median acceleration capacities of the above items are:

Ductwork and Dampers	0.97g
Batteries and Racks	1.01g
Relief Tank	1.19g
Transformer	1.39g

If plant specific analyses were conducted for these items, the uncertainty on median capacity would be greatly reduced, but we feel that any revised median value would probably fall within the $\pm 1\beta$ value of the initially estimated uncertainty. As discussed in Section 7.2.4 of the ZPSS, it is not necessary to model some of these components (such as the relief tank and ductwork) because their failure is not a consideration in the failure scenarios. The transformer failure is negated by the lower capacity of its insulators. Further, as seen by the Booleans in Section 7, core melt is dominated by two major structural failures, the containment building and the control building. Relative to these two major structural failure modes of 0.73g for soil failure beneath the containment building and 0.73g for shear failure of the auxiliary building shear wall, the above equipment capacities are relatively high. Therefore, plant specific analysis of the batteries and racks would not change the results and is unwarranted.

Seismic Fragility

7. What effect does the coarseness of the data points for the hazard and fragility curves have on the accuracy of the tails of the probability density function for frequency of core melt? (pp. 5, 6, and 94)

Response

The importance of greater discretization of acceleration and fragility curves could only be shown if parallel calculations were performed, which we have not accomplished. However, we believe there would be no significant difference in the mean values or in the distributions.

Seismic Fragility

8. What is the basis for the displacement versus acceleration curves shown in Figure 4-7 of Section 7.9.2? (pp. 56 and 75)

Response

As discussed in the report, base slab uplift was determined based on a nonlinear dynamic analysis of the containment building which was conducted by Sargent & Lundy (Reference 1). Initiation of base slab separation from the soil is predicted at slightly less than the DBE (0.17g), and a maximum uplift of 1.9 inches at approximately 0.7g. This is consistent with uplift predicted from other nonlinear dynamic analysis of reactor containment structures (Reference 2). From Reference 1, a relationship between moment and acceleration was available. Using both the tangent stiffness and secant stiffness approximation, the base rotation and contact area as a function of acceleration were developed from which the uplift as a function of acceleration was computed. The accelerations included a factor of 1.09 to account for the response spectrum developed by the time history compared to the median centered response spectrum used in this analysis.

The variability shown in Figure 4-7 is the composite variability expected from both randomness and uncertainty. In the range of base slab uplift of interest shown in Figure 4-7, the curves of displacement as a function of acceleration become steep in the higher acceleration range, but they are not vertical. It should not be implied that displacements increase without bound. For massive structures such as the reactor building to be overturned as rigid bodies by earthquake excitation is not considered credible. Although overturning moments can be generated which, if applied as a static moment, would predict overturning, the time duration of earthquake cycles is much too short to allow rigid body rotations of this magnitude to occur. Thus, the direction of excitation will reverse long before the structure can rotate an appreciable fraction of the total rotation necessary to cause instability. As an example, the rotation at the base slab for 1.9 inches of uplift is approximately 1.5×10^{-3} radians.

References

1. Amin, M. and Y. N. Chen, "Mean Ultimate Capacity of the Zion Station Containment Building," Sargent & Lundy, July 1980.

2. Kennedy, R. P., S. A. Short, D. A. Wesley, and T. H. Lee, "Effect on NonLinear Soil-Structure Interaction Due to Base Slab Uplift on the Seismic Response of a High Temperature Gas-Cooled Reactor (HTGR)," Nuclear Engineering and Design 38, (1976).

Seismic Fragility

9. What effect would consideration of a "best estimate" site specific ground response spectrum relative to the broad-banded spectrum used in the analysis have on the value of the factor, F (Section 7.9.3)? (pp. 43, 92, and 93)

Response

A "best estimate" site specific ground response spectrum is not available for Zion and the comparison of results with the broadbanded spectrum has not been made. Presumably, a site specific spectrum would have somewhat lower randomness since the broadbanded spectrum includes earthquakes from a wide range of site conditions and magnitudes. However, some additional uncertainty would be introduced by the use of a site specific spectrum which would tend to offset the reduced randomness. In the PSS, all the variability in the spectral shape parameter was attributed to randomness since the uncertainty associated with the spectrum was judged to be small. (The broad-banded spectrum was based on including a large majority of the applicable earthquake records available at the time it was developed.) It is judged the composite variability, β_C, would not be significantly changed. If the β_C associated with the spectral shape for a typical failure mode (for instance, the auxiliary building concrete shear walls, $\beta_C = 0.18$) were distributed equally between β_R and β_U, then $\beta_R = \beta_U \approx 0.13$. For this case, the total randomness, β_R, is reduced to 0.27, the total uncertainty, β_U, is increased to 0.31, and the total composite variability, β_C, remains unchanged at 0.41.

Seismic Fragility

10. How close do the electrical components, which were eliminated from Table 7.2-3, compare to the tested components that were used to develop the generic fragility data? (pp. 12, 26, and 78)

Response

The components tested in the Safeguards Program were typically from two or three manufacturers. It is not known if the Zion components were from any of the manufacturers that supplied safeguards equipment. In the safeguards program, a few components had lower than acceptable fragility and were not used. Since we did not know the similarity between those components tested in the safeguards program and those installed in Zion, we included the weak components from the safeguards program in the data base to develop generic fragilities.

Seismic Fragility

11. How do design and construction errors and aging affect the fragility curves and the subsequent systems analysis for the effects of seismic events? (pp. 10, 39, and 62)

Response

Design and construction errors were not treated explicitly in the Zion PSS. The possibility of design and construction errors was considered implicitly in many cases in establishing the uncertainty associated with a given failure mode, however.

In the case of primary coolant piping, the possibility of a large through wall flaw was considered as a lower bound on capacity and limit moment capacities of other piping were biased below the test data median to account for possible flaws. Unfortunately, essentially no data is available to quantify these effects for nuclear power plants. Although a number of discrepancies have been previously identified in nuclear power plants, the items identified to date have been modified as necessary or shown to have no safety implications. The code of Federal Regulations provides a strong incentive to continue the upgrade of discrepancies. Nevertheless, there is a possibility that design and construction errors which can affect the seismic capacity may exist.

It should be recognized that design and construction errors do not necessarily always result in a decrease in capacity. It is also possible to install higher strength bolts than specified, larger reinforcing bars or more closely spaced bars than required, or slip a decimal point in the conservative as well as in the unconservative direction of the analysis. However, the inspection and QA requirements for nuclear power plants are expected to produce fewer design and contruction errors than in typical civil and mechanical construction projects.

Some additional confidence exists in that structures and equipment are subjected to normal operating loads and static 1g vertical loads continually. In many cases, these loads are large, as for instance in the case of pressure, water hammer, and thermal loads in fluid systems, when compared to seismic loads. Low level dynamic loads resulting from cranes, forklifts, and other component handling equipment produce loads in structures which occur on a regular basis and which might serve to initiate some failures with very low capacities resulting from design and construction errors. Pressure tests of containment vessels, while producing different types of response than seismic, would likely provide an indication if significant construction errors exist in these structures. Finally, wind forces on structures produce lateral forces which may be at least comparable in magnitude to those developed by earthquakes, and at least for some wind velocities, occur on a much more frequent basis. Thus, although data on which to quantify accurate estimates of the effects of design and construction errors are not available, these are expected to be minimal, and are included to some degree in the uncertainties estimated for the Zion fragilities.

Aging effects were included quantitatively only in the strength of concrete. For pressure boundaries of equipment items such as pressure vessel nozzles or piping, the presence of possible flaws introduced by aging (thermal fatigue, intergranular stress corrosion cracking, etc.) was implicitly included for the primary coolant system by consideration of flawed weld joints as a lower bound on capacity. For other piping, limit moments derived from test data were biased on the low side of the median value to approximately account for flaws that may occur from aging.

Aging effects data on seismic resistance of electrical components are not available. The uncertainty bound on electrical equipment fragility tended to be greater than for mechanical components. The wider uncertainty was intended to address not only the generic treatment of electrical components but to some degree, address mild to moderate aging effects on electrical equipment fragility.

Most electrical equipment fabrication materials and procedures have been developed from field experience and environmental testing data to assure that insulation material will survive the equipment operating life and environment. Performance testing at least every 30 days assures that aging deterioration that would render a component inoperable is detected. While functional testing does not assure that aging has not deteriorated seismic resistance, it does identify aging problems in general and offers some assurance that gross aging degradation will not be present. Since the uncertainties bound is believed to account for mild to moderate aging degradation, the effect of aging on electrical equipment seismic fragility is believed to be implicitly included.

Seismic Fragility

12. Was the possibility of a LOCA followed by an aftershock, or the occurrence of a moderate earthquake when some safety related equipment is unavailable, considered in the analysis leading to the probability distribution of frequency of core melt? (pp. 13, 45, and 58)

Response

The frequency of a nonseismic induced LOCA followed by an earthquake would be too low to contribute or to be considered in the analysis. The frequency of a seismically induced LOCA or melt resulting from a transient, causing containment pressurization with time and without release, followed by an aftershock large enough to cause additional failures was not considered further also because of the low frequency of this meticulously timed scenario.

Seismic Fragility

13. Could the split of variability into randomness and uncertainty components be different than assumed in the analysis. If so, what would be the effect of a different split on the tails of the frequency of core melt density function? (pp. 43, 51, 52, 54, 55, 58, 59, 60, 65, 66, 67, and 72)

Response

Some difference in the split between randomness and uncertainty can be expected from different qualified individuals making the evaluations. In the Zion PSS, randomness was in general based on available data such as material strengths, earthquake characteristics, and comparisons of SRSS modal analysis results with absolute sum results. Uncertainty was based on estimates of our lack of knowledge. If most of the variability was judged to be either randomness or uncertainty, in many cases the total variability was lumped in either randomness or uncertainty. However, in all cases, the composite variability was judged to be a realistic value. Sensitivity analyses on results of somewhat different splits between randomness and uncertainty have not been conducted. However, the results are not expected to be significantly changed since the composite value would not be changed.

Seismic Fragility

14. In developing the values for the mode shape parameter for equipment, was the location of equipment relative to the location of the masses of the building model considered? If yes, how were they considered? (pp. 52, 62, 82, 84, and 89)

Response

Equipment response was considered to be uncoupled from structural response except in the case of the primary coolant system. The primary coolant system analysis conducted by Westinghouse included the containment structure in the model. Other components are light relative to the structure supporting them and uncoupling is justified.

Most equipment and piping under consideration are floor or wall mounted. The floors and walls are relatively stiff and in-structure response spectra for local modes are not expected to be significantly different from the spectra used in design.

Seismic Fragility

15. Why in Section 7.2.2 were maximum acceleration values assigned, while in the section on the hazard analysis (Section 7.9.1) maximum acceleration values were treated as being uncertain? (pp. 17 and 18)

Response

The upper bound of effective peak ground acceleration (EPA) was developed based on the existence of assumed upper bound on Modified Mercalli Intensity (I_{mm}) levels. Based on historic and geologic data, upper bounds for I_{mm} for the Zion site are considered realistic. However, fragility curves defining the frequencies of seismic induced failures for structures and equipment are much more easily incorporated in the overall risk assessment when the frequency of failure estimates are defined as a function of the EPA or similar mathematically quantifiable parameter. The EPA is the ground acceleration level at which a broad frequency content structural response spectrum should be anchored for the purposes of predicting structural damage. Site specific ground response spectra for the Zion site were not available for the PRA. For predicting structure and component damage at the Zion site, Structural Mechanics Associates, Inc. (SMA) has used the median broad frequency content structural response spectrum for alluvium sites defined in Reference 1. SMA has assumed 3 to 5 near-peak response excursions approaching the levels defined by this structural response spectrum anchored to the EPA.

This approach is most applicable when dealing with longer duration ground motions which contain a broad range of frequency contents such as the Taft recording from the 1952 Kern County, California, earthquake (local magnitude M_L = 7.2, range from causative fault = 40 km) or the highway test laboratory recording from the 1949 Olympia, Washington, earthquake (M_L = 7.0, range = 29 km). For such earthquakes, the EPA to which a broad frequency content response spectrum is anchored and the instrumental peak acceleration (IPA) should be essentially the same. Such records result in 3 to 5 structural response excursions approaching the levels defined by the structural response spectrum.

However, the frequency of strong ground motion at the Zion site will be mostly due to lower magnitude earthquakes ($M_L \leq 5.7$) and shorter ranges (less than 20 km). Ground motions from such earthquakes have characteristics like those recorded at the Gilroy Array from the 1979 Coyote Lake, California, earthquake (M_L = 5.7, range = 7 km) or the Melendy Ranch Barn record from the 1972 Bear Valley, California, earthquake (M_L = 4.7, range = 6 km). These records have narrow frequency content, and within the majority of the frequency range of interest (2 to 10 Hz) their structural response spectrum is seriously overpredicted by the Reference 1 broad frequency content median spectrum when this spectrum is anchored at the IPA (Instrumental Peak Ground Acceleration). Secondly, only a single cycle of strong structural response occurs from these records because of their limited duration and energy content. Thus, for these records the IPA cannot be used as a basis for predicting the level of structural response which is approached 3 to 5 times during an earthquake. Since structural damage predominantly depends upon multiple cycles of strong response, the IPA cannot serve as a good indicator of structural damage for these earthquakes. In these cases, damage is better described by an EPA which is much less than the IPA. This EPA represents the ground acceleration for an equivalent long duration record with broad frequency content which causes 3 to 5 response cycles and results in the same expected damage level as the actual record.

Effective Peak Versus Instrumental Peak and Sustained Peak Accelerations

SMA is currently engaged with Woodward-Clyde consultants in a research program sponsored by the Nuclear Regulatory Commission (NRC) to define effective ground motion parameters useful in predicting structural damage. This section briefly summarizes some of the tentative findings to date.

For the purpose of predicting elastic response of structures in the amplified acceleration frequency range (2 to 10 Hz), median broad frequency content response spectra such as those from Reference 1 are more accurately anchored to an EPA defined by:

$$A_E = 1.25 * A_{3F} \tag{1}$$

rather than the IPA. The quantity A_{3F} represents the thirdhighest acceleration peak of a filtered acceleration time-history record. The filter should be chosen to pass all frequency content below about 8 Hz and filter out all frequency content above 9 Hz. The quantity A_{3F} corresponds closely with what Nuttli (Reference 2) has defined as sustained peak acceleration.

Therefore, Equation (1), together with Nuttli's definition of sustained acceleration, can be used to define an EPA (A_E) to be used to estimate elastic response of a structure within the 2 to 10 Hz frequency range. However, elastic response is not a good measur of damage. Based upon current work, two ground motion timehistories with the same spectral acceleration values at the structure's natural frequency can lead to vastly different nonlinear response or damage for the same structure model. For instance, one can compare (Reference 3) the maximum nonlinear response (damage) from the Melendy Ranch Barn record (magnitude 4.7) with that computed from the Taft record (magnitude 7.2). It is found that the Melendy Ranch Barn record must be scaled to produce spectral accelerations between 1 and 2g's at the structural natural frequency to produce the same level of damage as a 0.5g spectral acceleration from the Taft record. Thus, for Melendy Ranch, the spectral acceleration must be 2 to 4 times as great as for Taft to produce the same level of structural damage. Similar conclusions are reached for the Coyote Lake records (magnitude 5.7) versus the Olympia record (magnitude 7.0) or Taft. Thus, for obtaining an EPA corresponding to a given level of structural damage, Equation (1) should be modified, as follows:

$$A_D = \frac{1.25}{F} * A_{3F} \tag{2}$$

The factor F must be established as a function of ground motion characteristics for a constant level of structural damage. For magnitudes greater than about 7.0 and ranges greater than about 40 km, F can be taken as unity and Equation (1) can be used to predict the EPA corresponding to structural damage. However, with magnitudes less than about 5.0 and ranges less than about 20 km, F should have a value greater than 2 for predicting structural damage. As a consequence, the EPA (A_D) should range from less than 0.6 to 1.25 times the sustained ground acceleration (A_{3F}) depending upon the earthquake magnitude and hypocentral range with the lower factor being appropriate to low magnitudes (less than about 5.0) and short hypocentral ranges (less than 20 km). The appropriate ratio of A_D/A_{3F} is strongly influenced by the duration of strong ground motion.

The SMA fragility curves for Zion are best anchored to the damage EPA defined by Equation (2). Because of the generally low magnitude of earthquakes which might result in strong motion at the Zion site, it is judged that the 90 percent confidence bounds on F are:

$$1.0 \leq F \leq 3.0 \tag{3}$$

Because of the tentative nature of the research conducted to date and the controversy of the subject of EPA versus IPA, it is recommended that F be conservatively selected for use in Equation (2) and that the EPA be defined by:

$$A_D \approx A_{3F} \tag{4}$$

with Nuttli's sustained peak acceleration being used to define A_{3F}.

Upper Bound Cutoff on Effective Peak Acceleration

The EPA is being used as a measure of damage to structures with a fundamental natural frequency in the 2 to 10 Hz frequency range. The I_{mm} is also a measure of damage. Although I_{mm} is a subjective scale, it probably correlates best to damage of conventional structures which generally have natural frequencies in the 0.3 to 3 Hz range. Because both EPA and I_{mm} are measures of damage capability of ground motion, these two quantities should be closely correlated with each other. Thus, one should be able to establish upper bounds on the EPA irrespective of frequency of exceedance if upper bounds exist on intensity.

Table 1 describes the earthquake effects (damage) corresponding to each of the I_{mm} scale levels. These damage descriptions can be used to define upper bounds on the EPA corresponding to a given I_{mm} level. Masonry A construction corresponds to earthquake resistant masonry structures designed to the Uniform Building Code (UBC) in California (Zone 4). Masonry B contruction is reinforced and represents well-engineered masonry structures in UBC Zones 0 or 1. Masonry C contruction represents well-constructed unreinforced masonry structures. The SMA methodology used to develop the fragility curves for structures and components at Zion will predict very substantial damage and/or at least partial collapse of 50 percent of these masonry structures for 3 to 5 cycles of the following EPA levels:

Masonry Type	50% Damage EPA Levels (g's)
C	0.25 - 0.3
B	0.4 - 0.5
A	0.6 - 0.8

Thus, very serious damage to a large number of Masonry A, B, and C structures would be predicted by the SMA methodology to correspond to EPA levels of less than 0.8, 0.5, and 0.3g's, respectively. The SMA methodology for predicting damage levels has been benchmarked against observed damage in past earthquakes in which substantial damage was observed for sustained ground motions corresponding to these levels. Based upon the damage descriptions in Table 1, serious damage to at least some Masonry A, B, C construction correspond to I_{mm} levels X, IX, and VIII, respectively. Comparing the EPA levels defined above for each of these levels of damage, one would estimate that I_{mm} of X would correspond to an EPA of 0.6 to 0.8g's or less, I_{mm} of IX corresponds to an EPA of 0.4 to 0.5g's or less, and I_{mm} of VIII corresponds to an EPA of 0.25 to

0.3g's or less based upon the described damage to masonry construction. Even considering uncertainty in the correlation between the two descriptors of damage (I_{mm} and EPA), an upper bound must exist on EPA for a given I_{mm} level. Otherwise, the two indicators of damage would be contradictory. Therefore, an upper bound on EPA can be estimated by assigning the EPA ground motion levels defined above to an intensity value one level lower than that for which a given type of masonry construction damage is considered appropriate. Thus:

Intensity, I_{mm}	Upper Bound EPA (g's)
IX	0.8
VIII	0.5
VII	0.3
VI	0.2

The EPA values given in this table are judged to represent conservative upper bounds for the corresponding intensity levels. These EPA levels would result in the prediction of substantially more damage than that from which the intensity level is defined.

If upper bound intensity levels are defined for the Zion site, then the EPA levels should also be limited to being below the upper bound levels defined above when the SMA fragility curves are used to predict structure and component damage. Unless these limits to EPA are applied, one would predict substantially more damage than could possibly correspond to a given upper bound intensity level. Even with these limits, it is judged that the level of predicted damage would correspond to at least one intensity level higher than the upper bound intensity level.

References

1. "A Study of Vertical and Horizontal Earthquake Spectra," WASH-1255, Nathan M. Newmark Consulting Engineering Services, prepared for USAEC, April 1973.

2. Nuttli, O. W., "The Relation of Sustained Maximum Ground Acceleration and Velocity to Earthquake Intensity and Magnitude," Paper S-73-1 published in State-of-the-Art for Assessing Earthquake Hazards in the United States, U.S. Army Engineer Waterways Experiment Station, Vicksburg, Mississippi, 1979.

3. Kennedy, R. P., S. A. Short, N. M. Newmark, "The Response of A Nuclear Power Plant to Near-Field Moderate Magnitude Earthquakes", Paper K 8/1, Sixth International Conference on Structural Mechanics in Reactor Technology, Paris, France, August 17-21, 1981.

Seismic Fragility

16. What is the basis for the median capacity of 1.16g for the reactor pressure vessel internals? (p. 67)

Response

The median acceleration capacity is derived from the product of the equipment capacity factor F_C, the equipment response factor, F_{ER}, and the structural response factor, F_{SR}. The derivation of F_C is described in Section 7.9.2 of the report, pages 5-14 and 5-15, and is 1.66. F_C is based on the ratio of the collapse moment of the CR Guide Tube of 1.86 My (determined from experimental data) and the response computed for a Housner spectrum anchored to 0.5g and equal to 1.12 My. Note that the ratio of the 0.5g Housner spectrum to the Zion ZPA of 0.17g is accounted for in the reponse factor.

The response factor accounts for:

a. Ratio of Housner spectrum anchored to 0.5g to the Zion ZPA of 0.17g. F = 2.94 (note that 0.18g was used in the original calculation resulting in a conservative bias of 5.8 percent).

b. Response for design damping of 2 percent versus median damping of 5 percent:

$$F_D = 1.22$$

c. Mode shape frequency and mode combination factors of 1.0 each.

d. A combination of earthquake components factor of 0.93 to account for slight unconservatism in the design basis earthquake component combination versus a median centered combination.

The resulting response factor is 3.33 (3.15 originally calculated).

The structural response factor of 1.3 is listed in Table 5-3 of the report.

Multiplying the three factors times the DBE peak ground acceleration of 0.17g results in a median ground acceleration capacity of 1.22 (1.16 originally calculated).

Seismic Fragility

17. What is the basis for the fragility parameter values for the control rod drive mechanisms? (p. 68)

Response

The control rod drive mechanism capacity is stated in Table 5-6 to be 3.33g and is the product of the capacity factor of 5.50, the equipment response factor of 2.74, and the structural response factor of 1.30. The strength factor of 5.50 is developed on page 5-20 of the report and is based upon the ratio of the CRD housing strength to the response to a generic design spectrum. The difference between the generic design spectrum and the Zion DBE 5 percent damped in-structure response spectrum is accounted for in the equipment response factor.

The equipment response factor of 2.74 is the product of the following factors:

a. Qualification Method - At the fundamental frequency of 5.8 Hz, the ratio of the generic design spectral acceleration to the 5 percent damped spectral acceleration for the Zion DBE at the RPV support is 1.76.

b. Spectral Shape Factor - The Zion DBE in-structure spectra were described to be a factor of 1.67 conservative relative to the spectra that would have resulted from the specified ground spectrum.

c. Damping Factor - In this case, the damping factor is 1.0 since a 5 percent median damped spectrum was used to develop the qualification method factor.

d. Mode Shape, Frequency, and Mode Combination - These factors are all unity.

ε. Earthquake Component Combination Factor - The appropriate factor is 0.93 for vertical components that respond predominantly in the horizontal direction. This results from comparing the design basis earthquake component combination criteria to estimated median centered response.

The structural response factor of 1.3 is listed in Table 5-3 of the report.

Multiplying the three factors times the DBE peak ground acceleration of 0.17g results in a median ground acceleration capacity of 3.33g.

Appendix B

Review of the Zion Probabilistic Safety Study
Seismic External Events

Prof. Ronald L. Street
University of Kentucky

Consulting Report to
Sandia National Laboratories

DRAFT

Evaluation of Overall Methodology

The overall methodology used in considering the seismic hazard at the Zion Nuclear Power Plant is, in my opinion, an appropriate approach and one in which I am in total agreement with. I disagree, however, with that portion of the study dealing with the seismogenic zones, the maximum historical earthquake, and the rate of activity.

In particular, I do not feel that the proposed Wisconsin Arch or Wisconsin Arch-Michigan Basin seismogenic zones can be justified on the basis of the known seismicity on the deep-seated geological structures. I also believe that the maximum historical earthquake in the area could have been a 5-1/2 m_{bLg} magnitude event, and that the epicenter of that event was appreciably closer to the power plant site than that which was apparently used in the study.

Detailed Review Of Section 7.2.2

"Seismicity"

a. In general it is my experience that relationships such as

$$m_b = 0.5 (I_o + 3.5)$$

should be avoided when doing site-specific studies because of the weak correlation between m_b and I_o as compared to m_b and other types of intensity data; i.e., m_b and the falloff-of-intensity with distance, m_b and the area within the intensity IV isoseism, and m_b and the total felt area associated with the event.

b. As discussed in the Detailed Review of Section 7.9.1, the maximum historical earthquake could have been as much as a 5.6 event, and I would feel more comfortable if the predicted maximum m_b values was chosen as 6.0 rather than 5.8.

c. I am in agreement with the appropriateness of the relationships given in this section for the estimation of a_s, the sustained level of acceleration.

d. I disagree with the portion of this section which utilizes the Wisconsin Arch and Wisconsin Arch-Michigan Basin seismogenic zones proposed in Section 7.9.1.

Detailed Review OF Section 7.9.1

"Seismic Ground Motion Hazard at Zion Nuclear Power Plant Site:

A. Seismogenic Zones

I disagree with this section of the report because of the suggested "Wisconsin Arch" and "Wisconsin Arch-Michigan Basin" seismogenic zones. In my opinion, seismogenic zones in central United States should be based on the observed distribution of seismicity or the deep-seated structures involving the crystalline basement within which most of the earthquakes in central United States occur.

Figure 7.9.1-1 is a plot of the seismicity listed by Nuttli (1979) for the area bounded by the latitudes of 41° and 45°N and the longitudes of 84° and 92°W. The earthquakes in the figure are plotted to scale in accordance to their epicentral intensities. Two earthquakes not shown in the figure, but which appear in Nuttli's (1979) catalog, are the events of February 9, 1899 and May 19, 1906. The first event is listed in Barstow et al. (1981) as not being an earthquake, while the second event was determined to be 800 kegs of blasting powder exploding at Pleasant Praire, Wisconsin.

Another difference between the seismicity plotted in Figure 7.9.1-1 and that listed by Nuttli (1979), is the epicentral location of the May 26, 1909 event. After reviewing the distribution of the earthquake effects, I feel that the epicenter near Aurora, Illinois as suggested by Docekal (1970) is more appropriate than 42.5°N/89.0°W used by Nuttli (1979) and Coffman and von Hake (1973). The results of my review of this event are discussed in detail in Part B of this section.

Illustrated along with the seismicity in Figure 7.9.1-1 is the outline of the proposed Wisconsin Arch seismogenic zone. The proposed zone does a poor job of accounting for the known seismicit in the area and, in my opinion, does not seem to be justified.

Figure 7.9.1-2 illustrates the same seismicity shown in Figure 7.9.1-1, but with an outline of the proposed Wisconsin Arch-Michigan Basin seismogenic zone. I have two problems with this zone. Firstly, as with the Wisconsin Arch zone, the proposed Wisconsin Arch-Michigan Basin zone does not correlate well with the known seismicity. And secondly, based on the gravity data of the region about northeastern Illinois and southern Michigan, the proposed seismogenic zone cuts across major basement structure and suggests to me that such a zone is unlikely.

As an alternative to the proposed Wisconsin Arch and Wisconsin Arch-Michigan Basin seismogenic zones, I would suggest a zone more on the order of the one outlined in Figure 7.9.1-3. This zone is similar to the Northern Illinois zone proposed by Nuttli and Herrmann (1978), but unlike their zone, the outlined zone in the figure has been extended northwards to include the seismic activity that seems to be spatially associated with the southern border of the Wisconsin Dome. In addition, the outlined zone correlates somewhat with the proposed Wisconsin Arch zone except that by considering a larger area -- particularly in the southerly direction -- all of the significant seismic activity has been indicated.

B. Seismic Parameters

1. Seismic Activity Rate

The rate of the seismic activity depends upon the choice of the boundaries of the seismogenic zones. And since I disagree with the seismogenic zones proposed in the previous section of the study, I also arrive at a different cumulative magnitude-recurrence curve.

Using the earthquakes in Table 7.9.1-1, which represent the seismic activity that occurred within the seismic zone shown in Figure 7.9.1-3 during the 95 year period of (1880-1975), and the method of plotting the observed cumulative rates of activity at the lower end of 0.5 unit magnitude intervals for those events of $m_{bLg} \geq 4.0$, I get a cumulative magnitude-recurrence curve very similar to that determined by Nuttli and Herrman (1978) for their proposed Northern Illinois source area. The open circles superimposed on Figure 7.9.1-4 (taken from Nuttli and Herrman, 1978), indicate the data points I determined.

2. Maximum Magnitude

In this part of Section 7.9.1 it is stated that the maximum historical earthquake to have occurred in the area had an estimated m_b magnitude of 5.3. The event apparently being referred to is the May 26, 1909 earthquake that Nuttli (1979) lists as a 5.3 event.

Figure 7.9.1-5 illustrates the distribution intensity data for the May 26, 1909 event based on a review of the newspaper articles in my files for this event. Superimposed on the figure is my interpretation of where the various isoseisms should be drawn. The isoseisms, along the northern portion of the map, are dashed to indicate the uncertainty resulting from the lack of information. Note that the greatest level of concentration occurred in and about the Aurora, Illinois area, and it is for this reason that I choose to use Docekal's (1970) epicentral coordinates rather than those of Nuttli (1979) and Coffman and von Hake (1973).

Given the distribution of the MM intensity data for an event, there are a number of empirical techniques that have been developed for the purpose of estimating m_b magnitudes for earthquakes in eastern North America. Using Nuttli's (1973) falloff-of-intensity with distance technique, I estimate that the m_b magnitude of the event to be 5.6. Using the area within the intensity IV isoseism (184,000 km^2) and the results of Nuttli et al. (1979), the m_b magnitude of the event is determined to be 5.5 $\pm$ 0.23 and using the 800,000 km^2 felt area listed by Nuttli (1979) and the results of Street and Lacroix (1979), the m_b magnitude of the event is determined to be 5.4 $\pm$ 0.30.

On the other hand, I estimate that the felt area to have been more on the order of the 445,000 km^2 given by Docekal (1970), which by Street and Lacrois (1979) is equivalent to a m_b of 5.1 $\pm$ 0.30.

In summary, I don't disagree with the 5.3 maximum historical earthquake, but there is a distinct possibility the m_b magnitude of the May 26, 1909 was as large as 5-1/2. And as a consequence, I would suggest that the best estimate of $m_{b,max}$ should be raised from 5.8 to 6.0.

C. Estimation of Seismic Ground Motion

The approach used in this section of the study to estimate peak acceleration as a function of earthquake magnitude and distance seems to be appropriate. The conclusions in this section, however, are dependent on the acceptance of Sections A and B with which I disagree.

Conclusions

The methodology used in that portion of this study that I reviewed, is both adequate and appropriate given the present level of knowledge of seismicity in Central United States. The difficulty that I have with the seismogenic zones, the rate of activity, and maximum historical earthquake's m_b magnitude and location, however, does have a bearing on the final results.

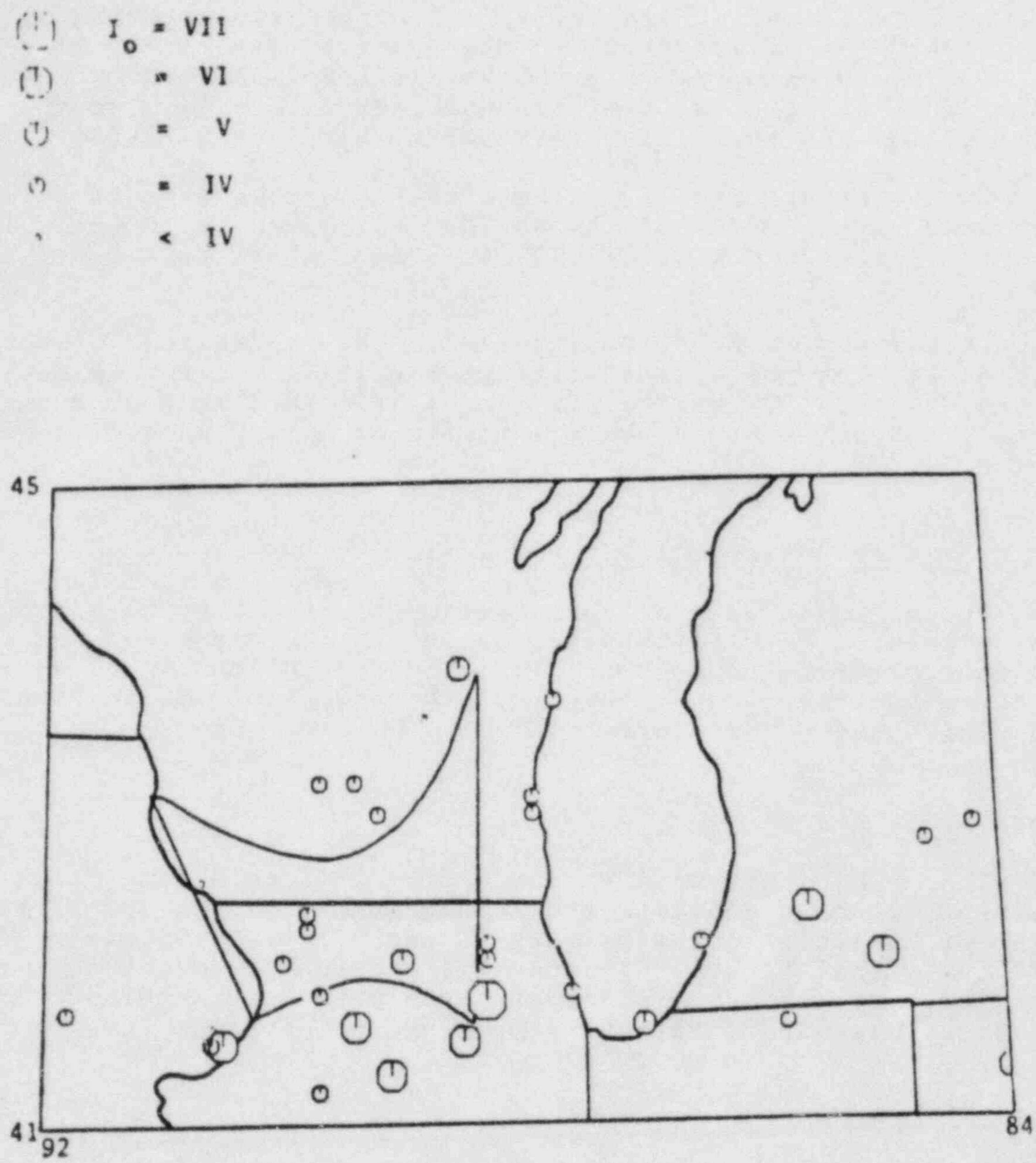


FIGURE 7.9.1-1

I_o = VII
= VI
= V
= IV
< IV

Figure 7.9.1-2

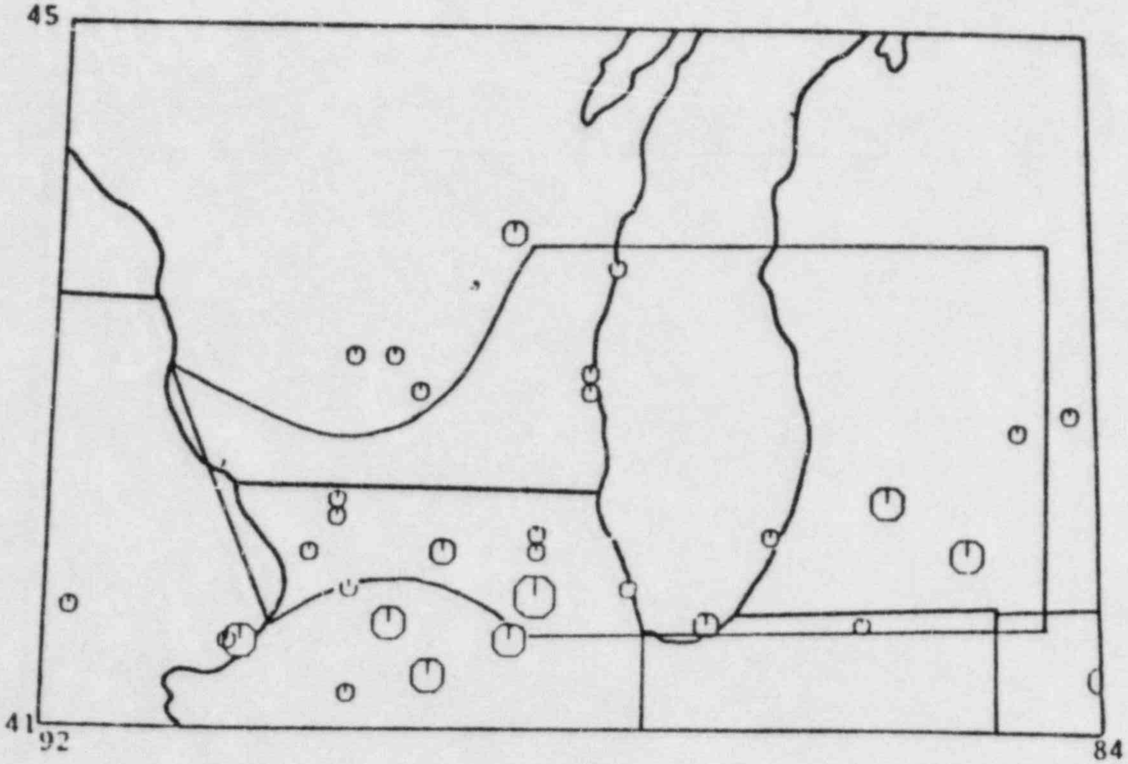


Figure 7.9.1-3

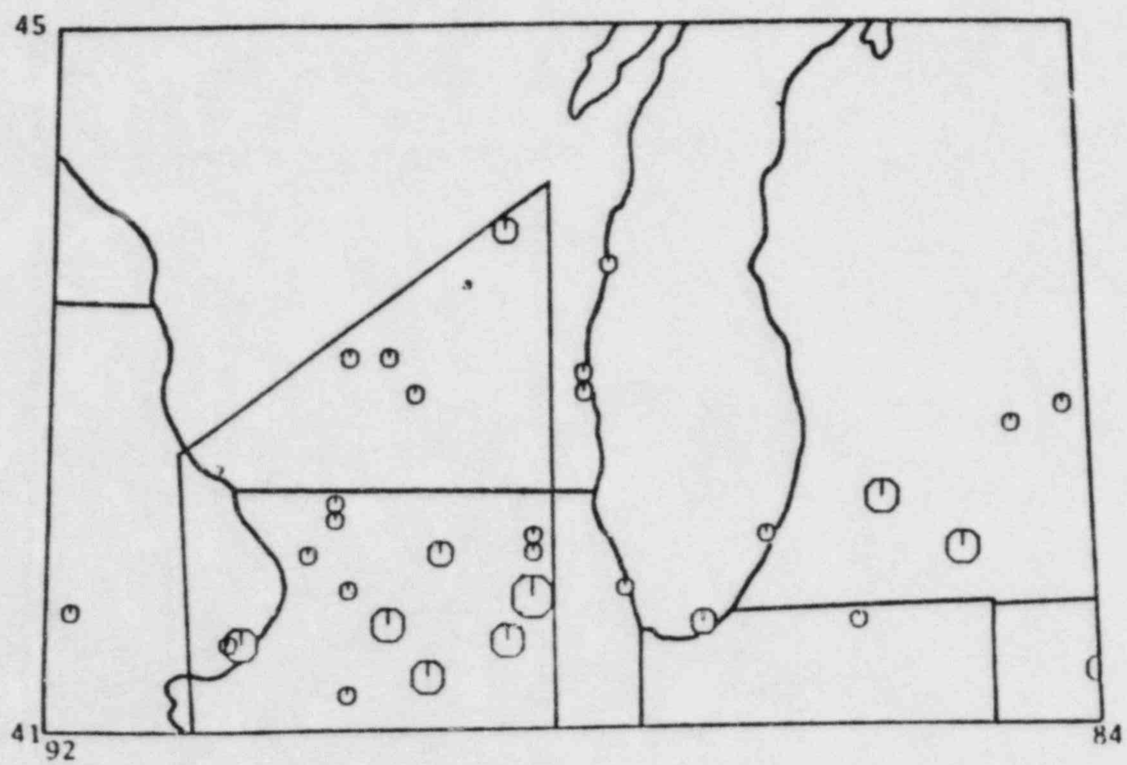
I_o = VII

= VI

= V

= IV

< IV



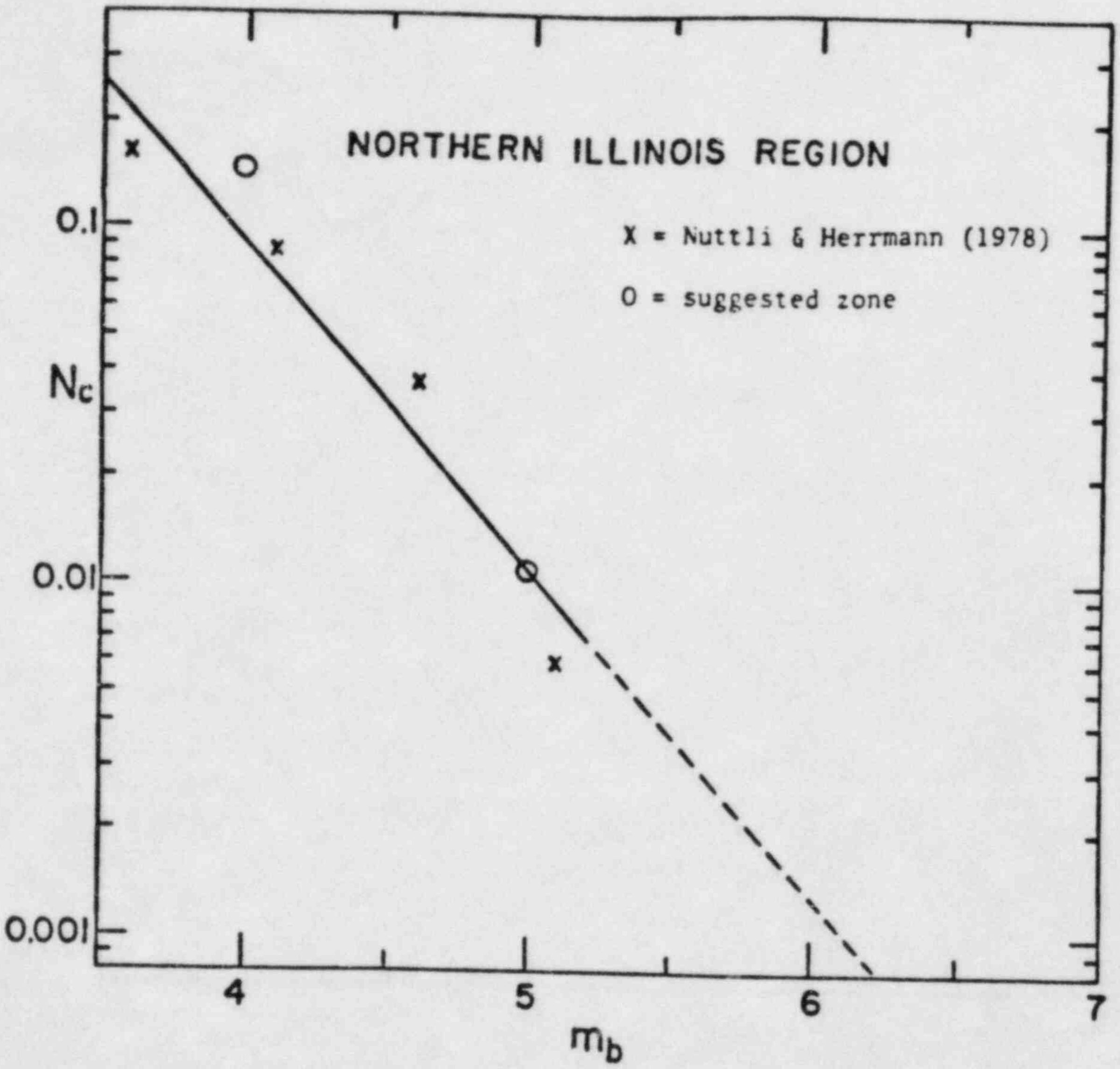


Figure 7. Cumulative magnitude-recurrence curve for the Northern Illinois region

(taken from Nuttli and Herrmann, 1978)

Figure 7.9.1-4

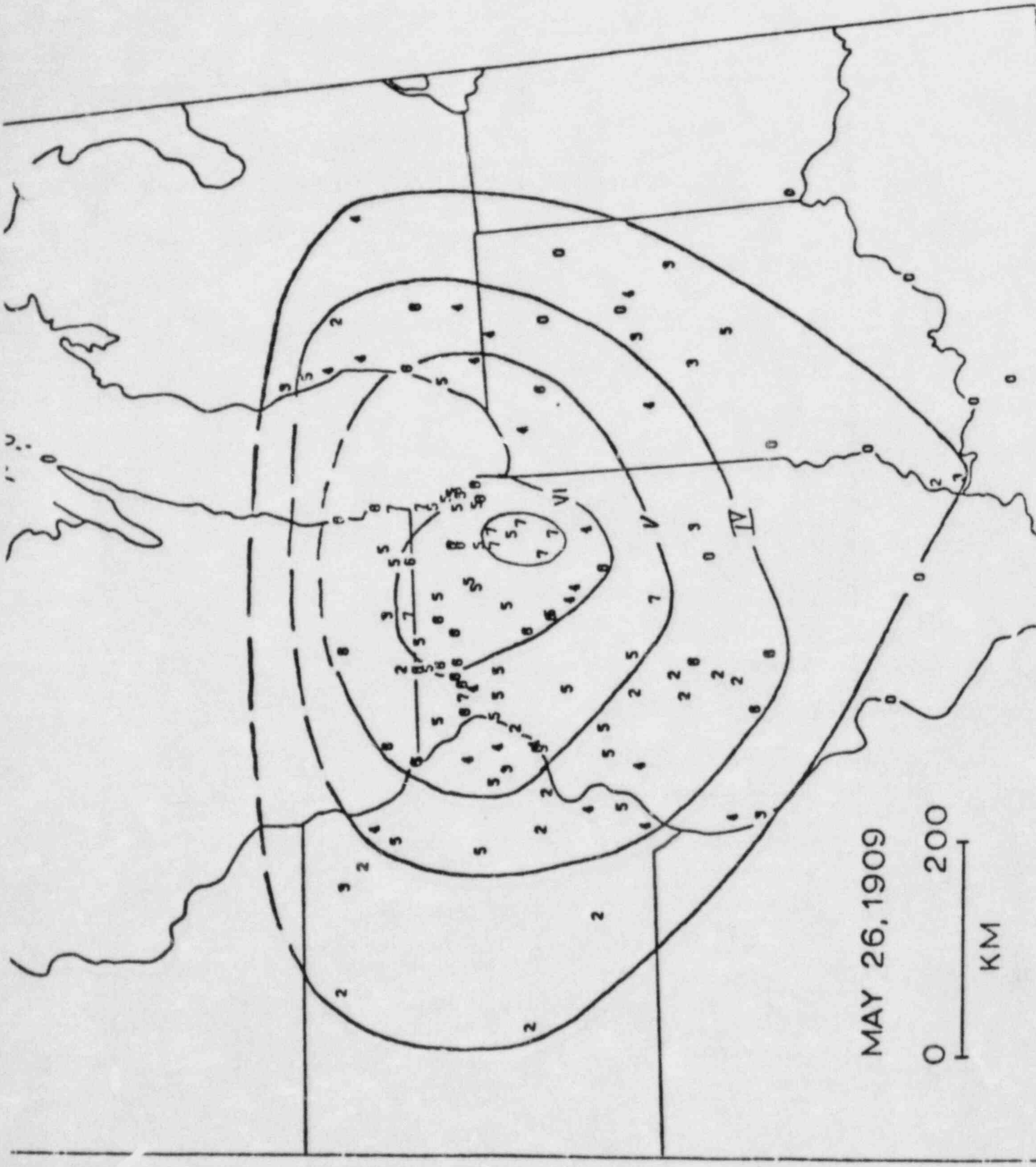


Figure [illegible].9.1-5

Table 7.9.1-1

Date Day-Mo.	Year	Magnitude m_{bLg}
27-05	1881	4.7
28-11	1907	3.8
28-11	1908	3.8
26-05	1909	5.3
22-10	1909	4.0
02-01	1912	4.7
25-09	1912	3.6
17-10	1913	3.6
07-10	1914	3.8
31-05	1916	3.0
22-02	1918	3.8
07-07	1922	4.2
03-03	1925	3.2
23-01	1928	3.8
10-06	1931	4.2
18-10	1931	3.4
07-12	1933	4.2
12-11	1934	4.7
05-01	1935	4.2
05-01	1935	3.4
12-02	1938	4.2
08-11	1938	3.0
08-11	1938	3.0
08-11	1938	3.0
24-11	1939	3.2
01-03	1942	4.0
16-03	1944	3.4
16-03	1947	3.6
06-05	1947	4.0
15-01	1948	3.9
20-04	1948	3.8
08-01	1957	3.6
15-09	1972	4.4

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Appendix C

Review of Zion Probabilistic Safety Study
Seismic External Events

Prof. Daniele Veneziano
Massachusetts Institute of Technology

Consulting Report to
Sandia National Laboratories

DRAFT

Evaluation of Overall Methodology

One should distinguish between the general approach as described in Section 0 of the Zion safety report and the specific methods used to implement such an approach. I basically agree with the former, which reflects state-of-the-art modeling of nuclear power plant safety, but I have a few reservations and questions about the implementation. These reservations and questions refer to specific steps of the analysis and will be expressed later in Section 2.

With regard to the general methodology, my only concern is the distinction between frequentist and nonfrequentist uncertainty, i.e., between probabilities with relative-frequency and with degree-of-belief interpretation. As a reference example, consider tossing a dice which may or may not be fair. Uncertainty about the outcome of a generic tossing is contributed (1) by uncertainty on the generic outcome given that the dice is fair or that it is loaded, and (2) by uncertainty as to whether the dice is fair or not. One way to classify these uncertainties as frequentist or nonfrequentist is to answer the question: Can uncertainty be reduced, e.g., by way of mechanical testing or statistical sampling? If it cannot, then uncertainty is all of the frequentist type. In the previous example, uncertainty as to whether the dice is fair or loaded can be eliminated by tossing the dice a large number of times and is therefore nonfrequentist. On the contrary, uncertainty on the generic outcome of a fair or of a loaded dice cannot be reduced and is therefore to be regarded as frequentist.

Alternatively, one can distinguish frequentist from nonfrequentist uncertainty by asking: Is uncertainty due to diversity within a statistical population of "objects" (of outcomes from dice tossing) or is it the result of ignorance about a specific object (a specific dice)? In the former case, which is verbally characterized by the adjective "generic," uncertainty is frequentist; in the latter case, for which the qualification "specific" is appropriate, uncertainty is nonfrequentist.

In the evaluation of risk from a nuclear power plant, one can separate frequentist from nonfrequentist uncertainty and express the results in the so-called probability-of-frequency format. However, I do not agree with the interpretation of uncertainty in the Zion study: one can regard as frequentist the uncertainty on the generic accelerogram given effective peak acceleration (EPA), whereas uncertainty on structural behavior and on the resistance on specific components to a given ground motion is nonfrequentist. In fact, the latter uncertainty can be reduced by inspecting and testing (e.g., proof loading) each component and can ultimately be eliminated by performing an ideal full-scale dynamic test of the entire facility.

Classification of resistance uncertainty as nonfrequentist implies an increase in the standard deviations β_U and a decrease in the standard deviations β_R in Table II.7.1, in such a way that $(\beta_U^2 + \beta_R^2)$ remains constant. The associated change in the fragility curves of a typical component is qualitatively shown in Figure 1: each curve becomes steeper and the dispersion of the family of curves increases, while the mean value curve remains the same. The frequency curves of the damage indices in Figures II.2.1 through II.2.5 undergo a similar transformation; in particular, their spacing increases, with higher 0.9 fractile curves and lower 0.5 and 0.1 curves. Dispersion of these curves is not expected to increase dramatically if radioactive release depends on failure of components for which the Zion study β_R is already smaller than β_U. Notice that the composite (mean) risk curves depend only on total uncertainty; therefore, the way in which total uncertainty is split has practical relevance only for decisions that depend on risk distribution characteristics other than the mean value. Because the mean risk curves do not depend on the interpretation of uncertainty and carry some weight in the final evaluation of safety, I suggest that they be added to Figures II.2-1 through II.2-5.

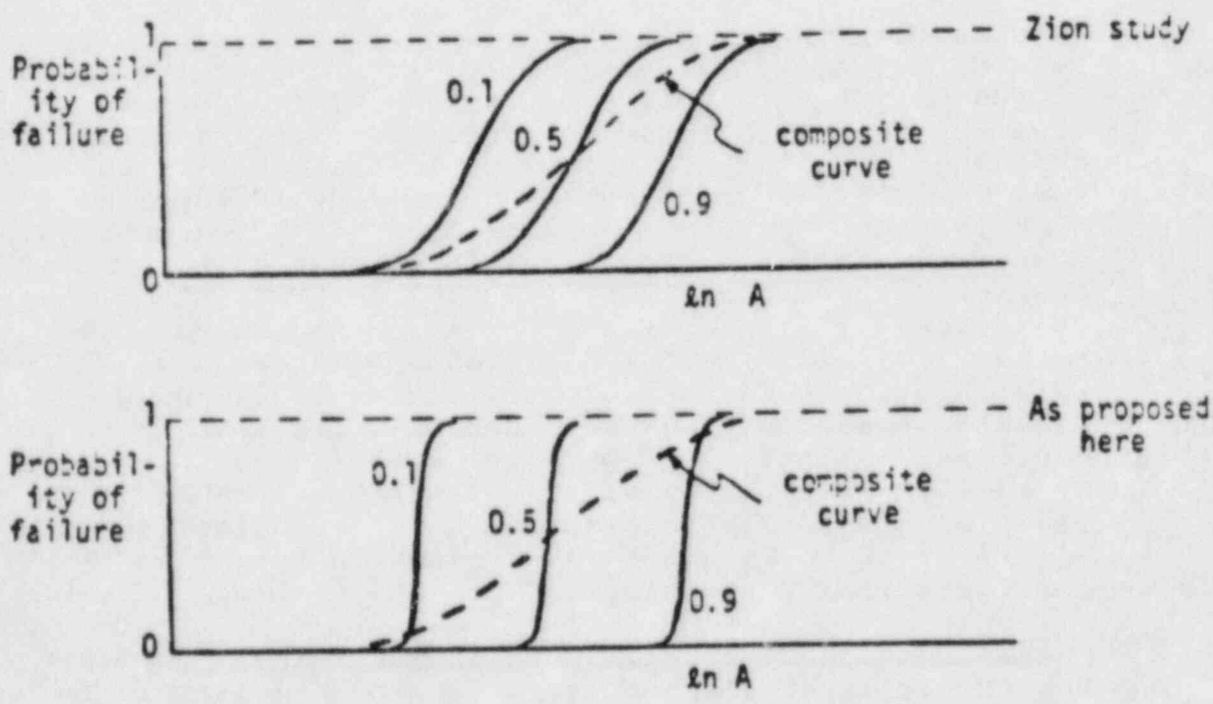

Figure 1. **Family of Fragility Curves for a Typical Component. Effect of changing the interpretation of resistance uncertainty.**

Detailed Review

The comments that follow refer separately to Sections II.7.1, 7.2.2, 7.9.1, 7.9.3, and 8.8.1 of the Zion Probabilistic Safety Study. Aspects of the analysis that affect several parts of the study are discussed with reference to the more relevant subsection. Specifically, upper bound acceleration is mentioned in Sections II.7.1, 7.2.2, 7.9.1, and 7.9.3, and selection of the appropriate site intensity measure (instrumental, sustained, or effective peak acceleration) is relevant to Sections 7.2.2, 7.9.1, and 7.9.3. Both issues are addressed here under "Section 7.9.3." A summary of the main findings will be given in Section 3.

Section II.7.1

Subsection II.7.1.1

- On the subject of upper bound to the effective peak acceleration (EPA), see "Section 7.9.3."

Subsection II.7.1.2

- I agree that the fragility curves reflect variability both of structural material properties and of the ground motion, given EPA. It should be recognized, however, that such material properties are sometimes dependent and that the ground motion is actually the same for all the components. Dependence is especially high for components of the same type and in general for components that are sensitive to the same gound motion characteristics (duration, spectral content, etc.). The existence of positive correlation is acknowledged in Section 0.13.8, but I have found no subsequent reference to it in the sections I have reviewed in detail. Dependence between component resistances decreases the safety of parallel systems and increases by smaller amounts the safety of series systems. For example, the Boolean expression for plant state SE on page II.7-18 corresponds to an essentially series configuration (V symbols).

- The evaluation of fragility is in many cases judgmental. For the critical components, it would be desirable to validate judgment through a few nonlinear dynamic analyses using representative historic ground (floor) motions.

Subsection II.7.1.2.1

- The assumption of "double lognormal" distribution for seismic resistances is difficult to validate. It would therefore be useful to demonstrate robustness of the results with respect to the assumed distribution shape. Without the support of analysis, it is difficult to accept statements such as (page 7.2-4) "such fragility curves will contain a great deal of uncertainty and, therefore, great precision in attempting to define the shape of these curves is unwarranted."

- Comments in Section 1 on the interpretation of resistance uncertainty are pertinent to estimation of β_R and β_U in Table II.7-1. The changes suggested in Section 1 would increase the dispersion among the fragility curves for each component while making each curve steeper.

- The criteria for excluding from Table II.7-1 some of the components and failure modes in Table 7.2-1 should be made more transparent. For example, why is "Foundation Slab Soil Failure" not retained?

Subsection II.7.1.3

- Figure II.7-4. There is inadequate explanation of how the core melt fragility curves in the figure were obtained. In theory, there should be one such fragility curve for each combination of mean component resistances (parameter a for each component) and seismic hazard curve in Figure II.7-1. All these (ideally infinite) curves should then be grouped (e.g., into 5 curves as in the case in Figure II.7-4). No such work is documented or mentioned in any of the sections I reviewed. In any case, the grouping of Figure II.7-4 is too coarse, especially at the low-resistance end.

Subsection II.7.1.4

- Figure II.7-5. Again, intermediate steps of calculation are not shown. In each of the 45 combinations of the 9 seismic hazard functions in Figure II.7-1 and the 5 fragility curves in Figure II.7-4, there is one value of the annual frequency of core melt. It would be helpful to show the histogram of these 45 values (of more values if Figure II.7-4 is revised to include more refined grouping) and compare this histogram with the smooth fits of Figure II.7-5.

Subsection II.7.1.5

- Table II.7-2. The discretization of acceleration levels is coarse and may lead to non-negligible errors in the calculation of the frequency of core melt. More refined convolution should be made of the rate density of EPA with the conditional core melt probability, for low to moderate acceleration values. It is unnecessary to extend calculations beyond 0.65 g since, according to Figure II.7-1, accelerations of this magnitude cannot occur (see comments in Section B on this topic).

Section 7.2.2

- For questions related to the selection of earthquake intensity and to the EPA upper bound, see "Section 7.9.3." I find some of the language on page 7.2-2 rather confusing, e.g., the qualification of the factor 1.23 as "a variable that depends on earthquake magnitudes."

Section 7.9.1

- In their study, Dames and Moore used acceleration upper bounds different from those in Figure II.7-1. For comments on the issue of upper bound and on the definition of effective peak acceleration, see "Section 7.9.3."

- No alternative assumption was finally made on the coefficients of the attenuation law and on the attenuation error variance. However, I believe that results would not be sensitive to reasonable variations of these parameters.

- Except for these points, modeling assumptions and results of the seismic hazard analysis in Section 7.9.1 seem to be appropriate.

Section 7.9.3

Definition of EPA. From Section 7.9.3, I understand the procedure of the Zion study as follows:

1. The damage of systems in the frequency range from 2 to 10 Hz is best correlated with spectral ordinates obtained by anchoring the response spectrum of strong, broad-band motions to an "effective peak acceleration" EPA, defined as (Equation 2 in Section 7.9.3):

$$EPA = \frac{1.25}{F} A_{3F} \quad (1)$$

in which A_{3F} is the acceleration at the top of page 3 of Section 7.9.3 and F is a quantity that depends on magnitude and distance.

2. A_{3F} in Equation (1) can be satisfactorily replaced with sustained peak acceleration (SPA), as defined by Nuttli (third highest acceleration peak). Similarity of the factor 1.25 in Equation (1) with the factor 1.23 in Equation (3) of Section 7.9.1 makes me believe that this replacement is in terms of SPA for a generic horizontal direction, not for the worst of two orthogonal directions.

3. For the Zion site, a conservative single-value estimate of F is taken to be F = 1.25. Therefore, a conservative definition of EPA for the Zion site is (Equation 4 of Section 7.9.3)

$$EPA = SPA \quad (2)$$

4. Seismic hazard is calculated in terms of EPA = SPA using Nuttli's median attenuation function in Equation (2) of Section 7.9.1.

5. Fragility curves in terms of EPA are obtained by assuming 3 to 5 cycles of linear response near the value of the response spectrum for long, broad-band earthquakes with peak acceleration EPA (i.e., with the spectrum anchored to the EPA); see first paragraph of Section 7.9.3.

I have some difficulty in reconciling this seismic hazard procedure with other statements in this report. Specifically,

1. If both hazard and fragility curves are in terms of Nuttli's SPA, then I see no reason why one should relate SPA to instrumental peak acceleration (IPA); i.e., the comments in the last part of page 9 in Section 7.9.1, starting from "To estimate peak acceleration" are irrelevant.

2. If, as stated at the top of page 10 in Section 7.9.1, Equation (2) in that section refers to the larger SPA for the two horizontal components of motion, then Equation (2) should be corrected by multiplying the right-hand sides by 0.9. I find no evidence that this was actually done.

3. The comment in the second paragraph of page 14 in Section 7.9.1, that the smaller damage potential of low-magnitude short-duration events is accounted for by limiting peak acceleration, is in contrast with my understanding from Section 7.9.3 that the difference in earthquake damage potential is the reason for replacing IPA with EPA, not for constraining the values of acceleration.

I should mention two more points related to the definition of EPA:

4. I agree with searching for quantities such as EPA that correlate with structural damage better than IPA. However, one must then face the problem of having to work with two different earthquake intensity measures, one for acceleration-sensitive equipment, the other for EPA-sensitive structures. If two such categories of components exist (and I believe they do), then seismic hazard should be defined jointly in terms of IPA and EPA. I should like to see some comments on this point in the report.

5. The report should show how seismic fragilities have indeed been obtained in terms of EPA and not, for example, in terms of IPA. How was the fact that 3 to 5 peaks occur near the maximum response value taken into consideration?

<u>Upper Bound EPA</u>. Treatment of the EPA upper bound is unsatisfactory on several grounds:

1. I cannot follow the argument that imposes limits to the EPA based on limits on I_{mm}. That argument is especially tenuous if EPA is defined as EPA = A_{3F} (Equation 4 of Section 7.9.3). In this case, the statement at the top of page 12 in Section 7.9.1 does not hold. I also find it objectionable to use damage to masonry construction in order to obtain limits on EPA for non-masonry structures.

2. If an upper bound to EPA exists, such bound should be included through truncation of the attenuation error distribution, not through correction of the final hazard curves.

3. There is no evidence in the report that the curves of Figure II.7.1 have actually been calculated. Rather, I have the impression that the transition between the untruncated curves and the vertical asymptotes have been obtained by direct graphical procedures.

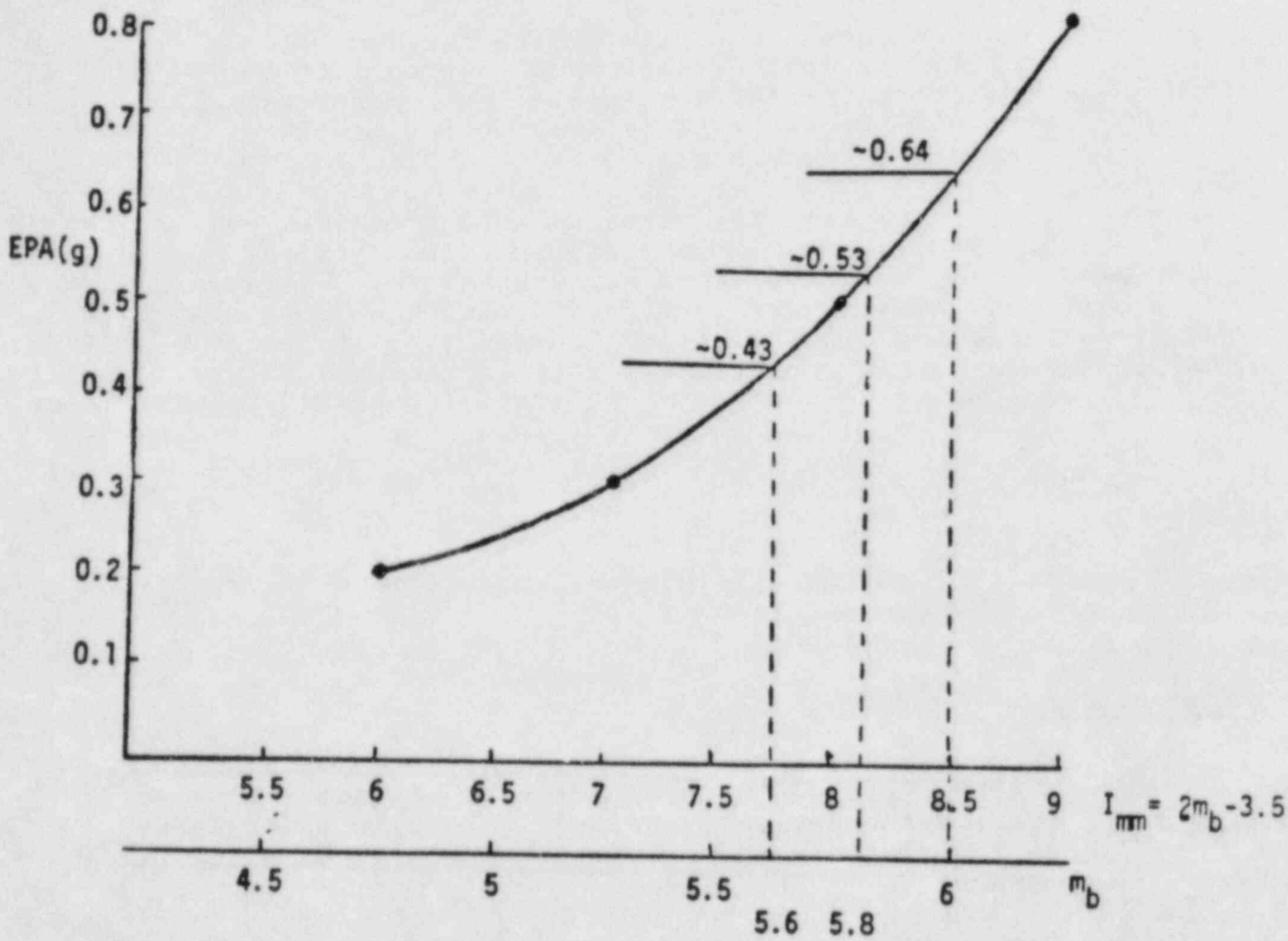

Fig. 2 - Upper bound EPA values in the Zion Probabilistic Safety Study (Fig. II.7.1)

4. I have tried to reconstruct the method used to obtain the upper bounds on EPA in Figure II.7.1. It seems that upper bound intensity values have been found from upper bound magnitudes using the relationship $I_{mm} = 2m_b - 3.5$ and that the associated upper bounds EPA have been calculated by interpolation of the values on page 6 of Section 7.9.3 (see Figure 2 of this report). However, the above relationship between I_{mm} and m_b was obtained by fitting dispersed data and does not apply to upper bounds.

5. Large uncertainty exists on the maximum value of EPA. It would therefore be appropriate to consider alternative values of this parameter through different truncations of the attenuation error distribution.

In summary, I agree that the values of EPA for given (m_b, Δ) be bounded, but I disagree with the way in which this was one in the Zion study. It is difficult to say whether reanalysis would produce higher or lower hazard estimates. For example, truncation of the attenuation error distribution would reduce seismic hazard, but inclusion of higher alternative truncation values would increase hazard, especially in the region of small exceedance probabilities.

Section 8.8.1

- I have no special comment on this section, except to iterate that one should use a more refined discretization of acceleration.

Conclusions

1. The general methodology is adequate and up-to-date. A more appropriate interpretation of uncertainty on the components resistance would lead to wider spread in the distribution of risk, but would not alter its mean value.

2. Implementation of the methodology is unsatisfactory in two major aspects:

   (a) The EPA upper bound should be incorporated in the seismic hazard analysis through truncation of the attenuation error distribution. This seems to be an important parameter and should be subjected to sensitivity analysis.

(b) Calculation of the fragility curves for core melt should be documented and curve grouping should be more detailed, especially in the range of low EPA values. Because no calculation detail is given in the Zion report, it is difficult to anticipate the effect of a more accurate work. However, I believe that this may lead to higher risk values.

3. Other areas that would benefit from improvement are:

   (a) Redefinition of EPA, accounting for the lower damage potential of small events near the Zion site (this lower potential should be better documented). Such a redefinition would lead to a reduction of seismic hazard in the case of zonations for which hazard is contributed primarily by close, low-magnitude events. One can also take advantage of the 0.9 reduction factor for randomization of the horizontal component of motion.

   (b) The use of EPA instead of IPA is inappropriate for acceleration-sensitive equipment.

   (c) More refined discretization of acceleration values for the calculation of the frequency of core melt.

4. I should like that comments be added about:

   (a) The degree of correlation of the fragility curves for different components and about the likely effect on the results. This effect depends on the configuration (mainly parallel or mainly series) of the fault trees for the various release category events.

   (b) The effect of changing the type of distribution (now lognormal) for the fragility curves.

In my opinion, Items 2a, 2b, 3a, and 3b are those of more critical importance.

Appendix D

Review of the Zion Probabilistic Safety Study
Seismic External Events

Prof. Erik H. Vanmarcke
Massachusetts Institute of Technology

Consulting Report to
Sandia National Laboratories

DRAFT

Evaluation of Overall Methodology

The overall analysis format involving a consecutive matrix operations on the vector(s) of initiating event probabilities is simple and attractive, and is quite appropriate for seismic risk evaluation.

In reference to the five main steps in the seismic safety analysis (as outlined on page II.7.1), I have some concern about the interplay of the first three steps (seismicity, fragility and plant logic), in particular, about the manner in which the analyses involving these three first steps were carried out. The central question is how the uncertainty inherent in the fragility curves is modeled and evaluated.

I would argue that a step is missing in the sequence comprising the seismic safety analysis. In between Step 1 (Seismicity) and Step 2 (Fragility), there should be a step labeled Seismic Response or Seismic Load Effect.

When an earthquake occurs, a ground motion characterized by peak acceleration (whether "instrumental", "effective", or "sustained" does not matter at this point) is experienced at the base of the structure. The dynamic seismic input causes many simultaneous response accelerations a_j at points j (locations of structural components or equipment support points) throughout the structure. These response motions are actually predominantly narrow-band, that is, they have a frequency content quite different from that of the input motion. The output-to-input acceleration ratios a_j/a may be seen as random variables whose marginal statistics depend on the seismic reponse, the randomness of the ground motion, the (uncertain) dynamic properties, etc. Seismic design is based on the seismic response, the randomness of the ground motion, the (uncertain) dynamic properties, etc. Seismic design is based on the predicted response accelerations a_j to which an appropriate safety factor is applied. This yields the mean or median capacity (or resistance) of component j in terms of acceleration. The actual capacity of component j is of course a random variable.

In the format of the seismic safety part of the Zion study, the uncertainty represented by the fragility curves originates from both the loading and the resistance, and the uncertainty about the (response-related) ratio a_j/a is incorporated in the fragility curves.

I believe that the introduction of an intermediate step (Seismic Response or Seismic Load Effect) in the seismic safety assessment would help clarify and resolve many issues related to modeling, interpretation and processing of component fragility curves, in particular:

a. Variability: The components of uncertainty related to seismic input (owing to complexity of accelerograms) and response could be separated from those related to capacity or resistance (measurable by component testing).

b. Probability Models: Much is known about probability density functions of seismic load effects. Input accelerations are approximately Gaussian, the acceleration responses of linear systems are also Gaussian; their peaks follow known extreme value distributions; the acceleration response of an elastoplastic system has a distribution with a spike at the acceleration yield limit, etc. The point is that it would no longer be necessary to adopt the sweeping assumption that all random variables involved have a lognormal distribution. We know better.

c. Failure Criteria: It would no longer be necessary to express all fragility curves in terms of peak acceleration (a definite drawback of the present format). Depending on the function (or rather, malfunction) of each component, the fragility curve might be in terms of maximum (response) acceleration, sustained peak acceleration, relative displacement, or even energy absorption capacity.

d. Correlation: Patterns of correlation (different for random load and resistance factors) are not adequately accounted for in the present format of converting component fragility curves into system fragility curves by using plant logic diagrams. Clearly, the component to-system conversion is accomplished (quite artificially) in the "resistance domain" by assuming statistical independence between the random variables that control the width of component fragility curves. In reality, for a given input acceleration , the response accelerations $_j$ are fairly strongly correlated. The associated component resistances are perhaps more nearly independent. Depending on the relative variability of load effects and resistances, the real system condition may be closer to one or another of the two extreme conditions of perfect dependence and perfect independence. Structural system reliability theory offers techniques for bounding the system reliability when information about correlation is available.

Detailed Review Comments

Section II.2.2, Paragraph 1

1. The results in Table II.2-1 are indeed striking. Earthquake risk (Major Seismic Event) is the dominant contributor to predicted health and safety risk, and it ranks second among contributions to the mean frequency of core melt. This is all the more surprising in view of the fact that the Zion plant is located in an area of relatively low seismicity.

Section II.7.1.1: Seismicity

2. Page II.7-1, fifth from last line: "...such a curve... would adequately characterize the seismic activity at the site, where we able to draw it". This is an overstatement. Such a curve provides no information, for example, about duration of shaking or frequency content, although these may have a significant impact on seismic response, performance or damage.

3. Detailed comments about the seismicity study are presented as part of the review of Section 7.9.1. My main concern with this summary (and also that in Section 7.2.2) is that it does not faithfully restate the conclusions and reproduce the results of the Dames & Moore study. Nowhere in the Dames & Moore analysis are rigid bounds imposed on effective peak acceleration; this asymptotic behavior at low risk levels is, however, the single most striking feature of seismicity curves of Figure II.7-1 (or Figure 7.2-1). The last sentence in Section II.7.1.1 does not adequately explain the logic which led from Section 7.9.1 to the exceedance curves used to evaluate core damage probabilities and final seismic "frequency-of-probability" curves.

Section II.7.1.2: Fragility

4. Figure II.7-2: It is preferable to label the different fragility curves with fractions (which sum to one) rather than with cumulative frequencies. The format of display in Figure II.7-4 is correct in this regard. The use of cumulative frequencies is especially confusing if there is a chance that the different curves in a family might overlap. (This could easily happen if the fragility curves are permitted to have significantly different " " values).

5. Last paragraph of Section II.7.1.? makes it clear that acceleration is not necessarily the best response parameter in terms of which to define fragility curves; for example, relative displacement might be superior in some cases.

6. The choice of the lognormal distribution is expedient but not necessarily consistent with available information. Seismic response is more nearly normal than lognormal (Seismic excitations are approximately normal (with mean zero), and any linear system preserves this normality; hence, the response time histories are normal.) The absolute maximum of the random response of a linear system follows an extreme value distribution about which much is known. Hence, the sweeping assumption of lognormality is justified mainly on account of analytical convenience (i.e., it facilitates analysis of products of independent random variables).

Section II.7.1.4: Seismic Core Melt Frequencies

7. Table II.7-2: The total dominance of plant state SE may be attributable to the assumption of the 0.65g upper bound acceleration (shown in Figure II.7-8). Evidently, only the ceramic insulators and the service water pumps have a chance of experiencing accelerations exceeding their predicted tolerable limit.

Section 7.2.2

8. The comments made under Item (3) also apply to Section 7.2.2.

Section 7.9.1: Seismic Hazard Analysis

9. Section 7.91. is not as carefully worded as other parts of the report. This is particularly evident in the introductory section. The following minor corrections and word changes are suggested:

   p.1, 2nd paragraph, line 2, replace "...should be derived" by "...will be performed".

   p.1, 2nd paragraph, last sentence: leave out

   p.1, 3rd paragraph, line 1: replace "effort" by "resources"

   p.1, 3rd paragraph, line 9:...The assumptions <u>adopted</u>...

p.12, last line: replace "in" by "is".

p.14, fourth from last line: replace "analyses" by "analysis"

Table 3, column labeled .50, second to last number: should be $.48 \times 10^{-5}$ instead of $.48 \times 10^{-4}$.

The remaining comments about Section 7.9.1 are technical and relate to assumptions made and their likely impact on the final results.

10. An unstated assumption throughout Section 7.9.1 is that the "seismogenic zone" approach has been used rather than alternate methodology based solely on historical seismicity. In view of the range of assumptions about zone geometry, I judge that the range of results adequately covers what would be predicted by alternate methodology.

11. Page 3, Seismic Hazard Model, Item 1: I question the statement: "...the average predicted rates of occurrence in these zones". The words "predicted" and "accurately" should be dropped. Incidentally, uncertainty about mean occurrence rates is neglected and I agree it is unlikely to have much of an impact on overall uncertainty.

12. Page 3, Seismic Hazard Model, Item 2, "...truncated exponential distribution...": To set the record straight, I would like to mention that this model was first proposed in a report not referenced in the Dames & Moore Study [Vanmarcke, E. H. and Cornell, C. A., "Analysis of Uncertainty in Earthquake Ground Motions and Structural Response," M.I.T. Department of Civil Engineering Report R69-24, April 1969].

13. Page 3, Seismic Hazard Model, Item 3, "...local soil conditions.": Local soil conditions at the Zion site are not explicitly accounted for, as has been common in nuclear plant seismic design. Recently, however, at a number of nuclear plant sites, successful attempts have been made to identify and isolate the systematic amplification effect which local soil has on incoming seismic waves.

14. The assignment of uncertainty to the attenuation laws ($\sigma_{\ln a} = 0.6$) is reasonable. Alternate assumptions could have been tested (with appropriate weights attached), but I expect this would not have had much impact on the final results. The same may be said about the choice of the lower limit on magnitude ($m_b = 4$).

15. The comment (on page 6, Item 2, line 4) "...even if peak accelerations are high..." is revealing. It implies recognition that accelerations are indeed highly variable. Many seismologists and earthquake engineers would say that this is equally true at high as at low values of m_b (or Mercalli Intensity), and that any rigid upper bound on peak acceleration is unrealistic.

16. Uncertainty about "b-value" (on page 7): The three-valued discretization (mean and mean $\pm$ one standard deviation) appears inadequate as it obviously does not cover the tails of the distribution.

17. Discretization of $m_{b,max}$ (on page 7 and 8): The double-triangular distribution has an upper bound of 6.2; it is then converted into a three-valued probability mass function whose largest value is $m_{b,max}=6$. The resulting error in seismic risk calculations may not be negligible (in the low probability range) if the rigid bound on effective acceleration were to be relaxed.

18. It is stated on page 8 that "It was felt by the seismological consultant that there is some negative correlation between b-values and values of $m_{b,max}$." This is the apparent justification for assuming complete probabilistic dependence between b and $m_{b,max}$. It would be interesting to see some results based on the assumption that b and $m_{b,max}$ vary independently. Also, it might have been preferable to quantify the seismological consultant's judgment in terms of a (discretized) joint probability distribution implying partial correlation.

19. Consideration of alternative attenuation laws (Equations 5 and 6) is adequate.

Sections 7.9.1 and 7.9.3: Treatment of Peak Acceleration

20. Nuttli's data in Figure 4 (Section 7.9.1) indicate that the 1.37 value for the ratio of sustained to peak acceleration applies to the magnitude range $m_b \geq 6.0$. The 1.37 value is in fact adopted for all magnitudes. Note, however, that the upper magnitude bound adopted in the study equals $m_{b,max} = 6.0$ (with probability 0.28), while the m_b magnitude follows a truncated exponential distribution; it follows that the condition $m_b \geq 6.0$ (to which the 1.37 value corresponds)

is in fact assigned zero probability of occurrence. the 1.37 value is therefore subject to question. The 0.9 factor mentioned on page 10 of Section 7.9.1 (leading to the factor 1.37 x 0.9 = 1.233 ≈ 1.25 in Section 7.9.3) is acceptable.

21. The influence of the choice of a_{max} is understated, for example on page 13 in Section 7.9.1.: "In general, the variation in hazard resulting from the use of alternate estimates of peak acceleration is within the variation resulting from different hypotheses on seismogenic zones." It is quite obvious from Table 3 in Section 7.9.1 that calculated probabilities are more sensitive to a_{max} than to zonation in the critical "high acceleration-low probability" range of the seismicity curves. It is this range of the curves which most influences the calculated risk of earthquake-induced core damage.

22. While I agree with SMS'a assessment of the inadequacy of peak acceleration to represent damage or damage potential (because factors such as ground motion duration and inelastic behavior are unaccounted for), I feel that the proposed acceleration reduction factors and especially the upper bounds are introduced in the wrong place. Such bounds (with probabilities attached) should perhaps appear in the fragility curves. There is little evidence of the existence of a firm limit on acceleration for a given Mercalli Intensity category. In any case, if such limits are introduced, they should be in evidence as part of the input to the seismic hazard analysis, and not appear as an after-the-fact adjustment of the output.

23. In any case, the presence of these acceleration correction factors and imprecise bounds points to the urgent need to implement improved earthquake ground motion descriptions which explicitly account for duration (in addition to a measure of intensity such as peak acceleration) and to apply analysis procedures which predict seismic response measures more directly correlated with performance and damage. Much of this is within the state-of-knowledge of earthquake engineering.

Section 8.8

24. The overall analysis format involving consecutive matrix operations is straightforward and attractive. As I pointed out before, my main concern is with the zeros in the last few columns of Table 8.8-1 (which shows the nine "seismic initiating event" probability vectors).

Evaluation of Final Results

In my detailed comments in the preceding section, I have tried to uncover all the main assumptions made in the seismic risk portion of the Zion Probabilistic Safety Study. Whenever possible, I included an expression of judgment about the appropriateness of these assumptions, and about their likely impact on the final results. In my opinion, the results expressed in terms of mean annual risk of core damage (or mean risk to public health and safety) are not particularly sensitive to reasonable variations in the many assumptions made except for the assumption discussed in the next two paragraphs.

The critical assumption relates to the imposition of an upper bound of effective peak acceleration. In conventional seismic risk work, this is a highly unusual step. If this assumption were relaxed it will probably lead to moderate increases in final mean seismic risk estimates.

The problem is exacerbated by inconsistent reporting of exactly how the acceleration bounds are introduced. In Section 7.9.1, probabilities are assigned to three alternate acceleration limits, including $a_{max} = \infty$ (assigned a 20% probability). Based on SMA's statement in Section 7.9.3, the alternative " $_{max} = \infty$ " is apparently left out in the (undocumented) final synthesis that results in the family of seismicity curves shown in the introductory section (Figure II.7-1) and Section 7.2.2 (Figure 7.2-1), and presented in Table 8.8-1. The key feature of these curves is their asymptotic behavior at effective accelerations of 0.45, 0.55, and 0.65 g. Contrary to what is stated in the second sentence of Section 7.2.2, Table 3 in Section 7.9.1 does not imply this kind of asymptotic decay since the seismicity curves corresponding to "$a_{max} = \infty$ " should be given a 20 percent weight in the final synthesis.

The other assumptions and procedures, mainly those dealing with the treatment of uncertainty in the fragility curves and with the coarse discretization of random variables in the seismic hazard analysis, are unlikely to have much impact on mean risk rates, but they will affect (in ways hard to predict) the family of "frequency-of-probability" curves shown in Figures II.2-1 through II.2-5.

APPENDIX E

Response to
Commonwealth Edison Comments
on the
Draft

NUREG/CR-3300, Volume 1

1. Introduction

Copies of the draft of NUREG/CR-3300 were provided to Commonwealth Edison for review and comment. Their comments were provided in Reference 5. Included in the reference were many comments pointing out errors in our analyses or taking issue with assumptions on which the analyses were based. In general, the comments were accurate and appropriate and several resulted in significant revisions to the NUREG. There are, however, issues on which differences remain. In the interest of brevity, we do not include here Commonwealth Edison comments with which we concur, or which address topics not important in terms of analytical results. However, where there are differences on issues significant to the evaluation of plant damage state frequencies, the pertinent portions of Reference 5 are presented; each individual comment is then followed by our response to it. The only editing we have done to the utility comments is to amend their references to specific sections and pages of our review, several of which have changed between the draft and final documents.

2. Zion Service Water System Fault Tree

2.1 Commonwealth Edison Comments

The equation on p. 1.5-658 of the ZPSS (which is correct), assumes that three of the six SW pumps are required for system success. Sandia interprets the Zion FSAR to mean that only two pumps are required. This is not correct. The FSAR indicates that on a licensing basis, two pumps are considered necessary for each unit. Realistically, however, only three pumps are needed in total for the two units. A success criterion of three SW pumps was therefore assumed in the ZPSS and is retained here.

Sandia incorporates common cause failures of the SW pumps into their review of the SW system analysis. We agree that common cause failures of the SW pumps should have been included in the ZPSS. We do not agree, however, that the application of the beta factor method presented in the Sandia report (p. 2-43) is appropriate. The SW system normally has four pumps operating and two in standby. The two standby pumps should not be considered candidates for the same common cause events that could affect the four operating pumps. Certainly, the beta factor of 0.014 used by Sandia is not supported by any evidence for such a large number of concurrent pump failures, especially considering the different status of some of the pumps.

In a review of Atwood (Reference 1), we find that in the data base applicable to the SW pumps there are no events involving failure of more than two pumps. Also, multiple pump failures to run involving even two pumps are rare. We therefore believe that a beta factor of 0.014 for the failure of the four operating pumps in a 24-hour period is overly conservative. If this highly pessimistic value is assumed, the system unavailability computed by Sandia (4.6×10^{-7}) results. If only three pumps are assumed to be affected by the common cause event, the computed SW system unavailability assuming a three-pump success criterion would be reduced by about two orders of magnitude since the additional independent failure of a standby SW pump would be necessary for system failure.

We believe that the likelihood of all four operating SW pumps failing to run is extremely low. The operator recovery probability would be a function of the underlying cause of the multiple failures and also of the time available for recovery.

Conservatively assuming a beta factor of 0.014 for failure of all four operating pumps and a 10 percent chance that the operator will be unable to restart even one of the four pumps in the time allotted, the SW system unavailability due to common cause now becomes 4.6×10^{-8}. Adding this to the SW system unavailability due to pipe breaks reported in the ZPSS (2.15×10^{-8}) gives a total SW system unavailability of 6.8×10^{-8}. This increase in system unavailability does not significantly affect plant risk despite the highly conservative assumptions made regarding the use of the beta factor.

2.2 Sandia Response

In the main body of our review, we do consider that the success criterion for the SWS is two pumps. Because the utility deems that three are, in fact, necessary for system success, we have added Section 4.9 to this review in which we consider the three pump criterion. We continue to use the report of Atwood on ß-factors (Reference 1) however. We believe it is the best available analysis of common mode pump failures experienced in the industry. In some cases no data exist, but Atwood applied a consistent method to derive plausible failure rates for those situations. Until a better analysis is done, we see no reason not to use his. We also must point out that the 0.014 ß-factor used in the ZPSS is purely subjective and that, without a detailed common mode failure analysis, it is purely speculative to conjecture how many of the pumps could be affected by a given common mode fault.

3. Service Water For Loss of Off-site Power

3.1 Commonwealth Edison Comments

On p. 2-43, Sandia attempts to compute the SW system unavailabilities for various degraded electric power states. These calculations contain several errors. In particular, the equations on p. 2-45 assume that as many as four service water pumps or three diesel generators may simultaneously be out for service. These violations of plant technical specifications are clearly not realistic. Also, although Sandia correctly recognizes that the swing diesel may connect to either Bus 147 or 247 with equal likelihood, they account for this incorrectly in their analysis. It is not equally likely for two or three diesels to be available for Unit 2 when the electric power state for Unit 1 is "no diesels available" (Sandia's Case 1). The most likely event in which no diesels would be available at Unit 1 would be when the swing diesel closes onto bus 247. The event that the swing diesel has failed and is unavailable is much less likely. Similar errors are made in Sandia's computations for their cases 3 and 5.

In addition, Sandia's equations assume that two SW pumps are needed for success. As noted earlier, we believe that three SW pumps are required. However, loss of SW following loss of off-site power is only of interest because of the potential for an RCP seal LOCA without RCS makeup. Once off-site power is restored, SW and RCS makeup can be easily restored. The likelihood of off-site power restoration is high enough (.9999), and the allowed recovery time from a seal LOCA long enough (realistically, several hours*), that the loss of SW following loss of off-site power sequences are not of interest.

3.2 Sandia Response

Four points raised above by Commonwealth Edison need to be addressed. First, the utility correctly asserts that, in our draft, we erred in calculating the diesel generator and service water pump unavailabilities due to maintenance. The analysis now starting on p. 2-45 of this report is correct. Secondly, we concede our initial handling of the availability of buses 147 and 247 was incorrect. Section 2.4.1.11 (and others affected by it) have been appropriately amended. Beginning on p. 2-24, we derive the probability that bus 247 is available, given that bus 147 is not, and apply this probability throughout the report. Thirdly, as stated above in Section 2.2 of this appendix, we now treat the three pump SWS success criterion as a sensitivity issue in Section 4.9 of the report.

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*See Section 6.1 of this appendix.

As to the fourth point, we continue to disagree with the utility. In Section 3.1 above, the utility states that restoration of off-site power is very likely and that the time to a seal LOCA is long. Based on our information, we disagree, and this is discussed more fully below in Section 6.2 of this appendix.

4. Failure of Component Cooling Water (CCW), SEFC

4.1 Commonwealth Edison Comments

The Sandia review adopts the ZPSS mean frequency for loss of CCW as an initiator, 9.4×10^{-4} per year. Sandia attributes this frequency to two types of scenarios: (1) single pipe breaks which could disable the CCW system; and (2) combinations of pump failures which would constitute CCW system failure.

A. Pipe Breaks. The contribution of single pipe breaks to CCW system failure was calculated in the ZPSS by assuming a mean pipe break frequency of 8.6×10^{-10} per hour per pipe segment. The ZPSS identified 30 pipe segments whose failure could potentially result in CCW failure. This gives a mean CCW failure frequency due to pipe breaks of 2.1×10^{-4} per year. This pipe break frequency is conservative in that it includes all rupture sizes. In order for a pipe break to disable the system, it must be large enough to overcome the system makeup capacity, which is reported by Sandia at 800 gpm. Based on a brief review of pump capacities, this value appears reasonable.

Since a review of LER data indicates that more than 94 percent of pipe breaks begin as slow leaks which can be detected and isolated (WASH-1400, p. III-77), the CCW failure frequency given above is very conservative. Also, even if a sufficiently large pipe break at a location low enough to drain the system were to occur, the break could be isolated without loss of system function and the system could be refilled with makeup water within a pessimistic assumed core uncovery time of 90 minutes. Since leak indications (low flow, low discharge pressure, and sump level alarms for the area served by CCW) are available, failure to recover from the break is unlikely. Either of the CCW system headers can be isolated without loss of system function. Assuming a human error rate of 0.044 (see ZPSS p. 1.5-693) for failing to recover the CCW system within 30 minutes* in the event of a large break, the frequency of an unrecovered loss of CCW due to large pipe breaks is given by

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*Note that the 30 minutes is a very conservative recovery time. More recent work discussed in Section 6.1 indicates that the time until a severe seal LOCA occurs following a loss of CCW is more likely to be about 10 hours.

$$2.1 \times 10^{-4} \times .06 \times .044 = 5.5 \times 10^{-7}/\text{year}$$

In one case, however, the location of the break may have an impact on the likelihood of successful recovery. In particular, a break in the pump suction header could cause the CCW pumps to fail within a short period of time due to lack of water. In this case, the CCW system would not be recoverable unless the operator acted quickly enough to shut down the CCW pumps before they failed. Five pipe segments capable of causing such an event have been identified with a total failure frequency of 3.4×10^{-5} per year. Using 0.5 as the chance that the operator fails to secure the running and standby CCW pumps in time, the contribution of suction piping rupture to system failure is 1.7×10^{-5} per year.

However, this rate assumes that an 8.6×10^{-10} per hour frequency of pipe rupture per section is applicable. Accounting for those pipe breaks which would be detected as leaks before a break occurred (i.e., 94 percent), but neglecting the fact that not all break sizes exceed the makeup capacity of the system, results in a frequency of non-recovered CCW system failure initiated by suction pipe breaks of

$$1.7 \times 10^{-5} \text{ per year} \times .06 = 1 \times 10^{-6} \text{ per year}$$

Combining the two types of pipe break contributions calculated above gives

$$\phi_{pipe} = 5.5 \times 10^{-7} + 1 \times 10^{-6} = 1.6 \times 10^{-6} \text{ per year}$$

This assessment is conservative because the CCW system is less susceptible to pipe breaks than most plant systems. The system has good chemistry control to prevent corrosion, does not experience wide temperature fluctuations, is a closed loop system, and has been in operation for a number of years--plenty of time to sort out errors in installation, construction, design, and fabrication.

B. Pump Failures. Sandia assumes that the contribution of multiple component failures (primarily pump failures) to CCW failure is equal to the difference between the loss of CCW initiator frequency given in the ZPSS (9.4×10^{-4} per year), and the contribution of single CCW pipe breaks to this event. The original loss of CCW initiator frequency was calculated based on data and engineering judgment using a two-stage Bayesian approach. Now that a more detailed look at the system has been undertaken, however, it is more appropriate to add the frequency of CCW system failure due to pump failures to the pipe break initiated failure frequency to obtain a revised loss of CCW initiator frequency.

The contribution of multiple pump failures to CCW system failure was computed in Section 2.4.1.10. To determine the frequency of an unrecovered loss of CCW involving pump train failures as an initiating event, the results obtained in our response to Section 2.4.1.10 must be reevaluated using a mission time of one year rather than 24 hours. For the two different assumed system success criteria, the results are as follows:

$$\varnothing pump = 1.7 \times 10^{-6}/year; \quad \varnothing pump = 8.8 \times 10^{-6}/year$$

(1 of 5) (2 of 5)

Extensive recovery actions such as pump repairs which could be undertaken in the several hours available before core uncovery have not been considered in this analysis. An additional conservatism in the above analysis is the use of the same beta factor for the loss of CCW initiating event as for CCW system unavailability. The operating CCW pumps would need to fail almost simultaneously in order to cause an initiating event, since otherwise timely shutdown of the plant would be achieved and a long CCW recovery time would be available.

As mentioned in our response to Section 2.4.1.10, we believe that the CCW success criterion for the loss of CCW sequence should be one pump rather than two. The multiple pump failure contribution for a one pump success criterion can therefore be added to the pipe break contribution to obtain a revised unrecovered loss of CCW initiating event frequency of

$$\varnothing CCW = \varnothing pipe + \varnothing pump = 1.6 \times 10^{-6} + 1.7 \times 10^{-6}$$

$$= 3.3 \times 10^{-6} \text{ per year}$$

This event frequency is sufficiently low to be dropped from the list of major contributors to release category 8B, and is illustrative of the results which can be obtained by accounting for even minimal recovery; i.e., assuming an RCP seal leak rate of 300 gpm per pump and a correspondingly short recovery time. The effects of additional recovery actions which would be possible using a more realistic recovery time have not been quantified.

4.2 Sandia Response

We must disagree with the utility presentation on pipe break frequency. The presented ZPSS mean datum for frequency of breaks in pipes of greater than a three-inch diameter is, as they state, 8.6×10^{-10} per hour per segment. This corresponds well with the WASH-1400 median datum of 1×10^{-10} per hour per segment with an error factor of thirty (Table III.2-1 of WASH-1400). Our strong disagreement concerns the reference to p. III-77 of WASH-1400. First, the utility states that it is LER data but, in fact, that page presents data from US non-nuclear utility experience. Secondly, although the 94 percent factor does represent the fraction of pipe failures which first leaked, the subsequent WASH-1400 analysis takes this into account. That is, the median frequency of 1×10^{-10} per hour per segment is already discounted for the 94 percent which do leak first and thus represents that fraction which do not, but simply rupture. Hence, to use a six percent discounting again, as the utility suggests, either double counts for leak-before-break or redefines the basis of their 8.6×10^{-10} per hour per segment datum (where the prior in the ZPSS are the WASH-1400 values, see Item 47 on p. 1.5-78 of ZPSS where the fifth and ninety-fifth percentiles of WASH-1400 are used as the twentieth and eightieth percentiles for the ZPSS).

At the same time, we recognize that the pipe break frequency is probably conservative (see Section 3.2.1 of the report). The value is more appropriate for high pressure piping, which is also well tested. The CCW piping is at lower pressure, but we do not know the degree of its testing. Until better data exist for piping such as that found in the CCW system, however, we believe the data we used are proper.

Furthermore, in the reanalysis by the utility of the CCW pipe break scenario (Section 4.1 of this appendix above), they use a non-recovery factor of 0.044 and cite p. 1.5-693 of the ZPSS as its reference. Inspection of this page reveals that such a datum is given as failure of the operator to recover local failures within thirty minutes. The analysis, though, is for the AFW system, e.g., to take local control of the turbine pump. We fail to see the relevance of AFW recovery potential to that of CCW pipe break recovery potential. A more thorough analysis should be performed for operator actions subsequent to a CCW pipe break.

As to that portion of this sequence due to CCW pump failures, the utility now presents us with a third frequency for the initiating events of loss of CCW (see Section 3.2.1 of the report for the other two). Obviously, this confuses us and, because the utility does not refute the 9.4×10^{-4}/yr value we used from the ZPSS, we see no reason to amend our analysis. We should note that we gave the operator nearly 95 percent probability of recovering this particular sequence.

5. Failure of DC Bus 111, Failure of Auxiliary Feedwater, TEFC

5.1 Commonwealth Edison Comments

Failure of DC buses as an initiating event was treated in ZPSS Section 1.3.4.13.5. A bounding argument was used there to show that DC bus failures would have little effect on the plant matrix compared to other reactor trip initiators. However, the dependence between the DC buses and the PORVs was not taken into account. Consequently, credit was taken for feed and bleed cooling even though one of the two PORVs could not be opened in the event of DC bus failures.

DC Bus 111 powers one motor-driven AFW pump and thus could contribute to AFW failure, but does not power a PORV. Conversely, DC Bus 011-1 powers one of the two PORVs, but not the AFW pumps. DC Bus 112 powers a motor-driven AFW pump and a PORV. Failure of Bus 112 could thus contribute to failure of the AFW system and also potentially disable feed and bleed cooling.

The initiating event frequency computed in the ZPSS (and used by Sandia) was for loss of any one of six DC buses, but the DC bus dependence identified by Sandia applies to only one bus at Unit 1. Consequently, the results presented by Sandia must be reduced by a factor of 1/6. Also, recovery actions to restore DC power are simple and could be implemented in short order. In fact, for all the DC bus failures acknowledged in the ZPSS and cited by Sandia, recovery was extremely rapid and simple. Immediate recovery of the deenergized bus is, therefore, very likely. Even assuming a very conservative non-recovery factor of 1/10, Sandia's estimate of this sequence's frequency would become

$$\lambda_{DC} = 6.4 \times 10^{-5} \times 1/6 \times 1/10 = 1.1 \times 10^{-6}/\text{year} \quad .$$

This sequence is thus a small contributor to release category 8B, and therefore to plant risk.

Finally, ongoing work by Westinghouse suggests that reactor cooling by feed and bleed with only one PORV open is feasible if initiated rapidly (20 minutes up to 1 hour) or if DC power (thus Auxiliary Feedwater) are recovered within about 2 hours. The criteria for success of feed and bleed cooling are dependent on a number of factors which include:

1. Time of steam generator dryout
2. Number of PORVs available
3. Time at which PORVs are opened
4. Number of trains of SI available

The effect of these factors on feed and bleed success for the Zion plant is discussed below.

The time at which steam generator dryout occurs affects the success of feed and bleed in that the decay heat load which must be removed following steam generator dryout decreases with time. For the Zion plant this time can range from approximately 45 minutes to approximately 67 minutes.

Analyses have shown that with minimum safety injection available, opening of both PORVs is sufficient for success of feed and bleed provided they are opened prior to steam generator dryout. These results indicate that the sooner the PORVs are opened the better. For events in which only one PORV is available the opening time of the PORV is more crucial.

The flow capacity of one PORV combined with minimum safety injection is insufficient to remove decay heat at low enough pressures to allow safety injection to replace lost inventory and prevent core uncovery. Partial core uncovery is expected to occur. As the core liquid level drops and the upper portion of the rods heat up the steam passing the uncovered portions of the core will be superheated. During this phase the temperature of the fuel rods and cladding depends primarily on the decay heat level. Preliminary calculations indicate that the operation of one PORV with minimum safety injection should be sufficient to preclude significant fuel or clad damage, when the steam generator dryout time is sufficiently long, as would be the case for the loss of DC power event or DC power is recovered. The potential for feed and bleed cooling with only one PORV available reduces even further the impact of the dependence identified by Sandia between the DC power system and the PORVs.

5.2 Sandia Response

The utility is correct in pointing out to us that we erred in identifying bus 112 as bus 111 in the draft, but our analysis was based on failure of bus 112 as the initiating event. As to the initiating event frequency calculated in the ZPSS, our interpretation as to whether it applies to one bus or to six (we thought the former) is actually immaterial. As discussed on p. 3-11 of the report, the actual Zion experience, per bus, is nearly that of the initiating event frequency we used for our point estimate.

We have allowed for operator recovery in the report, which we did not do in the draft. Our 90 percent recovery probability may be non-conservative however. The sequence involves total failure of AFW, and the operator may not be sufficiently apprised of the situation because of the peculiarity in annunciation in the Zion control room. The Zion DC Ground Location Procedure (ZED 3, dated September 9, 1980) denotes the failures caused by, and annunciation ensuing from, failure of DC bus 112. On p. 171 of this document, it states that stop valve 1FCV-MS57 fails open on loss of DC power from bus 112 and its position indication is lost on the main control board. This valve allows steam to enter the AFW pump turbine. What is peculiar is that failure of this bus also energizes the turbine pump fail-to-start alarm. Thus, the operator, by procedure, knows that the alarm is "false," but for this sequence, it is true! This will certainly add more stress to the situation.

As to the feed and bleed discussion by the utility, we concur that, if the ongoing work by Westinghouse does confirm that one PORV is sufficient for the bleed, then this sequence requires further evaluation. The feed and bleed criterion we use in the report, i.e., both PORVs necessary, is the criterion supplied in the ZPSS. We have no newer reference and avidly await the reporting of the Westinghouse research.

6. <u>Loss of Off-site Power: Loss of Component Cooling Water: Failure to Restore Power in 4 Hours, SEFC (also Sections 3.2.4, 3.2.5, 3.2.7, and 3.2.11)</u>

6.1 Commonwealth Edison Comments

Five new dominant sequences involving loss of off-site power (LOP) and failure of the CCW are postulated by Sandia. The Sandia analysis differs from the ZPSS in that two CCW pumps rather than one are assumed to be required for system success, RCP seal LOCAs are considered, CCW pump failures are included in the system unavailability calculations, and a different distribution of electric power recovery times is used. All five Sandia sequences involving LOP and failure of the CCW system are discussed in this section of our response.

A. <u>Electric Power Recovery</u>. The ZPSS study team has already responded to comments from Brookhaven National Laboratory (BNL) on the electric power recovery distribution (Reference 2). We feel that generic power recovery data are not applicable to the important loss of off-site power sequences at Zion. In the ZPSS, recovery of off-site power is of interest primarily in those sequences in which there is a severe degradation of on-site power supplies and also a loss of core cooling capability. No such instances are found in the generic data base. In virtually all cases in the data base, on-site power was available from one or more diesel generators and there was no immediate concern about loss of core cooling or loss of reactor coolant inventory. In some events, there are even indications that partial off-site power service was available and could have been connected to the plant if the on-site equipment had malfunctioned. Consequently, we believe that in the loss of off-site power sequences of interest in the ZPSS there would be a far more vigorous effort to restore off-site power than in the events observed to date, resulting in shorter recovery times than those indicated by the generic data.

Our evaluation of the generic data base also indicates that those sites experiencing extended power outages tend to be located in areas subject to regional grid instability problems (e.g., St. Lucie and Turkey Point) or unique supply line routing (e.g., Pilgrim), or have experienced localized outages directly attributable to some characteristic of the site (e.g., Millstone and Palisades). This provides further indication that Zion power recovery times should be lower than the average for the plant population as a whole.

Overall, we believe that the Zion site characteristics, Commonwealth Edison's experience, and the characteristics of the event sequences of interest are sufficient to support the power recovery time distribution used in the ZPSS. This distribution explicitly accounts for normal shift manning, conflicting concerns about the failed diesel generators and the restoration of normal power, standard plant practice for switchyard operations, and expected operator performance under conditions not covered by specific procedures but involving operations familiar to plant personnel. It is conservative to the extent that it models the diesel generator trouble investigation and switchyard response actions as purely sequential events for a single operator when, in fact, it is possible that different operators could respond to both locations simultaneously.

The recovery time probabilities computed by Sandia from generic plant data are contrasted with the plant specific values from the ZPSS in Table 1. The 60-minute non-recovery probability from the ZPSS has been revised slightly, from .042 to .046, as a result of comments by BNL (Reference 2). Also, the mean loss of off-site power initiating event frequency has been increased slightly, from 0.057 to 0.068 events per site calendar year, again due to BNL comments. By adding in the transient induced loss of off-site power events applicable to the loss of CCW accident sequences in question, a new loss of off-site power initiating event frequency for this analysis is obtained:

Loss of Off-site Power	=	.068
Loss of Main Feedwater + LOP	=	1.8×10^{-3} x 2 Units
Turbine Trip + LOP	=	1.3×10^{-3} x 2 Units
Reactor Trip + LOP	=	1.3×10^{-3} x 2 Units
		.0768 per site calendar year

By comparing the Sandia dominant sequence results with results obtained using the ZPSS non-recovery probabilities but the revised loss of off-site power initiating event frequency (0.068), we can see the impact of the generic non-recovery probabilities assumed by Sandia (see Table 2). The results clearly demonstrate the importance of the assumed off-site power recovery distribution. Only Sandia dominant accident sequence 4 still has an appreciable frequency of occurrence when the ZPSS non-recovery probabilities are used. The other sequences are clearly not dominant when the Zion plant specific power recovery distribution is used, even assuming for the moment that Sandia's CCW success criterion and other assumptions are appropriate.

B. Sandia's Dominant Accident Sequence 4. The assumptions made by Sandia in estimating the frequency of dominant sequence 4 with its 1-hour non-recovery duration will now be considered in more detail.

First, note that in the equations developed by Sandia for LOP with CCW pump failure (p. 2-40), not all AC power may be lost because some of the diesels may still be available. With power available to either bus 148 or 149, Unit 1 RHR pumps can be operated in the low pressure injection mode if the reactor coolant system has been depressurized using the steam generators and the AFW system. This alternative cooldown scheme is described in the recent Westinghouse Owners Group Emergency Response Guidelines (Reference 3), but is not taken into account by Sandia.

An additional conservatism in the Sandia estimate of the frequency of dominant accident sequence 4 is the assumption that loss of CCW for 1 hour would result in core damage. Assuming that the RCP seal O-rings would remain intact for several hours, a realistic estimate for RCP seal leakage is between 10 and 13 gpm per pump (Reference 4), rather than 300 gpm. Even assuming an upper bound leak rate of 300 gpm per pump, recovery from loss of CCW without core damage should be possible for a 75-minute period rather than just 1 hour. For more realistic leak rates, the 1- to 4-hour off-site power recovery time frame for sequence 4 would allow plenty of time to restore CCW and safety injection once off-site power is restored and thus prevent core damage.

Table 1

AC Electric Power Nonrecovery Probabilities

Nonrecovery Time t	ZPSS Probability	Sandia Probability (generic)
30 Minutes	.292	.52
60 Minutes	.046	.38
4 Hours	1.1×10^{-4}	.25
8 Hours	$< 10^{-4}$	.[illegible]0

In addition, recovery of a failed diesel generator is not considered in the Sandia analysis of sequences involving loss of off-site power and loss of CCW. However, the chances are good (P = 0.775) that a failed diesel generator would be recovered within 6 hours (see ZPSS Section 1.3.2.4). Even with only a 75-minute recovery time, corresponding to an assumed 300 gpm RCP seal leak rate, there is a 35 percent chance of recovering a failed diesel generator. Since more than one diesel generator would be failed in the sequences of interest, the chance that at least one of these would be quickly recoverable is even greater.

Finally, we do not agree that two CCW pumps are required for successful RCP seal cooling in sequence 4 as assumed by Sandia. The operator would have plenty of time to shed unnecessary CCW loads (e.g., the spent fuel pool) even if no procedures specifically directing him to do so are available, and one CCW pump would then be sufficient for system success (see our response to Section 2.4.1.10).

With a one-pump success criterion, the frequency of Sandia's dominant accident sequence 4 would be reduced by at least one order of magnitude (to less than 2×10^{-6} per year). Even assuming that the operator does not isolate unnecessary loads so that two CCW pumps are required, the other identified conservatisms (i.e., Sandia's failures to consider diesel generator recovery, the low seal leak rate, and the low pressure injection cooling mode) combine to assure that sequence 4 does not contribute significantly to plant damage state SEFC or to plant risk.

Conservatively assuming a 10 percent chance that the initial plant configuration is such that one CCW pump is not sufficient without operator intervention and that the operator fails to isolate unnecessary loads, the frequency of Sandia's dominant sequence 4 corrected for off-site power recovery as in Table 2 (1.8×10^{-5} per year) would be further reduced to 1.8×10^{-6} per year. This then would be a small contributor (i.e., about 4 percent) to release category 8B.

We believe, however, that a realistic RCP seal leak rate is much less than the 300 gpm per pump assumed by Sandia, so several hours would be available for recovery. After offsite power is recovered, which is assumed to occur before 4 hours in this sequence, reactor coolant system makeup can be easily accomplished since power will be available for the idle CCW pumps. Therefore, assuming realistic leak rates, Sandia's sequence 4 is a negligible contributor to risk.

Table 2

Loss of Off-site Power and Component Cooling Water Dominant Sequence Possibilities

Sequence	Sandia Result	Result with ZPSS Plant Specific Recovery Interval Probabilities*
3. (4- to 8-hour recovery)	4.6-5	4.2-8
4. (1- to 4-hour recovery)	4.0-5	1.8-5
5. (> 8-hour recovery with containment fan failure)	1.8-5	< 2.3-8
7. (> 8-hour recovery)	7.9-6	< 9.9-9
11. (> 8-hour recovery with failure of containment sprays and fan coolers)	4.7-6	< 5.9-9

*To obtain, multiply Sandia results by

$$\frac{(.0768)}{(.061)} \times \frac{\text{(ZPSS recovery interval probability)}}{\text{(generic recovery interval probability)}}$$

NOTE: Exponential notation is indicated in abbreviated form; i.e., 4.6-5 = 4.6×10^{-5}.

6.2 Sandia Response

We must address five specific issues raised by Commonwealth Edison in their comments presented in Section 6.1 of this appendix. First as has been addressed earlier in this appendix, we originally erred in our calculation of the availability of power at bus 247, given that power was unavailable on bus 147. We have corrected this for this final report.

Second, there is disagreement on recovery of off-site power. We use data developed by the Electric Power Research Institute,[7] which is based on historical events, whereas, the ZPSS developed what is deemed a plant specific recovery analysis. Based on the description of the recovery model found in Section 1.3.2 of the ZPSS, we are surprised to see the statement in the ZPSS that "we express the following histogram as a representative distribution for the time to restore power ..." with no basis offered for the histogram. It cannot be plant specific data; Zion has never experienced a loss of off-site power event. There was apparently no attempt to review and use data from actual offsite power occurrences which would suggest that the curve in Figure 1.3.2-2 (Page 1.3-24) of the ZPSS regarding recovery of offsite power, is optimistically steep. The bottom line is that the non-recovery value for 60 minutes would appear to be a factor of ~5-10 too small in com- parison with industry experience.

An analysis of diesel generator recovery was also conducted in the ZPSS although it is not apparent that these data were ever used. Since the recovery was found to be small as reported in other PRAs, use of these data would make little difference in the final sequence frequency. Now, however, the utility apparently wishes to use these data.

As to power recovery, we further raise the point that, because it is at the center of many grids, we feel that the frequency used in the ZPSS for loss of off-site power is acceptable, although it is at the low end of industry experience. On the other hand, should such an event occur at Zion, we can speculate that the problem could be massive, or else the power would not be lost in the first place (unless the problem was in the plant part of the transmission yard). Thus, we could conjecture that recovery at Zion would be worse than industry averages. Other than speculation, we have no basis for assuming so and, therefore, for this reason and those stated above, we believe it is proper to continue to use the EPRI data.

The third issue raised in the utility comments in Section 6.1 of this appendix is that of rapid RCS cooldown via the steam generators to a pressure regime in which the RHR pumps could inject. We are aware that this deliberate overcooling transient is being considered for the emergency response guidelines of the three PWR vendors and believe it is quite worthy of further investigation as another potential means of averting core damage. We have not seen, nor has Commonwealth Edison offered, plant specific analyses as to whether or not Zion could perform this without incurring problems. Timing considerations would need to be addressed such as when does the decision to do the rapid cooldown have to be made. Additionally, we do not know if the steam generator tubes can maintain their integrity in such a mode nor if this mode requires changes in the AFW success criterion (and hence failure probability). Furthermore, if the cooldown is rapid and the RCS is not allowed to soak to remove latent heat, we do no know if a bubble will form in the vessel head. All these concerns are probably irrelevant if the situation is that, without the cooldown, core damage is a certainty. We point them out, however, to show that much plant specific analysis is required before the mode should be credited fully as an alternate capability. (We should also mention that RHR will also fail in the recirculation mode due to lack of heat exchanger cooling.)

As to the fourth issue, the timing and size of a transient-induced reactor coolant pump seal LOCA, we must disagree with the utility based on information currently available to us. The ZPSS used the 300 gpm in thirty minutes, and until more test data become available, the NRC agrees with the assumption in the ZPSS (Reference 6). We continue to use the 300 gpm in thirty minutes in this report.

The fifth and final issue concerns the success criterion we use for the CCW system: two of five pumps must operate. As stated earlier in this appendix, we used two of six pumps as the success criterion for the service water system. Apparently, Zion needs only one of five CCW pumps most of the time (when the flow to the spent fuel pool is low) and can succeed for a long duration at other times (when the flow to the spent fuel pool is higher, but such flow can be isolated). At the same time, the utility states that three of six SW pumps are needed (see Section 2.1 of this appendix). We reexamined the effect these two criteria have on our list of the revised dominant accident sequences at Zion. The analysis and its effects are discussed in Section 4.9 of this report. As can be seen there, the "2-2" criteria is not much different than the "1-3." Furthermore, we understand that the utility is reexamining its service water criterion of three pumps in specific situations, such as loss of off-site power. Zion may not initially require three SW pumps because, for example, the containment fan load could be isolated. Core damage has not yet occurred, and decay heat is being removed through the steam generators. Because the frequency difference between the two criteria sets is not great and the issue is being reexamined by the utility, we present the criteria issue as a sensitivity (Section 4.9 of the report) and do not alter the rest of the report.

7. Summary

7.1 Commonwealth Edison Comments

The Sandia frequency estimates for all five dominant accident sequences involving LOP and loss of CCW have been shown to be substantially conservative. Some conservatisms (especially involving off-site power recovery) have been quantified and additional qualitative conservatisms have been identified. This analysis is sufficient to assure that none of the five sequences considered here are dominant with respect to plant risk or core-melt frequency.

7.2 Sandia Response

Obviously based on our above responses, we respectively disagree. We believe that our analysis is an accurate representation of the core melt potential at Zion, given the present data available to us. In the main report, we present uncertainties for the internal event sequences, and as can be seen, in some cases, these are quite large. To reduce these uncertainties, resolve success criteria questions, and to possibly allow for other credible modes of plant operation, we believe other studies should be performed and have reflected these concerns in our responses in this appendix.

8. ZPSS Fire Analysis

8.1 Commonwealth Edison Comments

The review and evaluation of the ZPSS fire analysis (given in Section 7.3) is presented in Section 4.6 of the Sandia report. In general, we believe that their sensitivity analysis is too conservative.

Before proceeding to discuss Sandia's specific comments on the fire analysis, it is worth noting that the Sandia reviewers viewed their comments as a sensitivity analysis, not a definitive assessment of the impact of their proposed changes. Therefore, the results of the Sandia revised fire analysis do not appear among the "Revised Zion Dominant Accident Sequences" in Section 3 of the report.

Specific comments by the Sandia reviewers are discussed below.

A. The Zion fire analysis only analyzed two plant areas - the auxiliary electrical equipment room and the cable spreading room. Other important plant areas were either qualitatively assessed (e.g., Auxiliary Building Zone 11-3.0) or not addressed at all in the Zion PRA (e.g., component cooling water (CCW) pump area). (p. 4-30)

We agree with Sandia that we should have explicitly delineated our reasons for not quantifying the fire risk from areas other than those addressed in Section 7.3 of the ZPSS. However, the discussion in that section does mention that the overall risk from those areas is thought to be dominated by the risk from the cable spreading room (CSR) and the auxiliary electrical equipment (AEE) room. As mentioned in the Sandia review, the small contribution from the unquantified areas would be further reduced when the plant modifications for compliance with Appendix R to 10CFR50 are taken into account.

B. The Zion fire analysis did not address seal LOCA events caused by the loss of CCW. (p. 4-30)

As pointed out by Sandia, the possibility of a LOCA due to RCP seal failure was not considered in the fire analysis of the ZPSS. Fires outside of the cable spreading room which are capable of causing RCP seal failure are deemed to be unlikely due to the physical separation between the redundant trains and the available fire protection features. The areas outside the CSR where critical cables are closest together are in the component cooling water and service water (SW) pump areas. For a critical set of pumps to fail, an extremely large fire would have to occur.

The potential for seal failure from fires in the cable spreading room is addressed in Item 4.

C. The Zion fire analysis did not consider that power to both electrical AFW pumps and to the steam regulating valve of the steam-driven AFW pump all run through the same cable spreading room. (p. 4-30)

The turbine-driven auxiliary feedwater pump can be manually operated locally in the absence of electric power to the control circuits of the pump train since the steam regulating valve is an air-operated valve which opens upon loss of power, and the pump has been tested in this operating mode. The availability of the steam-driven AFW pump decreases the impact of cable spreading room fires even if the cables to all AFW pumps are affected.

D. The Zion fire analysis did not consider the loss of service water or component cooling water by fire in combination with an unavailability of redundant components due to maintenance. (p. 4-30)

The failure of the CCW and SW systems due to fires in areas other than the cable spreading room is discussed above. In the CSR, the power cables for the CCW and the SW pumps span a distance of about 14 feet horizontally and are 7 feet below the ceiling. Thus, a very large fire would be needed to simultaneously fail either the CCW or the SW cables. In addition, a hot gas layer under the ceiling is very unlikely to be as much as 7 feet thick.

The Sandia reviewer was unaware of the exact routing of the cables and tried to avoid conservatism by assigning a value of 0.05 to the chance of a fire being large enough and located in a critical area so as to damage a vital set of these cables. Based on the configuration of the critical cable trays, we think that this factor is conservative.

Another source of conservatism in the Sandia analysis is the assumption that all the containment cooling features may be lost due to a single fire. This is why the reviewer concludes that damage states SE and TE may result. However, the diesel-driven containment spray pump depends on the service water system only for engine cooling. The mean unavailability of this pump, even including the possibility that the service water pumps may be lost, is still less than 0.1. This shows that the frequencies assigned by the reviewer to damage states SE and TE as a result of fires are overestimated by at least an order of magnitude.

E. The Zion fire analysis assumed correct operator actions with a mean probability of 2.5 x 10^{-2} even under high-stress fire conditions. (p. 4-30)

For the AEE room fire analysis, the Sandia reviewer has assigned a larger frequency to operator error than was used in the ZPSS. We do not agree with this assumption and believe that the frequency we used is sufficiently conservative. First, the operators would most likely find out in a short time that the source of abnormalities on the control board was an AEE room fire, because the AEE room is next to the control room. Upon fire detection, the operators would not be likely to rely on the main control board as the only source of information. In addition, the operators would most likely have several hours to mitigate the consequences of an AEE room fire since those initiating events which have short core melt times are very unlikely to occur due to that cause. Based on these observations, we believe that the ZPSS human error analysis for AEE room fires is reasonable.

F. The analyses that were reported in the ZPSS have not considered equipment or cable damage by hot gas layers or failure of cabling at temperatures below autoignition temperatures. (p. 4-30)

The possibility of a hot gas layer under the ceiling causing damage to cables prior to ignition was not recognized as an issue until after the ZPSS was published. In addition, the change in accident frequency estimates due to the recent research on this issue may not be significant because the test results to date are not directly applicable to the Zion plant for the following reasons:

1. The tests were conducted in a relatively small room. The temperature of a hot gas layer in the Zion CSR would be lower due to the larger size of the room.

2. The damaged cable trays in the tests were very close to the ceiling. The hot gas layer temperature decreases with distance from the ceiling.

3. The test source fire was large and its frequency of occurrence will therefore be low.

In addition, the overall accuracy of the ZPSS fire analysis can only be assessed if the impact of conservative as well as non-conservative assumptions is quantified. The treatment of various heat sinks is an example of an area where the analysis has been conservative. This may counteract any non-conservatism resulting from the omission of hot gas layers in the analysis.

8.2 Sandia Response

Commonwealth Edison addresses six points made in the draft review of the ZPSS fire analysis. The first considered our concern that only two plant areas were analyzed in the ZPSS. We believe that potential fire areas should be identified in a systematic method and further that the method and its results should be well documented in the PRA itself. That is, perhaps the ZPSS did analyze the dominant fire areas, but the study results have not demonstrated this by a consistent application of a well-conceived method.

The second Commonwealth Edison comment addressed our statement that the ZPSS fire analysis did not consider seal LOCA events. While we concur that the physical separation of the redundant trains preventing seal LOCAs makes the event unlikely, we still believe that a systematic effort should have been made to identify all potential fire areas and scenarios.

Thirdly, the claim by the utility that the turbine-driven AFW pump can be locally controlled does not negate our assertion that the ZPSS did not consider that power to all three AFW pumps (two motor, one turbine) runs through the CSR. An analysis of the reliability of the operator to take local control of the turbine pump during a CSR fire should still be performed. Such was performed for internal events (see p. 1.5-693 of the ZPSS).

We have three responses to the fourth utility comment concerning the loss of service or component cooling water by fire. First, they do not address our central claim, that the failure of some pumps due to fire while others are in maintenance was not considered in the ZPSS. We believe it should be. Secondly, as to the hot gas layer, such a layer often goes nearly to the floor in fire tests. Only if a door is open to allow a less thick layer to accumulate, can it be deemed unreasonable that the layer will not be seven feet thick. Thirdly, the discussion of the diesel-driven containment spray pump confuses us. If the diesel engine of the pump requires service water for cooling, then failure of service water fails this pump. If it does not require service water to operate, then it should be so considered throughout the ZPSS. Because the dependency of the pump on service water is identified in the internal events analyses, we maintain our claim that potential SE and TE damage states may result from a single fire.

The fifth Commonwealth Edison comment addresses our concern that 2.5×10^{-2} is possibly non-conservative for failure of the operator to correctly act under high-stress conditions. They assert that the operators "would not be likely to rely on the main control board as the only source of information" given an AEE room fire. This statement is inconsistent with all other assumptions regarding operator response to accidents. Furthermore, whether he relies on the main board or not, we believe that a more thorough analysis of operator actions is warranted for this fire scenario. It should be noted that the 0.025 probability is approximately half that of the operator failing to take local control of the AFW turbine pump given an internal event. We believe a fire situation is more stressful and requires more operations.

The sixth utility comment concerns the omission in the ZPSS of the fire failure mechanism due to a hot gas layer, specifically, on how the inclusion of the layer might affect the analysis of the CSR fire. They state that the effects of a hot gas layer would be less in the CSR because it is a larger room than that used for the tests. In fact, the effects could be more if hot pockets formed. In addition, although the temperature of the hot gas layer decreases with distance from the ceiling, as they state, the decrease is not dramatic. If the cables are in the layer, they will most likely fail. Furthermore, the test fire was not large as the utility states, and thus its frequency of occurrence is not necessarily low. Lastly, they indicate that the omission of heat sinks in the ZPSS results in the analysis presented therein being conservative. This assertion may be so, but we note that, for short fires, heat sinks may not have sufficient time to react to transient conditions.

As stated in Section 4.6 of this review, we believe the ZPSS fire analysis lacks sufficient depth and clarity. The purpose of our review is not to perform an independent PRA, but rather to review dominant sequences and search for omissions. Because we have found deficiencies in the analysis and sufficient documentation was lacking to adequately amend the deficiencies, we choose to present the fire analysis review as a sensitivity issue and not as part of the dominant sequence review.

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NRC FORM 335
(11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG/CR-3300, Vol. 1
SAND83-1118/1 of 2

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Review and Evaluation of the Zion Probabilistic Safety Study: Plant Analysis

5. DATE REPORT COMPLETED

MONTH	YEAR
March	1984

DATE REPORT ISSUED

MONTH	YEAR
May	1984

6. *(Leave blank)*

7. AUTHOR(S) D. L. Berry, T. L. Brisbin, D. D. Carlson, et al (SNL); J. W. Reed, M. W. McCann (JRB Associates, Inc.)

8. *(Leave blank)*

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
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444 Castro Street, Suite 501
Mountain View, CA 94041

10. PROJECT/TASK/WORK UNIT NO.

11. FIN NO.
A1125

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Division of Safety Technology
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

13. TYPE OF REPORT
Technical Evaluation Report

PERIOD COVERED *(Inclusive dates)*

14. *(Leave blank)*

15. SUPPLEMENTARY NOTES

16. ABSTRACT *(200 words or less)*

This report describes the review of the internal and external event plant analyses of the Zion Probabilistic Safety Study (ZPSS). The review was conducted by Sandia National Laboratories. The purpose of the review was to search for areas in the ZPSS where omissions and critical judgments were made which could impact the quantitative results. The review identified several of these areas.

17. KEY WORDS AND DOCUMENT ANALYSIS
Zion Unit 1
Zion Unit 2
Commonwealth Edison Company
Probabilistic Risk Analysis

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS *(This report)*
Unclassified

20. SECURITY CLASS *(This page)*

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (11-81)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

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