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Evaluation of Severe Accident Risks: Zion, Unit 1

Main Report

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

In support of the Nuclear Regulatory Commission's (NRC) assessment of the risk from severe accidents at commercial nuclear power plants in the U.S. reported in NUREG-1150, a revised calculation of the risk to the general public from severe (core meltdown) accidents at the Zion Power Station, Unit 1, has been completed. This power plant, located 64 km (40 mi) north of Chicago, Illinois, is operated by the Commonwealth Edison Company.

The emphasis in this risk analysis was to determine the distribution of risk, and to discover the uncertainties that account for the breadth of this distribution. This risk assessment was limited to severe accidents initiated by internal events.

The offsite risk from internal initiating events was found to be of the same order of magnitude as the risk estimates reported about a decade ago in the Reactor Safety Study (RSS) (WASH-1400). However, the pressurized water reactor (PWR) analyzed in the RSS was Surry, and an updated risk assessment for Surry also performed in support of NUREG-1150 at Sandia National Laboratories (SNL) (refer to Volume 3 of this report) found the revised risk estimate to be generally below the RSS estimates. The higher risk estimates for Zion (compared with Surry) are due to the higher core damage frequency and higher population distribution around the Zion site.

Loss of coolant accidents following pump seal failure were estimated to have a relatively high frequency of occurrence when compared with the frequency of other possible accidents at Zion. However, the likelihood of early containment failure following reactor pressure vessel (RPV) failure was found to be low for all accidents with the exception of containment bypass events. Even though the likelihood of early containment failure is low (together with bypass events) dominates the risk estimates. The uncertainties in risk are therefore largely due to uncertainties in predicting the frequency of loss-of-coolant accident (LOCA) events and the likelihood of early containment failure and bypass during a severe accident.

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FOREWORD

This is one of numerous documents that support the preparation of the final NUREG-1150 document by the U.S. Nuclear Regulatory Commission (NRC) Office of Regulatory Research. Figure 1 illustrates the documentation of the accident progression, source term, consequence, and risk analyses. The direct supporting documents for the first draft of NUREG-1150 are given in Table 1. They were produced by the three interfacing programs that performed the work: the Accident Sequence Evaluation Program (ASEP), the Severe Accident Risk Reduction Program (SARRP), and the Probabilistic Risk Assessment (PRA) Phenomenology and Risk Uncertainty Evaluation Program (PRUEP).

The Accident Frequency Analysis, and its constituent analyses, such as the Systems Analysis and the Initiating Event Analysis, are reported in NUREG/CR-4550. Originally, NUREG/CR-4550 was published without the designation "Draft for Comment." Thus, the current version of NUREG/CR-4550 is designated Revision 1. The label Revision 1 is used consistently on all volumes, including Volume 2 which was not part of the original documentation. NUREG/CR-4551 was originally published as a "Draft for Comment." While the current version could have been issued without a revision indication, all volumes of NUREG/CR-4551 have been designated Revision 1 for consistency with NUREG/CR-4550.

The material contained in NUREG/CR-4700 in the original documentation is now contained in NUREG/CR-4551; NUREG/CR-4700 is not being revised. The contents of the volumes both in NUREG/CR-4550 and NUREG/CR-4551 have been altered. In both sets of documents now, Volume 1 describes the methods utilized in the analyses, Volume 2 presents the elicitation of expert judgment, Volume 3 presents the analyses for Surry, Volume 4 concerns the analyses for Peach Bottom, and so on. Note that the Zion volume of NUREG/CR-4551, now Volume 7, was Volume 5 in the original Draft for Comment version of NURG/CR-4551, published in February 1987. The Zion plant was not treated in the original Draft for Comment version of NUREG/CR-4700.

In addition to NUREG/CR-4550 and NUREG/CR-4551, there are several other reports published in association with NUREG-1150 that explain the methods used, document the computer codes that implement these methods, or present the results of calculations preformed to obtain information specifically for this project. These reports include:

> NUREG/CR-5032, SAND87-2428, "Modelling Time to Recovery and Initiating Event Frequency for Loss of Off-Site Power Incidents at Nuclear Power Plants," R.L. Iman and S.C. Hora, Sandia National Laboratories, Albuquerque, NM, January 1988.
>
> NUREG/CR-4840, SAND88-3102, "Recommended Procedures for Simplified External Event Risk Analyses," M.P. Bohn and J.A. Lambright, Sandia National Laboratories, Albuquerque, NM, December 1988.
>
> NUREG/CR-5174, SAND88-1607, "A Reference Manual for the Event Progression and Analysis Code (EVNTRE)," J.M. Griesmeyer and L.N. Smith, Sandia National Laboratories, Albuquerque, NM, 1989.

NUREG/CR-5380, SAND88-2988, "A User's Manual for the Post Processing Program PSTEVNT," S.J. Higgins, Sandia National Laboratories, Albuquerque, NM, 1989.

NUREG/CR-5360, SAND89-0943, "XSOR Codes User's Manual," H.-N. Jow, W.B. Murfin, and J.D. Johnson, Sandia National Laboratories, Albuquerque, NM, 1989.

NUREG/CR-4624, BMI-2139, "Radionuclide Release Calculations for Selected Severe Accident Scenarios," Volumes I-V, R.S. Denning et al., Battelle Columbus Division (BCD), Columbus, OH, 1986.

NUREG/CR-5062, BMI-2160, "Supplemental Radionulclide Release Calculations for Selected Severe Accident Scenarios," M.T. Leonard et al., Battelle Columbus Division, Columbus , OH, 1988.

NUREG/CR-5331, SAND89-0072, "MELCOR Analyses for Accident Porgression Issues," S.E. Dingman et al., Sandia National Laboratories, Albuquerque, NM, 1989.

NUREG/CR-5253, SAND88-2940, "A User's Guide to PARTITION: A Program for Defining the Source Term/Consequence Analysis INterfaces in the NUREG-1150 Probabilistic Risk Assessments," R.L. Iman, J.C. Helton, and J.D. Johnson, Sandia National Laboratories, Albuquerque, NM, 1989.

NUREG/CR-5382, SAND88-2695, "Incorporation of Consequence Analysis Results into the NUREG-1150 Probailistic Risk Assessments," J.C. Helton et al., Sandia National Laboratories, Albuquerque, NM, 1989.

NUREG/CR-5282, BNL-NUREG-52181, "Estimation of Containment Pressure Loading due to Direct Containment Heating for the Zion Plant," N.K. Tutu, C.K. Park, C.A. Grimshaw, and T. Ginsberg, Brookhaven National Laboratory, Upton, NY, June 1989.

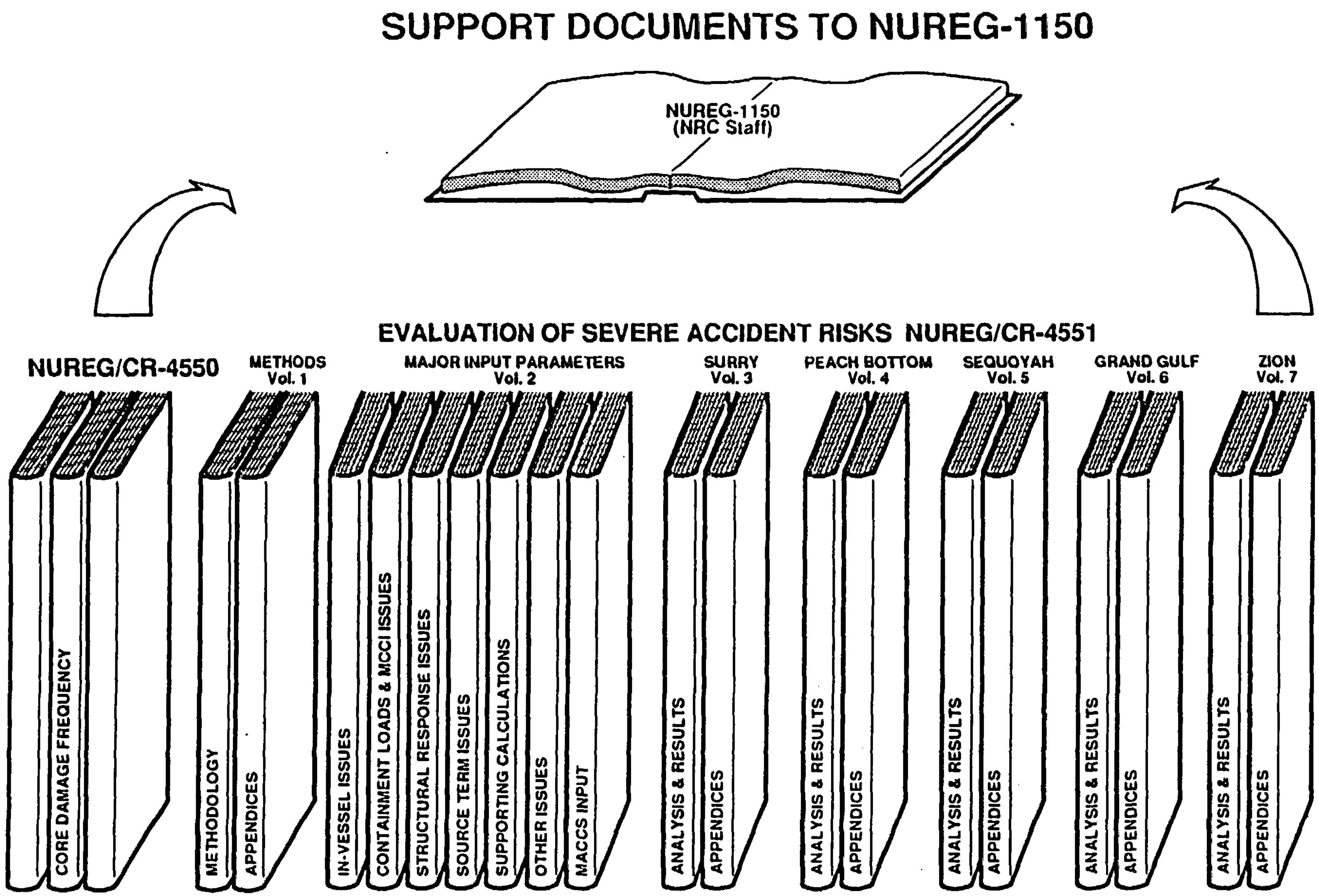


Figure 1. Back-End Documentation for NUREG-1150.

Table 1. NUREG-1150 Analysis Documentation

Original Documentation

NUREG/CR-4550	NUREG/CR-4551	NUREG/CR-4700
Analysis of Core Damage Frequency From Internal Events	Evaluation of Severe Accident Risks and the Potential for Risk Reduction	Containment Event Analysis for Potential Severe Accidents
Vol. 1 Methodology	Vol. 1 Surry Unit 1	Vol. 1 Surry Unit 1
2 Summary (Not Published)	2 Sequoyah Unit 1	2 Sequoyah Unit 1
3 Surry Unit 1	3 Peach Bottom Unit 2	3 Peach Bottom Unit 2
4 Peach Bottom Unit 2	4 Grand Gulf Unit 1	4 Grand Gulf Unit 1
5 Sequoyah Unit 1		
6 Grand Gulf Unit 1		
7 Zion Unit 1		

Revised Documentation

NUREG/CR-4550, Rev. 1, Analysis of Core Damage Frequency	NUREG/CR-4551, Rev. 1, Eval. of Severe Accident Risks
Vol. 1 Methodology	Vol. 1 Part 1, Methodology; Part 2, Appendices
2 Part 1 Expert Judgment Elicit. Expert Panel	2 Part 1 In-Vessel Issues
Part 2 Expert Judgment Elicit. Project Staff	Part 2 Containment Loads and MCCI Issues
	Part 3 Structural Issues
	Part 4 Source Term Issues
	Part 5 Supporting Calculations
	Part 6 Other Issues
	Part 7 MACCS Input
3 Part 1 Surry Unit 1 Internal Events	3 Part 1 Surry Analysis and Results
Part 2 Surry Unit 1 Internal Events App.	Part 2 Surry Appendices
Part 3 Surry External Events	
4 Part 1 Peach Bottom Unit 2 Internal Events	4 Part 1 Peach Bottom Analysis and Results
Part 2 Peach Bottom Unit 2 Int. Events App.	Part 2 Peach Bottom Appendices
Part 3 Peach Bottom Unit 2 External Events	
5 Part 1 Sequoyah Unit 1 Internal Events	5 Part 1 Sequoyah Analysis and Results
Part 2 Sequoyah Unit 1 Internal Events App.	Part 2 Sequoyah Appendices
6 Part 1 Grand Gulf Unit 1 Internal Events	6 Part 1 Grand Gulf Analysis and Results
Part 2 Grand Gulf Unit 1 Internal Events App.	Part 2 Grand Gulf Appendices
7 Zion Unit 1 Internal Events	7 Part 1 Zion Analysis and Results
	Part 2 Appendices

ACRONYMS AND INITIALISMS

AFWS	auxiliary feedwater system
APB	accident progression bin
APET	Accident Progression Event Tree
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transients Without Scram
BCD	Batelle Columbus Division
BMT	basemat melt-through
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
CCDF	complementary cumulative density function
CCWS	component cooling water system
CCI	core-concrete interaction
CF	containment failure
CFCS	containment fan cooling system
CH	chronic health effect weight
CHR	containment heat removal
CSS	containment spray system
ECCS	emergency core cooling system
EH	early health effect weight
EPZ	emergency planning zone
HPIS	high pressure injection system
HPME	high pressure melt ejection
INEL	Idaho National Engineering Laboratory
IPE	individual plant examination
LHS	Latin Hypercube Sampling
LOCA	loss-of-coolant accident
LOSP	loss of offsite power
LPIS	low pressure injection system
LWR	light water reactors
MAAP	Modular Accident Analysis Program computer code
MACCS	MELCOR Accident Consequence Code System
NRC	Nuclear Regulatory Commission
PDF	probability density function
PDS	plant damage state
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PRCC	partial rank correlation coefficient
PRUEP	Phenomenology and Risk Uncertainty Evaluation Program
PWR	pressurized water reactor

RCP	reactor coolant pump
RCS	reactor cooling system
RPV	reactor pressure vessel
RSS	Reactor Safety Study
RWST	refueling water storage tank
SARRP	Severe Accident Risk Reduction Program
SBO	station blackout
SGTR	steam generator tube rupture
SNL	Sandia National Laboratories
SRV	safety relief valve
STCP	Source Term Code Package
SW	service water
TAF	top of active fuel
TCDF	total core damage frequency
TMI	Three Mile Island
V	interfacing systems LOCA
VB	vessel breach
ZPSS	Zion Probabilistic Safety Study

ACKNOWLEDGEMENTS

Many people at Brookhaven National Laboratory (BNL) and elsewhere contributed to the work reported in this volume. The authors are particularly grateful to M. B. Sattison and K. W. Hall of Idaho National Laboratory (INEL) who performed the analysis of the core damage frequency for Zion. Considerable interaction between the authors and staff at INEL was necessary to ensure correct definition of the plant damage states and propagation of uncertainties. Staff at both Laboratories worked closely together to produce the final risk estimates reported in this volume.

One of the objectives of performing the Zion analysis at BNL was to determine if the models being developed at Sandia National Laboratories (SNL) as part of the programs reported to NUREG-1150 could be exported to another Laboratory and successfully applied to the risk analysis of a nuclear power plant. This objective was accomplished and the authors are grateful to the SNL steam members who were very helpful in the implementation of the codes at BNL. In particular, the authors wish to acknowledge R. J. Breeding (SNL) who was the Surry team leader. The analysis reported in this volume utilized to a large extent the SNL Surry plant model suitably modified for Zion specific design features. Dr. Breeding was very helpful in the successful application of the models to the Zion analysis. In addition, Dr. Breeding gave permission to reproduce significant documentation related to model description from the Surry (Volume 3) report for use in this volume. This documentation was included in order to make this volume self contained. We also acknowledge the support of E. Gorham and F. Harper who were the SNL team leaders for the work performed at SNL on this project. The authors are also grateful for the support of several NRC staff members. In particular, the guidance and support of M. Cunningham and J. A. Murphy is gratefully acknowledged. Finally, C. Conrad, Lev Neymotin and K. Ryan were a great help in preparing this documentation.

SUMMARY

S.1 Introduction

The United States Nuclear Regulatory Commission (NRC) has recently completed a major study to provide a current characterization of severe accidents risks from light water reactors (LWRs). This characterization is derived from integrated risk analyses of five plants. The summary of this study, NUREG-1150,[1] has been issued in final form.

The risk assessments on which NUREG-1150 is based can generally be characterized as consisting of four analysis steps, an integration step, and an uncertainty analysis step:

1. Accident frequency analysis: the determination of the likelihood and nature of accidents that result in the onset of core damage.

2. Accident progression analysis: an investigation of the core damage process, both within the reactor vessel before it fails and in the containment afterwards, and the resultant impact on the containment.

3. Source term analysis: an estimation of the radionuclide transport within the reactor coolant system (RCS) and the containment, and the magnitude of the subsequent releases to the environment.

4. Consequence analysis: the calculation of the offsite consequences, primarily in terms of health effects in the general population.

5. Risk integration: the assembly of the outputs of the previous tasks into an overall expression of risk.

6. Uncertainty analysis: the propagation of the uncertainties in the initiating events, failure events, accident progression branching ratios and parameters, source term parameters through the first three analyses above, and the determination of which of these uncertainties contributes the most to the uncertainty in risk.

This volume presents the details of the last five of the six steps listed above for the Zion Power Station, Unit 1. The first step is described in NUREG/CR-4550, Volume 7.[2] This risk assessment is restricted to severe accidents initiated by internal events which might occur while the plant is at full power. Two other plants analyzed as part of NUREG-1150 included accidents initiated by external events (fire and seismic initiators) in addition to internal events. In a follow-on study risk assessments will be performed for two plants (a BWR and PWR) for accidents that might occur while the plants are at low power or shutdown. This follow-on study will include both internal and external initiators.

S.2 Overview of Zion Generating Station, Unit 1

Zion, Unit 1 is one of the two 1050-MW (net) reactors operated by the Commonwealth Edison Company. The site for the Zion station[3] is located on the western shore of Lake Michigan, on the outskirts of the city of Zion and is about 64 km (40 mi) north of Chicago, Illinois.

The nuclear steam supply system for each of the Zion units is a four loop Westinghouse pressurized water reactor (PWR). Each reactor is rated at 3,250-MW (thermal). The emergency core cooling systems (ECCSs), which are located in the auxiliary building, are totally independent for each of the two units and consist of redundant high pressure injection trains, and passive accumulators for each unit. Hot leg as well as cold leg injection capability exists. The ECCS takes suction from the containment sump through special pipes to the inlet of the low pressure injection pumps during recirculation.

The plant auxiliary cooling systems consist of a shared component cooling water system (a closed system) and a shared service water system. The auxiliary feedwater system (AFWS), serving the secondary side of the steam generators, is separate for each unit. Each unit has three pumping trains, each capable of feeding all four steam generators. Two of the trains are fed by separate, redundant, 100% capacity, motor driven pumps, while the third train is fed by a redundant 200% capacity steam turbine driven pump.

Electrical power is supplied through multiple offsite power sources. Backup diesel generators are available for safety related loads in the event that offsite power is lost. Batteries are available for supplying DC power in the event of such a loss. The diesel generator consists of five machines, two per unit, with the fifth being a swing diesel capable of tying into a third bus on either unit as demand arises. Safeguards actuation systems consist generally of standard Westinghouse logic networks with sequential diesel generator loading.

The balance of the plant equipment is not unique from a safety standpoint. The turbine-generators are Westinghouse tandem compound units. Six stages of feedwater heating are provided. Each unit uses a single pass, deaerating type condenser. Once through cooling is provided using Lake Michigan as a source of cooling water.

Each reactor system is housed in an individual containment building. These structures consist of post tensioned concrete shells over 0.006 m (1/4-in) thick steel liners. The containment volume is approximately 7.7×10^4 m^3 (2.7×10^6 ft^3), and the design pressure is 0.43 MPa (62 psia). Each containment is served by both fan cooler and containment spray systems. These systems provide redundant and diverse containment heat removal capability. There are a total of five fan cooler units per containment operating in parallel, with each one being rated at one-third the required capacity for accident conditions. During normal operation a maximum of four units are required to remove the design heat load. For post accident operation a minimum of three units must function to satisfy safeguards requirements. The containment spray system is divided into three independent 100% capacity subsystems with no common headers. Of the three spray pumps, two are motor driven and the third is diesel driven. All three pumps take suction from the refueling water storage tank (RWST) and discharge into the spray rings located around the inside of the containment dome.

S.3 Description of the Integrated Risk Analysis

Risk is determined by combining the results of four constituent analyses: the accident frequency, accident progression, source term, and consequence analyses. Uncertainty in risk is determined by assigning distributions to important variables, generating a sample from these variables, and propagating each observation of the sample through the entire analysis. The sample for Zion consisted of 150 observations involving variables from the first three constituent analyses. The risk analysis synthesizes the results of the four constituent analyses to produce measures of offsite risk and the uncertainty in that risk. This process is depicted in Figure S.1, which was reproduced from Volume 3 of this report. This figure shows, in the boxes, the computer codes utilized. The interfaces between constituent analyses are shown between the boxes. However, the accident frequency analysis portion of the Zion assessment was performed in a somewhat different manner than the remaining plants in the NUREG-1150 study. Reference should therefore be made to NUREG/CR-4550, Volume 7 for a more detailed description of the approach used for Zion rather than Figure S.1. A mathematical summary of the process, using a matrix representation, is given in Section 1.4 of this volume.

The accident frequency analysis uses event tree and fault tree techniques to investigate the manner in which various initiating events can lead to core damage and the frequency of various types of accidents. Experimental data, past observational data, and modeling results are combined to produce frequency estimates for the minimal cut sets that lead to core damage. A minimal cut set is a unique combination of initiating event and individual hardware or operator failures. The minimal cut sets in a PDS provide a similar set of initial conditions for the subsequent accident progression analysis. Thus, the PDS form the interface between the accident frequency analysis (which is reported in NUREG/CR-4550, Volume 7) and the accident progression analysis reported in this volume. The outcome of the accident frequency analysis is a frequency for each PDS or group of PDSs for each observation in the sample.

The accident progression analysis uses large, complex event trees to determine the possible ways in which an accident might evolve from each PDS. The definition of each PDS provides enough information to define the initial conditions for the accident progression event tree (APET) analysis. Past observations, experimental data, mechanistic code calculations, and expert judgment were used in the development of the model for accident progression that is embodied in the APET and in the selection of the branch probabilities and parameter values used in the APET. Due to the large number of questions in the Zion APET and the fact that many of these questions have more than two outcomes, there are far too many paths through the APET to permit their individual consideration in subsequent source term and consequence analysis. Therefore, the paths through the trees are grouped into accident progression bins (APBs), where each bin is a group of paths through the event tree that define a similar set of conditions for source term analysis. The properties of each accident progression bin define the initial conditions for the estimation of a source term. The result of the accident progression analysis is a probability for each APB, conditional on the occurrence of a PDS, for each observation in the sample.

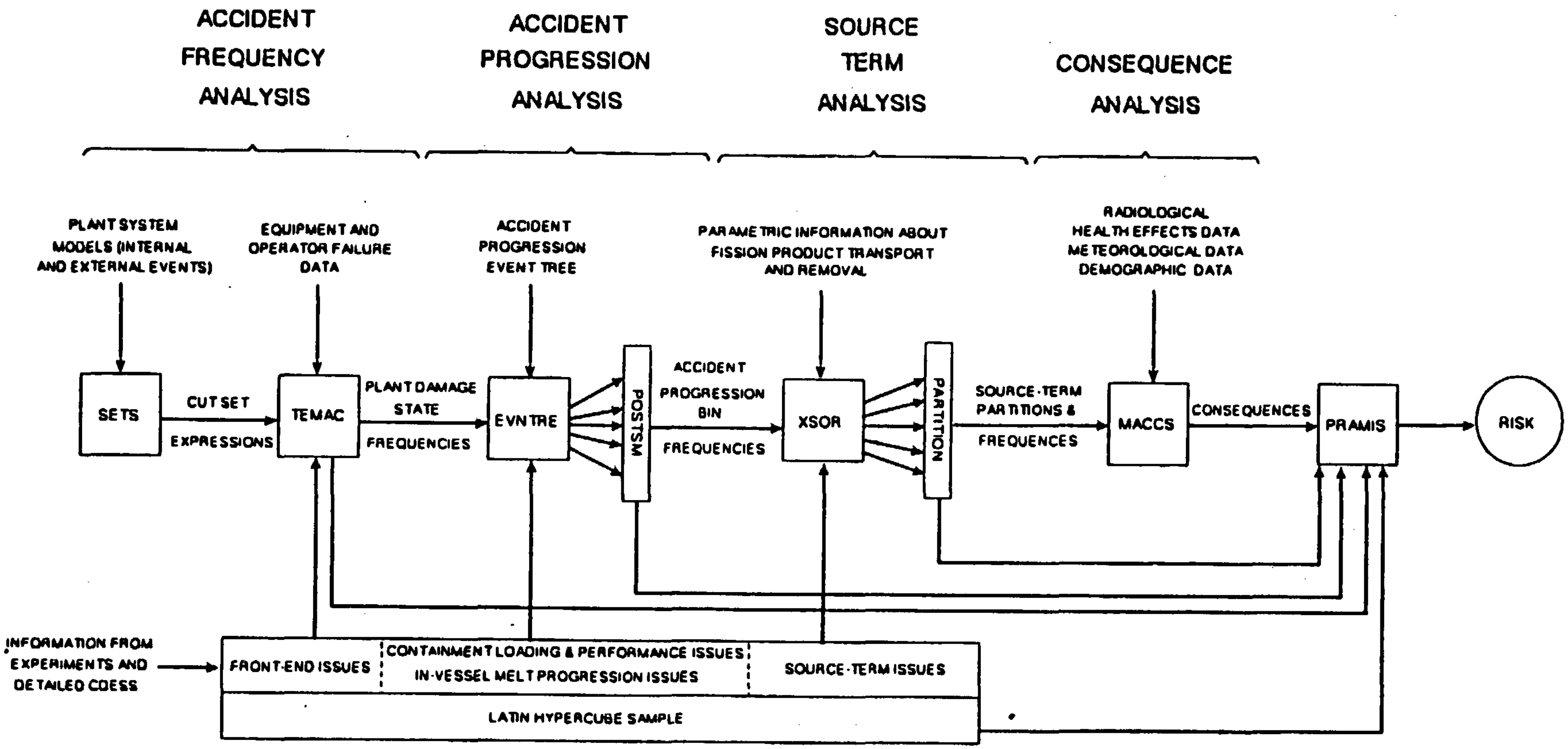


Figure S.1 Overview of Integrated Plant Analysis in NUREG-1150
(Reproduced from Volume 3)

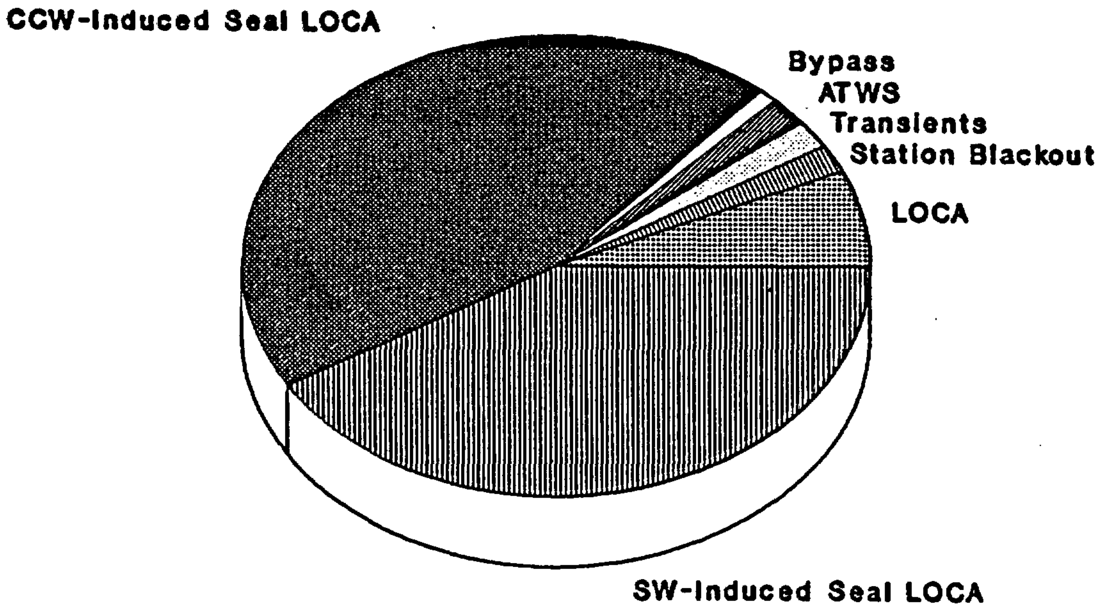
A source term is calculated for each APB with a non-zero conditional probability for each observation in the sample by ZISOR, a fast-running parametric computer code. ZISOR is not a detailed mechanistic model; it is not designed to be a realistic simulation of fission product transport, physics, and chemistry. Instead, ZISOR integrates the results of many detailed codes and the conclusions of many experts. Most of the parameters used to calculate fission product release fractions in ZISOR are sampled from distributions provided by an expert panel. Because of the large number of APBs, it is necessary to use a fast-executing code like ZISOR.

The number of APBs for which source terms are calculated is so large that it is not computationally practical to perform a consequence calculation for every source term. As a result, the source terms had to be combined into source term groups. Each source term group is a collection of source terms that result in similar consequences. The process of determining which APBs go to which source term group is called partitioning. It involves considering the potential of each source term group to cause early fatalities and latent cancer fatalities. The result of the source term calculation and subsequent partitioning is that each APB for each observation is assigned to a source term group.

A consequence analysis is performed for each source term group, generating both mean consequences and distributions of consequences. As each APB is assigned to a source term group, the consequence are known for each APB of each observation in the sample. The frequency of each PDS for each observation is known from the accident frequency analysis, and the conditional probability of each APB is determined for each PDS group for each observation in the sample, both frequency and consequences are determined. The risk analysis consists of assembling and analyzing all these separate estimates of offsite risk.

S.4 Results of the Accident Frequency Analysis

The accident frequency analysis for Zion is documented elsewhere.[2] This section only summarizes the results of the accident frequency analyses since these form the starting point for the analyses that are covered here. Figure S.2 displays the contributors to the mean core damage from accidents initiated by internal events at Zion. It is clear from Figure S.2 that accidents involving induced reactor coolant pump seal LOCAs dominate the estimated core damage frequency. After completion of the draft revision 1 analysis for Zion Unit 1, Commonwealth Edison made commitments to the NRC to make plant and procedure changes to address the major contributor to the core damage frequency. The impact of these changes would be a reduction in the core damage frequency of approximately 80%. With these changes seal LOCAs contribute significantly less to the core damage frequency. The risk estimates performed for Zion do not reflect these changes. The impact on mean risk of these changes was assessed as a sensitivity study.



Total Mean Core Damage Frequency: 3.4E-4

Note: see text for benefits of recent modifications:

S.2 Contributors to Mean Core Damage Frequency from Internal Events at Zion

S.5 Accident Progression Analysis

S.5.1 Description of the Accident Progression Analysis

The accident progression analysis is performed by means of a large and detailed event tree called the APET. This event tree forms a high-level model of the accident progression, including the response of the containment to the loads placed upon it. The APET is not meant to be a substitute for detailed, mechanistic computer simulation codes; rather, it is a framework for integrating the results of these codes together with experimental results and expert judgment. The detailed, mechanistic codes require too much computer time to be run for all the possible accident progression paths. Furthermore, no single available code treats all the important phenomena in a complete and thorough manner that is acceptable to all those knowledgeable in the field. Therefore, the results from these codes, as interpreted by experts, are summarized in an event tree. The resulting APET can be evaluated quickly by computer so that the full diversity of possible accident progressions can be considered and the uncertainty in the many phenomena involved.

The APET treats the progression of the accident from the onset of core damage to the core-concrete interaction (CCI). The APET accounts for all the events that may lead to the release of fission products due to the accident, even though some of the events may not occur until several days after the accident begins. The Zion APET consists of 72 questions, most of which have more than two branches. There are seven time periods considered in the tree. The recovery of offsite power is considered both before vessel failure as well as after vessel failure. The possibility of arresting the core degradation process before failure of the vessel is explicitly considered. Core damage arrest may occur following the recovery of offsite power or when depressurization of the RCS allows injection by an operating system (HPIS or LPIS) that previously could not function. Containment failure is considered at vessel breach (due to vessel blowdown, hydrogen combustion, direct containment heating, and steam explosions), after vessel failure (due to hydrogen combustion), and after several days (due to basemat meltthrough or eventual overpressure if containment cooling is not restored). Five mechanisms, four of them inadvertent, for depressurizing the vessel before failure are included in the APET.

The APET is so large and complex that it cannot be presented graphically and must be evaluated by computer. A computer code, EVNTRE, has been written for this purpose. In addition to evaluating the APET, EVNTRE sorts the myriad possible paths through the tree into a manageable number of outcomes called the APBs.

S.5.2 Results of the Accident Progression Analysis

Results of the accident progression analysis for internal initiators at Zion are summarized in Figures S.3 and S.4. Figure S.3 shows the mean distribution among the summary accident progression bins for the summary PDS groups. Technically, this figure displays the mean probability of a summary APB conditional on the occurrence of a PDS group. Since only mean values are shown, Figure S.3 gives no indication of the range of values encountered. Figure S.4 shows the distributions of the expected conditional probability for early containment failure (CF) given a PDS group. Early CF means CF at or before vessel breach (VB).

Figure S.3 indicates the mean probability of the possible outcomes of the accident progression analysis. The width of each box in the figure indicates how likely each accident progression outcome is for each type of accident. Except for the Bypass initiators, no failure of the containment is the most likely outcome for accidents initiated by internal events.

If CF does occur, late failure is more likely than failure at or before VB. Late failure may be due to hydrogen ignition some hours after VB, basemat meltthrough, or eventual overpressure after several days if CHR is not restored. Of these three late failure modes, basemat meltthrough is the most likely for internal initiators.

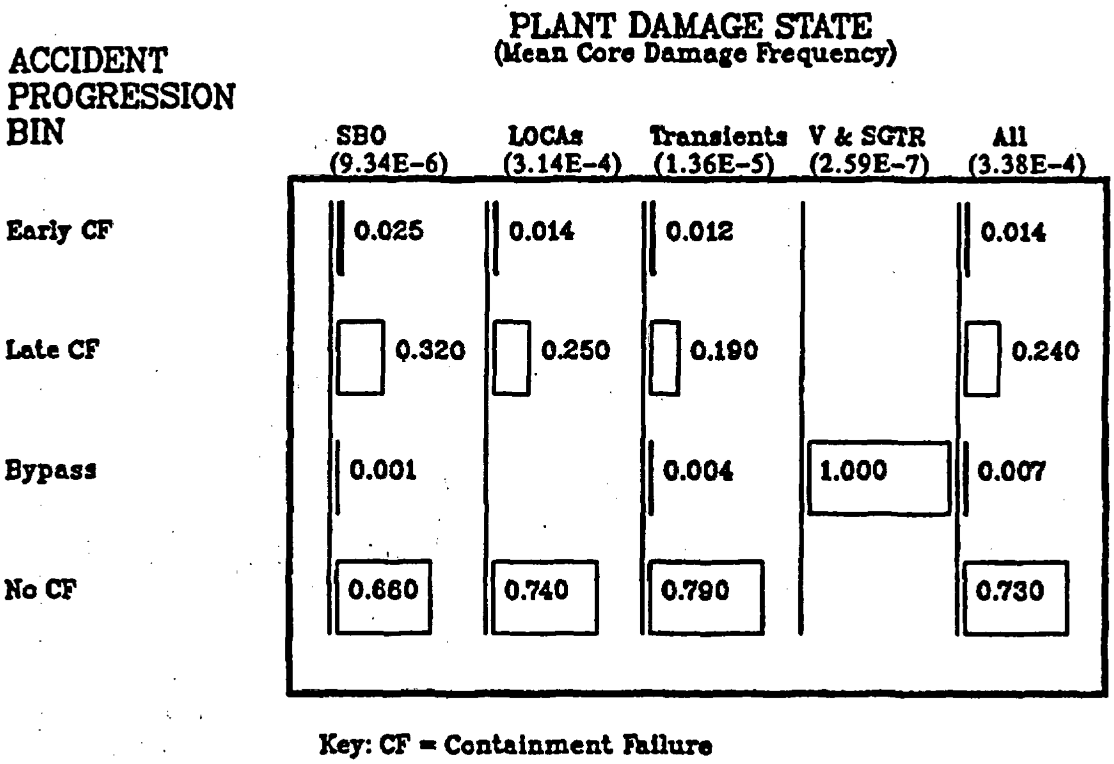
S.6 Source Term Analysis

S.6.1 Description of the Source Term Analysis

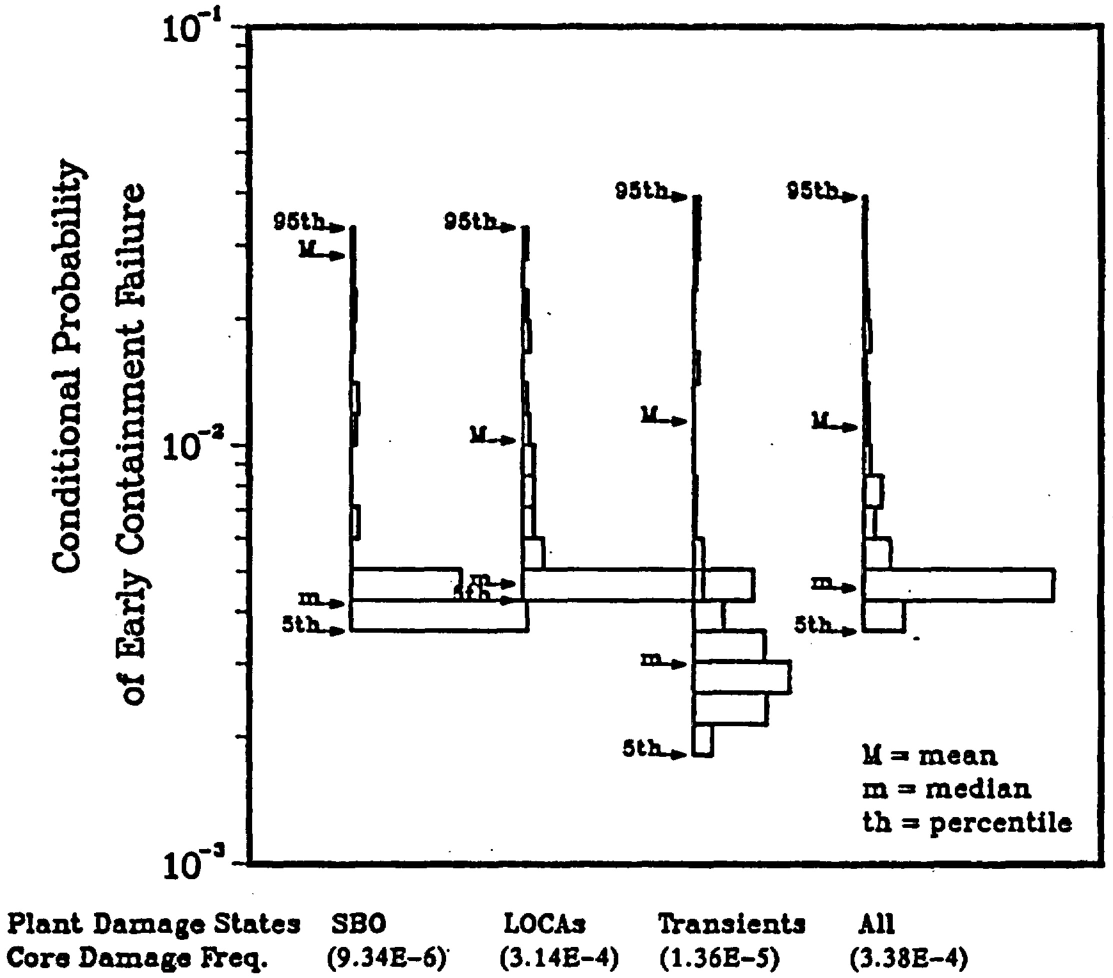
The source term for a given bin consists of the release fractions for the nine radionuclide classes for the early release and for the late release, and additional information about the timing of the releases, the energy associated with the releases, and the height of the releases. It includes the information required for the calculation of consequences in the succeeding analysis. A source term is calculated for each APB for each observation in the sample. The nine radionuclide classes are: inert gases, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium, and barium.

The source term analysis is performed by a relatively small computer code, ZISOR. The purpose of this code is not to calculate the behavior of the fission products from their chemical and physical properties and the flow and temperature conditions in the reactor and the containment. Instead, ZISOR provides a means of incorporating into the analysis the results of the more detailed codes that do consider these quantities. This approach is needed because the detailed codes require too many computer resources to be able to compute source terms for the numerous APBs and the 150 observations that result from the sampling approach used in NUREG-1150.

ZISOR is a fast-running, parametric computer code used to calculate the source terms for each APB for each observation for Zion. Since there are normally about a hundred bins for each observation, and 150 observations in the sample, the need for a source term calculation method that requires few computer resources for one evaluation is obvious. ZISOR provides a framework for synthesizing the results of experiments and mechanistic codes, as interpreted by experts in the field. The reason for "filtering" the detailed code results through the experts is that no code available treats all the phenomena in a manner generally acceptable to those knowledgeable in the field. Thus, the experts are used to extend the code results in areas where the codes are deficient and to judge the applicability of the model predictions. They also factor in the latest experimental results and modify the code results in areas where the codes are known or suspected of oversimplifying. Since the majority of the parameters used to compute the source term are derived from distributions determined by an expert panel, the dependence of ZISOR on various detailed codes reflects the preferences of the experts on the panel.



S.3 Conditional Probability of Accident Progression Bins at Zion



S.4 Conditional Probability Distributions for Early Containment Failure at Zion

It is not possible to perform a separate consequence calculation for each of the approximately 20,000 source terms computed for the Zion integrated risk analysis. Therefore, the interface between the source term analysis and the consequence analysis is formed by grouping the source terms into a much smaller number of source term groups. These groups are defined so that the source terms within them have similar properties, and a single consequence calculation is performed for the mean source term for each group. This grouping of the source terms is performed with the PARTITION program, and the process is referred to as "partitioning".

The partitioning process involves the following steps: definition of an early health effect weight (EH) for each source term, definition of a chronic health effect weight (CH) for each source term, subdivision (partitioning) of the source terms on the basis of EH and CH, a further subdivision on the basis of the time the evacuation starts relative to the start of the release, and calculation of frequency-weighted mean source terms.

The result of the partitioning process is that the source term for each accident progression bin is assigned to a source term group. In the risk computations, each accident progression bin is represented by the mean source term for the group to which it is assigned, and the consequences are calculated for that mean source term.

S.6.2 Results of the Source Term Analysis

When all the internally initiated accidents at Zion are considered together, plots of the type shown in Figures S.5 and S.6 are obtained. These plots show statistical measures of the 150 curves (one for each observation in the sample) that give the frequencies at which release fractions are exceeded. Figures S.5 and S.6 summarize the complementary cumulative distribution functions (CCDFs) for two representative radionuclide groups (iodine and strontium). The mean frequency of exceeding a release fraction of 0.10 for iodine is on the order of 4×10^{-6}/yr. The mean exceedance frequency for release of 0.10 of the core strontium is somewhat lower. The highest fractional releases are computed for early containment failures (refer to Figure S.7) and bypass accidents. The releases for late containment failures, most of which are basemat meltthrough, are quite small. Releases associated with no containment failure occur because of leakage paths and are also very small (refer to Figure S.8).

S.7 Consequence Analysis

S.7.1 Description of the Consequence Analysis

MACCS is used to calculate offsite consequences for each of the source term groups defined in the partitioning process. MACCS tracks the dispersion of the radioactive material in the atmosphere from the plant and computes its deposition on the ground. MACCS then calculates the effects of this radioactivity on the population and the environment. Doses and the ensuing health effects from 60 radionuclides are computed for the following pathways: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, inhalation of resuspended ground contamination, ingestion of contaminated water, and ingestion of contaminated food. MACCS treats atmospheric dispersion by using multiple,

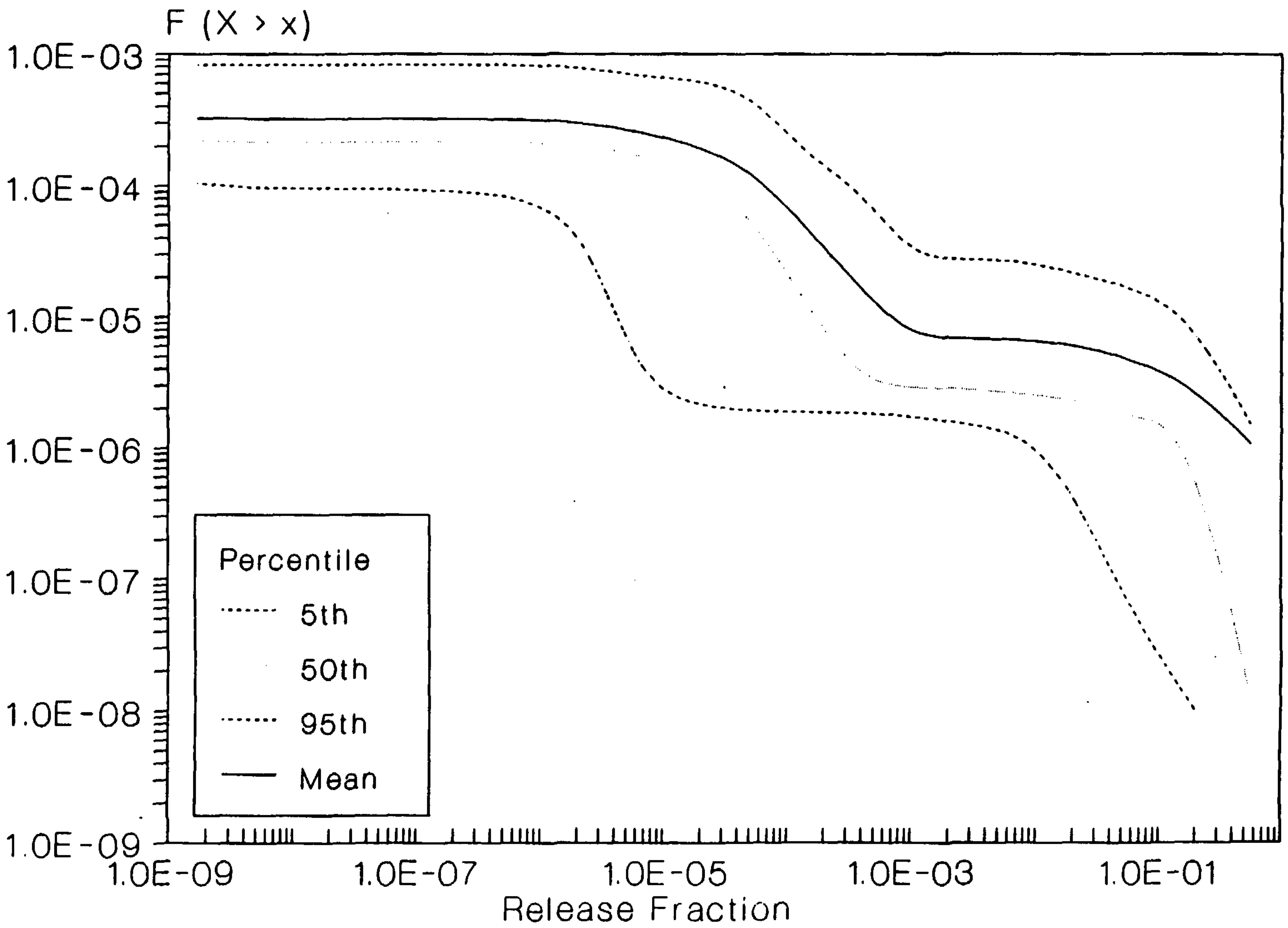


Figure S.5 CCDF for Iodine Release

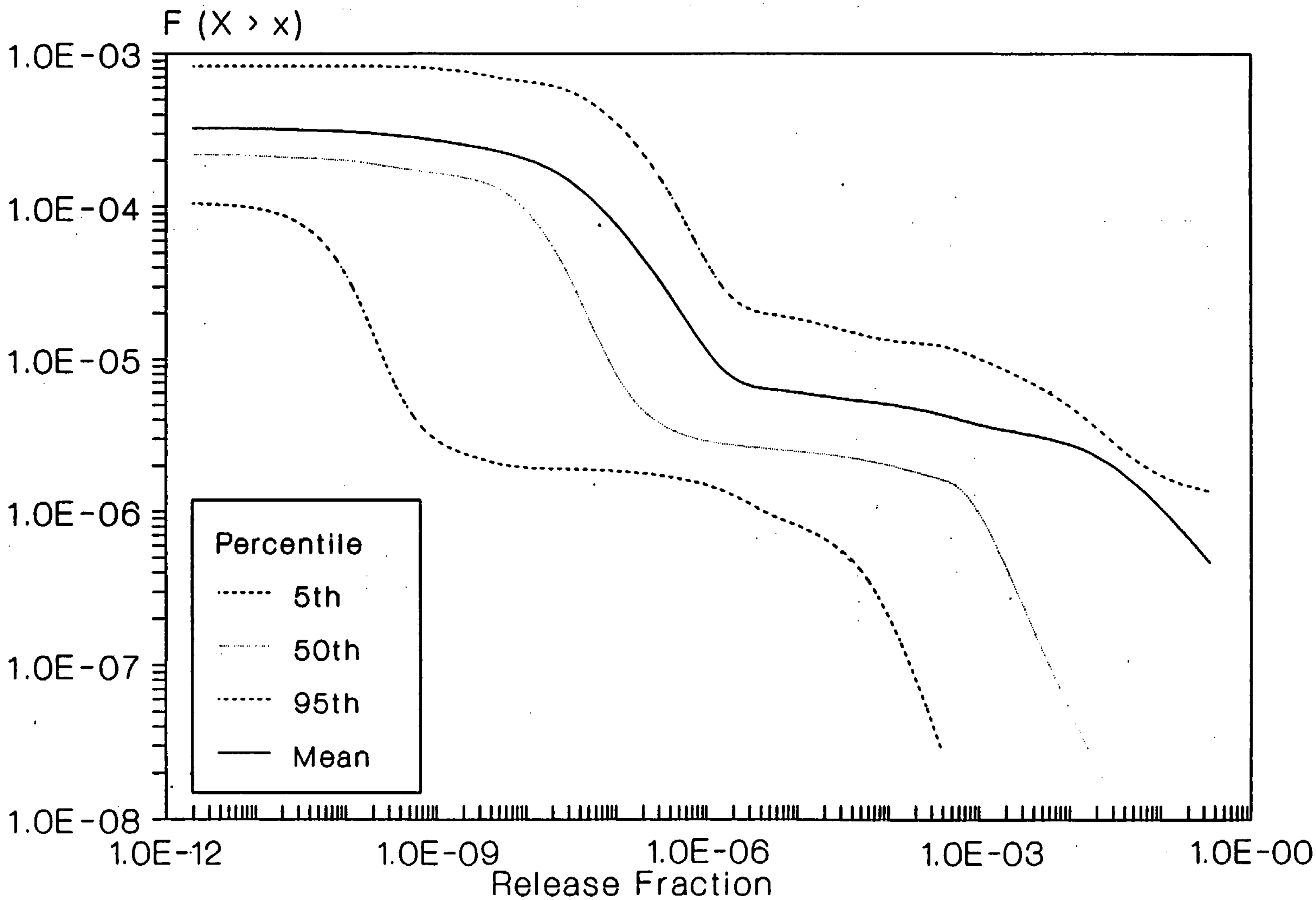


Figure S.6 CCDF for Strontium Release

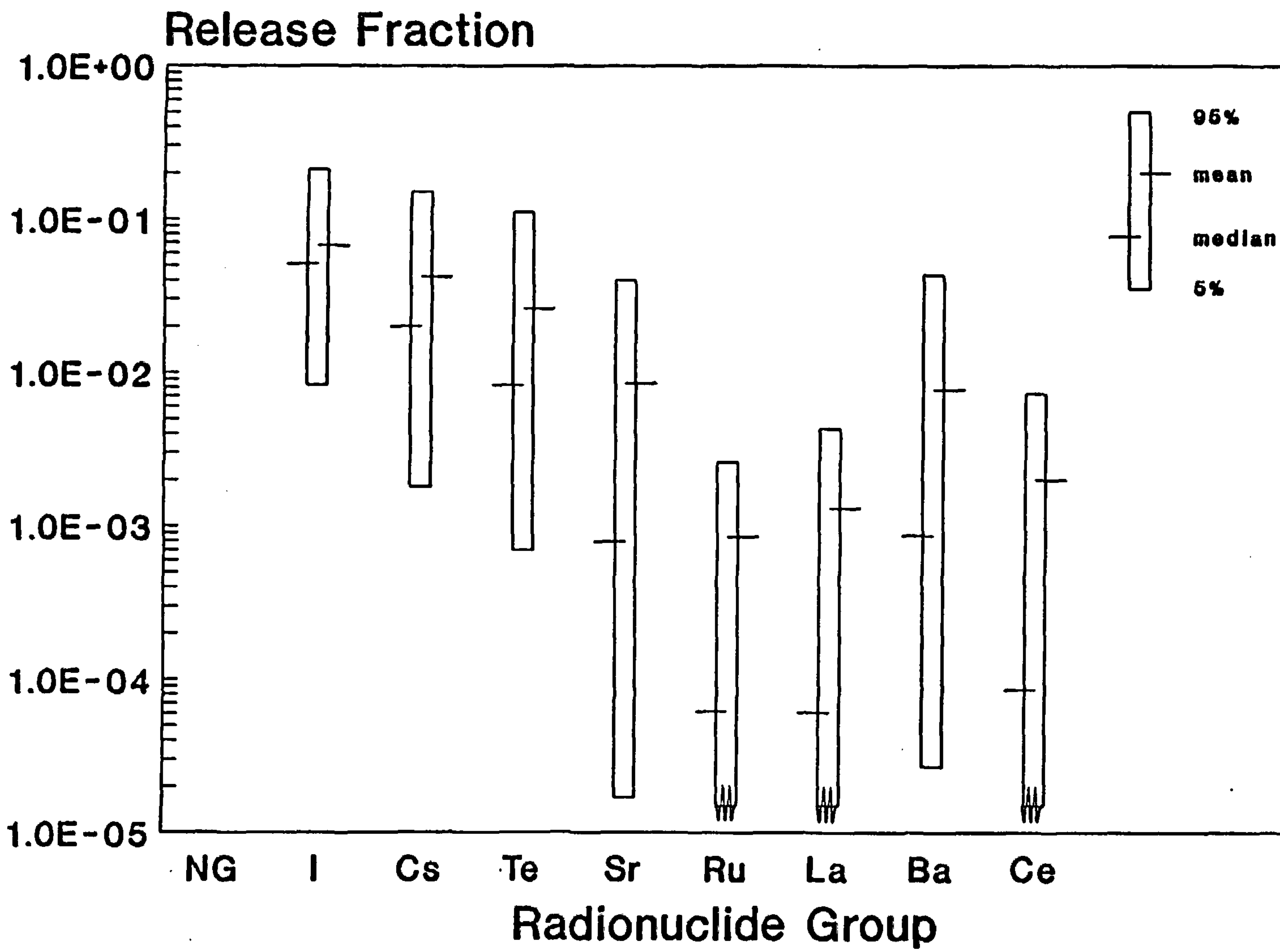


Figure S.7 Source Term Distribution for Early Failure at Zion

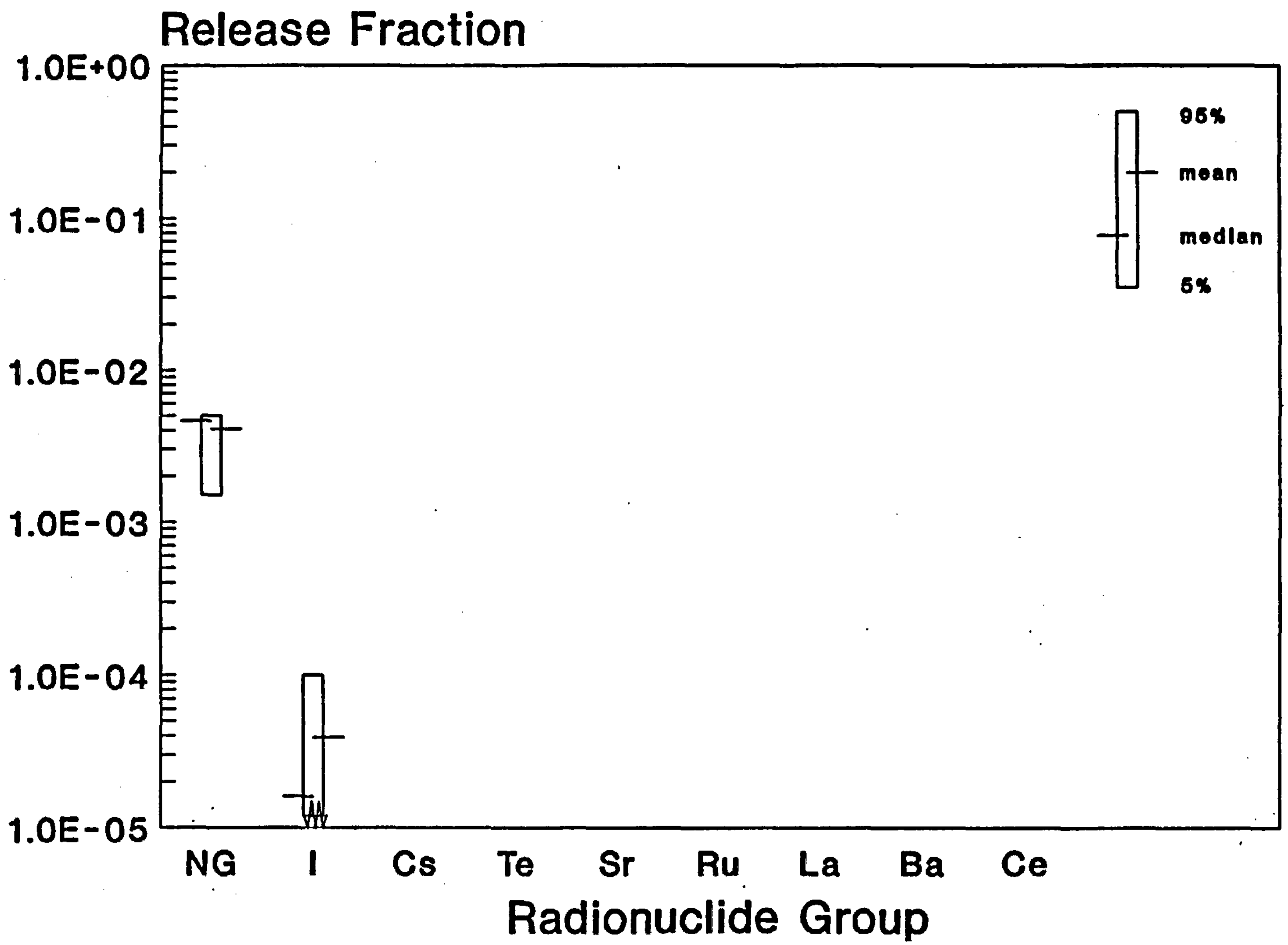


Figure S.8 Source Term Distributions for No Containment Failure at Zion straight-line Gaussian plumes. Each plume can have a different direction, duration, and initial radionuclide concentration. Cross-wind dispersion is treated by a multi-step function. Dry and wet deposition are treated as independent processes. The weather variability is treated by means of a stratified sampling process.

For early exposure, the following pathways are considered: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, and inhalation of resuspended ground contamination. For the long-term exposure, MACCS considers the following four pathways: groundshine, inhalation of resuspended ground contamination, ingestion of contaminated water, and ingestion of contaminated food. The direct exposure pathways, groundshine, and inhalation of resuspended ground contamination, produce doses in the population living in the area surrounding the plant. The indirect exposure pathways, ingestion of contaminated water and food, produce doses in those who ingest food or water emanating from the area around the accident site. The contamination of water bodies is estimated for the washoff of land-deposited material as well as direct deposition. The food pathway model includes direct deposition onto the crop species and uptake from the soil.

Both short-term and long-term mitigative measures are modeled in MACCS. Short-term actions include evacuation, sheltering, and emergency relocation out of the emergency planning zone. Long-term actions include relocation and restrictions on land use and crops. Relocation and land decontamination, interdiction, and condemnation are based on projected long-term doses from groundshine and the inhalation of resuspended radioactivity. The disposal of agricultural products and the removal of farmland from crop production are based on ground contamination criteria.

The health effects models link the dose received by an organ to morbidity or mortality. The models used in MACCS calculate both short-term and long-term effects to a number of organs.

Although the variables thought to be the largest contributors to the uncertainty in risk are sampled from distributions in the accident frequency, accident progression, and source term analyses, there is no analogous treatment of uncertainties in the consequence analysis. Variability in the weather is fully accounted for, but the uncertainty in other parameters, such as the dry deposition velocity or the evacuation rate, is not considered.

The MACCS consequence model calculates a large number of different consequence measures. This report gives results for the following six consequence measures: early fatalities, total latent cancer fatalities, population dose within 50 miles, population dose for the entire region, early fatality risk within 1 mile, and latent cancer fatality risk within 10 miles. For NUREG-1150, 99.5% of the population is assumed to evacuate and 0.5% of the population continues normal activity. For Zion, the evacuation delay time between warning and the beginning of evacuation is assumed to be 2.3 hours.

S.7.2 Results of the Consequence Analysis

The results presented in this section depend on the occurrence of a source term group. That is, if a release takes place with release fractions and other characteristics as defined by one of the source term groups, then the tables and figures in this section give the consequence expected. This section contains no indication at all about the frequency with which these consequences may be expected. Implicit in the results given in this section is that 0.5% of the population does not evacuate and that there is a 2.3 hour delay between the warning to evacuate and the actual start of the evacuation.

S.8 Integrated Risk Analysis

S.8.1 Determination of Risk

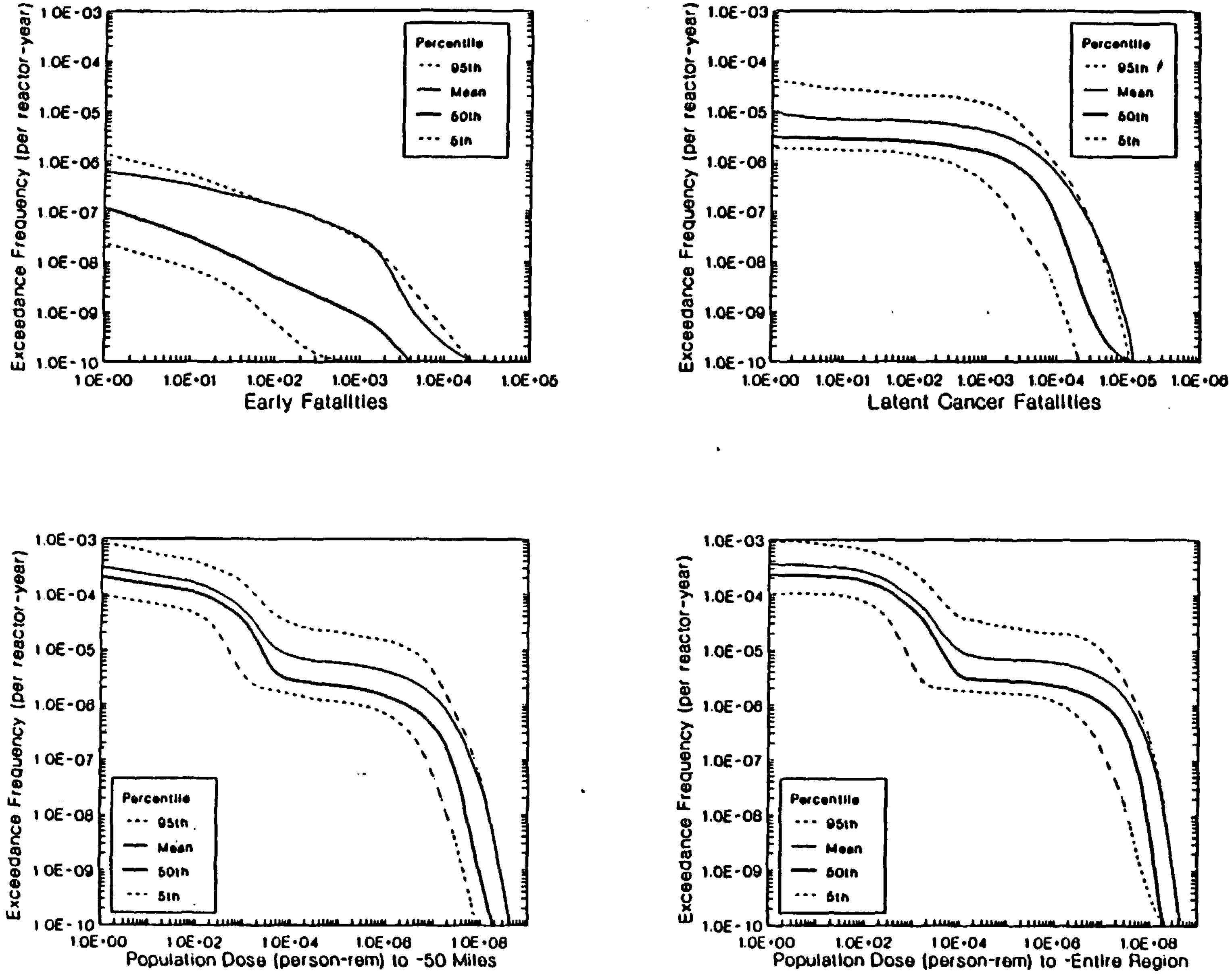
Risk is determined by bringing together the results of the four constituent analyses: the accident frequency analysis, the accident progression analysis, the source term analysis, and the consequence analysis. This process is described in general terms in Section S.2 of this summary, and in mathematical terms in Section 1.4 of this volume. Specifically, the accident frequency analysis produces a frequency for each PDS group for each observation, and the accident progression analysis results in a probability for each APB, conditional on the occurrence of the PDS group. The absolute frequency for each bin for each observation is obtained by summing the product of the PDS group frequency for that observation and the conditional probability for the APB for that observation over all the PDS group.

For each APB for each observation, a source term is calculated; this source term is then assigned to a source term group in the partitioning process. Then the consequences are computed for each source term group. The overall result of the source term calculation, the partitioning, and the consequence calculation is that a set of consequence values is identified with each APB for each observation. Because the absolute frequency of each APB is known from the accident frequency and accident progression results, both frequency and consequences are known for each APB. The risk analysis consists of assembling and analyzing all these separate estimates of offsite risk.

S.8.2 Results of the Risk Analysis

Figure S.9 shows the basic results of the integrated risk analysis for internal initiators at Zion. This figure shows four statistical measures of the families of the CCDFs for early fatalities, latent cancer fatalities and population dose within 50 miles and for the entire region. The CCDFs display the relationship between the frequency of the consequence and the magnitude of the consequence. Since there are 150 observations in the sample for Zion, the actual risk results at the most basic level are 150 CCDFs for each consequence measure. Figure S.9 displays the 5th percentile, median, mean, and 95th percentile for these 150 curves, and shows the relationship between the magnitude of the consequence and the frequency at which the consequence is exceeded, as well as the variation in that relationship.

The 5th and 95th percentile curves provide an indication of the spread between observations, which is often large. This spread is due to uncertainty in the sampled variables, and not to differences in the weather at the time of the accident. As the magnitude of the consequence measure increases, the mean curve typically approaches or exceeds the 95th percentile curve. This results when the mean is dominated by a few observations, which often happens for large values of the consequences. Only a few observations have nonzero exceedance



Note: As discussed in Reference 4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses

S.9 Frequency Distributions of Offsite Consequence Measures at Zion - Internal Initiators determined for each observation in the sample by summing the product of the frequencies for these large consequences. Taken as a whole, the results in Figure S.9 indicate that large consequences are relatively unlikely to occur.

Although the CCDFs convey the most information about the offsite risk, summary measures are also useful. Such a summary value, denoted expected risk, may be frequencies and consequences for all the points used to construct the CCDF. This has the effect of averaging over the different weather states as well as over the different types of accidents that can occur. Since the complete analysis consisted of a sample of 150 observations, there are 150 values of expected risk for each consequence measure. These 150 values may be ranked and plotted as histograms, which is done in Figures S.10 - S.12.

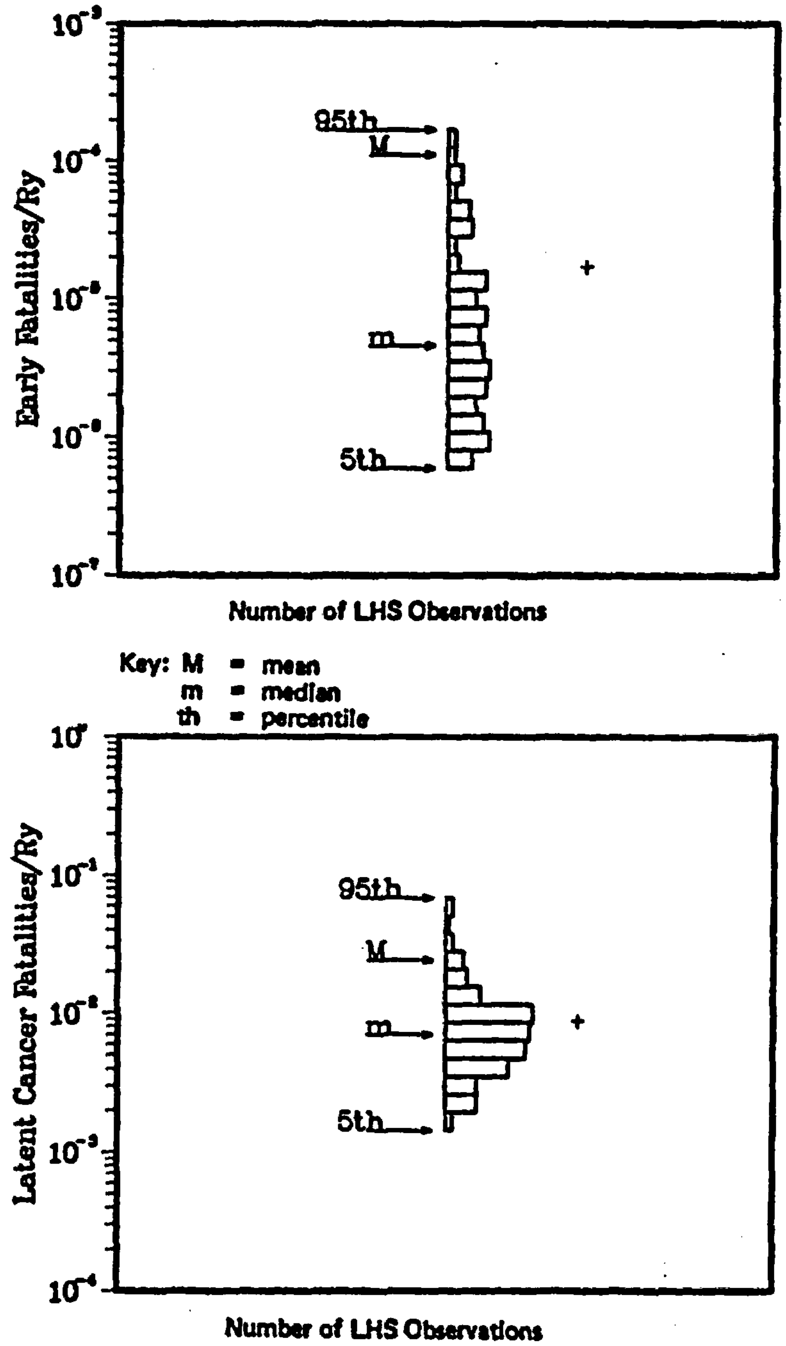
The plots in Figures S.10 - S.12 show the variation in the expected risk for internal initiators for four consequence measures. Where the mean is close to the 95th percentile, a relatively small number of observations dominate the mean value. This is more likely to occur for the early fatality consequence measures than for the latent cancer fatality or population dose consequence measures due to the threshold effect for early fatalities.

The safety goals are written in terms of individual fatality risks. The plots in Figure S.12 for individual early fatality risk and individual latent cancer fatality risk show that essentially the entire risk distribution for Zion falls below the safety goals.

S.8.3 Important Contributors to Risk

Figure S.13 shows the major plant damage state contributors to the mean early and latent cancer fatality risk estimates for Zion. The major accident progression bin contributors to the same risk estimates are given in Figure S.14. Figure S.13 indicates that induced seal LOCAs are the major contributors to both the early and latent risk estimates. Figure S.14 demonstrates that accident progression bins resulting in early containment failure dominate both risk measures.

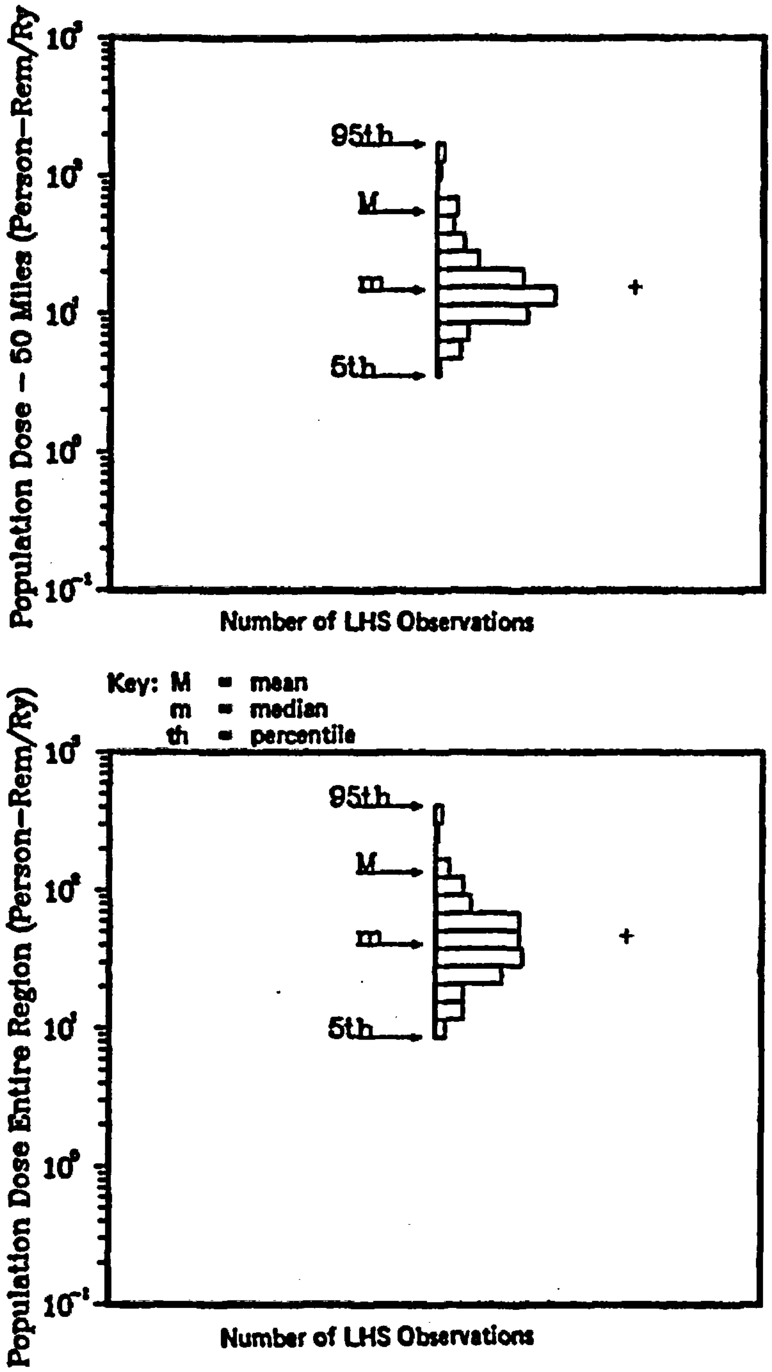
Accidents involving induced reactor coolant pump seal LOCAs dominate the estimated core damage frequency. After completion of the draft revision 1 analysis for Zion Unit 1, Commonwealth Edison made commitments to the NRC to make plant and procedure changes to address the major contributor to the core damage frequency. The impact of these changes would be a reduction in the core damage frequency of approximately 80%. With these changes seal LOCAs contribute significantly less to the core damage frequency. The Zion risk estimates reported in this volume do not reflect these changes. However, a sensitivity study was performed to assess the impact of the changes in the mean Zion risk estimates. The result of the sensivity analysis was a reduction in the frequency of those plant damage states involving CCW and SW induced seal LOCAs. This inturn significantly reduced all of the mean risk measures as shown in Figures S.10 - S.12.



Notes: As discussed in Reference 4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

"+" shows recalculated mean value based on plant modifications discussed in text.

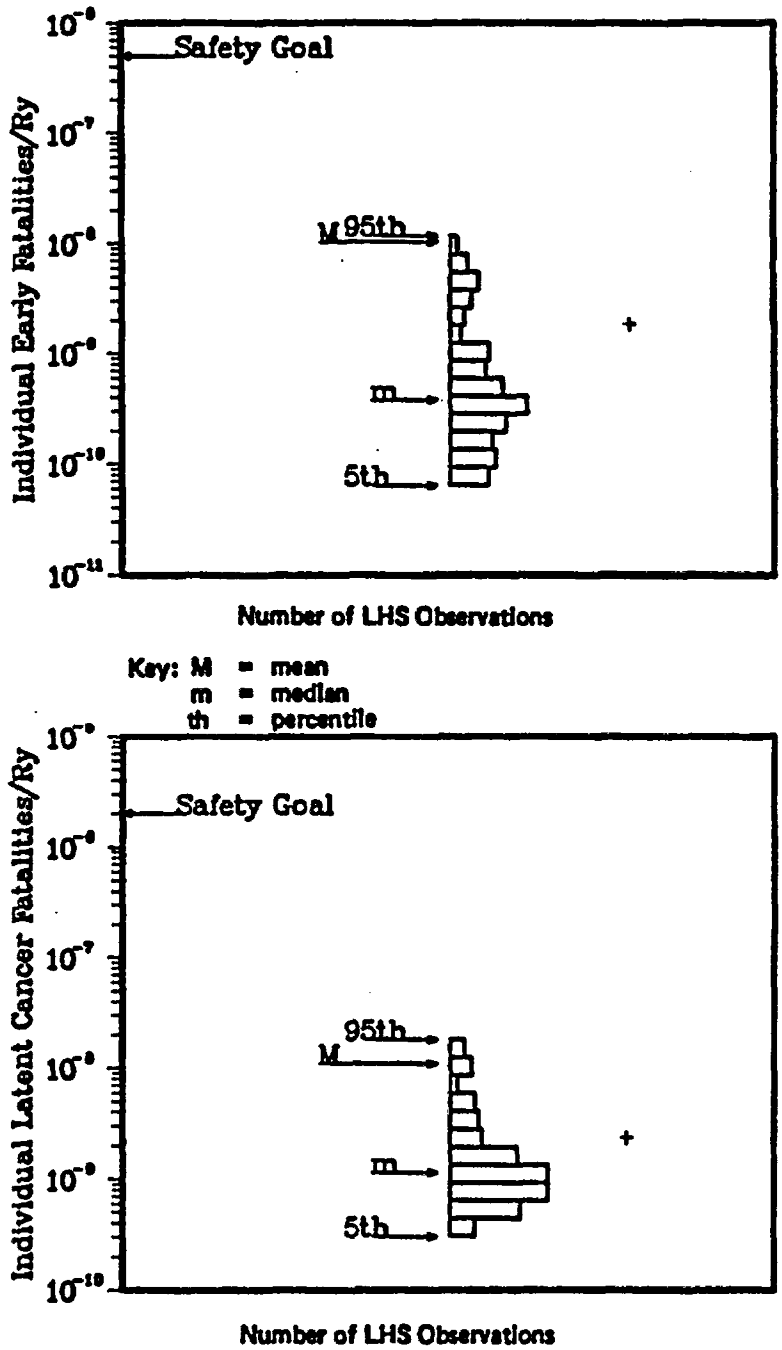
S.10 Early and Latent Cancer Fatality Risks at Zion - Internal Initiators



Notes: As discussed in Reference 4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

"+" shows recalculated mean value based on plant modifications discussed in text.

S.11 Population Dose Risks at Zion - Internal Initiators



Notes: As discussed in Reference 4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of other health effects not studied in the risk analyses.

"+" shows recalculated mean value based on plant modifications discussed in text.

S.12 Individual Early and Latent Cancer Fatality Risks at Zion - Internal Initiators

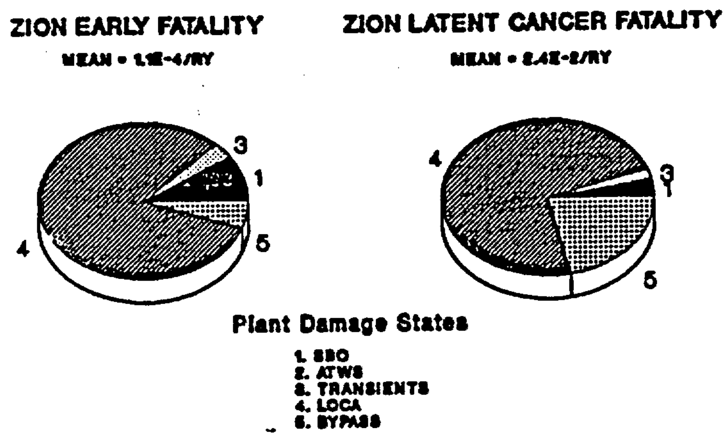


Figure S.13 Major Contributors (Plant Damage States) to Mean Early and Latent Cancer Fatality Risks at Zion-Internal Initiators

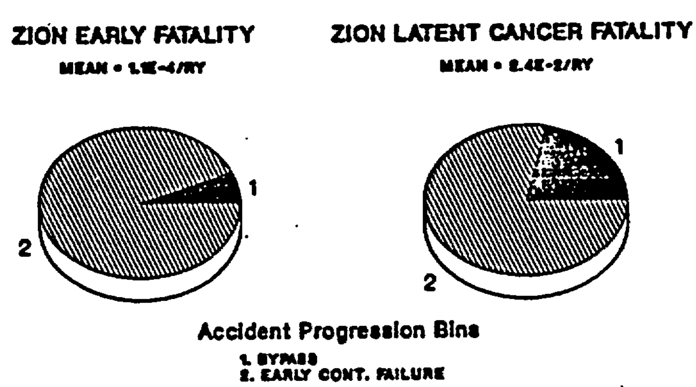


Figure S.14 Major Contributors (Accident Progression Bins) to Mean Early and Latent Cancer Fatality Risks at Zion-Internal Initiators

Table S.1
Risk Importance Analysis: Partial Rank Correlation Coefficient

Variable Description	Early Fatalities	Early Injuries	Probability of One Fatality	Latent Cancer Fatalities (50 Mi)	Latent Cancer Fatalities (Entire Region)	Safety Goals: Individual Early Fatalities	Safety Goals: Societal Individual Latent Cancer Fatalities	Pop. Dose (50 Mi)
CCWS Initiating Event		0.49	0.55					
CCWS Hardware Recovery	0.36	0.58	0.67	0.40	0.45	0.37	0.49	0.45
Early Containment[1] Failure	0.45	0.56	0.62	0.60	0.62	0.50	0.56	0.62
FCOR[2]	0.42	0.42	0.36	0.56	0.56	0.46	0.48	0.55
FCONV[3]	0.40					0.41		
FISG*FOSG[4]	0.45	0.35		0.74	0.68	0.47	0.47	0.69
R^2	0.67	0.74	0.78	0.79	0.78	0.70	0.71	0.78

Notes:
1 - Early containment failure was most influenced by steam explosions and direct containment heating.
2 - Fraction of initial inventory of nuclide group release from the fuel in-vessel.
3 - Containment transport fraction for releases prior to or at vessel breach.
4 - Fraction of fuel release to the environment through the steam generator.

S.8.4 Important Contributors to the Uncertainty in Risk

In order to identify important contributors to the uncertainty in risk estimation, a regression analysis was performed using the Partial Rank Correlation Coefficient (PRCC) as a measure of importance. Table S.1 summarizes the results of the regression analysis. Six variables were identified as important contributors to the uncertainties of the various risk estimations. These seven variables were selected based on calculated PRCC values greater than 0.3 for several risk indices.

Two accident frequency issues were identified as important. The frequencies of these issues are directly related to the total core damage frequency. The failure of CCWS means that there is no seal cooling nor emergency coolant injection. Late AC power recovery negatively contributes to the latent cancer fatalities and the total cost estimation. One containment issue was identified as important. The frequency of in-vessel steam explosion (alpha-mode containment failure) was found to be an important contributor to the uncertainty of all the risk indices. There were three source term parameters (or source term issues) identified as being important in terms of their contribution to the risk uncertainties.

S.9 Insights and Conclusions

Reactor Coolant Pump Seal LOCA. Accidents involving induced reactor coolant pump seal LOCAs dominate the estimated core damage frequency. After completion of the draft revision 1 analysis for Zion Unit 1, Commonwealth Edison made commitments to the NRC to make plant and procedure changes to address the major contributor to the core damage frequency. The impact of these changes would be a significant reduction in the core damage frequency. The Zion risk estimates reported in this volume do not reflect these changes. However, a sensitivity study was performed to assess the impact of the changes in the mean Zion risk estimates. The result of the sensitivity analysis was a reduction in all of the mean risk measures as indicated in Figures S-10 to S-12.

Depressurization of the RCS. Depressurization of the RCS before the vessel fails is important in reducing the loads placed upon the containment at vessel breach and in arresting core damage before VB. For accidents in which the RCS is at the PORV setpoint pressure during core degradation, the effective mechanisms for pressure reduction are temperature-induced failure of the hot leg or surge line, temperature-induced failure of the RCP seals, and the sticking open of the PORVs.

Containment Failure. If a core damage accident proceeds to the point where the lower head of the reactor vessel fails, the containment is unlikely to fail at this time. This is partially due to the depressurization of the RCS before vessel failure and partially due to the strength of the Zion containment relative to the loads expected. If the containment does fail, it is more likely to fail many hours after VB than at VB. The mode and time of failure depends upon the availability of CHR. If CHR is recovered within a day or so, basemat meltthrough is the most probable failure mode. If CHR is not recovered within days, an overpressure failure is possible.

Bypass Accidents. Bypass accidents can potentially result in a large early release and in the Surry Analysis (refer to Volume 3 of this report) were found to be important contributors to risk. However, because of the low frequency of bypass events when compared with the core damage frequency they were not found to be dominant contributors to risk at Zion.

Fission Product Releases. There is considerable uncertainty in the release fractions for all types of accidents. For most accidents, the central portions of the release fraction distributions are below most release fraction estimates made several years ago. While the upper portions of the release fraction distributions are comparable with the values of the RSS,[3] many of these distributions now extend to release fractions several orders of magnitude lower than those of the RSS.

Uncertainty in Risk. Considerable uncertainty is associated with the risk estimates produced in this analysis. The largest contributors to this uncertainty are the initiating events, early containment failure modes and the uncertainty in some of the parameters that determine the magnitude of the fission product release to the environment.

Comparison with the Safety Goals. For both distributions for individual fatality probability, the 95th percentile value for annual risk falls more than an order of magnitude below the safety goal.

References

1. USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Final Report, NUREG-1150, December 1990.

2. M. B. Sattison and K. W. Hall, "Analysis of Core Damage Frequency: Zion, Unit 1," Idaho National Engineering Laboratory, NUREG/CR-4550, Volume 7, Revision 1, EGG-2554, May 1990.

3. USNRC, "Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG 75/014), October 1975.

4. H. J. C. Kouts, et al., "Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)," NUREG-1420, August 1990.

1. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) has recently completed a major study to provide a current characterization of severe accident risks from light water reactors. The characterization was derived from the analysis of five plants. The report of that work, NUREG-1150[1], is based on extensive investigations by NRC contractors. Several series of reports document these analyses and their results in detail.

The risk assessments can generally be characterized as consisting of four analysis steps, an integration step, and an uncertainty step.

> Systems analysis, the determination of the likelihood and nature of accidents that result in the onset of core damage;
>
> Accident progression and containment analysis, an investigation of the core damage process both in and outside the reactor vessel and the resultant impact on the containment;
>
> Source term analysis, an estimation of the radionuclide releases associated with the progression of the accident; and
>
> Consequence analysis, the calculation of the offsite consequences in terms of health effects and financial impact.
>
> Risk integration, the combination of the outputs of the previous tasks into an overall expression of risk.
>
> Estimate of the uncertainty in the risk calculation due to uncertainty in the knowledge of important physical and chemical phenomena.

This report is one of seven volumes of the NUREG/CR-4551 series which explain the supporting analyses for the last five items listed above, covering the progression of the accident once core damage is initiated, through to an integrated estimate of overall risk and uncertainty in risk. This particular volume describes the inputs used in these analyses and the results obtained for one of the five nuclear power plants analyzed in NUREG-1150. The plant under consideration in this report is the Zion Nuclear Power Plant Unit 1. The methods utilized in these analyses are described in detail in Volume 1 of this report and are only briefly discussed here.

1.1 Background and Objectives of NUREG-1150

Assessment of risk from the operation of nuclear power plants, involves determination of the likelihood of various accident sequences and their potential off-site consequences. In 1975, the NRC completed the first comprehensive study of the probabilities and consequences of core meltdown accidents--the Reactor Safety Study (RSS)[2]. This report showed that the probabilities of such accidents were higher than previously believed, but that the consequences were significantly lower. The product of probability and consequence--a measure of the risk of core melt accidents--was estimated to be quite low. Since that time, many risk assessments of specific plants have been performed. In general, each of these has progressively reflected at least some of the advances that have been made in the ability to predict accident behavior.

In order to investigate the significance of more recent developments in a comprehensive fashion, it was concluded that the current efforts of research programs being sponsored by the NRC should be coalesced to produce an updated representation of risk for operating nuclear power plants; this led to the formulation of NUREG-1150 which gives a detailed analysis for five such plants. Zion is one of the five plants that has undergone detailed reanalysis, carried out by Brookhaven National Laboratory (BNL), as part of this program. The work on the other four plants: Surry, Grand Gulf, Peach Bottom, and Sequoyah, is being carried out at Sandia National Laboratory (SNL).

The overall objectives of the NUREG-1150 program are:

- To provide a current assessment of the severe accident risks of five nuclear power plants of different design, which:

    Provides a snapshot of risks reflecting plant design and operational characteristics, related failure data, and severe accident phenomenological information available as of March 1988;

    Updates the estimates of the NRC's 1975 risk assessment, the Reactor Safety Study[2];

    Includes quantitative estimates of risk uncertainty, in response to the principal criticism of the Reactor Safety Study; and

    Identifies plant-specific risk vulnerabilities for the five studied plants, supporting the development of the NRC's individual plant examination (IPE) process;

- To summarize the perspectives gained in performing these risk analyses, with respect to:

    Issues significant to severe accident frequencies, containment performance, and risks;

    Risk-significant uncertainties that may merit further research;

    Comparisons with NRC's safety goals; and

    The potential benefits of a severe accident management program in reducing accident frequencies; and

- To provide a set of probabilistic risk assessment (PRA) methods for the prioritization of potential safety issues and related research.

These objectives required special considerations in the selection and development of the analysis methods. This report describes those special considerations and the solutions implemented in the analyses supporting NUREG-1150.

1.2 Overview of the Zion Plant, Unit 1

Zion, Unit 1 is one of two 1050-MW (net) reactors operated by the Commonwealth Edison Company. The site for the Zion station[3] is located on the western shore of Lake Michigan, on the outskirts of the city of Zion and is about 64 km (40 mi) north of Chicago, Illinois.

The nuclear steam supply system for each of the Zion units is a four loop Westinghouse pressurized water reactor (PWR). Each reactor is rated at 3,250-MW (thermal). The emergency core cooling systems (ECCSs), which are located in the auxiliary building, are totally independent for each of the two units and consist of redundant high pressure injection trains, redundant low pressure injection trains, and passive accumulators for each unit. Hot leg as well as cold leg injection capability exists. The ECCS takes suction from the containment sump through special pipes to the inlet of the low pressure injection pumps during recirculation.

The plant auxiliary cooling systems consist of a shared component cooling water system (a closed system) and a shared service water system. The auxiliary feedwater system (AFWS), serving the secondary side of the steam generators, is separate for each unit. Each unit has three pumping trains, each capable of feeding all four steam generators. Two of the trains are fed by separate, redundant, 100% capacity, motor driven pumps, while the third train is fed by a redundant 200% capacity steam turbine driven pump.

Electrical power is supplied through multiple offsite power sources. Backup diesel generators are available for safety related loads in the event that offsite power is lost. Batteries are available for supplying DC power in the event of such a loss. The diesel generator consists of five machines, two per unit, with the fifth being a swing diesel capable of tying into a third bus on either unit as demand arises. Safeguards actuation systems consist generally of standard Westinghouse logic networks with sequential diesel generator loading.

The balance of plant equipment is not unique from a safety standpoint. The turbine-generators are Westinghouse tandem compound units. Six stages of feedwater heating are provided. Each unit uses a single pass, deaerating type condenser. Once through cooling is provided using Lake Michigan as a source of cooling water.

Each reactor system is housed in an individual containment building. These structures consist of post tensioned concrete shells over 0.006 m (1/4-in) thick steel liners. Figure 1.1 shows a section through one of the Zion containments. The volume is approximately 7.7×10^4 m^3 (2,700,000 ft^3, and the design pressure is 0.43 MPa (62 psia). Each containment is served by both fan cooler and containment spray systems. These systems provide redundant and diverse containment heat removal capability. There are a total of five fan cooler units per containment operating in parallel, with each one being rated at one-third the required capacity for accident conditions. During normal operation a maximum of four units are required to

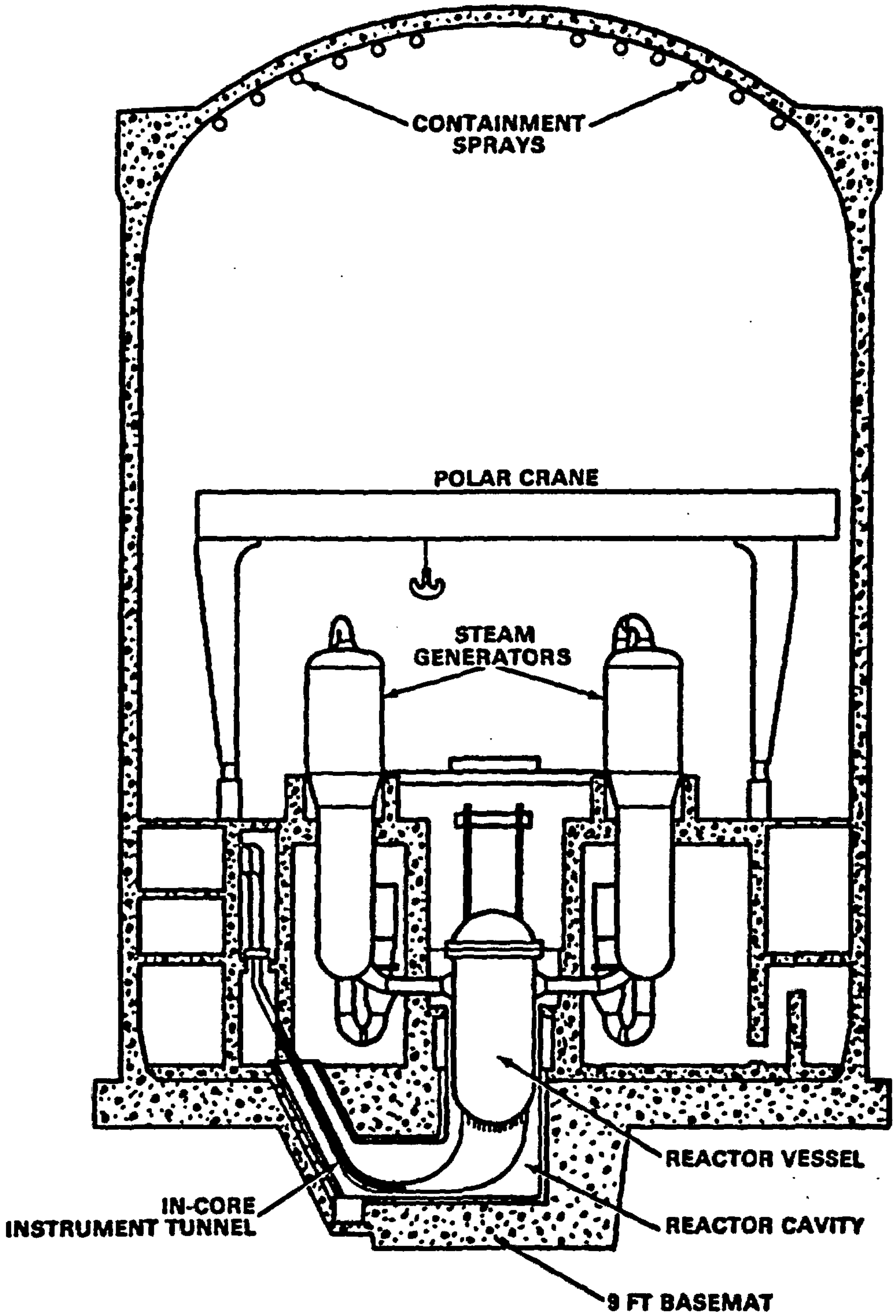


Figure 1.1 Cross-Section of the Zion Containment

remove the design heat load. For post accident operation a minimum of three units must function to satisfy safeguards requirements. Since the units are engineered safeguards they are located outside of the missile shield. The containment spray system is divided into three independent 100% capacity subsystems with no common headers. Of the three spray pumps, two are motor driven and the third is diesel driven. All three pumps take suction from the refueling water storage tank (RWST) and discharge into the spray rings located around the inside of the containment dome.

Section 2.1 of this volume contains more detail on the plant's features important to the progression of the accident and to the containment's performance.

1.3 Changes since the Draft Report

In 1980 the Zion Probabilistic Safety Study[4] (ZPSS) was initiated to establish the level of risk associated with the operation of this facility, which is located near a high population center. The ZPSS was a significant advance from the RSS in terms of assessing core meltdown phenomena and containment response, however, the source terms were still based on the RSS methodology.

In February 1987, a preliminary version of the Zion study performed at BNL was published for comment.[5] This analysis provided one of the technical bases for the first draft of NUREG-1150. Since that time, numerous comments both upon NUREG-1150 methods in general, and upon specifics of the Zion models have been received from industry, the public, and various peer review panels. In the light of methodological and modelling deficiencies identified in the draft analysis by both internal and external review, extensive modifications to the preliminary Zion study were undertaken in support of the second draft of NUREG-1150. The integrated analysis of Zion is a collaborative effort involving BNL and Idaho National Engineering Laboratory (INEL).

The preliminary Zion analysis for the first draft of NUREG-1150 differed in various respects from the first draft versions of the remaining reference plant studies. In particular, the accident sequence analysis was considerably truncated and the accident progression evaluation did not benefit directly from the process of expert opinion elicitation. The approach adopted was one in which the expert-based elicitation information generated for the analysis of Surry (another reference plant studied for NUREG-1150) was adapted, through scaling considerations, to provide input to the Zion study. The shortcomings of such an approach, both in general technical terms and for specific calculable issues, were recognized early by the analysts at BNL and confirmed by external review.

The primary improvements to the draft analysis are: a) greater extent and detail of the systems analysis, b) the generation of a plant-specific phenomenological data base by implementation of mechanistic severe accident codes, c) the formal elicitation of expert judgment on Zion-specific issues and, d) improvement of the methodological framework for the probabilistic propagation of uncertainties.

Modification of the Zion systems analysis was performed at INEL and was based upon reanalysis of the 100 frequency-dominant sequences as identified in the ZPSS[4], upon modelling modifications proposed in the SNL reviews ofthe ZPSS[6,7] upon current plant emergency procedures and upon integration of judgmental information generated on specific uncertainty issues as part of the expert elicitation and internal elicitation processes. The Zion system analysis is described in Volume 7 of NUREG/CR-4550.[8]

A dominant contributor to risk uncertainty, as identified in the draft Zion study, was the issue of pressure loads to the containment building at the time of vessel breach. To better characterize the associated uncertainties, a series of calculations, using mechanistic severe accident analysis codes, was performed at BNL to estimate the potential pressure loads associated with phenomena such as direct containment heating and hydrogen deflagration. These calculations are described in Reference 9 and were made available to the containment loads expert panel to help them characterize the loads at reactor pressure vessel (RPV) failure.

Quantification. A major change since the previous analyses is the expert elicitation process used to quantify variables and parameters thought to be large contributors to the uncertainty in risk. The process of eliciting expert opinion is a cornerstone of the NUREG-1150 uncertainty analysis methodology. The elicitation process relative to a given 'uncertainty issue' results ultimately in a probabilistic characterization of uncertainty as to the magnitudes of some family of input parameters to the risk models. For the Zion study, 13 such issues were identified covering a broad range of phenomenological areas including system reliability, and source term evaluation.

To ensure that expert opinion was obtained in a manner consistent with the state of the art in this area, specialists in the process of obtaining expert judgments in an unbiased fashion were involved in designing the elicitation process, explaining it to the experts, and training them in the methods used. The experts were given several months between the meeting at which the problem was defined and the meeting at which their opinions were elicited so that they could review the literature, discuss the problem with colleagues, and perform independent analyses. The results of the elicitation of each expert were carefully recorded, and the reasoning of each expert and the process by which their individual conclusions were aggregated into the final distribution are thoroughly documented.

Accident Progression Analysis. Not only was the Accident Progression Event Tree (APET) for Zion completely rewritten for this analysis (following the Surry APET) for this analysis, but the capabilities of EVNTRE, the code that evaluates the APET, were considerably expanded. The major improvements to EVNTRE were the ability to utilize user functions and the ability to treat continuous distributions. A user function is a FORTRAN subprogram which is linked with the EVNTRE code. When referenced in the APET, the user function is evaluated to perform calculations too complex to be handled directly in the APET. In the current Zion APET, the user function is called to determine the mode of containment failure and to compute the pressure rise in containment due to hydrogen deflagrations. These problems were handled in a much simpler fashion in the previous analysis. The current method explicitly treats the failure modes due to pressure rises that are fast with respect to the depressurization rates from small failures of the containment.

The event tree used for the analysis for the 1987 draft of NUREG-1150 could only treat discrete distributions. For example, for the containment failure pressure, only values of 110, 149, 175, and 275 psia were possible. In the analysis reported here, a continuous distribution is used for containment failure pressure, so the values are not constrained to these four values. Use of continuous distributions removes a significant constraint from the expert elicitations and better represents uncertainties. Continuous distributions were also used in the parametric evaluation of source terms.

Another major change in the accident progression analysis is in the binning or grouping of the results of evaluating the APET. In the first analysis, all results were placed in one of about 20 previously defined bins. There were many pathways through the tree that did not fit well into these previously defined bins. For the current analysis, a flexible bin structure, defined by the characteristics important to the subsequent source term analysis was used. This eliminates a major problem in the original analysis process.

The event tree that forms the basis of this analysis was completely rewritten. In addition to utilizing a user function for added flexibility, the APET now considers offsite electric power recovery in the period between the onset of core damage and vessel failure. This led to a significant portion of the station blackout accidents terminating not with vessel breach but in an arrested core damage state similar to TMI-2. Additional means of depressurizing the RCS are now in the event tree. These additional mechanisms, along with the higher probabilities for some of them that resulted from the expert elicitations, mean that the likelihood is small that an accident that is at full system pressure at the onset of core damage will still be at that pressure when the vessel fails. Accidents in which core damage begins with a low pressure injection system (LPIS), or both LPIS and high pressure injection system (HPIS) operating are treated in the current APET whereas they were omitted in the previous version. If an event occurs to reduce the RCS pressure in these situations, core damage may be arrested before the vessel fails, leading, by another path, to an arrested core damage state similar to that of TMI-2.

<u>Source Term Analysis</u>. While the basic parametric approach used in the original version of ZISOR, the code used to compute source terms, has been retained in the present version of ZISOR, the code has been completely rewritten with a different orientation. The previous version was designed primarily to produce results that could be compared directly with the results of the Source Term Code Package (STCP). Discrete values for the parameters that differed from those that produced results close to STCP results were then used in the sampling process, with the probabilities for each value or level determined by a small panel of experts. Thus, the first version of ZISOR determined uncertainty in the amount of fission products released for the limited number of predefined bins from the STCP as a base.

The current version of ZISOR is quite different. First, it is not tied to the STCP in any way. It was recognized before the new version was developed that most of the parameters would come from continuous distributions defined by an expert panel. Thus, the current version does not rely on results from the STCP or any other specific code. The experts utilized the results of one or more codes in deriving their distributions, but ZISOR itself merely combines the parameters defined by the expert panel. Second, ZISOR now treats any consistent accident progression state defined by the 12 characteristics that constitute an accident progression bin for Zion. It is not limited to a small number of pre-defined bins as it was in the original version.

Finally, a new method to group the source terms computed by ZISOR has been devised. A source term is calculated for each accident progression bin for each observation in the sample. As a result, there are too many source terms to perform a consequence calculation for each and the source terms have to be grouped before the consequence calculations are performed. The "clustering" method utilized in the previous analysis was somewhat subjective and not as reproducible as desired. The new "partitioning" scheme developed for grouping the source terms in this analysis eliminates these problems.

Consequence Analysis. The consequence analysis for the current NUREG-1150 does not differ no markedly from that for the previous verison of NUREG-1150 as does the accident progression analysis and the source term analysis. Version 1.4 of MACCS was used for the original analysis, while version 1.5 is used for this analysis. The major difference between the two versions is in the data used in the lung model. Version 1.4 used the lung data contained in the original version of "Health Effects Models for Nuclear Power Plant Accident Consequence Analysis",[10] whereas version 1.5 of MACCS uses the lung data from Revision 1 (1989) of this report.[11] Other changes were made to the structure of the code in the transition from 1.4 to 1.5, but the effects of these changes on the consequence values calculated are small.

Another difference in the consequence calculation is that the NRC specified evacuation of 99.5% of the population in the evacuation area for this analysis, as compared with the previous analysis in which 95% of the population was evacuated.

Risk Analysis. The risk analysis combines the results of the accident frequency analysis, the accident progression analysis, the source term analysis, and the consequence analysis to obtain estimates of risk to the offsite population and the uncertainty in those estimates. This combination of the results of the constituent analyses was performed essentially the same way for both the previous and the current analyses. The only differences are in the number of variables sampled and the number of observations in the sample.

1.4 Structure of the Analysis

The analysis of the Zion plant for NUREG-1150 is a Level 3 probabilistic risk assessment composed of four constituent analyses:

1. Accident frequency analysis, which estimates the frequency of core damage for all significant initiating events;

2. Accident progression analysis, which determines the possible ways in which an accident could evolve given core damage;

3. Source term analysis, which estimates the source terms (i.e., environmental releases) for specific accident conditions; and

4. Consequence analysis, which estimates the health and economic impacts of the individual source terms.

Each of these analyses is a substantial undertaking in itself. By taking care to carefully define the interfaces between these individual analyses, the transfer of information is facilitiated. At the completion of each constituent analysis, intermediate results are generated for presentation and interpretation. An overview of the assembly of these components into an integrated analysis is shown in Figure 1.2.

The NUREG-1150 plant studies are fully integrated probabilistic risk assessments in the sence that calculations leading to both risk and uncertainty in risk are carried through all four components of the individual plant studies. The frequency of the initiating event, the conditional probability of the paths leading to the consequence, and the value of the consequence itself can then be combined to obtain a risk measure. Measures of uncertainty in risk are obtained by repeating the calculations, using a Monte Carlo technique (Latin Hypercube Sampling),[12] many times with different values for important parameters. This provides a distribution of risk estimates that is a measure of the uncertainty in risk.

It is important to recognize that a probabilistic risk assessment is a procedure for assembling and organizing information from any sources; the models actually used in the computational framework of a probabilistic risk assessment serve to organize this information, and as a result, are rarely as detailed as most of the models that are actually used in the original generation of this information. In order to capture the uncertainties, the first three of the four constituent analyses attempt to utilize all available sources of information for each analysis component, including past observational data, experimental data, mechanistic modelling and, as appropraite or necessary, expert judgment. This requires the use of relatively quick running models to assemble and manipulate the data developed for each analysis.

To facilitate both the conceptual description and the computational implementation of the NUREG-1150 analyses, a matrix representation[13,14] is used to show how the overall integrated analysis fits together and how the progression of an accident can be traced from initiating event to offsite consequences.

<u>Accident Frequency Analysis</u>. The minimal cut sets, obtained from the systems analysis, are grouped into plant damage states (PDSs), where all minimal cut sets in a PDS provide a similar set of conditions for the subsequent accident progression analysis. Thus, the PDSs form the interface between the systems analysis and the accident progression analysis.

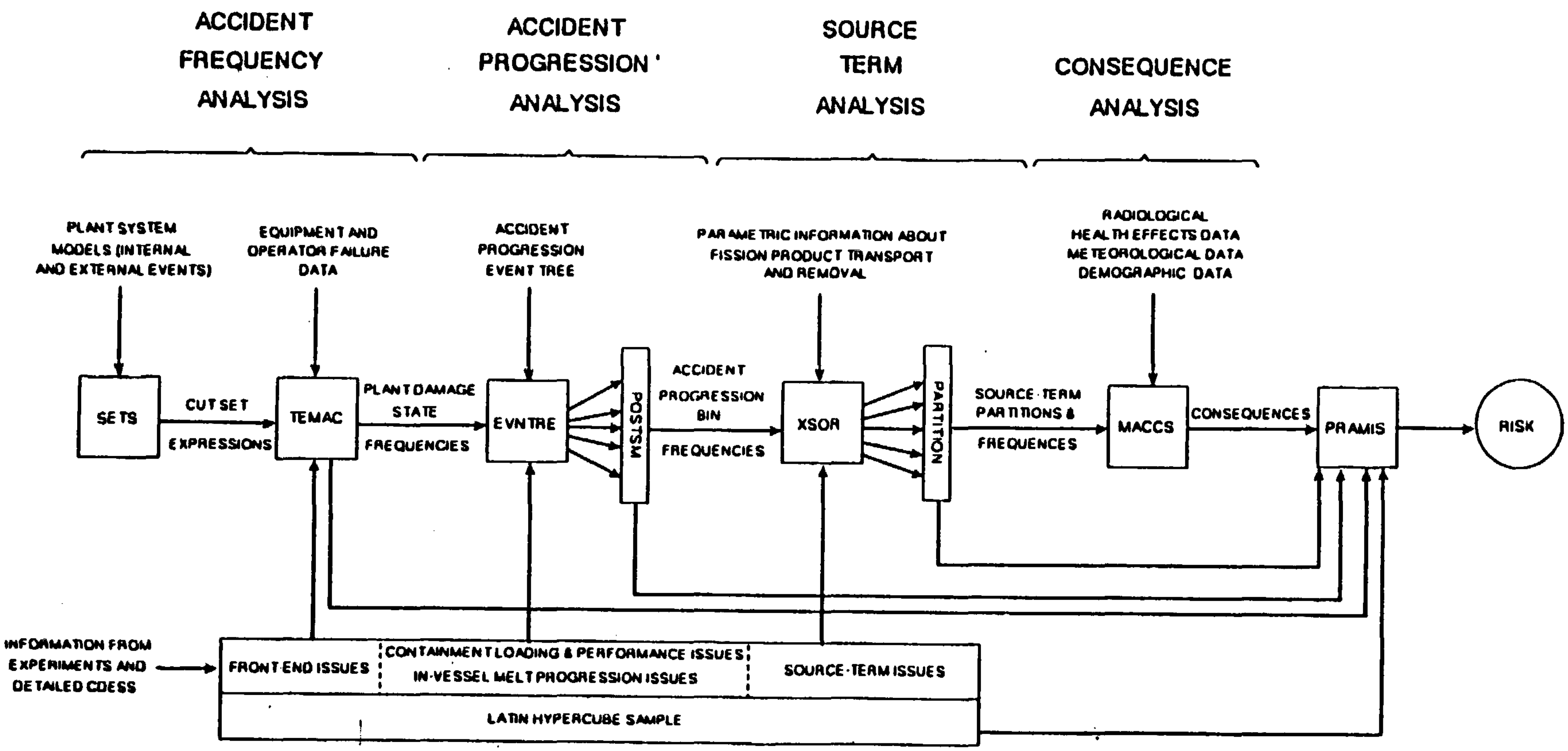


Figure 1.2 Overview of Integrated Plant Analysis in NUREG-1150

With use of the transition matrix notation, the accident progression analysis may be represented by

$$fPDS = fIE\ P(IE \rightarrow PDS), \qquad \text{(Eq. 1.1)}$$

where fPDS is the vector of frequencies for the PDSs, fIE is the vector of frequencies for the initiating events, and P(IE→PDS) is the matrix of transition probabilities from initiating events to PDSs. Specifically,

$fIE = [fIE_1, \ldots, fIE_{nIE}]$,
fIE_i = frequency (yr^{-1}) for initiating event i,
nIE = number of initiating events,
$fPDS = [fPDS_1, \ldots, fPDS_{nPDS}]$,
$fPDS_j$ = frequency (yr^{-1}) for plant damage state j,
nPDS = number of plant damage states,

$$P(IE \rightarrow PDS) = \begin{bmatrix} pPDS_{11} & \cdots & pPDS_{1,nPDS} \\ \vdots & & \vdots \\ pPDS_{nIE,1} & \cdots & pPDS_{nIE,nPDS} \end{bmatrix}$$

and

$pPDS_{ij}$ = probability that initiating event i will lead to plant damage state j.

The elements $pPDS_{ij}$ of P(IE→PDS) are conditional probabilities: given that initiating event i has occurred, $pPDS_{ij}$ is the probability that plant damage state j will also occur. The elements of P(IE→PDS) are determined by the analysis of the minimal cut sets. In turn, both the cut sets and the data used in their analysis come from earlier studies that draw on many sources of information. Thus, although the elements $pPDS_{ij}$ of P(IE→PDS) are represented as though they are single numbers, in practice these elements are functions of the many sources of information that went into the accident frequency analysis.

Accident Progression Analysis. The accident progression analysis uses event tree techniques to determine the possible ways in which an accident might evolve from each PDS. Specifically, a single event tree is developed for each plant and evaluated with the EVNTRE computer program.[15] The definition of each PDS provides enough information to define the initial conditions for the APET analysis. Due to the large number of questions in the Zion APET and the fact that many of these questions have more than two outcomes, there are far too many paths through each tree to permit their individual consideration in subsequent source term and consequence analysis. Therefore, the paths through the trees are grouped into accident progression bins, where each bin is a group of paths through the event tree that define a similar set of conditions for source term analysis. The properties of each accident progression bin define the initial conditions for the estimation of the source term.

Past observations, experimental data, mechanistic code calculations, and expert judgment were used in the development and parameterization of the model for accident progression that is embodied in the APET. The transition matrix representation for the accident progression analysis is

$$fAPB = fPDS\ P(PDS \rightarrow APB), \qquad \text{(Eq. 1.2)}$$

where fPDS is the vector of frequencies for the PDSs defined in Eq. 1.1, fAPB is the vector of frequencies for the accident progression bins, and $P(PDS \rightarrow APB)$ is the matrix of transition probabilities from PDSs to accident progression bins. Specifically,

$fAPB = [fAPB_1, \ldots, fAPB_{nAPB}]$,

$fAPB_k$ = frequency (yr^{-1}) for accident progression bin k,

nAPB = number of accident progression bins,

$$P(PDS \rightarrow APB) = \begin{bmatrix} pAPB_{11} & \cdots & pAPB_{1,nAPB} \\ \vdots & & \vdots \\ pAPB_{nPDS,1} & \cdots & pAPB_{nPDS,nAPB} \end{bmatrix}$$

and

$pAPB_{jk}$ = probability that plant damage state j will lead to accident progression bin k.

The properties of fPDS are given in conjunction with Eq. 1.1. The elements $pAPB_{jk}$ of $P(PDS \rightarrow APB)$ are determined in the accident progression analysis by evaluating the APET with EVNTRE for each PDS group.

Source Term Analysis. The source terms are calculated for each APB with a non-zero conditional probability for each observation in the sample by a parametric computer code entitled ZISOR. ZISOR is not a detailed mechanistic model and makes no pretense of being a realistic simulation of fission product transport, physics, and chemistry. Instead, ZISOR integrates the results of many detailed codes and the conclusions of many experts. The experts, in turn, based many of their conclusions on the results of calculations with codes such as the Source Term Code Package[16,17] and MAAP. Most of the parameters are sampled from distributions provided by an expert panel. Because of the large number of APBs, use of a fast-executing code like ZISOR is necessary.

The number of APBs for which source terms are calculated is so large that it is not practical to perform a consequence calculation for every source term. That is, the consequence code, MACCS,[18,19,20] required so much computer time to calculate the consequences of a source term that the source terms had to be cominbed into source term groups. Therefore, all APBs, with similar radiological potential, are grouped into a small subset of representative release groups. The frequency of the each group is the sum of the frequencies of all the APBs which make up the group. The grouping process is called partitioning and is discussed in detail in Volume 1 of this report.

The transition matrix representation of the source term calculation and the grouping process is

$$fSTG = fAPB\ P(APB \rightarrow STG) \qquad \text{(Eq. 1.3)}$$

where fAPB is the vector of frequencies for the accident progression bins defined in Eq. 1.2, fSTG is the vector of frequencies for the source term groups, and P(APB→STG) is the matrix of transition probabilities from accident progression bins to source term groups. Specifically,

$$fSTG = [fSTG_1, \ldots, fSTG_{nSTG}],$$

$$fSTG_L = \text{frequency } (yr^{-1}) \text{ for source term group } \ell,$$

$$nSTG = \text{number of source term groups},$$

$$P(APB \rightarrow STG) = \begin{bmatrix} pSTG_{11} & \cdots & pSTG_{1,nSTG} \\ \vdots & & \vdots \\ pSTG_{nAPB,1} & \cdots & pSTG_{nAPB,nSTG} \end{bmatrix}$$

and

$pSTG_{kL}$ = probability that accident progression bin k will be assigned to source term group ℓ.

$$= \begin{cases} 1 & \text{if accident progression bin k is assigned source term group 1} \\ 0 & \text{otherwise.} \end{cases}$$

The properties of fAPB are given in conjunction with Eq. 1.2. Note that the source terms themselves do not appear in Eq. 1.4. The source terms are used only to assign an APB to a source term group. The consequences for each APB are computed from the average source term for the group to which the APB has been assigned.

Consequence Analysis. The consequence analysis is performed for each source term group by the MACCS program. The results for each source term group include estimates for both mean consequences and distributions of consequences. When these consequence results are combined with the frequencies for the source term groups, overall measures of risk are obtained.

The consequence analysis differs from the preceding three constituent analyses in that uncertainties are not explicitly treated in the consequence analysis, with the exception of weather variability sampling.

In the transition matrix notation, the risk may be expressed by

$$rC = fSTG\ cSTG, \qquad \text{(Eq. 1.4)}$$

where fSTG is the vector of frequencies for the source term groups defined in Eq. 1.3, rC is the vector of risk measures, and cSTG is the matrix of mean consequence measures conditional on the occurrence of individual source term groups. Specifically,

$rC = [rC_1, \ldots, rC_{nC}]$,

rC_m = risk (consequence/yr) for consequence measure m,

nC = number of consequence measures,

$$cSTG = \begin{bmatrix} cSTG_{11} & \cdots & cSTG_{1,nC} \\ \vdots & & \vdots \\ cSTG_{nSTG,1} & \cdots & cSTG_{nSTG,nC} \end{bmatrix}$$

and

$cSTG_{\ell m}$ = mean value (over weather) of consequence measure m conditional on the occurrence of source term group ℓ.

The properties of fSTG are given in conjunction with Eq. 1.3. The elements $cSTG_{\ell m}$ of cSTG are determined from consequence calculations with MACCS for individual source term groups.

Computation of Risk. Equations 1.1 through 1.4 can be combined to obtain the following expression for risk:

$$rC = fIE\ P(IE \rightarrow PDS)\ P(PDS \rightarrow APB)\ P(APB \rightarrow STG)\ cSTG. \qquad \text{(Eq. 1.5)}$$

This equation shows how each of the constituent analyses enters into the calculation of risk, starting from the frequencies of the initiating events and ending with the calculation of consequences. Evaluation of the expression in Eq. 1.5 is performed with the PRAMIS[21] and RISQUE codes.

The description of the complete risk calculation so far has focused on the computation of mean risk (consequences/year) because doing so makes the overall structure of the NUREG-1150 PRAs more easy to comprehend. The mean risk results are derived from the frequency of the initiating events, the conditional probabilities of the many ways that each accident may evolve, and the probability of occurrence for each type of weather sequence at the time of an accident. The mean risk, then, is a summary risk measure.

More information is conveyed when distributions for consequence values are displayed. The form typically used for this is the complementary cumulative distribution function (CCDF). CCDFs are defined by pairs of values (c,f), where c is a consequence value and the f is the frequency with which c is exceeded. Figure 1.3 is an example of a CCDF. The construction of CCDFs is described in Volume 1 of this report. Each mean risk result is the outcome from reducing a curve of the form shown in Figure 1.3 to a single value. While the mean risk results are often useful for summaries or high-level comparisons, the CCDF is the more basic measure of risk because it displays the relationship between the size of the consequence and frequency exceedance. The nature of this relationship, i.e., that high consequence events are much less likely than low consequence events is lost when mean risk results alone are reported. This report utilizes both mean risk and CCDFs to report the risk results.

Propagation of Uncertainty through the Analysis. The integrated NUREG-1150 analyses use Monte Carlo procedures as a basis for both the uncertainty and the sensitivity analyses. This approach utilizes a sequence:

$$X_1,\ X_2,\ \ldots,\ X_{nV} \qquad \text{(Eq. 1.6)}$$

of potentially important variables, where nV is the number of variables selected for consideration. Most of these variables were considered by a panel of experts representing the NRC and its contractors, the academic world, and the nuclear industry. For each variable treated in this manner, two to six experts considered all the information at their disposal and provided a distribution for the variable. Formal decision analysis techniques[22] (also in Volume 2 of this report) were used to obtain and record each expert's conclusions and to aggregate the assessments of the individual panel members into summary distribution for the variable. Thus, a sequence of distributions

$$D_1,\ D_2,\ \ldots,\ D_{nv}, \qquad \text{(Eq. 1.7)}$$

is obtained, where D_i is the distribution assigned to variable X_i.

From the distributions, obtained by aggregating expert panel elicitation results, a stratified Monte Carlo technique, Latin Hypercube Sampling[12,23] is used to obtain the variable values that will actually be propagated through the integrated analysis. The result of generating a sample from the variables in Eq. 1.6 with the distributions in Eq. 1.7 is a sequence:

$$S_i = [X_{i1},\ X_{i2},\ \ldots,\ X_{i,nV}],\ i = 1,\ 2,\ \ldots,\ nLHS, \qquad \text{(Eq. 1.8)}$$

of sample elements, where X_{ij} is the value for variable X_j in sample element i and nLHS is the number of elements in the sample. The expression in Eq. 1.5 is then determined for each element of the sample. This creates a sequence of results of the form:

$$rC_i = fIE_i\ P_i(IE \rightarrow PDS)\ P_i(PDS \rightarrow APB)\ P_i(APB \rightarrow STG)\ cSTG, \qquad \text{(Eq. 1.9)}$$

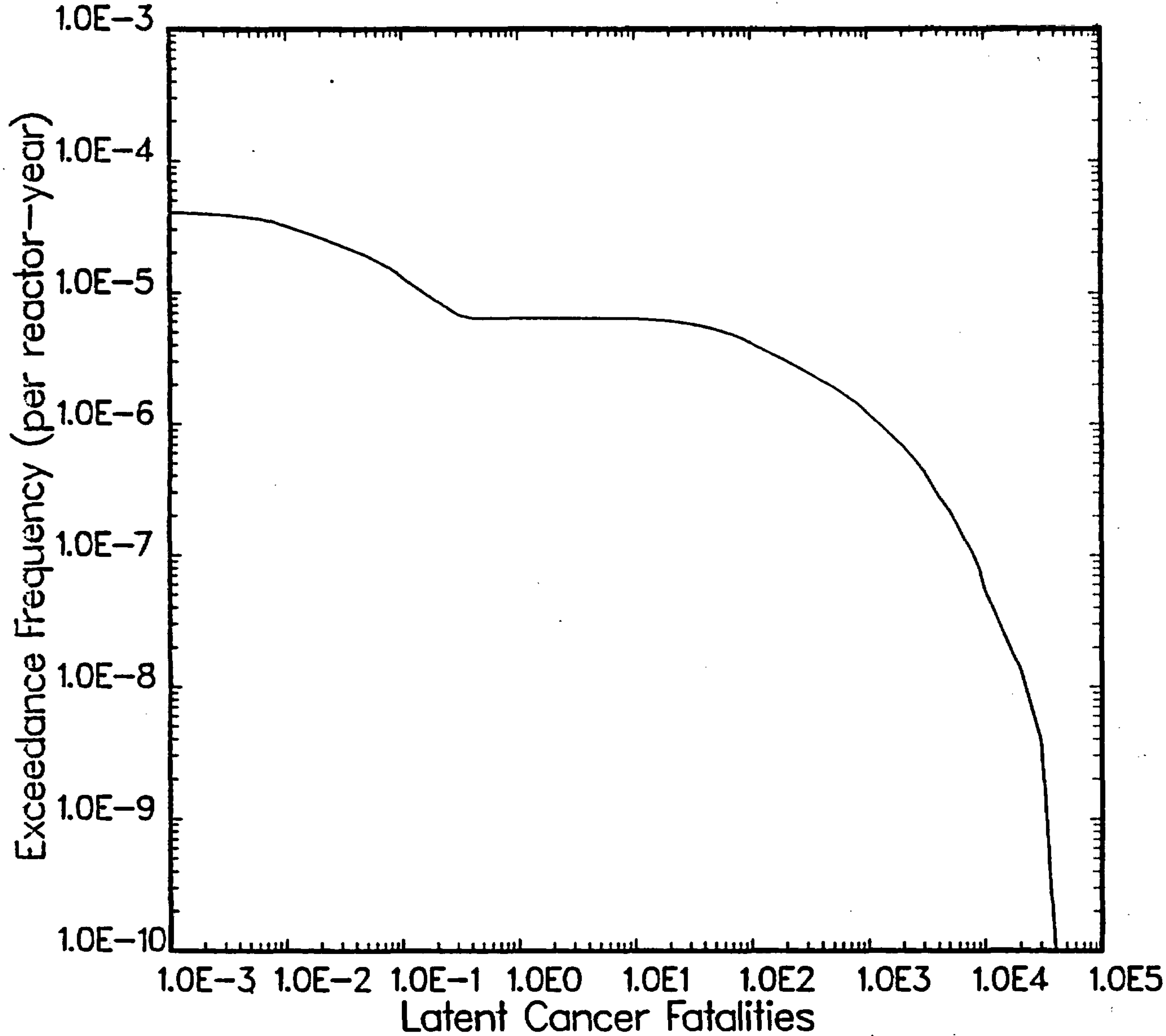


Figure 1.3 Example Risk CCDF

where the subscript i is used to denote the evaluation of the expression in Eq. 1.5 with the ith sample element in Eq. 1.8. The uncertainty and sensitivity analyses in NUREG-1150 are based on the calculations summarized in Eq. 1.9. Since P(IE→PDS), P(PDS→APB) and P(APB→STG) are based on results obtained with EVNTRE and ZISOR, determination of the expression in Eq. 1.9 requires a separate evaluation of the cut sets, the APET, and the source term model for each element or observation in the sample. The matrix cSTG in Eq. 1.9 is not subscripted because the NUREG-1150 analyses do not include consequence modelling uncertainty other than the stochastic variability due to weather conditions.

1.5 Organization of this Report

This report is published in seven volumes as described briefly in the Foreword. The first volume of NUREG/CR-4551 describes the methods used in the accident progression analysis, the source term analysis, and the consequence analysis, in addition to presenting the methods used to assemble the results of these constituent analyses to determine risk and the uncertainty in risk. The second volume describes the results of convening expert panels to determine distributions for the variables thought to be the most important contributors to uncertainty in risk. Panels were formed to con- sider in-vessel processes, containment structural response, molten core-concrete interactions, and source term issues. In addition to documenting the results of these panels for about 30 important parameters, Volume 2 includes supporting material used by these panels and presents the results of distributions that were determined by other means.

Volumes 3 through 6 present the results of the accident progression analysis, the source term analysis, and the consequence analysis, and the combined risk results for Surry, Peach Bottom, Sequoyah, and Grand Gulf, respectively. These analyses were performed by Sandia National Laboratories. Volume 7 presents analogous results for Zion, performed by Brookhaven National Laboratory.

This volume of NUREG/CR-4551, Volume 7, presents risk and constituent analysis results for Unit 1 of the Zion Power Station, operated by Commonwealth Edison in Illinois. Part 1 of this volume presents the analysis and the results is some detail; Part 2 (A and B) consists of the appendices which contain further detail. Following a summary and an introduction, Chapter 2 of this volume presents the results of the accident progression analysis for internal initiating event. Chapter 3 presents the result of the source term analysis, and Chapter 4 gives the result of the consequence analysis. Chapter 5 summarizes the risk results, including the contributors to uncertainty in risk, and Chapter 6 contains the insights and conclusions of the complete analysis.

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2. ANALYSIS OF THE ACCIDENT PROGRESSION

This chapter describes the analysis of the progression of the accident, starting from the uncovering of the top of active fuel (TAF) and continuing for about 24 h. The main tool for performing the accident progression analysis is a large and complex event tree. The methods used in this accident progression analysis are presented in NUREG/CR-4551, Vol. 1. The accident progression analysis starts with information received from the accident frequency analysis: frequencies and definitions of the plant damage states (PDSs). The results of the accident progression analysis are passed to the source term analysis and the risk analysis.

Section 2.1 reviews the plant features that are important to the accident progression analysis and the containment response. Section 2.2 summarizes the results of the accident frequency analysis, defines the PDSs, and presents their frequencies. Section 2.3 contains a brief description of the accident progression event tree (APET). A detailed description of the APET is contained in Appendix A. Section 2.4 describes the way in which the results of the evaluation of the APET are grouped together into bins. This grouping is necessary to reduce the information resulting from the APET evaluation to a manageable amount while still preserving the information required by the source term analysis. Section 2.5 presents the results of the accident progression analysis.

2.1 Zion Plant Features Important to Accident Progression

The major differences in the Zion and Surry designs consist of: (1) the Zion power is rated at 3,250 MW (thermal) (four loop reactor coolant system) whereas that of Surry is 2,441 MW (thermal) (three loop reactor coolant system), (2) Zion has a large, dry containment (7.7×10^4 m^3 [2.7×10^6 ft^3] free volume) in comparison to the Surry containment (5.1×10^4 m^3 [1.8×10^6 ft^3]), which is normally maintained at sub-atmospheric pressure (approximately 0.07 MPa [10 psia]), and (3) the reactor cavity design for Zion is characterized by smooth passageways which would conduct both gases and debris from the cavity to the lower compartment. In the case of the Surry reactor cavity, discontinuities exist which would tend to trap the debris and allow the gas to escape. Figure 1.1 shows a cross-section of the Zion containment.

The branch points and probabilities for the Zion accident progression tree reflect consideration of a number of plant-specific features that could have important effects on the progression of a severe accident. More detailed description of the Zion plant-specific features regarding the containment failure pressure, containment safeguards systems, and the reactor cavity geometry are given in Sections 2.1.1 through 2.1.3.

2.1.1 The Zion Containment Structure

The Zion containment is in the shape of a cylinder with a shallow, domed roof, and a flat foundation slab. The cylindrical portion is pre-stressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post-tensioning system. The foundation slab is conventionally reinforced with bonded, reinforced steel. The entire structure is lined with a leak-tight, 6 x 10^{-3} m (1/4-inch) welded steel plate. The design pressure of the building is 0.43 MPa (62 psig).

The ultimate capacity of the containment is an important factor when determining the likelihood and timing of containment failure during a severe accident. The pressure and size of the break at containment failure can also influence the quantity of fission products released to the environment. The internal pressure capacity of the Zion containment has been analyzed by Sargent and Lundy[1] and at Brookhaven National Laboratory(BNL).[2] The Sargent and Lundy study of the Zion containment predicted an ultimate internal pressure capacity of 1.03 MPa (149 psia) at which the failure mode is yielding of the hoop tendons. The failure criterion was one of 1% strain in the steel tendons anywhere in the containment shell. In the subsequent analysis performed at BNL, containment failure at Zion was predicted to be due to loss of shear capacity at the basemat-cylinder intersection, the failure occuring at a pressure load of 0.87 MPa (126 psia). A Los Alamos National Laboratory study[3] reported two containment failure pressures for Zion corresponding to shear failure near the basemat-cylinder intersection and to hoop tendon yielding. These were 0.97 and 1.04 MPa (140 and 151 psia), respectively. These analyses were given to the NUREG-1150 Structural Expert Panel to help them develop the Zion containment failure distributions.

2.1.2 The Containment Heat Removal Systems

The containment fan cooler and spray systems provide redundant and diverse containment heat removal capability for Zion. The initiation pressure for fan coolers and sprays are 0.14 and 0.26 MPa (20 and 37 psia), respectively.[4] The containment fan cooler system is designed to remove heat from the containment building during both normal operation and in the event of a design basis accident. The containment fan cooler units are an engineered safeguard system. Five fan coolers are provided for the containment. Each cooler is rated at one-third the required capacity for design basis accident conditions.

The containment spray system, on the other hand, is designed to limit the pressure in the containment atmosphere to below the containment design pressure and to reduce the radiological releases to the 10CFR100 limits. Three completely redundant containment spray system trains are provided for each unit, with each system rated at 100% capacity for design basis accident conditions. One of the spray trains has a diesel engine driven spray pump for added diversity. All three containment spray pumps take suction from the refueling water storage tank (RWST) and discharge into the spray rings located around the inside of the containment dome. Should spray be required during the recirculation phase of the accident, two of the three spray subsystems can be supplied with water from the containment sump via the residual heat removal pumps which deliver water to the discharge lines of the two motor operated spray pumps. Spray pump operation is therefore not necessary during the recirculation phase. Both motor-driven pumps and all motor operated valves can be supplied with power from the emergency diesel generators in the event of a loss-of-offsite power (LOSP). Failure of a single diesel or emergency bus will affect one subsystem only.

2.1.3 Sump and Cavity Arrangement

The progression of the accident following the reactor pressure vessel (RPV) failure is strongly affected by the reactor cooling system (RCS) pressure before RPV failure, the cavity geometry, and water availability. The design of the reactor cavity, in particular, can have an important impact on the accident progression due primarily to: (1) the degree to which the core debris is inhibited from being dispersed following ejection from the reactor vessel, (2) the ability of water in the containment to reach the core debris, and (3) the ability to transfer heat from the cavity to the containment atmosphere.

In the Zion containment, an instrument tunnel connects the cavity to the lower compartment of the containment, which could provide a potential path for high pressure discharge of the corium out of the cavity if the accident scenario involves core meltdown with the primary system at high pressure. On the other hand, the initial operation of the containment sprays (if available) will ensure a flooded cavity during the accident and the amount of water in the cavity may be sufficient to quench the debris at normal decay heat levels. This is in contrast to the Surry cavity design which does not communicate directly with the containment sump and the potential path for corium discharge is more tortuous.

2.2 Interface with the Core Damage Frequency Analysis*

2.2.1 Definition of Plant Damage States

Information about the many different accidents that lead to core damage is passed from the core damage frequency analysis to the accident progression analysis by means of PDSs. Because most of the accident sequences identified in the core damage frequency analysis will have accident progressions similar to other sequences, these sequences have been grouped together into plant damage states. All the sequences in one PDS should behave similarly in the period following the uncovering of the TAF. For Zion, the PDS is denoted by an eight-letter indicator that defines eight characteristics that largely determine the progression of the accident.

Table 2.2-1 lists the eight characteristics used to define the PDSs for Zion. For each characteristic the possible values are given underneath. For example, the first characteristic denotes the RCS pressure at the time core damage begins (assumed to be approximately when the TAF is uncovered). Table 2.2-1 shows that there are eight possibilities for this characteristic: T for transient or no break; A, S_1, S_2, and S_3 for the four sizes of break which do not bypass the containment; G and H for steam generator tube ruptures (SGTRs), and interfacing systems LOCA (V) for the large bypass pipe failure.

* These are generic definitions. Related discussions are applicable to most pressurized water reactors (PWRs). Adopted from NUREG/CR-4551, Vol. 3, and modified for the Zion plant wherever necessary.

Table 2.2-1

Zion Plant Damage State Characteristics

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1. Status of RCS at Onset of Core Damage
   - T - no break (transient)
   - A - large break in the RCS pressure boundary
   - S_1 - medium break in the RCS pressure boundary
   - S_2 - small break in the RCS pressure boundary
   - S_3 - very small break in the RCS pressure boundary
   - G - SGTR
   - H - SGTR with loss of secondary system integrity
   - V - large break in an interfacing system

2. Status of ECCS
   - B - operated in injection and now operating in recirculation
   - I - operated in injection only
   - R - not operating, but recoverable
   - N - not operating, not recoverable
   - L - LPIS available in both injection and recirculation modes

3. Containment Heat Removal
   - B - operated in injection and now in recirculation
   - I - operated in injection only
   - R - not operating, but recoverable
   - N - never operated, not recoverable
   - C - available in injection and in recirculation
   - A - available in injection only

4. AC Power
   - Y - available
   - P - partially available
   - R - not available, but recoverable
   - N - not available, not recoverable

5. Contents of RWST
   - Y - injected into containment
   - R - not injected, but could be injected if power recovered
   - N - not injected, cannot be injected in the future

6. Heat Removal from the Steam Generators
   - X - at least one AFWS operating, SGs not depressurized
   - Y - at least one AFWS operating, SGs depressurized
   - S - S-AFWS failed at beginning, E-AFWS recoverable
   - C - S-AFWS operated until battery depletion, E-AFWS recoverable, SGs not depressurized
   - D - S-AFWS operated until battery depletion, E-AFWS recoverable, SGs depressurized
   - N - no AFWS operating, no AFWS recoverable

Table 2.2-1 (Continued)

7. Cooling for RCP Seals
   - Y - operating
   - R - not operating, but recoverable
   - N - not operating, not recoverable

8. Status of Containment Fan Coolers
   - Y - operating
   - R - not operating, but recoverable
   - N - not operating, not recoverable

The first characteristic in the PDS is not necessarily an indication of the initiating event. It is an indicator of the RCS integrity at the time the core uncovers. That is, if the initiating event is a transient, say loss of offsite power, but a reactor coolant pump (RCP) seal failure occurs before the onset of core degradation, then there is a small hole in the RCS pressure boundary at the time that core damage begins, which is the time the accident progression analysis begins. The PDS for this accident would begin with S3 to reflect the fact that there is a small hole in the RCS when this analysis starts. It is the plant condition at the onset of core damage that is important for the accident progression analysis, not what the original initiator may have been.

Thus, the first character in the PDS indicates the condition of the RCS at the onset of core degradation. As a hold over from the use of this character to indicate the original initiator, "T" is used to indicate no break (transient). An S2 break is between 0.01 (0.5) and 0.05 (2) m(inches) in diameter; an S3 break is less than 0.01 m (0.5 inches) in diameter. A and S1 breaks are considered together in the accident progression analysis since both result in low pressure in the RCS. SGTRs are S3 size. Almost all pump seal failures result in a leak area equivalent to an S2 break. A stuck-open power-operated relief valve (PORV) is equivalent to an S2 break. Event V is such a well known and unique type of accident that the subsequent six characteristics are usually not written out.

The second characteristic concerns the status of the ECCS. Recoverable means that the ECCS will operate if or when electric power is recovered. The value "L" for the second characteristic is used when the low pressure injection system (LPIS) is available to inject when the core is uncovered but cannot because the RCS pressure is too high. "L" implies that the high pressure injection system (HPIS) is failed.

The third characteristic concerns the status of the containment spray system (CSS). Recoverable means that the CSS will operate if or when electric power is recovered. The value "B" for the third characteristic is used when the sprays are operating during both injection and recirculation modes. Even if there is no heat removal, it is important to know whether the sprays are operating because they reduce the aerosol concentrations in the containment atmosphere.

The fourth characteristic concerns the status of AC power. Recoverable means that power can be restored within the timeframe of the accident, roughly 24 h.

The fifth characteristic concerns the status of the water in the RWST. It is important for analyzing the accident progression to know whether or not the water from the RWST is inside the containment. If the RWST water is in the containment then it will fill the sumps and the reactor cavity. The value "N" for this characteristic is used when some failure prevents the injection of the RWST, such as when the water from the RWST has been injected into the RCS but has ended up outside the containment. This occurs in event V subgroup when the water is injected into the RCS but flows out through the break into the auxiliary building, and thus is not available inside the containment.

The sixth characteristic concerns the heat removal from the steam generators. There are six possible values for this characteristic since the auxiliary feedwater system (AFWS) may operate for some time in a blackout accident, and the secondary system may or may not be depressurized by the operators. The following abbreviations are used in describing the sixth characteristic in Table 2.2-1:

E-AFWS - Electric-motor-driven auxiliary feedwater system (two 50% motor driven pumps); and
S-AFWS - Steam-turbine-driven auxiliary feedwater system (100% turbine driven pumps).

The seventh characteristic concerns cooling for the RCP seals. Recoverable means that cooling will become available if or when electric power is recovered.

The eighth characteristic concerns the availability/operability of the containment fan cooling system (CFCS). The Zion CFCSs are safety-grade and are thus potentially available during an accident. Without the operation of the spray systems, the CFCS can remove the thermal load from the design basis accidents.

2.2.2 Plant Damage State Frequencies

This subsection presents the core damage frequencies for the PDSs and PDS groups. The Zion analysis did not consider external initiators.

Table 2.2-2 indicates how the 57 Zion PDSs were placed into five groups. These 57 PDSs are those with mean frequencies of 1E-9 or higher, and they account for most of the total core damage frequency (TCDF).

PDS Group 1 consists of three blackout PDSs. In these accidents, offsite power is lost and the diesel generators fail to start or run. The steam-turbine-driven AFWS operates until the batteries are depleted. Without power for instruments and controls, it is assumed that the AFWS fails. Battery depletion is assumed to take about 4 hours. During this time the RCP seals may fail or the PORVs may stick open. Thus the three PDSs in this group have the RCS in different conditions when core damage begins.

For one of the PDSs in this group, the RCS is intact at the time that the core is uncovered. Another two of the PDSs have S2-size breaks (due to failure of the RCP seals). The difference between the "T" and "S" PDS in Group 1 is whether or not there is cooling for the RCP seals.

PDS Group 2 consists of 38 loss-of-coolant accidents (LOCA) PDSs. Seven of the PDSs have an A-size break, and five of the PDSs have an S1-size break, which are essentially the same in this portion of the analysis. The rest of the PDSs have an S2-size break. Four of the PDSs in this group have the LPIS operating. In PDSs S2L, not all of the breaks depressurize the RCS enough to allow the LPIS to inject; thus some fraction of the accidents will progress to vessel failure at a pressure too high to allow the LPIS to inject unless a large temperature induced break occurs or the primary system is deliberately depressurized.

Table 2.2-2 Plant Damage State Cut Sets for Zion

PDS #	Group Name	Plant Damage State	Point Estimate	5th Percentile	Median	Mean	95th Percentile
1	Station Blackout	TRRRSRR	4.7E-06	2.8E-06	4.7E-06	5.6E-06	8.3E-06
		S2RRRRYRR	4.2E-07	5.1E-08	4.2E-07	6.1E-07	1.3E-06
		S2RRRRSRR	8.2E-08	4.8E-08	8.2E-08	9.8E-08	1.5E-07
2	LOCAs	S2NBYYYNY	1.2E-04	2.8E-05	9.0E-05	1.6E-04	5.7E-04
		S2NBYYYNN	1.2E-04	3.0E-05	9.1E-05	1.2E-04	2.7E-04
		S2IBYYYYY	8.0E-06	1.6E-06	8.0E-06	1.1E-05	1.6E-05
		S1IBYYYYY	5.4E-06	5.4E-06	5.4E-06	5.4E-06	5.4E-06
		AIBYYYYY	4.9E-06	4.9E-06	4.9E-06	4.9E-06	4.9E-06
		ANBYYYYY	1.4E-06	1.4E-06	1.4E-06	1.4E-06	1.4E-06
		S2NBYYYNR	3.8E-07	7.0E-08	3.8E-07	5.6E-07	1.2E-06
		S2IIYYYNR	3.3E-07	6.5E-08	3.3E-07	4.1E-07	6.8E-07
		S2NIYYYNY	1.9E-07	4.4E-08	1.4E-07	2.6E-07	9.1E-07
		S2NIYYYNN	2.3E-07	9.4E-08	2.2E-07	2.6E-07	5.3E-07
		S2IBYYYNY	3.0E-07	2.5E-08	1.2E-07	1.9E-07	4.3E-07
		S2IIYYYYR	3.9E-08	7.7E-09	3.9E-08	5.1E-08	7.7E-08
		S1NBYYYNY	2.1E-08	2.1E-08	2.1E-08	2.1E-08	2.1E-08
		S1NBYYYNN	2.1E-08	2.1E-08	2.1E-08	2.1E-08	2.1E-08
		ANBYYYNN	2.1E-08	2.1E-08	2.1E-08	2.1E-08	2.1E-08
		ANBYYYNY	2.1E-08	2.1E-08	2.1E-08	2.1E-08	2.1E-08
		S2LBYYYNR	2.1E-08	1.7E-08	2.1E-08	2.0E-08	2.3E-08
		S2NNYNYNR	1.4E-08	1.7E-09	1.4E-08	2.0E-08	4.3E-08
		S2LBYYNYY	1.9E-08	1.9E-08	1.9E-08	1.9E-08	1.9E-08
		S2NIYYYNR	1.2E-08	1.5E-09	1.2E-08	1.7E-08	3.7E-08
		S2INYYYNR	3.9E-09	7.9E-10	3.9E-09	5.0E-09	8.5E-09
		S2IIYYNNR	1.1E-08	2.3E-09	1.1E-08	1.4E-08	2.4E-08
		S2IIYYYY	1.3E-08	1.3E-08	1.3E-08	1.3E-08	1.3E-08
		S2NNYNYNN	1.0E-08	1.5E-09	8.7E-09	1.1E-08	2.3E-08
		S2NBYYNNY	6.6E-09	1.9E-09	5.4E-09	1.0E-08	2.9E-08
		S2LBYYYNY	1.3E-08	6.9E-10	4.9E-09	9.9E-09	3.3E-08
		S2NBYYNNN	6.6E-09	2.8E-09	5.7E-09	8.8E-09	1.8E-08
		AIIYYYYY	7.8E-09	7.8E-09	7.8E-09	7.8E-09	7.8E-09
		S1IIYYYYY	7.8E-09	7.8E-09	7.8E-09	7.8E-09	7.8E-09
		S2LBYYYYN	7.7E-09	7.7E-09	7.7E-09	7.7E-09	7.7E-09
		S2NNYNYNY	5.6E-09	2.5E-10	2.7E-09	7.0E-09	2.9E-08

Table 2.2-2 (Continued)

PDS #	Group Name	Plant Damage State	Point Estimate	5th Percentile	Median	Mean	95th Percentile
2	LOCAs (contd.)	S2INYYYNR	3.9E-09	7.9E-10	3.9E-09	5.0E-09	8.5E-09
		S2NIYYNNN	3.6E-09	2.1E-09	3.6E-09	4.3E-09	6.4E-09
		ANIYYYYY	2.3E-09	2.3E-09	2.3E-09	2.3E-09	2.3E-09
		S1RIYYYYR	2.1E-09	2.1E-09	2.1E-09	2.1E-09	2.1E-09
		ARIYYYYR	2.1E-09	2.1E-09	2.1E-09	2.1E-09	2.1E-09
		S2IIYYNYR	1.6E-09	3.1E-10	1.6E-09	2.1E-09	3.1E-09
		S2NIYYNNR	1.0E-09	6.0E-10	1.0E-09	1.2E-09	1.8E-09
3	Transients	TIAYYNYR	5.2E-06	1.0E-06	5.2E-06	6.8E-06	1.0E-05
		TLCYNNYY	6.1E-06	6.1E-06	6.1E-06	6.1E-06	6.1E-06
		TRRRRSRR	4.7E-06	2.8E-06	4.7E-06	5.6E-06	8.3E-06
		TICYYNYY	4.5E-07	8.9E-08	4.5E-07	5.9E-07	8.9E-07
		TINYYNYR	3.3E-08	6.6E-09	3.3E-08	4.4E-08	6.6E-08
		TICYYNYR	1.9E-08	3.8E-09	1.9E-08	2.5E-08	3.8E-08
		TLCYNNYR	1.0E-08	1.0E-08	1.0E-08	1.0E-08	1.0E-08
		TLAYNNYY	8.5E-09	8.5E-09	8.5E-09	8.5E-09	8.5E-09
		TLAYNNYR	2.5E-09	2.5E-09	2.5E-09	2.5E-09	2.5E-09
		TNAYNNYN	1.0E-09	3.9E-09	1.0E-09	1.8E-09	3.9E-09
		TNCYNNYY	1.1E-09	4.9E-11	5.4E-10	1.4E-09	5.7E-09
		TNCYNNYN	1.1E-09	5.9E-11	4.2E-10	8.4E-10	2.8E-09
4	SGTRs	HICYNXYY	1.3E-06	1.3E-06	1.3E-06	1.3E-06	1.3E-06
		GNCYNYNN	2.9E-08	1.1E-08	2.9E-08	5.1E-08	1.1E-07
		GNCYNYNY	2.9E-08	1.1E-08	2.9E-08	5.1E-08	1.1E-07
		GNCYNNYY	5.6E-09	5.6E-09	5.6E-09	5.6E-09	5.6E-09
		GNCYNYYN	2.2E-09	2.2E-09	2.2E-09	2.2E-09	2.2E-09
5	Event V	V	1.1E-07	1.1E-07	1.1E-07	1.1E-07	1.1E-07

This table was reproduced from Table 5-2 of the Zion Level 1 report by M. B. Sattison and K. W. Hall entitled, "Analysis of Core Damage Frequency: Zion; Unit 1," NUREG/CR-4550, Volume 7, Revision 1, May 1990. The information in this table reflects additions to the plant damage states that resulted from a review of the original analysis. These additions were of relatively low frequency and therefore did not affect the estimated total core damage frequency of 3.4×10^{-4} per reactor year. However, the frequency of some of the plant damage state groups did change. For example, the addition of PDS HICYNXYY changed the frequency of SGTR from 1.6×10^{-7} to 1.4×10^{-6} per reactor year. The frequencies of the PDSs given in Figures S.3, S.4, 2.5-2, and 2.5-3 in this report reflect the original PDS state estimates (summarized in Table 4.11-2 of NUREG/CR-4550, Volume 7, Revision 1, May 1990). However, the final risk estimates presented in Chapter 5 reflect the revised PDS frequencies given above.

Group 3 is denoted Transients. In PDS TL, only the LPIS is available. The AFWS is failed and Bleed and Feed does not work because the HPIS is failed. Also the operators have failed to depressurize before the onset of core damage in these PDSs.

Group 4 consists of four PDSs that are initiated by SGTRs and which do not have scram failures. HICY-NXYY is an SGTR with stuck-open safety relief valves (SRVs) in the secondary system.

Group 5 consists solely of Event V. This is a large break in the low pressure piping following the failure of the two check valves that isolate the low pressure piping from the RCS. The break is outside containment in the auxiliary building, so the break both fails the RCS pressure boundary and bypasses the containment.

2.2.3 High-Level Grouping of Plant Damage States

Section 2.2.2 describes the "super" groups considered in the Zion analysis; no regrouping was necessary as the definitions of each had been previously agreed upon for all plants under study in NUREG-1150.

2.2.4 Variables Sampled in the Accident Frequency Analysis

The variables used in the accident frequency analysis are listed and defined in Table 2.2-3. The second column in this table gives the range of the distribution for the variable (first column) and the third column indicates the type of distribution used. The fourth column shows whether the variable is correlated with any other variable, and the last column is a description of the variable. More complete descriptions and discussion of these variables may be found in the Zion accident frequency analysis report (NUREG/CR-4550, Vol. 7). This report also gives the source or the derivation of the distributions for all these variables.

2.3 Description of the Accident Progression Event Tree

This section describes the APET that is used to perform the accident progression analysis for Zion. The APET itself forms a high-level model of the accident progression. The APET is too large to be drawn out in a figure as smaller event trees usually are. Instead, the APET exists only as a computer input file.

The APET is not meant to be a substitute for detailed, mechanistic codes such as the STCP, CONTAIN, MELCOR, and MAAP. Rather, it is an integrating framework for synthesizing the results of these codes together with expert judgment on the strengths and weaknesses of the codes. The detailed, mechanistic codes require too much computer time to run for all the possible accident progression paths. Therefore, the results from these codes are represented in the Zion APET, which can be evaluated very quickly. In this way, the full diversity of possible accident progressions can be considered and the uncertainty in the many phenomena involved can be includes.

Table 2.2-3 Variables Sampled in the Accident Frequency Analysis

Variable	Range	Distribution	Correlation	Description
V-Freq.	7.3E-11 - 1.5E-06	Lognormal	None	Frequency (1/reactor-yr) of Interfacing Systems LOCA. Provided by INEL.
CCW/SW	8.9E-06 - 1.3E-02	Lognormal	None	Frequency (1/reactor-yr) of CCW/SW Failure. Aggregated over Expert Elicitation, (Zion-specific).
CCW/Av.	0.05 - 0.5	User Distribution	None	Conditional Probability of CCW/SW Availability (Zion-Specific). Provided by INEL.
PS-LOCA	0.01 - 0.25	User Distribution	None	Conditional Probability of Pump Seal LOCA (Zion-specific). Provided by INEL.
EP-Rec.	0.1 - 0.3	User Distribution	None	Conditional Probability of Electric Power Recovery (Zion-specific). Provided by INEL.
RWST-Ref.	0.1 - 1.0	User Distribution	None	Conditional Probability of Reactor Water Storage Tank Refil (Zion-specific). Provided by INEL.
CCW-PR	0.0 - 1.0	Uniform	None	Conditional Probability of CCW/SW Pipe Rupture. Unelicited.

User Distributions have three discrete levels.

The following section contains a brief overview of the Zion APET. Details, including a complete listing of the APET and a discussion of each question, may be found in Appendix A of this volume. Section 2.3.2 is a summary of how the APET is quantified, that is, how the many numerical values for branching ratios and parameters were derived. Section 2.3.3 presents the variables that were sampled in the accident progression analysis for Zion.

2.3.1 Overview of the APET

The APET for Zion considers the progression of the accident from the time the TAF in the core is uncovered (which is assumed to be the onset of core damage) through RPV failure and inlcuding any ex-vessel interactions such as core-concrete interaction (CCI). Although ex-vessel interactions will progress for days, the end of this analysis was usually set at 24 h.

While every effort has been made to develop a general event tree that can be applied to any large volume containment, the tree does include Zion specific design features. Therefore, as each plant has some unique features, some revision of this tree will be required for other plants.

Table 2.3.1 lists the 72 questions in the Zion APET. This APET is broken into seven time periods. The mnemonic branch abbreviations for most branches start with a character or characters which indicate the time period of the question. The time periods and their abbreviations are:

B	Initial	Questions 1 through 11 determine the conditions at the beginning of the accident.
E	Early	Questions 12 through 30 concern the accident progression from the uncovering of the TAF to just before vessel breach (VB). Questions 14 through 18 concern events or actions which may depressurize the RCS before VB. The possibility that core degradation may be arrested and VB prevented is considered in Question 23.
I	Inter-mediate	Questions 31 through 42 determine the progression of the accident immediately before and at VB, including the possibility of containment failure at VB.
I2	Late Inter-mediate	Question 43 determines the status of the sprays shortly after VB, during the RCS release.
L	Late	Questions 44 through 50 determine the progression during CCI.
L2	Very Late	Questions 51 through 64 determine the accident progression in the period following CCI, including the possibility of containment failure due to hydrogen combustion.
F	Final	Questions 65 through 72 determine the final status of the containment.

The duration of each period will vary depending upon the type of accident being modeled.

This APET does not contain any questions to resolve core-vulnerable sequences. These are PDSs which have failure of containment heat removal only. The continual deposition of decay heat in the containment by operation of the ECCS in the recirculation mode is predicted to lead to eventual containment failure after a few days. Containment failure, in turn, may lead to ECCS failure. The Zion PDSs with frequencies exceeding 1.0E-7/year did not contain any accidents of this type.

In the period after CCI, the concentrations of hydrogen, oxygen, steam, and carbon dioxide in the containment atmosphere are tracked grossly by means of a "User Function". Hydrogen combustion in the period before vessel failure is now generally considered to present no threat to large, dry containments. In this analysis, the pressure rise at VB was determined by a group of experts, the Containment Loads Panel. They provided distributions for pressure rise at VB and there is no way to ascertain how much hydrogen was generated at VB, and how much was consumed at that time.

Thus, tracking hydrogen through the VB event is fairly difficult to do in a manner that is consistent with the opinions of the experts. For this reason, hydrogen is tracked only after VB.

In several places in the evaluation of the APET, a User Function is called. This is a FORTRAN function subprogram which is executed at that point in the evaluation of the APET. The user function allows computations to be carried out which are too complex to be treated directly in the event tree. The user function itself is listed in Appendix A, and the manipulations performed by the user function at each question that utilizes the user function are described below. The user function is called to:

Determine containment failure and the mode of failure
- Questions 42 and 64

Add hydrogen produced during CCI to hydrogen already in containment
- Question 59

Determine if the containment atmosphere is flammable
- Question 61

Determine the pressure rise from a late hydrogen burn
- Question 63

2.3.2 Overview of the APET Quantification

This section presents a list of the question in the Zion APET and discusses the types of questions and their quantification briefly. A detailed discussion of each question may be found in Appendix A.

In addition to the number and name of the question, Table 2.3-1 shows how the question was sampled if the distribution came from an internal source, and how the question was evaluated or quantified. In the sampling column, an entry of DS indicates that the sampling is from a distribution provided by one of the expert panels, or from the electric power recovery distribution. The item sampled may be either the branching ratios or the parameter defined at that

Table 2.3-1

Top Event Questions in the Zion APET

Question Number	Question	Sampling	Quantification
1.	Size/Location of RCS Break when the Core Uncovers?	SF	PDS
2.	For SGTR, are the Secondary System SRVs Stuck Open?	SF	PDS
3.	Status of ECCS?	SF	PDS
4.	Status of Sprays?		PDS
5.	Status of Fan Coolers?		PDS
6.	Status of AC Power?		PDS
7.	RWST Injected into Containment?		PDS
8.	Heat Removal from the Steam Generators?	SF	PDS
9.	Did the Operators Depressurize the Secondary before the Core Uncovers?	SF	PDS
10.	Cooling for RCP Seals?	SF	PDS
11.	Initial Containment Leak or Isolation Failure?		SysAn
12.	Event V - Break Location under Water?	SF	Note 1
13.	RCS Pressure at the Start of Core Degradation?		SARRP
14.	Do the PORVs Stick Open?	SF	Note 2
15.	Temperature-Induced RCP Seal Failure?	ZO	Note 3
16.	Is the RCS depressurized before breach by opening the Pressurizer PORVs?		SARRP
17.	Temperature-Induced Hot Leg or Surge Line Break?	DS	In-Vessel
18.	Temperature-Induced SGTR?	DS	In-Vessel
19.	Is AC Power Available Early?	SF	Data
20.	After Power Recovery, Is Core Cooling Re-Established Promptly?		SysAn
21.	Rate of Blowdown to Containment?		Summary
22.	Vessel Pressure just before VB?		SARRP
23.	Is Core Damage Arrested? No VB?	SF	SARRP
24.	Early Sprays?		Summary
25.	Early Fan Coolers?		Summary
26.	Early Containment Heat Removal?		Summary
27.	Baseline Containment Pressure before VB?		SARRP
28.	Time of Accumulator Discharge?		Summary
29.	Fraction of Zr Oxidized In-Vessel during Core Degradation?	DS	In-Vessel
30.	Amount of Zr Oxidized In-Vessel during Core Degradation?		Summary

Table 2.3-1 (continued)

Top Event Questions in the Zion APET

Question Number	Question	Sampling	Quantification
31.	Amount of Water in the Reactor Cavity at VB?		Summary
32.	Fraction of Core Released from the Vessel at Breach?	DS	In-Vessel
33.	Amount of Core Released from the Vessel at Breach?		Summary
34.	Does an Alpha Event Fail both Vessel and Containment?	SF	Note 4
35.	Type of VB?	ZO	In-Vessel
36.	Does the Vessel become a "Rocket" and Fail the Containment?		SARRP
37.	Size of Hole in Vessel (after ablation)?	ZO	SARRP
38.	Total Pressure Rise at VB? Large Hole Cases	DS	Loads
39.	Total Pressure Rise at VB? Small Hole Cases	DS	Loads
40.	Does an Ex-Vessel Steam Explosion Occur?		SARRP
41.	Containment Failure Pressure?	DS	Struct.
42.	Containment Failure and Type of Failure?	DS	Struct.
43.	Sprays after VB?		SARRP
44.	Is AC Power Recovered Late?	DS	Data
45.	Late Sprays?		Summary
46.	Late Fan Coolers?		Summary
47.	Late Containment Heat Removal?		Summary
48.	Amount of Core available for CCI?		Summary
49.	Is the Debris Bed in a Coolable Configuration?		SARRP
50.	Does Prompt CCI Occur?		Summary
51.	Is AC Power Recovered Very Late?	DS	Data
52.	Very Late Sprays?		Summary
53.	Very Late Fan Coolers?		Summary
54.	Very Late Containment Heat Removal?		Summary
55.	Does Delayed CCI Occur?		Summary
56.	Baseline Containment Pressure Very Late?		SARRP
57.	How much Hydrogen and Carbon Dioxide is produced during CCI?		SARRP
58.	How Much Hydrogen Burns or Leaks Out of Containment?		SARRP
59.	Add H_2 produced by CCI to H_2 already in Containment?		UFUN
60.	Amount of Steam in Containment after CCI?		SARRP

Table 2.3-1 (continued)

Top Event Questions in the Zion APET

Question Number	Question	Sampling	Quantification
61.	Is the H_2 Concentration Flammable?		UFUN
62.	Does Ignition Occur?		SARRP
63.	Resulting Pressure Rise?		UFUN
64.	Containment Failure and Type of Failure?	DS	Struct.
65.	Sprays after Very Late CF?		SARRP
66.	Fan Coolers after Very Late CF?		Summary
67.	Containment Heat Removal after Very Late CF?		Summary
68.	Eventual Basemat Melt-through?		SARRP
69.	Eventual Overpressure Failure of Containment?		SARRP
70.	Basemat Melt-through before Overpressure Failure?		SARRP
71.	Final Containment Condition?		Summary
72.	Time of Core Damage?		Summary

Note 1. Whether the location of the break in the low pressure piping would be under water in Event V at the time that the core was uncovered was determined by a special panel which only considered this problem for the first draft version of this analysis. Since there was no new information provided, there was no reason to change the conclusions reached by this group. See the discussion of Question 12 in Appendix A.

Note 2. There is little or no data on the failure rate of PORVs when passing gases at temperatures considerably in excess of their design temperature. The quantification was arrived at by discussions between the systems analyst and the SARRP analyst. See the discussion of Question 14 in Appendix A.

Note 3. In the systems analysis, a special panel was convened to consider the issue of the failure of Reactor Coolant Pump Seals. The quantification of this question is not as detailed as that done in the systems analysis, but relies on the information produced by the panel. See the discussion of Question 15 in Appendix A.

Note 4. The Alpha mode of vessel and containment failure was considered by the Steam Explosion Review Group a few years ago. The distribution used in this analysis is based on information contained in the report of this group. See the discussion of Question 34 in Appendix A.

question. For questions which are sampled and which were quantified internally, the entry ZO in the sampling column indicates that the question was sampled zero-one, and the entry SF means the questions was sampled with split fractions. The difference may be illustrated by a simple example. Consider a uniform distribution from 0 to 1 for a branching ratio. If the sampling is zero-one, half the time the branching ratio in question will be 1 and the other half of the time it will be 0. If the sampling is split fraction, the branching ratio will take on a selection of fractional values between 0 and 1 such that their average is 0.5. The implications of ZO or SF sampling are discussed in the methodology volume.[5]

If the sampling column is blank, the branching ratios for that question, and the parameter values defined in that that question, if any, are fixed. The branching ratios of the PDS questions change to indicate which PDS is being considered. Some of the branching ratios depend on the relative frequency of the PDSs which make up the PDS group being considered. These branching ratios change for every sample observation, but may do so for some PDS groups and not for others. If the branching ratios change from observation to observation for any one of the seven PDS groups, SF is placed in the sampling column for the PDS questions.

The abbreviations in the quantification column of the Table 2.3-1 are given below, with the number of questions which have that type of quantification indicated.

Type of Quant.	Number of Questions	Comments
PDS	10	Determined by the Plant Damage State.
SysAn	2	Determined by the Systems Analysis.
Other	4	See Notes 1 through 4 in Table 2.3-1.
SARRP	19	Quantified internally in this analysis. (Severe Accident Risk Reduction Program)
Summary	21	The branch taken at this question follows directly from the branches taken at previous questions.
Data	3	The probability of electric power recovery is determined by distributions derived from electric power recovery data for this plant.
UFUN	3	Calculated in the User Function.
In-Vessel	5	Distributions from the In-Vessel Expert Panel.
Loads	2	Distributions from the Containment Loads Expert Panel.
Struct.	3	Distributions from the Structural Expert Panel.
N.A.	0	Questions not applicable to Zion.

In some cases, a question may have more than one function, so the entry under Quantification in Table 2.3-1 can be only indicative. For example, Questions 42 and 64 are listed as being quantified by distributions generated by the Structural Expert Panel. The actual situation is more complicated. In these questions, a portion of the user function is evaluated which determines whether the containment fails using the failure pressure defined in Question 41. If the failure pressure is lower than the load pressure, then the containment fails and the mode of failure is determined using the random number defined in Question 41 and a table of conditional failure mode probabilities contained in the user function. This table was also generated by the Structural Expert Panel. So the quantification entry for Questions 42 and 64 could have been either UFUN or Struct.

2.3.3 Variables Sampled for the Accident Progression Analysis

About 70 variables, listed in Table 2.3-2, were sampled for the accident progression analysis. That is, every time the APET was evaluated by EVNTRE, the original values of about 70 variables were replaced with values selected for the particular observation under consideration. These values were selected by the LHS program from distributions that were defined before the APET was evaluated. Most of these distributions were determined by expert panels. Some are branch fractions, others are parameter values for use in calculations performed by user functions in EVNTRE.

In Table 2.3-2, the first column gives the variable identifier, and the question (and case if appropriate) in which the variable is used.

The second column gives the range of the distribution for the variable. An entry of "0.0-1.0" in this column indicates that the variable took on fractional values between 0.0 and 1.0. An entry of "Zero/One" in this column indicates that the variable was sampled Zero-One; i.e., it took on only the values 0.0 or 1.0.

The third column indicates the type of distribution used. The entry "Experts" for the distribution indicates that the distribution came from an expert panel and the entry "Internal" indicates that the distribution was determined internally by the project staff. Plots of the aggregate expert distributions are contained in Volume 2 of this report. For Zero-One variables, an indication of the probability of each state is given in this column.

The fourth and fifth columns show whether the variable is correlated with any other. "Rank 1" indicates a rank correlation of 1.0. For further information on each of the variables listed see the detailed discussion of the indicated APET question in Appendix A.

The RCS pressure at VB variables, RCSPR-VB2 and RCSPR-VB3, (Question 22), are sampled Zero-One. The distribution column gives the fraction of the time each of the pressure ranges is chosen. Low is below 200 psia, Im indicates the intermediate pressure range, from 200 to 600 psia. The High pressure range extends from 600 to 2000 psia, but is nominally about 1000 to 1500 psia. Setpoint pressure refers to the PORV and SRV settings, about 2500 psia.

Table 2.3-2

Variables Sampled in the Accident Progression Analysis

Variable Question & Case	Range	Distribution	Corre-lation	Correlation with	Description
V-UWATER Q12	0.0 - 1.0	Uniform	None		Probability that the break location will be underwater when the radioactive releases begin, given Event V.
PORV-OPN Q14 C1	0.0 - 1.0	Uniform	None		Probability that at least one PZR PORV or RCS SRV sticks open, given that the RCS is intact and the PORVs or SRVs are cycling.
RCP-SL-P2 Q15 C2	Zero-One	Fail 0.71	Rank 1	RCP-SL-Pn	Probability of a T-I failure of the RCP seals, given core damage, RCS at setpoint pressure, and no cooling for the RCP seals.
RCP-SL-P3 Q15 C3	Zero-One	Fail 0.65	Rank 1	RCP-SL-Pn	Probability of a T-I failure of the RCP seals, given core damage, RCS at high pressure, and no cooling for the RCP seals.
RCP-SL-P4 Q15 C4	Zero-One	Fail 0.60	Rank 1	RCP-SL-Pn	Probability of a T-I failure of the RCP seals, given core damage, RCS at intermediate or low pressure, and no cooling for the RCP seals.
TI-SGTR Q18 C1	0.0 - 0.12	Experts Median = 0.01	None		Probability of a T-I SGTR, given core damage, RCS at set-point pressure, and no cooling for the steam generators.
TI-HOTLG1 Q17 C1	0.0 - 1.0	Experts Median = 0.77	Rank 1	TI-HOTLG2 FR-ZROXn	Probability of a T-I failure of the hot leg or surge line given core damage, AFWS failure, and the RCS intact at setpoint pressure.
TI-HOTLG2 Q17 C2	0.0 - 1.0	Experts Median = 0.04	Rank 1	TI-HOTLG1 FR-ZROXn	Probability of a T-I failure of the hot leg or surge line, given core damage, AFWS failure, and an S_3 break in the RCS.

Table 2.3-2 (continued)

Variables Sampled in the Accident Progression Analysis

Variable Question & Case	Range	Distribution	Corre- lation	Correlation with	Description
RCSPR-VB2 Q22 C2	Zero-One	0.20 Low 0.80 Im	Rank 1	RCSPR-VB3	RCS pressure just before vessel breach, given an initiating or induced S_2 break.
RCSPR-VB3 Q22 C3	Zero-One	0.33 Low 0.34 Im 0.33 High	Rank 1	RCSPR-VB2	RCS pressure just before vessel breach, given an initiating or induced S_3 break.
CDARREST5 Q23 C5	0.8 - 1.0	Uniform	Rank 1	CDARRESTn	Probability that core damage can be arrested before VB, given the conditions of Case 5. (Also used for Case 8.)
CDARREST6 Q23 C6	0.0 - 1.0	Quadratic Median = 0.67	Rank 1	CDARRESTn	Probability that core damage can be arrested before VB, given the conditions of Case 6.
CDARREST7 Q23 C7	0.0 - 1.0	Uniform	Rank 1	CDARRESTn	Probability that core damage can be arrested before VB, given the conditions of Case 7.
FR-ZROX1 Q21 C1	0.0 - 1.3	Experts Median = 0.44	Rank 1	TI-HOTLGn FR-ZROXn	Fraction of equivalent core Zr oxidized in-vessel given that the RCS is at setpoint pressure and the accumulators discharge before or after core melt.
FR-ZROX2 Q21 C2	0.0 - 1.3	Experts Median = 0.50	Rank 1	TI-HOTLGn FR-ZROXn	Fraction of equivalent core Zr oxidized given that the RCS is at setpoint pressure and the accumulators discharge during core melt.
FR-ZROX3 Q21 C3	0.0 - 0.80	Experts Median = 0.32	Rank 1	TI-HOTLGn FR-ZROXn	Fraction of equivalent core Zr oxidized in-vessel given that the RCS is at high pressure and the accumulators discharge before or after core melt.

Table 2.3-2 (continued)

Variables Sampled in the Accident Progression Analysis

Variable Question & Case	Range	Distribution	Corre-lation	Correlation with	Description
FR-ZROX4 Q29 C4	0.0 - 0.85	Experts Median = 0.38	Rank 1	TI-HOTLGn FR-ZROXn	Fraction of equivalent core Zr oxidized given that the RCS is at high pressure and the accumulators discharge during core melt.
FR-ZROX5 Q29 C5	0.0 - 1.2	Experts Median = 0.48	Rank 1	TI-HOTLGn FR-ZROXn	Fraction of equivalent core Zr oxidized in-vessel given that the RCS is at intermediate pressure and the accumulators discharge before or after core melt.
FR-ZROX6 Q29 C6	0.0 - 1.2	Experts Median = 0.52	Rank 1	TI-HOTLGn FR-ZROXn	Fraction of equivalent core Zr oxidized in-vessel that that the RCS is at intermediate pressure and the accumulators discharge during core melt.
FR-ZROX7 Q29 C7	0.0 - 1.2	Experts Median = 0.45	Rank 1	TI-HOTLGn FR-ZROXn	Fraction of equivalent core Zr oxidized given that the RCS is at low pressure and the accumulators discharge before core melt.
FR-HPME Q32	0.0 - 0.60	Experts Median = 0.3	None		Fraction of core which participates in HPME at VB.
VB-ALPHA Q34 C1	0.0 - 1.0	Experts Median = 0.01	None		Probability that an Alpha mode CF occurs, given that the RCS is at low pressure. (One tenth this value is utilized for Case 2.)
TYPE-VB1 Q35 C2	Zero-One	Experts HPME 0.79 BtmHd 0.08 Pour 0.13	Rank 1	TYPE-VB2	Type of VB given that the RCS is at setpoint pressure.

Table 2.3-2 (continued)

Variables Sampled in the Accident Progression Analysis

Variable Question & Case	Range	Distribution	Correlation	Correlation with	Description
TYPE-VB2 Q35 C3	Zero-One	Experts HPME 0.60 BtmHd 0.27 Pour 0.13	Rank 1	TYPE-VB1	Type of VB given that the RCS is at high pressure. (Also used for Case 4.)
VBHOLSIZ Q37 C1	Zero-One	0.1 Large 0.9 Small	None		Size of the hole in the vessel after ablation given high pressure melt ejection.
PRISE-LO Q38 C3	0.4 - 4.5 ba	Experts Median - 1.75	None		Pressure rise at VB given that the RCS is at low pressure or the mode of VB is Pour.
PRISE-VB1 Q38 C5	1.0 - 8.0 ba	Experts Median - 4.49	Rank 1	PRISE-VBn	Pressure rise at VB given Med. RCS pressure, high fraction melt ejected, large hole, wet cavity.
PRISE-VB2 Q38 C6	0.9 - 6.0 ba	Experts Median - 3.65	Rank 1	PRISE-VBn	Pressure rise at VB given Med. RCS pressure, medium fraction melt ejected, large hole, wet cavity.
PRISE-VB3 Q38 C7	0.6 - 5.3 ba	Experts Median - 2.7	Rank 1	PRISE-VBn	Pressure rise at VB given Med. RCS pressure, low fraction melt ejected, large hole, wet cavity.
PRISE-VB4 Q38 C8	1.7 - 8.0 ba	Experts Median - 5.02	Rank 1	PRISE-VBn	Pressure rise at VB given Med. RCS pressure, high fraction melt ejected, large hole, dry cavity.
PRISE-VB5 Q38 C9	1.6 - 6.1 ba	Experts Median - 4.02	Rank 1	PRISE-VBn	Pressure rise at VB given Med. RCS pressure, medium fraction melt ejected, large hole, dry cavity.
PRISE-VB6 Q38 C10	1.0 - 5.3 ba	Experts Median - 2.88	Rank 1	PRISE-VBn	Pressure rise at VB given Med. RCS pressure, low fraction melt ejected, large hole, dry cavity.

Table 2.3-2 (continued)

Variables Sampled in the Accident Progression Analysis

Variable Question & Case	Range	Distribution	Corre-lation	Correlation with	Description
PRISE-VB7 Q38 C11	1.0 - 10.2 ba	Experts Median = 5.62	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpoint RCS pressure, wet cavity, high fraction melt ejected, large hole.
PRISE-VB8 Q38 C12	1.0 - 7.8 ba	Experts Median = 4.55	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpoint RCS pressure, wet cavity, med. fraction melt ejected, large hole.
PRISE-VB9 Q38 C13	0.7 - 6.2 ba	Experts Median = 3.15	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpoint RCS pressure, wet cavity, low fraction melt ejected, large hole.
PRISE-VB10 Q38 C14	1.7 - 9.9	Experts Median = 6.18	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpoint RCS pressure, dry cavity, high fraction melt ejected, large hole.
PRISE-VB10 Q38 C15	1.7 - 8.1	Experts Median = 4.97	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpoint RCS pressure, dry cavity, med. fraction melt ejected, large hole.
PRISE-VB10 Q38 C16	1.0 - 6.3	Experts Median = 3.36	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpoint RCS pressure, dry cavity, low fraction melt ejected, large hole.
PRISE-VB13 Q39 C2	0.8 - 7.7 ba	Experts Median = 3.81	Rank 1	PRISE-VBn	Pressure rise at VB given Med. RCS pressure, high fraction melt ejected, small hole, wet cavity.
PRISE-VB14 Q39 C3	0.83 - 5.3 ba	Experts Median = 3.13	Rank 1	PRISE-VBn	Pressure rise at VB given Med. RCS pressure, medium fraction ejected, small hole, wet cavity.
PRISE-VB15 Q39 C4	0.6 - 4.9 ba	Experts Median = 2.44	Rank 1	PRISE-VBn	Pressure rise at VB given Med. RCS pressure, low fraction melt ejected, small hole, wet cavity.

Table 2.3-2 (continued)

Variables Sampled in the Accident Progression Analysis

Variable Question & Case	Range	Distribution	Corre-lation	Correlation with	Description
PRISE-VB16 Q39 C5	1.1 - 7.6 ba	Experts Median = 4.25	Rank 1	PRISE-VBn	Pressure rise at VB given medium RCS pressure, high fraction melt ejected, small hole, dry cavity.
PRISE-VB17 Q39 C6	0.9 - 5.8 ba	Experts Median = 3.48	Rank 1	PRISE-VBn	Pressure rise at VB given medium RCS pressure, medium fraction melt ejected, small hole, dry cavity.
PRISE-VB18 Q39 C7	0.65 - 5.1 ba	Experts Median = 2.62	Rank 1	PRISE-VBn	Pressure rise at VB given medium RCS pressure, low fraction melt ejected, small hole, dry cavity.
PRISE-VB19 Q39 C8	1.0 - 9.7 ba	Experts Median = 5.04	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpoint RCS pressure, wet cavity, high fraction melt ejected, small hole.
PRISE-VB20 Q39 C9	0.9 - 7.9 ba	Experts Median = 4.05	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpoint RCS pressure, wet cavity, medium fraction melt ejected, small hole.
PRISE-VB21 Q39 C10	0.65 - 6.2 ba	Experts Median = 2.9	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpoint RCS pressure, wet cavity, low fraction melt ejected, small hole.
PRISE-VB22 Q39 C11	1.5 - 9.2 ba	Experts Median = 5.62	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpoint RCS pressure, dry cavity, high fraction melt ejected, small hole.
PRISE-VB23 Q39 C12	1.3 - 7.5 ba	Experts Median = 4.51	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpooint RCS pressure, dry cavity, medium fraction melt ejected, small hole.
PRISE-VB25 Q39 C13	0.9 - 6.0 ba	Experts Median = 3.13	Rank 1	PRISE-VBn	Pressure rise at VB given high or setpooint RCS pressure, dry cavity, low fraction melt ejected, small hole.
POWERREC Q20, Q44, Q51			None		Variable used to select the probability that offsite power will be recovered in a specified time interval given that it was not recovered in a previous time interval.

Both the T-I hot leg failure variables (Question 17) and all the fraction of Zr oxidized variables (Question 21) are correlated with each other as the experts concluded that the oxidation of a lot of zirconium before VB would result in high temperatures which in turn would make hot leg or surge line failure more likely. This reasoning included the T-I SGTR as well as the hot leg break, and it was intended that variable TI- SGTR would be correlated with TI-HOTLG and FR-ZROX. Due to an oversight, this correlation was omitted. As T-I SGTRs were very infrequent, the omission of this correlation was not significant.

The type of vessel failure variables (Question 35) are sampled Zero-One and the entries under "Distribution" indicate the probability of each type of VB. HPME indicates ejection of the melt at high pressure through a hole that is small relative to the cross-section of the vessel. BtmHd indicates a gross failure of the entire bottom head of the vessel, and Pour indicates a slow release of the melt driven primarily by gravity. For the hole size (Question 37), large means greater than 0.4 m^2 (nominally 2.0 m^2) and small means smaller than 0.4 m^2 (nominally 0.1 m^2).

For the numerous pressure rise at VB variables (Questions 38 and 39), wet cavity means the cavity contains at least the accumulator water or that the cavity is full. The fraction of the core ejected at VB (Question 32) was placed into three groups in Question 33. High fraction ejected means greater than 40%, medium fraction ejected means between 20% and 40%, and low fraction ejected means less than 20%.

The failure mode, as a function of failure pressure, was determined by the structural expert panel. The containment failure mode variable, CF- MODE (Question 42), is a random variable used to determine the failure mode in the user functions. The method used to select the failure mode for each observation is explained in Volume 1, and the results of the expert panel on the failure pressure and failure mode for Zion may be found in Volume 2.

The last variable listed is used to select the probability that offsite power will be recovered in a specified time interval given that it was not recovered in a previous time interval. Distributions were developed for 12 cases, each with different start and end times, corresponding to different classes of accidents. The variable POWERREC defines a quantile for these distributions and the associated recovery probabilities are used in the analysis. Additional information can be found in Appendix A of this volume.

2.4 Description of the Accident Progression Bins

As each path through the APET is evaluated, the result of that evaluation is stored by assigning it to an Accident Progression Bin. This bin describes the evaluation in enough detail that a source term (release of radionuclides) can be calculated for it. The accident progression bins are the means by which information is passed from the accident progression analysis to the source term analysis. A bin if defined by specifying the attribute or value for each of 12 characteristics or quantities which define a certain feature of the evaluation of the APET. Section 2.4.1 describes the 12 characteristics, and the values that each characteristic can assume. The binner, which follows directly after the APET in the data file which forms the input to EVNTRE, is listed in Appendix A. It may be found following Question 72. A detailed description of each case in the binner is given in Appendix A. Section 2.4.2 contains a discussion of rebinning, a process that takes place between evaluating the APET (in which binning takes place) and the source term analysis. Section 2.4.3 describes a reduced set of binning characteristics which is used in presenting the results.

2.4.1 Description of the Bin Characteristics

The binning scheme for Zion utilizes 12 characteristics. That is, there are 12 types of information required to define a path through the APET. A bin is defined by specifying a letter for each of the 12 characteristics, where each letter for each characteristic has a meaning which will be defined below. For a characteristic, the possible states are termed attributes. The Zion binning characteristics are:

Characteristic	Abbreviation	Description
1	CF-Time	Time of Containment Failure
2	Sprays	Periods in which Sprays Operate
3	CCI	Occurrence of Core-Concrete Interactions
4	RCS-Pres	RCS Pressure before VB
5	VB-Mode	Mode of Vessel Breach
6	SGTR	Steam Generator Tube Rupture
7	Amt-CCI	Amount of Core Available for CCI
8	Zr-Ox	Fraction of Zr Oxidized In-Vessel
9	HPME	Fraction of the Core in high pressure melt ejection
10	CF-Size	Size or Type of Containment Failure (CF)
11	RCS-Hole	Number of Large Holes in the RCS after VB
12	CD-Time	Time of Core Damage

Most of this information, organized in this manner, is needed by ZISOR to calculate the fission product source terms. Characteristic 5, Mode of VB, is not used by ZISOR, but has been retained because it provides interesting output information about the APET outcome, or the paths taken through the APET. ZISOR obtains the information it needs concerning high pressure melt ejection (HPME) from Characteristic 9, Fraction of the Core in HPME.

The remainder of this section consists of a brief description of each characteristic, followed by an explanation of an example bin. A listing of each attribute or value for each of the 12 characteristics is given in Table 2.4-1.

Characteristic 1 primarily concerns the time of containment failure. There are seven attributes. Five of these attributes concern the time of containment failure. Two of the attributes concern Event V, a special failure mode. SGTRs are considered separately in Characteristic 6.

Characteristic 2 concerns the periods in which the sprays operate. The division into the eight attributes is a straightforward sorting out of the various combinations of time periods.

Characteristic 3 concerns the core-concrete interactions. There are six possibilities which cover the meaningful combinations of prompt CCI, delayed CCI, and no CCI, with the amount of water in the cavity. The amount of water in the cavity may be divided into three cases. If the cavity was dry at VB and the accumulators have discharged before breach, the cavity is dry at the start of CCI. If the cavity was dry at VB and the accumulators discharge at breach, the cavity will be about one quarter full. If the sprays operate before breach, then the cavity will be full.

Characteristic 4 concerns the pressure in the reactor vessel before VB; there are four levels. The pressures shown in parentheses in Table 2.4-1 are approximate pressures just before VB. The RCS pressure during most of the core degradation period is often less than this value.

Characteristic 5 concerns the mode of VB; there are six possibilities, including no VB. Direct heating of the containment always occurs to some extent if there is HPME, so there is no simple way to distinguish whether direct containment heating occurs.

Characteristic 6 concerns steam generator tube rupture. There are only three possibilities: no SGTR, SGTR, and SGTR with the SRVs on the secondary system stuck open. SGTR is considered separately from the other containment failure modes since it can occur in addition to the other failure modes. That is, occurrence of an SGTR before VB does not preclude containment failure at VB or late containment failure. As the SGTR creates a bypass of the containment which may have no removal mechanisms, it is important to treat this escape path separately.

Characteristic 7 concerns how much of the core not in HPME that is available to participate in the core-concrete interaction. The fractions 0.3 and 0.7 divide the range into three portions. The fourth attribute is no CCI. As ZISOR subtracts out the fraction of the core involved in HPME, the fraction of the core available for CCI is always set to Large when HPME occurs.

Characteristic 8 concerns the amount of the core zirconium which is oxidized in-vessel before VB. There are two possible values for this characteristic: low and high. The demarcation point between the two ranges is 40%.

Table 2.4-1. Description of Attributes for Each Binning Characteristic

Characteristic 1 - Containment Failure Time	
A - V-Dry	Event V, Break Location not Submerged
B - V-Wet	Event V, Break Location Submerged
C - Early-CF	Containment Failure before VB (Isolation Failure not followed by Containment Failure)
D - CF-at-VB	Containment Failure at VB
E - VLate-CF	Very Late Containment Failure (nominally a few hours after the start of CCI)
F - Final-CF	Containment Failure in the Final Period (nominally at least 24 h after VB)
G - No-CF	No Containment Failure (including STGRs)

Characteristic 2 - Sprays	
A - Sp-Early	The sprays operate only in the Early period.
B - Sp-E+I	The sprays operate only in the Early and Intermediate periods.
C - Sp-E+I+L	The sprays operate only in the Early, Intermediate, and Late periods.
D - SpAlways	The sprays Always operate during the periods of interest for fission product removal.
E - Sp-Late	The sprays operate only in the Late period.
F - Sp-L+VL	The sprays operate only in the Late and Very Late periods.
G - Sp-VL	The sprays operate only in the Very Late period.
H - Sp-Never	The sprays Never operate during the accident, or the sprays operate only during the Final period, which is not of interest for fission product removal.

Table 2.4-1 ctd. Description of Attributes for Each Binning Characteristic

Characteristic 3 - Core-Concrete Interactions	
A - Prmpt-Dry	CCI takes place promptly following VB. There is no overlying water pool to scrub the releases.
B - PrmptShlw	CCI takes place promptly following VB. There is a shallow overlying water pool to scrub the releases.
C - No-CCI	CCI does not take place.
D - PrmptDeep	CCI takes place promptly following VB. There is a deep overlying water pool to scrub the releases.
E - SDlyd-Dry	CCI takes place after a short delay. The debris bed is coolable, but the water in the cavity is not replenished. The delay is the time needed to boil off the accumulator water.
F - LDlyd-Dry	CCI takes place after a long delay. The debris bed is coolable, but the water in the cavity is not replenished. The delay is the time needed to boil off the water in a full cavity.

Characteristic 4 - RCS Pressure before VB	
A - SSPr	System Setpoint Pressure, 17.2 MPa (2500 psia)
B - HiPr	High Pressure, 6.9 to 13.8 MPa (1000 to 2000 psia)
C - ImPr	Intermediate Pressure, 1.4 to 6.9 MPa (200 to 1000 psia)
D - LoPr	Low Pressure, less than 1.4 MPa (200 psia)

Table 2.4-1 ctd. Description of Attributes for Each Binning Characteristic

Characteristic 5 - Mode of VB	
A - VB-HPME	High Pressure Melt Ejection occurs - direct heating always occurs to some extent.
B - VB-Pour	The molten core Pours out of the vessel, driven primarily by the effects of gravity.
C - VB-BtmHd	Gross failure of a large portion of the Bottom Head of the vessel occurs, perhaps as a result of a circumferential failure.
D - Alpha	An Alpha mode failure occurs - resulting in containment failure as well as vessel failure.
E - Rocket	A Rocket mode failure occurs - resulting in containment failure as well as vessel failure.
F - No-VB	No VB occurs.

Characteristic 6 - Steam Generator Tube Rupture	
A - SGTR	A SGTR occurs. The SRVs on the secondary system are not stuck open.
B - SGTR-SRVO	A SGTR occurs. The SRVs on the secondary system are stuck open.
C - No-SGTR	A SGTR does not occur.

Characteristic 7 - Amount of Core not in HPME available for CCI	
A - Lrg-CCI	A CCI occurs and involves a Large Amount of the Core (70-100%).
B - Med-CCI	A CCI occurs and involves a Medium amount of the Core (30-70%).
C - Sml-CCI	A CCI occurs and involves a Small amount of the Core (0-30%).
D - No-CCI	No CCI occurs.

Table 2.4-1 ctd. Description of Attributes for Each Binning Characteristic

Characteristic 8 - Zr Oxidation	
A - Lo-ZrOx	A Low amount of the core Zirconium was Oxidized in the vessel prior to VB. The implies a range from 0-40% oxidized, with a nominal value of 25%.
B - Hi-ZrOx	A High amount of the core Zirconium was Oxidized in the vessel prior to VB. This implies that more than 40% of the Zr was oxidized, with a nominal value of 65%.

Characteristic 9 - High Pressure Melt Ejection (HPME)	
A - Hi-HPME	A High fraction (> 40%) of the core was ejected under pressure from the vessel at failure.
B - Md-HPME	A Moderate fraction (20-40%) of the core was ejected under pressure from the vessel at failure.
C - Lo-HPME	A Low fraction (< 20%) of the core was ejected under pressure from the vessel at failure.
D - No-HPME	There was no HPME at vessel failure.

Table 2.4-1 ctd. Description of Attributes for Each Binning Characteristic

Characteristic 10 - Containment Failure Size	
A - Cat-Rupt	The containment failed by catastrophic rupture, resulting in a very large hole and gross structural failure.
B - Rupture	The containment failed by the development of a large hole or rupture; nominal hole size is 2.13 m^2 (7 ft^2).
C - Leak	The containment failed by the development of a small hole or a leak; nominal hole size is 0.03 m^2 (0.10 ft^2).
D - Shear	The containment failed by the development of a large hole or shear rupture at the cylinder-basemat junction area; nominal hole size is 2.13 m^2 (7 ft^2).
E - BMT	The containment failed by BMT.
F - Bypass	The containment did not fail but was bypassed by event V or an SGTR.
G - No-CF	The containment did not fail.

Characteristic 11 - Holes in the RCS	
A - 1-Hole	There is only One large Hole in the RCS following VB, so there is no effective natural circulation through the RCS after breach.
B - 2-Holes	There are Two large Holes in the RCS following VB, so there will be effective natural circulation through the RCS after breach.

Characteristic 12 - Timing of Core Damage	
A - E-CD	Onset of Core Damage within 2 h from accident initiation.
B - L-CD	Core Damage onset delayed by SGTR to 2 to 4 h after accident initiation.

Characteristic 9 concerns the amount of the core involved in HPME; there are four attributes. The possible range is divided into three portions by 20% and 40%, and no HPME is the fourth attribute.

Characteristic 10 concerns the size of the hole that results from containment failure or the type of containment failure. There are seven attributes. Unless otherwise specified, the failure location is the containment wall above ground. However, basemat melt-through (BMT) (Characteristic 11, attribute E) is classified as a very late (Characteristic 1, attribute F) leak for the calculations of the source terms, since the consequences of BMT are very small, as are the consequences of a very late leak. Event V and SGTR are classed as Bypasses and are not considered to be containment failures since the containment pressure boundary itself is intact. A catastrophic rupture is a failure of the containment pressure boundary that results in a very large hole (considerably greater than 2.13 m^2 [7 ft^2]) and extensive structural damage. A rupture is a hole on the order of 2.13 m^2 (7 ft^2), and a leak is a hole on the order of 0.03 m^2 (0.10 ft^2).

Characteristic 11 concerns the number of large holes in the RCS after breach. The experts on the source term panel who provided distributions for revolatilization from the RCS surfaces after breach gave different distributions depending on whether an effective natural circulation flow would be set up within the vessel. A significant flow could be expected only if there were two large, effective holes in the RCS; for example the hole in the bottom head resulting from vessel failure and a large temperature-induced hole in the hot leg. SGTRs, failure of the RCP seals, and Event V's would not count as large effective holes since effective natural circulation through the RCS would not result in these cases. S3-size holes are not considered large enough to result in effective natural circulation after VB.

Characteristic 12 concerns the timing of the onset of core damage. There are two attributes (early and late core damage), determined by the operation or operability of secondary heat removal. This characteristic has been included as an interface with ZISOR to assign proper delay times to bins which, although have the same definition of radionuclide release, belong to different PDSs.

A listing of the attributes for each characteristic can be found in Table 2.4-1. A typical bin might be FFADBCABDDBB, which, using the information presented in this table is:

F - Final-CF	Containment Failure in the Final Period
F - Sp-L+VL	Sprays only in the Late and Very Late periods.
A - Prmpt-Dry	Prompt CCI, Dry cavity
D - LoPr	Low Pressure in the RCS at VB
B - VB-Pour	Core material Poured out of the vessel at breach
C - No-SGTR	No Steam Generator Tube Rupture
A - Lrg-CCI	A Large fraction of the core was available for CCI
B - Hi-ZrOx	A High fraction of the Zr was Oxidized in-vessel
D - No-HPME	No High Pressure Melt Ejection

D - BMT	Basemat Melt-Through
B - 2-Holes	Two Holes in the RCS
B - L-CD	Core Damage delayed

2.4.2 Rebinning

The binning scheme utilized for the evaluation of the APET does not exactly match the input information required by ZISOR. The additional information in the initial binning is kept because it provides a better record of the outcomes of the APET evaluation. Therefore, there is a step between the evaluation of the APET and the evaluation of ZISOR known as "rebinning". In the rebinning, a few attributes in some characteristics are combined because there is no significant difference between them for calculating the fission product releases.

In the rebinning for Zion, only the 10th characteristic (mode of containment failure) is modified. The fourth and sixth attributes (D - BMT and F - No-CF) are combined into a new attribute D (No-CF) since the magnitude of the source terms and release is determined, for these bins, by the first characteristics alone. It should be noted that ZISOR ignores the attributes in this characteristic, past the third one (leak).

2.4.3 Summary Bins for Presentation

For presentation purposes in NUREG-1150, a set of "summary" bins has been adopted. Instead of the 12 characteristics and thousands of possible bins that describe the evaluation of the APET in detail, the summary bins place the outcomes of the evaluation of the APET into a few, very general number of groups. The four summary bins for Zion are:

Early CF (including Alpha, DCH, & isolation failure)
Late CF (including BMT)
Bypass
No CF

The order used in assigning results to these bins, however, is not the order given above. For assignment to reduced bins, the events are considered in the following order:

Bypass
Alpha
Early CF
Late CF
No CF

The reason that the reduced bins must have a definite priority is that all possible outcomes do not fit neatly into the four reduced bins. There are certain combinations of events which can be put in different places in the reduced bins and there are other combinations of events which do not fit well in any of the reduced bins.

A simple problem combination is Event V followed by an Alpha mode failure of the vessel and containment. Should this go in the Early CF reduced bin, or the

Bypass reduced bin? By the priority list above, it is placed in the Bypass reduced bin. The reason is that almost all of the fission products from the core before VB will have escaped to the auxiliary building through the bypass before VB. Thus this path determines most of the risk. Although SURSOR treats the CCI release as if all of it escapes through the ruptured containment, the early release is more important for determining offsite risk.

The placement in reduced bins of five other ambiguous combinations of events is discussed below.

Combination 1: V & B-Leak.

The Event V and Isolation Failure release (as calculated by ZISOR) is much closer to the release from Event V without Isolation Failure than it is to the release from accident with Isolation Failures which have no initial bypass of the containment. Therefore, this combination is placed in the Bypass reduced bin.

Combination 2: V & Alpha.

The release from Event V followed by an Alpha mode failure at VB (as calculated by ZISOR) is much closer to release from an Event V that is not followed by containment failure at VB than it is to the release from an accident that has initial bypass of the containment but which has an Alpha mode failure of the vessel and the containment. Thus, the V & Alpha combination is also classed as a Bypass.

Combination 3: V & CF-at-VB.

This combination is analogous to Combination 2, discussed above, except that the containment fails at VB for reasons other than an Alpha mode failure. It is also placed in the Bypass reduced bin.

Combination 4: SGTR & Alpha and SGTR & CF-at-VB.

ZISOR is set up to handle SGTRs in addition to other failures of the containment, so this is no problem for the source term calculation. As the SGTR largely determines the early release, and the early release is more important than the late release, these combinations of SGTR and containment failure at VB are placed in the Bypass reduced bin.

Combination 5: SGTR & CF-Late

This combination is analogous to Combination 4. Here the SGTR completely dominates the releases; there is no question that this combination should go to the Bypass reduced bin.

Thus, in assigning combinations of events in the APET to reduced bins, bypass failures (V and SGTR) take precedence no matter what else happens or does not happen. Alpha mode failures take precedence over other failure modes at VB, and over isolation failures.

2.5 Results of the Accident Progression Analysis

This section presents the results of evaluating the APET. As evaluating the APET produces a number of accident progression bins (APBs), the discussion is primarily in terms of APBs. Some intermediate results are also presented. Sensitivity analyses are discussed as well.

Section 2.5.1 presents the results for the internal initiators. Section 2.5.2 discusses the sensitivity analyses run for the internal intiators. The tables in this section and the results in Appendix A contain only a very small portion of the output obtained by evaluating the APETs. Complete listings giving average bin conditional probabilities for each PDS group, and listings giving the bin probabilities for each PDS group for each observation are available on computer media by request.

2.5.1 Results for Internal Intiators

This section presents the results for those accidents initiated by internal events at Zion. PDSs initiated by LOCAs (PDS Group 2) dominated (over 90%) the TCDF at Zion. Consequently the results for this PDS group are discussed in more detail than the other PDSs. The APET results for the other Zion PDS follow similar trends to the results obtained for the Surry analysis. Reference should therefore be made to the Surry volume (Volume 3 of this report) for more details on PDSs other than PDS Group 2. Detailed results from the Zion APET evaluation are given in Appendix A to this volume.

2.5.1.1 Results for PDS Group 1: SBO. This PDS group contains three PDSs. The PDS contributing the most to this group is initiated by a loss of offsite power. The diesel generators fail to respond, causing a station blackout condition. The auxiliary feedwater system fails to provide secondary cooling beyond the life of the batteries (6 h). The RCP seals remain intact. Electric power recovery attempts fail to restore power in time to provide feed and bleed cooling or auxiliary feedwater with the motor-driven pumps. The containment systems are failed due to the loss of power. The PDSs in this group are listed in Table 2.2-2 and contribute less than 2% to the TCDF for Zion.

The mean conditional probability of early containment failure for this PDS group is approximately 0.026. This conditional probability is about twice the early CF probability for the LOCA and transient PDSs but is still very low. The low probability is partially due to the depressurization of the RCS before vessel failure and partially due to the strength of the Zion containment relative to the loads expected. In order to illustrate the impact of potential depressurization mechanisms on the progression of a SBO PDS Figure 2.5-1 was constructed.

This figure shows various depressurization mechanisms incorporated in the Zion accident progression analysis. It also includes the corresponding RCS pressures for a SBO PDS. Even though, at the time of core uncovery, the RCS is at the system set-point pressure, a significant portion of it is finally depressurized to below 1.4 MPa (200 psia) at the time of vessel failure.

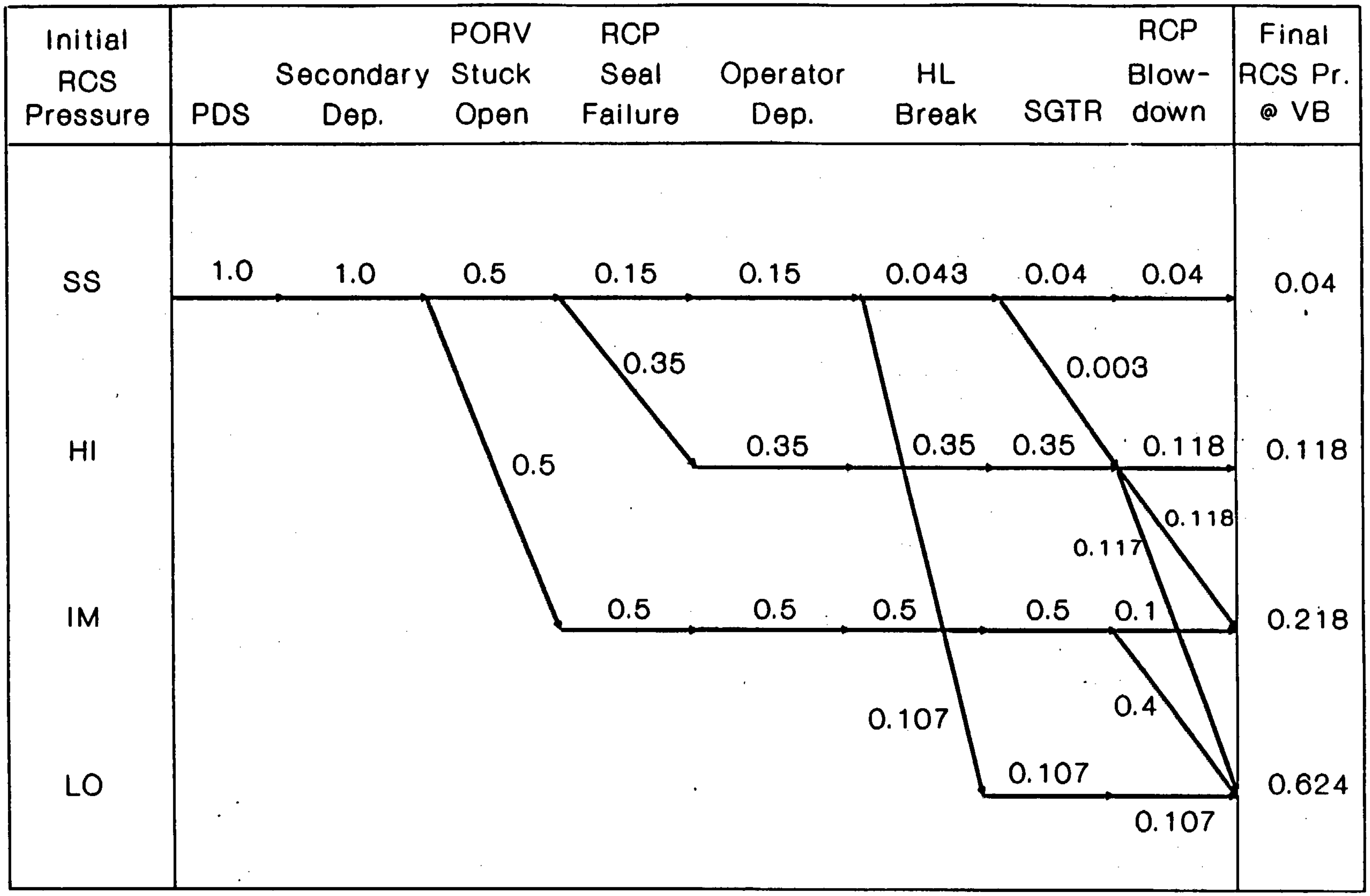


Figure 2.5-1 Depressurization Mechanisms and Corresponding RCS Pressure (Station Blackout Plant Damage State)

The depressurization mechanisms in the Zion accident progression analysis, like other studies for NUREG-1150, can be classified into three categories: operator actions, temperature induced failures, and component failure in a harsh environment. The operator actions include primary depressurization through feed and bleed and secondary depressurization using available steam generators and AFWS. There are various temperature induced primary systems failures due to the natural circulation of the primary system water inventory. The RCP seals, hot leg, and steam generator tubes are most vulnerable to high temperature. The primary system PORV is a component which may not survive the harsh environment created by a core melt accident.

The effect of the RCS being at low pressure at the time of VB is a reduction of containment loads (direct containment heating) caused by high pressure melt ejection (HPME). However, a low RCS pressure increases the potential for an in-vessel steam explosion, which inturn introduces the possibility of an "Alpha" failure mode. Thus, the relatively low conditional probability of early CF for the SBO PDS group is composed of HPME and Alpha failure modes.

2.5.1.2 Results for PDS Group 2: LOCAs. This PDS group is the largest contributor to the internal event core damage frequency at Zion and contains a large number of PDSs. However, two plant damage states (refer to Table 2.2-2) dominate the frequency of the PDS group (and therefore the total internal event core damage frequency). One of these dominant PDSs is initiated by a loss of service water with failure to recover it before a seal LOCA develops. Failure of the service water system causes a loss of cooling to the charging pumps and safety injection pumps. Core damage results due to the inability to maintain RCS inventory. Containment cooling is unavailable due to the loss of service water to the fan cooler units.

The other dominant plant damage state is intiated by a loss of component cooling water. A seal LOCA develops due to the loss of cooling and the loss of charging/injection. Attempts to recover cooling water before the onset of core damage fail. Containment heat removal is available.

Table 2.5-1 lists the ten most probable APBs for PDS group 2, and the five most probable APBs that have VB and early containment failure (CF). Most probable means most probable when the whole sample of 150 observations is considered; that is, the five most probable bins are the top five when ranked by mean probability conditional on the occurrence of the PDS group. In Table 2.5-1, the "Order" column gives the order of the bin when ranked by conditional probability. The "Prob." column lists mean APB probabilities conditional on the occurrence of the PDS group. That is, this table shows the results averaged over the 150 observations that form the sample. If Bin A occurred with a probability of 0.005 for each observation, its probability would be 0.005 in Table 2.5-1. If Bin B occurred with a probability of 0.75 for one observation and did not occur in the other 149 observations, its probability would also be 0.005. The column headed "No. Occur." gives the number of observations out of the 150 in the sample in which this APB occurred with a nonzero conditional probability.

Table 2.5-1
Results of the Accident Progression Analysis for Zion
Internal Initiators--PDS Group 2: LOCAs

Order	Bin	Prob*	No. Occur.	CF Time	Sprays	CCI	RCS Pres.	VB Mode	Amt CCI	Zr Ox	HPME	CF Size
Ten Most Probable Bins**												
1	GDDDBCAADGBB	0.156	117	No-CF	Always	PrmDeep	LoPr	Pour	Large	Lo	No	No-CF
2	GDDDBCABDGBB	0.149	109	No-CF	Always	PrmDeep	LoPr	Pour	Large	Hi	No	No-CF
3	GDCDBCDADGBB	0.136	117	No-CF	Always	No-CCI	LoPr	Pour	No-CCI	Lo	No	No-CF
4	GDCDBCDBDGBB	0.129	109	No-CF	Always	No-CCI	LoPr	Pour	No-CCI	Hi	No	No-CF
5	FDDDBCAADEBB	0.104	116	Final	Always	PrmDeep	LoPr	Pour	Large	Lo	No	BMT
6	FDDDBCABDEBB	0.099	109	Final	Always	PrmDeep	LoPr	Pour	Large	Hi	No	BMT
7	GDDDBCBADGBB	0.045	117	No-CF	Always	PrmDeep	LoPr	Pour	Medium	Lo	No	No-CF
8	GDDDBCBBDGBB	0.043	109	No-CF	Always	PrmDeep	LoPr	Pour	Medium	Hi	No	No-CF
9	FDDDBCBADEBB	0.011	116	Final	Always	PrmDeep	LoPr	Pour	Medium	Lo	No	BMT
10	FDDDBCBBDEBB	0.011	109	Final	Always	PrmDeep	LoPr	Pour	Medium	Hi	No	BMT
Five Most Probable Bins that have VB and Early CF**												
25	DAEDDCBADBBB	0.0032	27	CFatVB	Early	SDlyd-Dry	LoPr	Alpha	Medium	Lo	No	Rupture
29	DAEDDCBBDBBB	0.0023	27	CFatVB	Early	SDlyd-Dry	LoPr	Alpha	Medium	Hi	No	Rupture
34	DADDDCBADBBB	0.0014	24	CFatVB	Early	PrmDeep	LoPr	Alpha	Medium	Lo	No	Rupture
35	CDDDBCAADCBB	0.0012	74	Early-CF	Always	PrmDeep	LoPr	Pour	Large	Lo	No	Leak
37	CDDDBCABDCBB	0.0011	73	Early-CF	Always	PrmDeep	LoPr	Pour	Large	Hi	No	Leak

* Mean probability conditional on the occurrence of the PDS.
** A listing of all bins, and a listing by observation are available on computer media.

The remaining nine columns in Table 2.5-1 explain 9 of the 12 characteristics in the APB indicator. The sixth characteristic, SGTR, has been omitted since none of the 100 most probable bins for this PDS group had SGTR. The last two characteristics, RCS-Hole and Timing of Core Damage, were also omitted since they are of less interest than the others. The abbreviations for each APB characteristic are explained in Section 2.4 above.

The first part of Table 2.5-1 shows the first ten bins when they are ranked in order by probability. The four most probable bins all result in no CF, and all have the RCS at low pressure (less than 200 psia) at VB. All of the first 10 APBs that result in CF have BMT as the mode of CF.

The mean conditional probability of early CF for this PDS group is approximately 0.01. Three of the five most probable APBs with both VB and early CF are Alpha mode failures of the containment, the other two involve a low pressure pour. Early CF means CF before or at VB. CF before VB is improbable at Zion, almost all early CFs are CF at VB. The strength of the containment led to the conclusion that CF before VB due to hydrogen combustion is negligible.

Of the fraction of this PDS group which resulted in VB, all had the RCS at low pressure at VB. The fractions of this PDS group which are in the four pressure ranges just before VB (if it occurs) are:

RCS Pressure	Fraction of PDS in RCS Pressure Ranges Just Before VB
SSPr (2500 psia)	0.0
HiPr (600 to 2000 psia)	0.0
ImPr (200 to 600 psia)	0.096
Lo Pr (<200 psia)	0.904

A comparison of the above with the SBO PDS Group 1 results in Figure 2.5-1 indicates a higher fraction of the LOCA PDSs at low RCS pressure at the time of VB. The result is that Alpha mode failures are larger contributors to early CF in the LOCA PDSs than in the SBO PDSs.

2.5.1.3 Results for PDS Group 3: Transients. This PDS group contains 11 PDSs. The PDSs in this group are listed in Table 2.2-2 and contribute less than 5% to the TCDF for Zion. Transient in which the reactor scrams are included with ATWS events. One of the PDSs (with reactor scram) that contributes most to this PDS group is initiated by a loss of offsite power. One diesel generator for Unit 1 starts and runs to provide emergency power to bus 147. The auxiliary feedwater system fails to provide secondary cooling. The operators go on feed and bleed and cooling with the injection pump powered by bus 147, the plant cannot switch to the recirculation mode when the RWST is depleted. A recovery factor of 0.5 was applied to this sequence as credit for refilling the RWST and continuing in the injection mode. One of the ATWS PDS that contributes to this PDS group is initiated by a loss of main feedwater with the reactor at full power. The reactor fails to trip, but turbine trip occurs. The AFW system fails to provide sufficient feedwater to ensure secondary cooling, and the resulting pressure spike renders the safety injection system failed. With no means of removing heat from the reactor, core damage results.

The mean conditional probability of early CF for this PDS group is higher than the LOCA PDS group but lower than the SBO PDS group. This result is caused by the relative fraction of Transient PDSs that result in low RCS pressure at the time of vessel breach, which is between the SBO and LOCA PDS fractions. More details on the RCS pressures at the time of VB for the various PDSs are given in Appendix A.

2.5.1.4 Results for PDS Group 4: SGTR. This PDS group contains 4 PDSs, however one PDS dominates the core damage frequency for the group. This PDS is initiated by a steam generator tube rupture. The operators fail to depressurize the RCS below the main steam safety valve setpoints for the affected steam generator. Therefore, the loss of inventory from the RCS is never mitigated. The RWST is depleted with no means of recirculation, leading to core uncovery and core damage. The PDSs for this group are also listed in Table 2.2-2 and contribute less than 1% to the TCDF.

2.5.1.5 Results for PDS Group 5: Event V. This PDS group consists of interfacing systems LOCA. An interfacing systems LOCA (Event V) involves the failure of two series valves (motor-operated valves or combinations of motor-operated valves and check valves) separating the high pressure portions of the plant from the low pressure injection and residual heat removal systems. This PDS group is a very small contributor to the TCDF.

2.5.1.6 Core Damage Arrest and Avoidance of VB. It is possible to arrest the core damage process and avoid VB if ECCS injection is restored before the core degradation process has gone too far. Recovery of injection is due to one of two events. In the LOSP accidents, recovery of injection follows the restoration of offsite power. In other types of accidents, the ECCS is operating but no injection is taking place because the RCS pressure is too high. Any break in the RCS pressure boundary that allows the RCS pressure to decrease to the point where the ECCS inject is likely to arrest the core degradation process. The break may be an initiating break or a temperature-induced break or other failure that occurs after UTAF.

The potential for recovery prior to VB was found to be significant for the Surry analysis (refer to Volume 3 of this report). However, the dominant accident sequences at Zion (PDS Group 2: LOCAs) were not found to have a relatively high probability of recovery between UTAF and VB. Other PDSs were found to have similar recovery probabilities to the Surry analysis but as these PDSs are such small contributors to the TCDF their impact on the overall recovery probability was small.

2.5.1.7 Early Containment Failure. For those accidents in which the containment is not bypassed, the offsite risk depends on the probability that the containment will fail before or at VB. There are four possibilities:

1. Pre-existing containment leak;
2. Isolation failure;
3. CF before VB due to hydrogen combustion; and
4. CF at VB due to the events at VB.

Because the Zion containment was found to be quite strong by the structural experts who considered the issue, CF due to hydrogen burns before VB was not considered at Zion. It was estimated to be very unlikely that enough hydrogen would be generated in the vessel before VB to cause a pressure rise when dispersed to the containment and ignited, that would threaten the Zion containment. This failure mode was included in the APET used for the analysis for Draft NUREG-1150, and no CFs before VB were found. The main risk for non-bypass accidents at Zion, then, comes from CF at VB. Almost all early CFs are CF at VB. As used in Figure 2.5-2, early CF means CF before, at VB, or immediately following VB.

Figure 2.5-2 shows the probability distribution for early CF at Zion. The probability is conditional on core damage. There is no histogram for the Bypass collapsed PDS group. When the containment function is bypassed by Event V or an STGR, early CF ceases to be very important in determining the release of fission products and the offsite risk. Thus, the conditional probability of early CF was deliberately not plotted for the Bypass group. For accidents other than Bypass, the mean conditional probability of early CF is on the order of 0.01. This reflects the strength of the Zion containment relative the loads expected at VB and the probability that the vessel does not fail.

The uncertainties associated with various containment failure frequency estimates are generally within two orders of magnitude. These ranges are narrower than the preliminary Zion results[6] and the results obtained for the Surry analysis (refer to Volume 3 of this report). The Zion ranges are narrower than the Surry ranges because of the frequency of pre-existing leakage of 5E-3 used in the Zion analysis. This pre-existing leakage frequency essentially establishes a lower bound on the Zion containment failure frequency estimates. However, the Surry containment is a subatmospheric design, which means that it is continuously monitored so that any pre-existing leakage should be detected and corrected. The frequency of pre-existing leakage was therefore estimated to be much lower at Surry and consequently the lower bound of the Surry containment failure frequency range is much lower than the Zion range. The upper bounds of the Zion and Surry containment failure frequency ranges are similar.

2.5.1.8 Summary. Figure 2.5-3 shows the mean distribution among the summary APSs for the summary PDS groups. Only mean values are shows, so Figure 2.5-3 gives no indication of the range of values encountered. The distribution for early (at or before VB) failure of the containment is shown in Figure 2.5-2. Nonetheless, Figure 2.5-3 gives a good idea of the relative likelihood of the possible results of the accident progression analysis. Except for the Bypass initiators, either no containment failure or late containment failure are by far the most likely outcomes. The late failure may be due to hydrogen ignition some hours after VB, but is more likely to be due to BMT. Early CF is fairly unlikely, as was indicated by Figure 2.5-2. This is largely due to the robust nature of the Zion containment. Figure 2.5-3 shows the mean frequecies for the summary PDS groups and mean conditional probabilities for the summary APBs, where the mean is taken over all 150 observations in the sample.

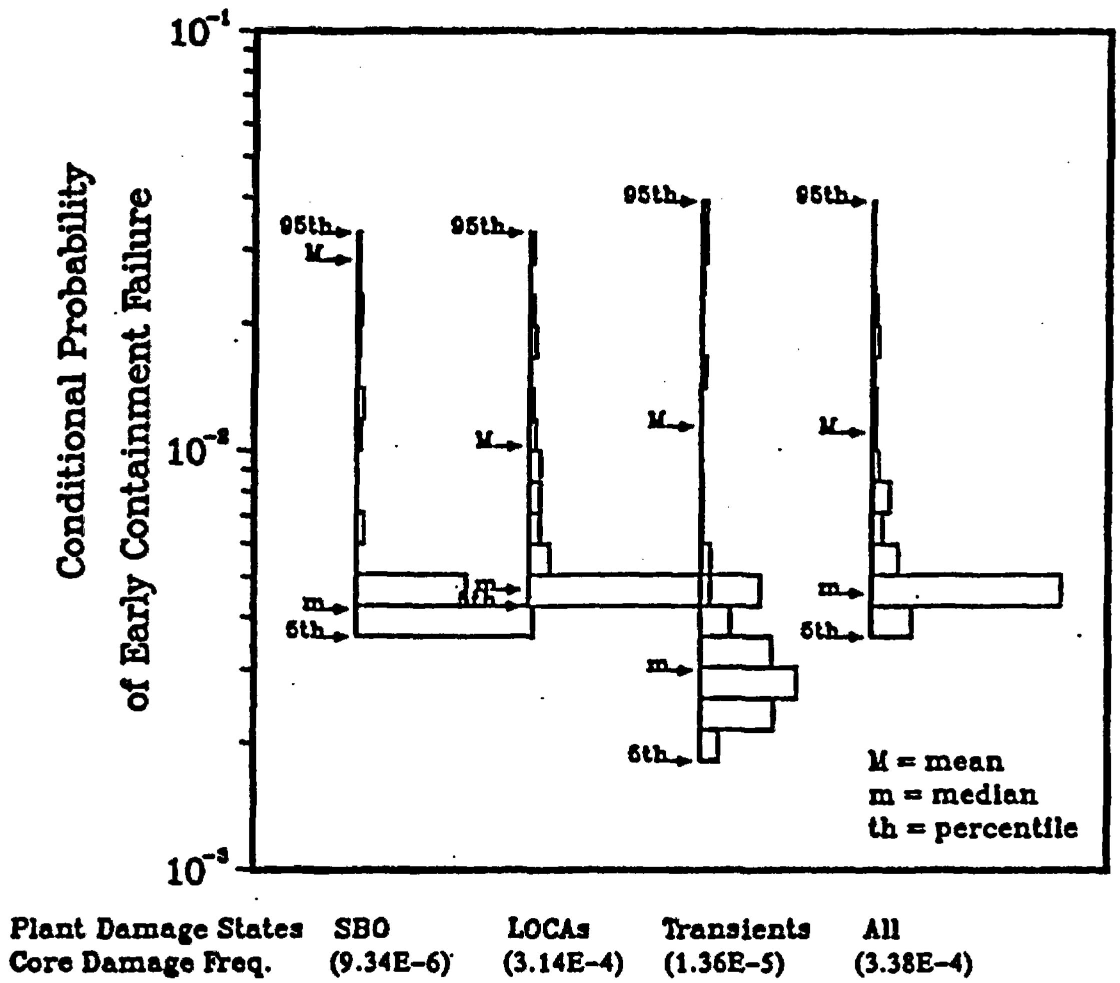


Figure 2.5-2 Conditional Probability Distributions for Early Containment Failure at Zion

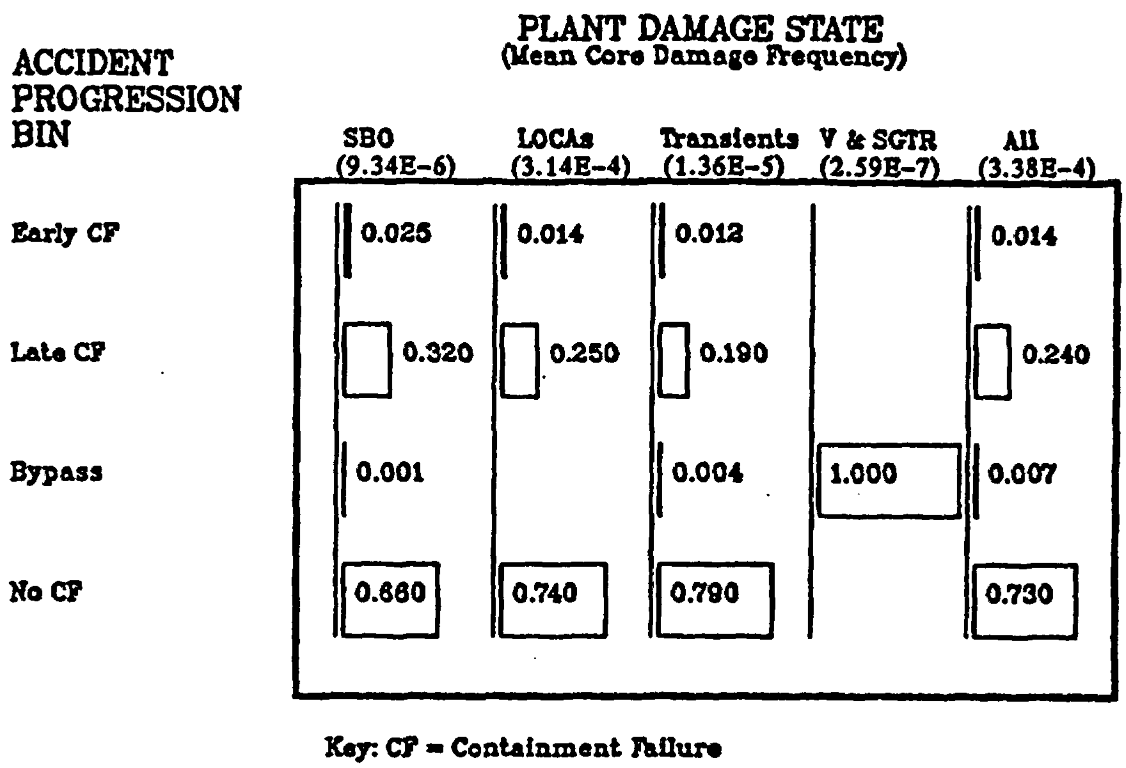


Figure 2.5-3 Conditional Probability of Accident Progression at Zion

2.5.2 Sensitivity Analyses for Internal Initiators

This section reports the results of two sensitivity analyses performed for the internally initiated accidents at Zion. The first explores the effects of different approaches to quantifying containment failure modes. The second addresses the sensitivity of the APET results to various RCS depressurization mechanisms.

2.5.2.1 Quantification of Containment Failure Mode In the Zion APET, two questions deal with containment failure probability and mode, one for failure at VB, one for late failure. The aggregated results provided by structural experts were in the form of curves (and tables) of conditional containment failure mode probabilities versus pressure loads in the Zion containment. However, for a given load, the precise containment failure mode is uncertain.

In order to quantify the two questions, an algorithm must be devised to deal with the expected hierarchy of failure modes. The Surry analysis (refer to Volume 3 of this report) and Appendix A.1 of this Volume present two different methods of quantification. In the Surry analysis, fast and slow pressurization algorithms are used, while in the Zion APET User Function, a method of comparison with containment failure joint distributions was adopted.

For both methods, the answer to the CF mode is arrived at by comparing pressure loads to containment capacity, and the mode of CF is determined on the basis of a random number generated by the LHS model. Thus, for agiven path through the tree, the CF mode is unique.

In order to compare the algorithms, the questions were isolated from the tree, and a one-thousand Monte Carlo sample was evaluated. Zion capacity and the highest Zion load* curves were used as input.

In addition, the results were compared to the evaluation of failure mode based only on structural expert split fraction probabilities, without using a random number selector. From the statistical point of view, this method was felt to provide a more exact answer. However, it cannot be used in the calculations of EVNTRE.

The results of the calculations are given below:

Algorithm	Early Failure	No Failure
Surry Fast Pressurization	0.20	0.80
Surry Slow Pressurization	0.20	0.80
Zion Joint Distribution	0.14	0.86
Split Fraction Approach	0.17	0.83

* Note: This curve corresponds to expert opinion loads for high DCH, dry cavity, high Zr oxidation, large hole, high RCS pressure conditions.

The above results indicate that the approaches give similar predictions. It was therefore concluded that the quantification of the questions of failure probability and failure mode for the Zion analysis was not very sensitive to the approach adopted.

2.5.2.2 Sensitivity of Results to RCS Depressurization Mechanisms Several questions in the Zion APET deal with various mechanisms (refer to Figure 2.5-1) which have the potential to depressurize the RCS after UTAF and prior to VB. A sensitivity study was therefore performed in which these depressurization mechanisms were removed. Details of the sensitivity calculation are given in Appendix A and show minor changes to the mean early CF probability when all PDSs are included. The lack of sensitivity is because the TCDF at Zion is dominated by the LOCA PDS group and a large fraction of the accident sequences in this group have initiating events which result in a low RCS pressure at VB. More sensitivity was expected for the SBO PDS group and the results for this group are given below:

Containment Failure Time	Base	Sensitivity
EF	0.026	0.05
LF	0.32	0.23
NF	0.66	0.72

The above results show that the conditional probability of early CF increased by a factor of about 2 when the depressurization mechanisms were removed for the SBO PDS group. However, the conditional probability of early CF remains low even for the sensitivity case because of the strength of the Zion containment relative to the loads expected.

2.6 Insights from the Accident Progression Analysis

The frequency of containment failure, especially early failure, is very low for the Zion plant. This is due to the fact that the Zion containment capacity estimated by the expert panel is high, and, in addition, the expected containment loads from the core melt accidents are not high enough to threaten the integrity of the containment during the early stages of an accident. In addition, a large fraction of the PDS are expected to result in low RCS pressure at the time of VB, which lowers the potential for loads associated with HPME. A low RCS pressure does, however, increase the probability of an in-vessel steam explosion, which introduces the possible of an alpha mode failure.

Therefore there are two major physical phenomena contributing to early containment failure, namely in-vessel steam explosions and direct containment heating. In-vessel steam explosions are more likely to occur when the primary system is at low-pressure. On the other hand, direct containment heating is associated with high-pressure sequences. The relative contributions from these two phenomena for each PDS are shown below.

PDS	α	DCH	Others	Total
SBO	5.910E-03	1.574E-02	3.183E-03	2.483E-02
(%)	(23.80)	(63.39)	(12.81)	(100.00)
Transients	3.505E-03	5.191E-03	3.058E-03	1.175E-02
(%)	(29.82)	(44.16)	(26.02)	(100.00)
LOCAs	7.983E-03	1.706E-03	4.166E-03	1.386E-02
(%)	(57.62)	(12.31)	(30.07)	(100.00)

An importance analysis of various uncertainty issues was performed through the use of a regression technique. The partial rank correlation coefficient (PRCC) is a measure of uncertainty importance. Figure 2.6-1 lists those variables identified by the importance analysis and shows the importance results graphically.

As discussed above, in-vessel steam explosions have the highest contribution to uncertainty in the early containment failure predictions. The variables related to the component cooling water system (CCWS) are highly negatively correlated with the containment bypass. This is mainly because the core damage frequency is dominated by the failure of the CCWS. Given the relatively small contribution of the interfacing systems LOCA to the total core damage frequency, the higher the CCWS failure contribution, the smaller the bypass contribution.

The containment failure probability, especially early failure, is low. The uncertainty associated with the frequency estimation is also small, mainly because of the residual risk due to the contribution of already existing leakage (containment isolation failure).

The most important physical phenomena contributing to the early containment failure predictions are in-vessel steam explosions, direct containment heating, and containment isolation failure.

2.7 References

1. Sargent and Lundy, "Containment Structural Capability of Light Water Nuclear Power Plants," IDCOR Program Technical Report 10.1, Technology for Energy Corporation, Knoxville, Tennesse, July 1983.

2. S. Sharma, et al., "Ultimate Pressure Capacity of Reinforced and Pre-Stressed Concrete Containments," NUREG/CR-4149, Brookhaven National Laboratory.

3. T. A. Butler and L. E. Fugelso, "Response to the Zion and Indian Point Containment Buildings to Severe Accident Pressures," NUREG/CR-2569, Los Alamos National Laboratory, 1982.

4. "Zion Station Final Safety Analysis Report," Commonwealth Edison Company, September 1981.

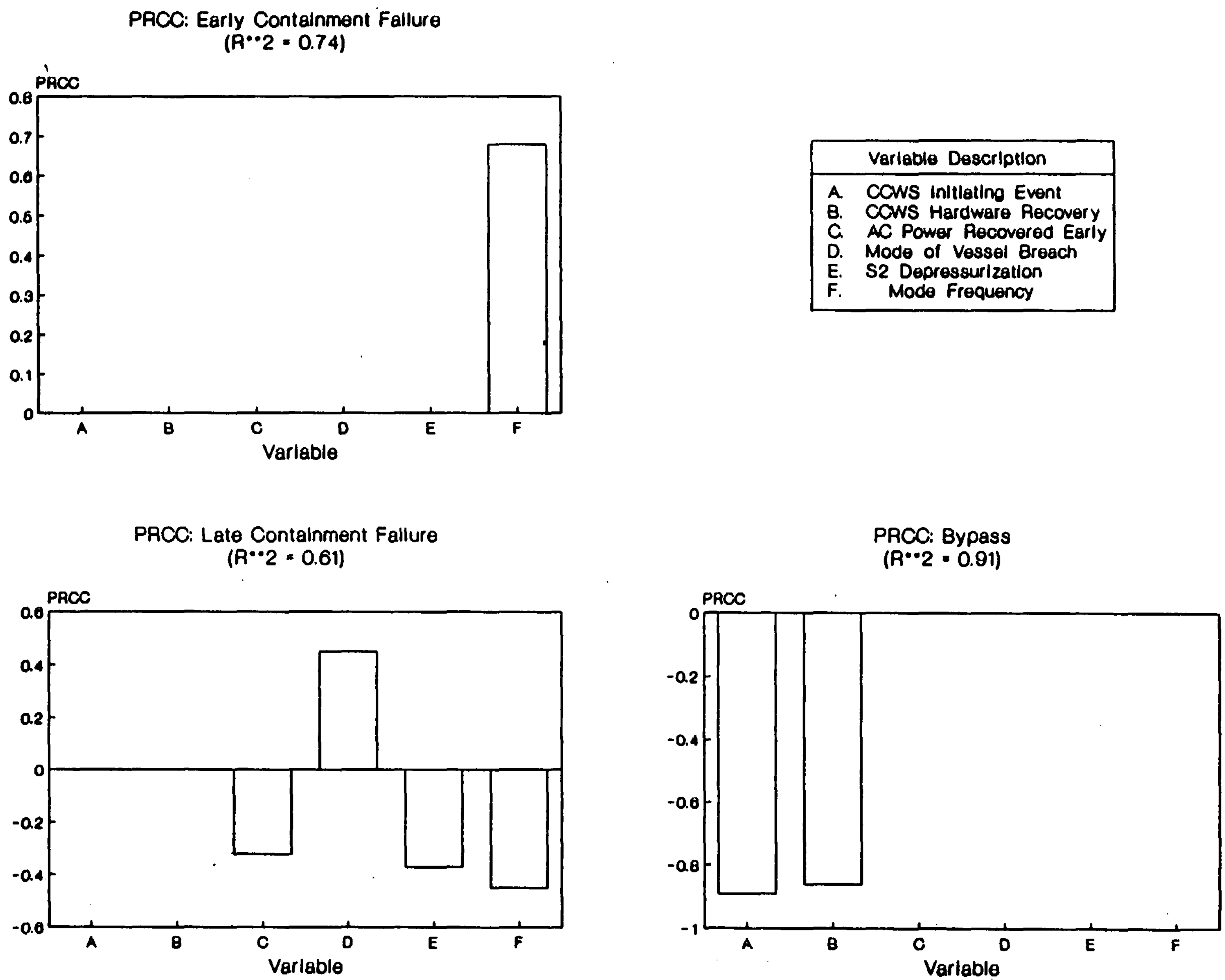


Figure 2.6-1 Importance Analysis: Containment Failure (Partial Rank Correlation Coefficient)

5. E. D. Gorham-Bergeron et al., "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration, and Uncertainty Analyses," Sandia National Laboratories, NUREG/CR-4551, Vol. 1, Draft Revision 1, SAND86-1309, to be published.*

6. M. Khatib-Rahbar et al., "Evaluation of Severe Accident Risks and Potential for Risk Reduction: Zion Power Plant," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 5, Draft for Comment, BNL-NUREG-52029.

* Available in the NRC Public Document Room, 2120 L Street, NW., Washington, DC.

3. RADIOLOGICAL SOURCE TERM ANALYSIS

The source term is the information passed to the next analysis so that the offsite consequences can be calculated for each group of accident progression bins. The source term for a given bin consists of the release fractions for the nine radionuclide groups for the early release and for the late release, and additional information about the timing of the releases, the energy associated with the releases, and the height of the releases.

The source terms for Zion are generated by a relatively small parametric computer code: ZISOR. The aim of this model is not to calculate the behavior of the fission products from their basic chemical and physical properties and the flow and temperature conditions in the reactor and the containment. Instead, the purpose is to represent the results of the detailed codes that do calculate the fission product behavior by application of the first principles of physics, chemistry, and thermodynamics.

A more complete discussion of the source term analysis, and of ZISOR in particular, may be found in NUREG/CR-5360.[1] The methods on which ZISOR is based are presented in NUREG/CR-4551, Volume 1,[2] and the source term issues considered by the expert panels are described more fully in NUREG/CR-4551, Volume 2,[3] Part 4.

Section 3.1 summarizes the features of the Zion plant that are important to the magnitude of the radionuclide release. Section 3.2 presents a brief overview of the ZISOR code, and Section 3.3 presents the results of the source term analysis. Section 3.4 discusses the partitioning of the thousands of source terms into groups for the consequence analysis. Section 3.5 concludes this section with a summary of the insights gained from the source term analysis.

3.1 Zion Plant Features Important to the Source Term Analysis

The nuclear reactor of Zion Unit 1 is a four-loop pressurized water reactor (PWR) contained in a containment constructed of post tensioned concrete with a welded steel liner forming the pressure boundary. Figure 1.1 shows a section through the Zion containment. More detail on the Zion plant in general is contained in Sections 1.2 and 2.1; this material is not repeated here.

The design pressure of the Zion containment is 0.43 MPa (62 psia), and the structural experts found the failure pressure to be generally between two and three times the times the design pressure. The relatively high failure pressure combined with the large size of the containment 7.7×10^4 m^3 (2.7×10^6 ft^3) implies that the containment is relatively resistant to failure by the events at VB or by hydrogen combustion before or after VB. This was confirmed by the results of the accident progression analysis.

In the Zion containment, shear rupture at the cylinder/basemat junction was identified as a potential failure mode. The location of this failure mode (below grade) could result in significant attenuation of the source term. Another potential failure mode is yielding of the hoop tendons. The most likely location of this failure mode is above grade. Since source terms for the hoop and shear rupture failure locations (above grade and below grade) were anticipated to differ significantly, the two failure modes were binned separately during the containment analysis.

Operation of the spray system is very effective in reducing the airborne concentration of aerosols. Other than the release of noble gases and some iodine evolution, the release of radioactive material to the atmosphere resulting from late containment leakage or BMT in which sprays have operated for an extended time would be very small. The source terms for the late containment failure accident progression bin are slightly higher than, but similar to, those of the no containment failure bin (where radioactive release is via containment leakage).

During an accident at Zion the accumulation of a relatively small amount of water on the containment floor will lead to an overflow into the reactor cavity. Consequently, for a relatively large number of accident sequences there will be a significant amount of water in the reactor cavity. As a result, there is a substantial likelihood of eliminating or mitigating the release of radionuclides from core-concrete interactions (CCIs).

3.2 Description of the ZISOR Code

This section describes the manner in which the source term is computed for each accident progression bin (APB). The source term is more than the fission product release fractions for each radionuclide class; it also contains information about the timing of the release, the height of the release, and the energy associated with the release. The next subsection presents a brief overview of the parametric model (XSOR) used to calculate the source terms. Section 3.2.2 discusses the model in some detail; a complete discussion of XSOR may be found in Reference 1. Section 3.2.3 presents the variables sampled in the source term portion of this analysis.

3.2.1 Overview of the Parametric Model

XSOR is a fast-running, parametric computer code used to calculate the source terms for each APB for each observation. As there are typically a few thousand bins for each observation, and 150 observations in the sample, the need for a source calculation method that requires a minimum of computer time for one evaluation is obvious. XSOR is not designed to calculate the behavior of the fission products from their basic chemical and physical properties and the flow and temperature conditions in the reactor and the containment. The purpose of XSOR is to provide a framework for integrating the results of the more detailed codes that do consider these quantities. Since many of the factors XSOR utilizes to calculate the release fractions were determined by a panel of experts, the results of the detailed codes enter XSOR "filtered" through the experts.

The 60 radionuclides (also referred to as isotopes, or fission products) considered in the consequence calculation are not dealt with individually in the source term calculation. Some different elements behave similarly enough both chemically and physically in the release path that they can be considered together. The sixty isotopes are placed in nine radionuclide classes as shown in Table 3.2.1. It is these nine classes which are treated individually in the source term analysis.

Table 3.2-1
Isotopes in Each Radionuclide Release Class

	Release Class	Isotopes Included
1.	Inert Gases	Kr-85, Kr-85M, Kr-87, Kr-88, Xe-133, Xe-135
2.	Iodine	I-131, I-132, I-133, I-134, I-135
3.	Cesium	Rb-86, Cs-134, Cs-136, Cs-137
4.	Tellurium	Sb-127, Sb-129, S\Te-127, Te-127M, Te-129, Te-129M, Te-131M, Te-132
5.	Strontium	Sr-89, Sr-90, Sr-91, Sr-92
6.	Ruthenium	C-58, Co-60, Mo-99, Tc-99M, Ru-103, Ru-105, Ru-106, Rh-105
7.	Lanthanum	Y-90, Y-91, Y-92, Y-93, Zr-95, Zr-97, Z\Nb-95, La-140, La-141, La-142, Pr-143, Nd-147, Am-241, Cm-242, Cm-244
8.	Cerium	Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241
9.	Barium	Ba-139, Ba-140

3.2.2 Description of ZISOR

In this section the form of XSOR used to model Zion (ZISOR) is described in detail. Since the largest consequences generally result from accidents in which the containment fails before VB or about the time of VB, the nomenclature and structure of ZISOR reflects failure at VB. There is an early release which occurs before, at, or a few tens of minutes after VB, and there is a late release which occurs several hours, after VB. In general, the early release is due to fission products that escape from the fuel while the core is still in the RCS, i.e., before vessel breach, and is often referred to as the RCS release. The late release is largely due to fission products that escape from the fuel during the CCI and is referred to as the CCI release. For situations where the containment fails many hours after vessel breach, the "early" release equation is still used, but the release is better termed the RCS release. The late release includes not only fission products released from the debris pool during CCI, but also material released from the fuel before VB which deposits in the RCS or the containment, and then is revolatilized after VB.

For radionuclide class i, the early or RCS release is calculated from the following equation in ZISOR:

```
ST(i) = FCOR(i)*[FISG(i)*FOSG(i)+(1-FISG(i))*FVES(i)*FCONV/DFE]
      + (1-FCOR(i))*(1-FREM)*FPME*FDCH(i)*FCONV/DFE
      + (1-FCOR(i))*(1-FREM)*(1-FPME)*FPART*FCCI(i)*FCONC(i)/DFL
      + [FCOR(i)*(1-FISG(i))*(1-FVES(i))+(1-FOCR(i))*FREM*FLATE(i)]
        *FCONC(i)/DFL
      + FCOR(I)*FISG(i)*(1-FOSG(i))*FLATE(i)
```

Where index i refers to the fission product groups listed in Table 3.2-1. The definitions of the parameters in this equation are given in Table 3.2-2. The first term is the source term contribution from the fuel releases during the in-vessel phase of an accident. The second through fourth terms of the equation represents the source term from direct containment heating, the releases from molten core concrete interactions, and the releases due to late revaporization of volatile species deposited in reactor coolant system, respectively. For steam generator tube rupture (SGTR) accidents with the secondary safety relief valves (SRV) stuck open, the fifth term of the equation is included to account for the late revaporization of volatile species deposited in the steam generators. For the iodine group, an additional term is added to account for the release of iodine remaining in the containment late in the accident which is then converted into organic iodine.

The equation can be represented by the tree diagram show in Figure 3.2-1. In the diagram, the branches that lead to environmental release are enclosed in a box. The amount of fission products released to the environment from each branch can be determined by the product of the parameters in the corresponding branch. The value of these parameters depends on the accident progression bins (as described in Section 2.4) defined by the APET analysis.

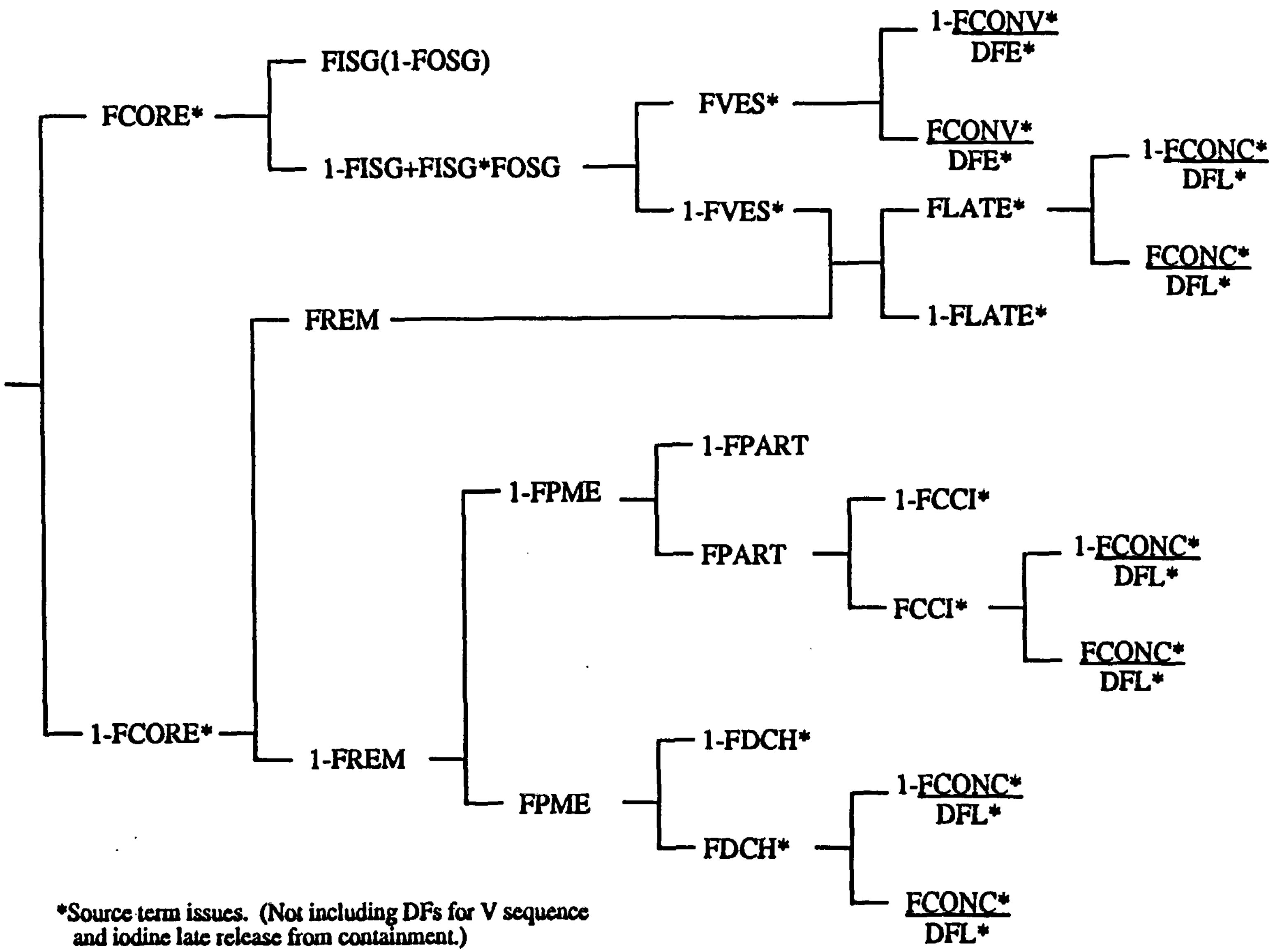
All the parameters in ZISOR are actually represented by distributions. As part of its input structure, ZISOR has a large data file which contains all the cumulative probability distributions for all the parameters in the LHS process, one value (between zero and one) is selected from the distributions for each of the parameters. The random number passed by the LHS program is used as a cumulative probability and parameter values corresponding to this probability are taken from the stored distributions.

The distributions of the most important parameters were evaluated by the source term expert panel. These parameters, together with the case structure used in the elicitation process, are listed in Table 3.2-3. For those parameters and cases for which expert evaluation was not available, distributions were internally generated at either Brookhaven National Laboratory (BNL) or Sandia National Laboratories (SNL).

The implementation of the parametric model described in Section 3.2.1 into ZISOR is straightforward. ZISOR accepts accident progression bins from APET analysis as input, then based on the random number passed by the LHS program, ZISOR assigns a value from the distributions to each of the parameters to calculate the source term.

Table 3.2-2 Parameter Definitions in the ZISOR Code

FCOR(i)	-	Fraction of initial inventory of nuclide group i release from the fuel in-vessel
FISG(i)	-	Fraction of fuel release transported to steam generator in an accident
FOSG(i)	-	Fraction of FISG released from steam generator to the environment
FVES(i)	-	Fraction of fuel release transported to the containment
FCONV	-	Containment transport fraction for releases prior to or at vessel breach
DFE	-	Decontamination factor of spray for in-vessel releases
FREM	-	Fraction of core remained in vessel after vessel breach, (value fixed, 0.05)
FPME	-	Conditional fraction of core involved in high pressure melt ejection accident, (three possible values, 0.85, 0.50, 0.15)
FDCH(i)	-	Fraction of FPME release to containment as radiological source terms
FPART	-	Conditional fraction of core involved in core concrete interactions (three possible values, 1.0, 0.85, 0.50)
FCCI(i)	-	Fraction release of nuclide group i from corium during molten core-concrete interactions
FCONC(i)	-	Containment transport fraction for ex-vessel release
DFL	-	Decontamination factor of spray for ex-vessel release and also the decontamination factor for the overlying (if any) water pool of CCI releases
FLATE(i)	-	Fractional releases of material deposited in RCS due to revaporization rate



*Source term issues. (Not including DFs for V sequence and iodine late release from containment.)

Figure 3.2-1 Tree Diagram for the Parametric Equation of ZISOR

Table 3.2-3 Source Term Issues

Issues		Case Structure
FCOR	(Fuel Release)	1. Low Zr Oxidation 2. High Zr Oxidation
FVES	(RCS Transport Fraction)	1. System Set Point 2. High Pressure (grouped with intermediate pressure by experts) 3. High or Intermediate Pressure 4. Low Pressure 5. Large Interfacing System LOCA
FCONV	(Containment Transport fraction for RCS Release)	1. Early Leak, Dry Containment 2. Early Leak, Wet Containment 3. Early Rupture 4. Late Rupture 5. V Sequence (set equal to that of Case 3)
FCONC	(Containment Transport fraction for CCI Release	1. Early Leak, Dry Containment 2. Early Leak, Wet Containment 3. Early Rupture 4. Late Rupture 5. V Sequence (set equal to that of Case 3)
FCCI	(Release fraction during MCCI)	1. Low Zr Oxidation, No water 2. High Zr Oxidation, No water 3. Low Zr Oxidation, Water present 4. High Zr Oxidation, Water present
DFSPR1	(Spray DF for RCS Release)	1. High Pressure, Early Containment Rupture 2. All other cases
DFSPRC	(Spray DF for CCI Release)	
VDF	(DF of Water Pool in Scrubbed V Sequence	

Table 3.2-3 Source Term Issues ctd.

Issues		Case Structure
FLATE	(Revolatization of FP deposited in RCS)	1. One Hole in RCS 2. Two Holes in RCS
LATEIL	(Fraction of I_2 remaining in Containment converted to Volatile form)	
FDCH	(Fraction of FPME released from Containment)	1. High RCS Pressure 2. Low RCS Pressure

In applying the parametric model to some accident scenario, the definition of some of the parameters has to be properly adjusted. For a V sequence in which the containment is bypassed early in the accident, FCONV is the reactor building transport fraction for release prior to vessel breach and FCONC(i) is the collective transport fraction of containment and reactor building for ex-vessel release. In the present calculations, FCONV and FCONC(i) of V sequences are assumed to have the same distribution as those of FCONV and FCONC(i) in the cases of early containment rupture. For a V sequence with a break located under the water, DFE and DFL are the decontamination factors of the water pool for in-vessel and ex-vessel releases, respectively. In ZISOR, it is also assumed that the spray has no effect on the source term release of the V sequence.

For an accident with SGTRs, the fission products released from the fuel can reach the environment through two paths. In SGTRs, a portion of the fuel releases will enter the steam generator and then release directly into the environment through the steam generator secondary side. The second release path will be directly to the containment after vessel failure and will be subjected to the retention in the containment. The actual amount of ex-vessel release that reaches the environment depends on the containment failure time and the failure mode.

It was identified in Sections 2.1 and 3.1 that shear rupture at the junction between the cylindrical wall and the basemat is a potential failure mode in the Zion containment. A failure at the cylinder wall/basemat junction would be about 3 m (10 ft) below ground. The failure would be characterized by circumferential cracks through which the pressure loading would be relieved. There are two possible paths for the radioactive material to be released to the environment, namely leakage through the gap between the soil and the containment wall, and migration or diffusion through the soil.

Since the migration or diffusion of fission products through the soil is a very slow process, the radioactive material released via this path can only reach the environment long after the accident. In the present source term analysis of Zion only the release via the first path is considered. The release of radioactive material through the gap between the containment wall and the soil occurs over a long duration due to the gradual blowdown of the containment. In ZISOR, the containment retention of fission products for shear failure mode is treated the same as that for the leakage failure mode.

The Zion containment could also fail due to basemat melt-through (BMT). In such a failure mode, the radioactive material can only reach the environment by migration or diffusion through the soil. The process is relatively slow and the radioactive material can only reach the environment long after the accident. In ZISOR, the environmental releases of source term in the case of BMT is assumed to be almost identical to that of the case of no containment failure, depending on the mode of VB and spray operation. For the cases of no containment failure, the containment transport fraction for both of the in-vessel and ex-vessel releases is 1.0×10^{-6} and the released fraction for noble gases and gaseous iodine is 0.005.

In ZISOR, the timing and other characteristics (e.g., energy and height) of the release of radioactive material to the environment following a nuclear power plant accident are based on the results of the APET analysis. The release characteristics used in ZISOR were obtained from Zion[4] and Seabrook[5] calculations. These constants can be found in RELCHAR subroutine of ZISOR. A listing of ZISOR is provided in Appendix B to this report.

3.2.3 Variables Sampled in the Source Term Analysis

Twelve variables are sampled for the source term analysis and are listed in Table 3.2-4.

The samples selected by LHS for these variables define quantiles in the parameter distributions; ZISOR internally interpolates from the distributions to obtain actual parameters values. For a given issue, perfect correlation is assumed for the nine release groups. Since sampling is performed on quantiles, all distributions input to the LHS program are uniform from 0 to 1. No correlation is imposed between issues in the process.

ZISOR defines a separate distribution for each radionuclide class for each variable unless otherwise noted. The different cases for each variable are noted in the description. Not all the cases considered by ZISOR are listed. Parameter values for other cases are determined internally, often from cases defined in Table 3.2-4. For example, no distribution for FCONV for late leaks is explicitly defined in ZISOR.

The distributions for the Zion-specific and PWR-generic issues elicited from the Source Term Expert Panel are fully described in Reference 3. Issues which were internally quantified by SNL are briefly discussed below:

VDF is the decontamination factor used for Event V when the release location is underwater. For these types of accidents, ZISOR sets DFE to the value of VDF. The distribution for VDF was determined by SNL. The range for VDF is from 1.6 to 5100; the median value is 6.2. VDF represents only scrubbing by passage through the pool of water overlying the break location. Any additional removal in the auxiliary building is accounted for by FCONV.

SPRDF refers to both the spray decontamination factor for the RCS release, DFSPV, and the CCI spray decontamination factor, DFSPC. There is only one value for each of these DFs; which applies to all radionuclide groups except inert gases. Different spray distributions are used for DFSPV for CF at VB for late CF. The value selected for DFSPC is always taken from the spray DF distribution for late CF. The one quantile from the LHS output is used to select both the RCS and CCI spray DF values, thus the spray DF distributions are complete correlated. The spray DF distributions were determined by SNL.

For the RCS release with CF at VB, there are two distributions for the spray DF. One applies if the RCS was at high pressure before VB. In this case most of the RCS release leaves the vessel at VB, and the sprays are not very effective. The range of this distribution is from 1.0 (no effect) to 2.8; the median value is 1.6. For the RCS release with CF at VB at low pressure the sprays are estimated to be very effective. The range of this distribution is from 2.3 to 2800; the median value is 40. The distribution for the CCI spray DF distribution ranges from 6.7 to 3200; the median value is 28.

Table 3.2-4

Variables Sampled in the Source Term Analysis

Variable	Description
FCOR	Fraction of each fission product group released from the core to the vessel before or at vessel breach. Two cases: high and low zirconium oxidation.
FVES	Fraction of each fission product group released from the vessel to the containment before or at vessel breach. Four cases: RCS at system setpoint pressure, RCS at high or intermediate pressure, RCS at low pressure, and Event V.
VDF	Decontamination factor for pool scrubbing for Event V when the break location is underwater at the time of the release. One distribution: applies to all radionuclide classes except inert gases.
FCONV	Fraction of each fission product group in the containment from the RCS release that is released from the containment in the absence of mitigating factors such as sprays. One distribution for each case: applies to all radionuclide classes except inert gases. Five cases: containment leak at or before vessel breach with sprays operating, containment leak at or before vessel breach with sprays not operating, containment rupture at or before vessel breach, very late containment rupture, and Event V. Note that FCONV does not account for fission product removal by the sprays. The case differentiation on spray operation is to account for differences in containment atmosphere temperature and humidity between the two cases.
FCCI	Fraction of each fission product group in the core material at the start of core-concrete interactions that is released to the containment. Four cases: low zirconium oxidation in the core and no overlying water, high zirconium oxidation in the core and no overlying water, low zirconium oxidation in the core with overlying water, and high zirconium oxidation in the core with overlying water.
FCONC	Fraction of each fission product group in the containment from the core-concrete interaction release that is released from the containment in the absence of mitigating factors such as sprays. Five cases as in FCONV.

Table 3.2-4 (continued)

Variables Sampled in the Source Term Analysis

Variable	Description
SPRDF	Decontamination factor for sprays. Quantified by SNL for all fission product groups (except noble gases). One distribution for each case. Three cases: RCS release at high pressure and CF at VB, RCS releases not covered by the first case, and CCI releases.
LATEI	Fraction of the iodine deposited in the containment which is revolatilized and released to the environment late in the accident.
FLATE	Fraction of the amount of each fission product group deposited in the RCS which revolatilized after VB and is released to the containment. Two cases: one large hole in the RCS, and two large holes in the RCS.
FDCH	Fraction of each fission product group in the core material at vessel breach in a direct containment heating event which is released to the containment. Two cases: vessel breach at high pressure (1000 to 2500 psia) and vessel breach at intermediate pressure (200 to 1000 psia).
FISG/FOSG	Fraction of each fission product group released from the reactor vessel to the steam generator, and from the steam generator to the environment, in a SGTR accident. Two distributions, FISG and FOSG, each of which has two cases: SGTRs in which the secondary SRVs reclose, and SGTRs in which the secondary SRVs stick open.
POOL-DF	Decontamination factor for a pool of water overlying the core debris during CCI. Two cases: a completely full cavity, and a partially full cavity.

LATEI refers to the evolution of iodine in volatile form from water in the containment late in the accident. Because of its volatile form (typically organic), all the iodine is released to the environment as it is unaffected by all the removal mechanisms (pool scrubbing, sprays, deposition, etc.). The release fraction determined by LATEI applies to all the iodine released from the fuel and retained in the containment in aqueous solution. This iodine is expected to be contained in the sump water. The sump water does not play the same role in heat removal that the suppression pool does in a boiling water reactor (BWR) so the results of the expert panel (which apply to BWRs only) were not utilized. Instead, the distribution obtained specifically for pressurized water reactors (PWRs) in the first draft of this report was used. The distribution used for LATEI ranges from 0.0 to 0.10; the median value is 0.05.

FISG and FOSG are the release fractions used for the RCS release for SGTR accidents. FISG is the fraction released from the core that enters the steam generator; and FOSG is the fraction entering the steam generator which is released from the steam generator to the environment.

As the material passing from the steam generator to the atmosphere bypasses the containment, FCONV and DFE are not applied to this release path. For the STGRs where the secondary system SRVs reclose, the distributions for FISG and FOSG were determined by SNL. For the SGTRs where the secondary system SRVs stick open, the distributions for FISG and FOSG were determined by an ad hoc expert panel. The panel provided distributions for the product FISG * FOSG for iodine, cesium, tellurium, and aerosols. There is no retention in the SGs for noble gases.

DFPSL is the decontamination factor for the late pool scrubbing of the CCI release. This DF is applied when the core debris is not coolable and CCI takes place under water. There are two distributions; one applies for a shallow pool and the other distribution applies when the sprays are (or were) operating and the cavity is full. For both the shallow and deep pool distributions, I, Cs, Ba, Ru, La, and Ce radionuclide groups, are considered separately from the Te and Sr radionuclide groups. The distributions were determined by the NUREG-1150 project staff.

3.3 Results of the Source Terms Analysis

This section presents the results of computing the source terms for the APBs produced by evaluating the APET. The APET's evaluation produced a large number of APBs, so, as in Section 2.5, only more likely and more important APBs are discussed here. However, source terms were computed for all the APBs for each of the 150 observations in the sample. The source term is composed of release fractions for the nine radionuclide groups for an early and a late release as well as release timing, release height, and release energy. As discussed above, the source terms are computed by a fast-running parametric computer code, ZISOR.

Section 3.3.1 presents the results for the internal initiators. Section 3.3.2 discussed the sensitivity analyses run for the internal initiators. The predictions of ZISOR are compared with STCP predictions for similar accident conditions in Section 3.3.3.

The tables in this section present only a very small portion of the output obtained by computing source terms for each APB. More detailed results are contained in Appendix B, and complete listings are available on computer media by request.

3.3.1 Results for Internal Initiators

Figure 3.3-1 shows the uncertainty distributions of the release fractions of nine source term groups for the early, late, and bypass, respectively. Corresponding statistical parameters are also summarized in Tables 3.3-1 through 3.3-4, respectively. In general, the release fractions due to the containment bypass events are higher than that of other release categories. The bypass category includes typical interfacing systems loss-of-coolant accidents (LOCAs) and temperature induced SGTRs, especially SGTR with the secondary safety relief valve stuck open. The early containment failure category includes those failures due to catastrophic rupture, rupture, shear, and leak at the time of vessel failure. The late failure category differs from the early failure category in only the timing of the containment failure, i.e., at the time of vessel failure versus about 24 h after vessel failure. The failure mode by BMT is included in the late containment failure category.

3.3.2 Sensitivity Analyses for Internal Initiators

For the uncertainties associated with those fission product groups related to in-vessel release (e.g., Iodine or Cesium) the uncertainty ranges are about two to three orders of magnitude. On the other hand, the uncertainties associated with ex-vessel release groups (i.e., refractories) are about twice as much as that of in-vessel groups. This may reflect the current state of knowledge that there are greater uncertainties associated with the ex-vessel physical phenomena, and therefore the experts elicited on these issues assigned correspondingly wider ranges.

A limited uncertainty sensitivity analysis was performed to determine the importance of each source term parameter in ZISOR using the probability distributions for the various source term uncertainty issues. The sensitivity analysis was performed by adding one variable at a time to determine the changes in the ranges of release fractions of the nine groups of fission product. The selected accident sequence was a station blackout. Other conditions are summarized in Table 3.3-5. Results are shown in Figures 3.3-2 through 3.3-4. Also shown in the figures are the point estimate values and the STCP code calculation results (denoted by x and o, respectively). The histograms shown in Figure 3.3-2 were constructed in such a way that, from left to right, a corresponding parameter was added as an uncertainty variable in the source term uncertainty analysis. The uncertainty contribution from each parameter was therefore included one at a time. For example, the first histogram in Figure 3.3-2 is the uncertainty distribution for the Iodine release fraction due to FCOR only. The next histogram shows the uncertainty contributions due to both FCOR and FVES. It is shown, in these figures, that the uncertainty contribution to the Iodine (representing in-vessel release) release fraction is mainly due to FVES, an in-vessel parameter. On the other hand, the uncertainty for the Ruthenium release, representing ex-vessel release, is mainly due to FCCI, an ex-vessel parameter.

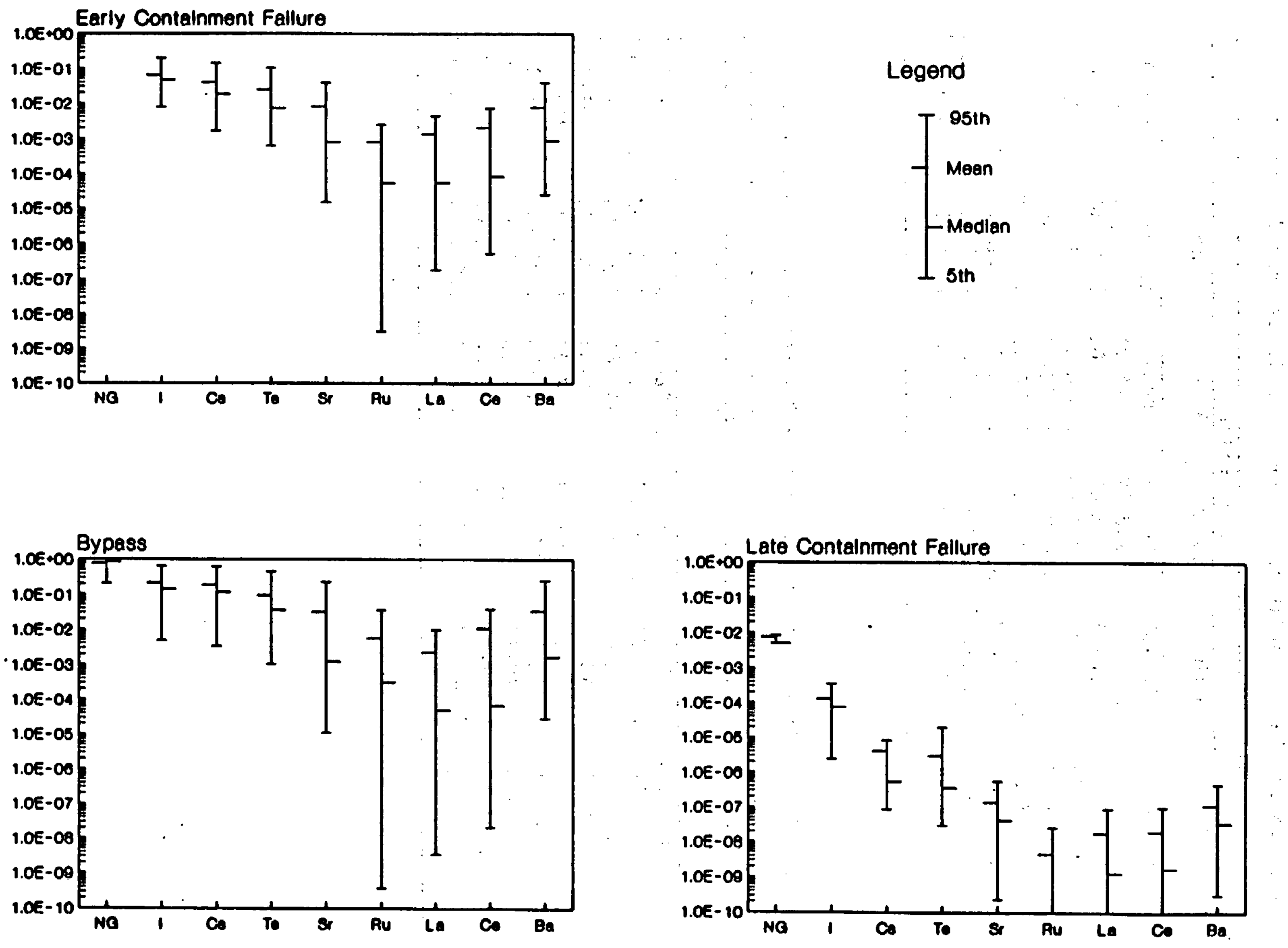


Figure 3.3-1 Source Term Uncertainty Distributions

Table 3.3-1. Source Term Statistics: Release for Early Containment Failure

Source Term	Percentile					
	Min	5-th	50-th	95-th	Max	Mean
NG	9.42E-01	9.78E-01	9.99E-01	1.00E+00	1.00E+00	9.96E-01
I	2.12E-03	7.86E-03	4.78E-02	2.02E-01	3.07E-01	6.47E-02
Cs	3.47E-04	1.59E-03	1.85E-02	1.41E-01	2.92E-01	4.00E-02
Te	1.34E-04	6.22E-04	7.24E-03	1.05E-01	2.99E-01	2.49E-02
Sr	8.65E-07	1.47E-05	7.91E-04	3.85E-02	2.79E-01	8.15E-03
Ru	1.03E-11	3.17E-09	5.29E-05	2.47E-03	4.76E-02	7.93E-04
La	1.12E-10	1.83E-07	5.35E-05	4.19E-03	9.41E-02	1.30E-03
Ce	1.76E-10	5.19E-07	7.90E-05	7.03E-03	1.33E-01	1.97E-03
Ba	1.44E-06	2.40E-05	8.33E-04	3.87E-02	2.75E-01	7.44E-03

Table 3.3-2. Source Term Statistics: Release for Late Containment Failure

Source Term	Percentile					
	Min	5-th	50-th	95-th	Max	Mean
NG	5.00E-03	5.00E-03	5.00E-03	8.40E-03	3.34E-01	7.53E-03
I	1.22E-08	2.45E-06	7.49E-05	3.44E-04	2.41E-03	1.30E-04
Cs	9.29E-10	8.45E-08	5.43E-07	8.24E-06	3.78E-04	4.03E-06
Te	4.75E-09	2.94E-08	3.59E-07	1.95E-05	8.36E-05	3.05E-06
Sr	2.73E-11	2.34E-10	4.18E-08	5.59E-07	1.89E-06	1.38E-07
Ru	8.39E-16	3.98E-15	8.04E-11	2.61E-08	1.26E-07	4.50E-09
La	7.87E-16	1.37E-13	1.28E-09	9.31E-08	2.53E-07	1.76E-08
Ce	4.59E-15	1.15E-11	1.77E-09	1.03E-07	3.53E-07	1.99E-08
Ba	4.41E-11	3.14E-10	3.57E-08	4.75E-07	1.50E-06	1.17E-07

Table 3.3-3. Source Term Statistics: Release for Bypass

Source Term	Percentile					
	Min	5-th	50-th	95-th	Max	Mean
NG	1.09E-01	2.08E-01	8.46E-01	9.85E-01	1.00E+00	7.65E-01
I	8.32E-05	4.89E-03	1.44E-01	6.33E-01	9.38E-01	2.12E-01
Cs	6.22E-05	3.27E-03	1.15E-01	6.20E-01	8.83E-01	1.88E-01
Te	6.08E-05	1.04E-03	3.62E-02	4.60E-01	6.73E-01	9.45E-02
Sr	4.91E-07	1.10E-05	1.24E-03	2.31E-01	6.93E-01	3.16E-02
Ru	1.79E-11	3.77E-10	3.18E-04	3.59E-02	1.43E-01	5.69E-03
La	1.80E-11	3.48E-09	4.92E-05	1.00E-02	7.30E-02	2.23E-03
Ce	2.66E-11	2.01E-08	6.74E-05	3.80E-02	5.06E-01	1.09E-02
Ba	6.89E-07	2.90E-05	1.67E-03	2.49E-01	6.93E-01	3.35E-02

Table 3.3-4. Source Term Statistics: Release for No Containment Failure

Source Term	Percentile					
	Min	5-th	50-th	95-th	Max	Mean
NG	1.09E-03	1.45E-03	4.57E-03	5.00E-03	5.00E-03	4.09E-03
I	1.05E-09	6.43E-07	1.58E-05	1.03E-04	1.49E-04	2.93E-05
Cs	5.00E-12	9.04E-09	1.11E-07	2.74E-07	4.39E-07	1.17E-07
Te	2.98E-10	4.61E-09	6.76E-08	1.61E-07	3.48E-07	7.13E-08
Sr	5.10E-12	4.42E-11	7.26E-09	9.66E-08	2.42E-07	2.24E-08
Ru	1.52E-16	7.07E-16	1.95E-11	7.95E-09	2.90E-08	1.20E-09
La	1.43E-16	9.65E-15	2.30E-10	1.32E-08	4.63E-08	3.01E-09
Ce	1.13E-15	2.01E-12	3.19E-10	1.84E-08	6.45E-08	3.84E-09
Ba	1.03E-11	6.34E-11	6.02E-09	9.07E-08	2.42E-07	2.01E-08

Table 3.3-5
Additional Conditions for the Source Term Bins Analyzed

1.	Containment Failure Near RPV Failure
2.	No Spray
3.	Prompt Dry MCCI
4.	RPV Pressure at System Set Point
5.	With HPME
6.	No STGR
7.	Large Amount of MCCI
8.	Low ZR Oxidation
9.	Low HPME
10.	Containment Rupture
11.	Two Large Holes in RCS

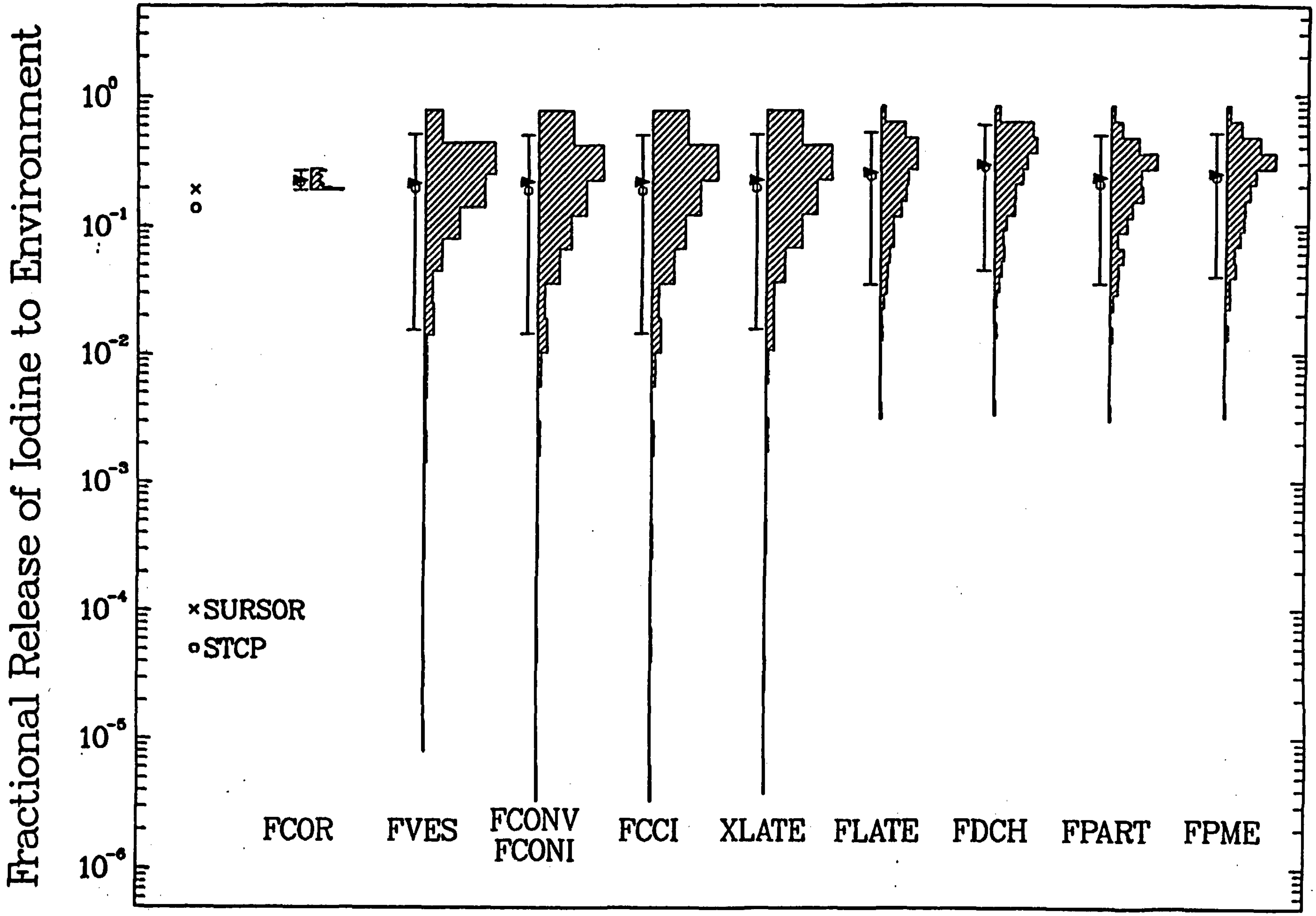


Figure 3.3-2 Fractional Release of Iodine to the Environment

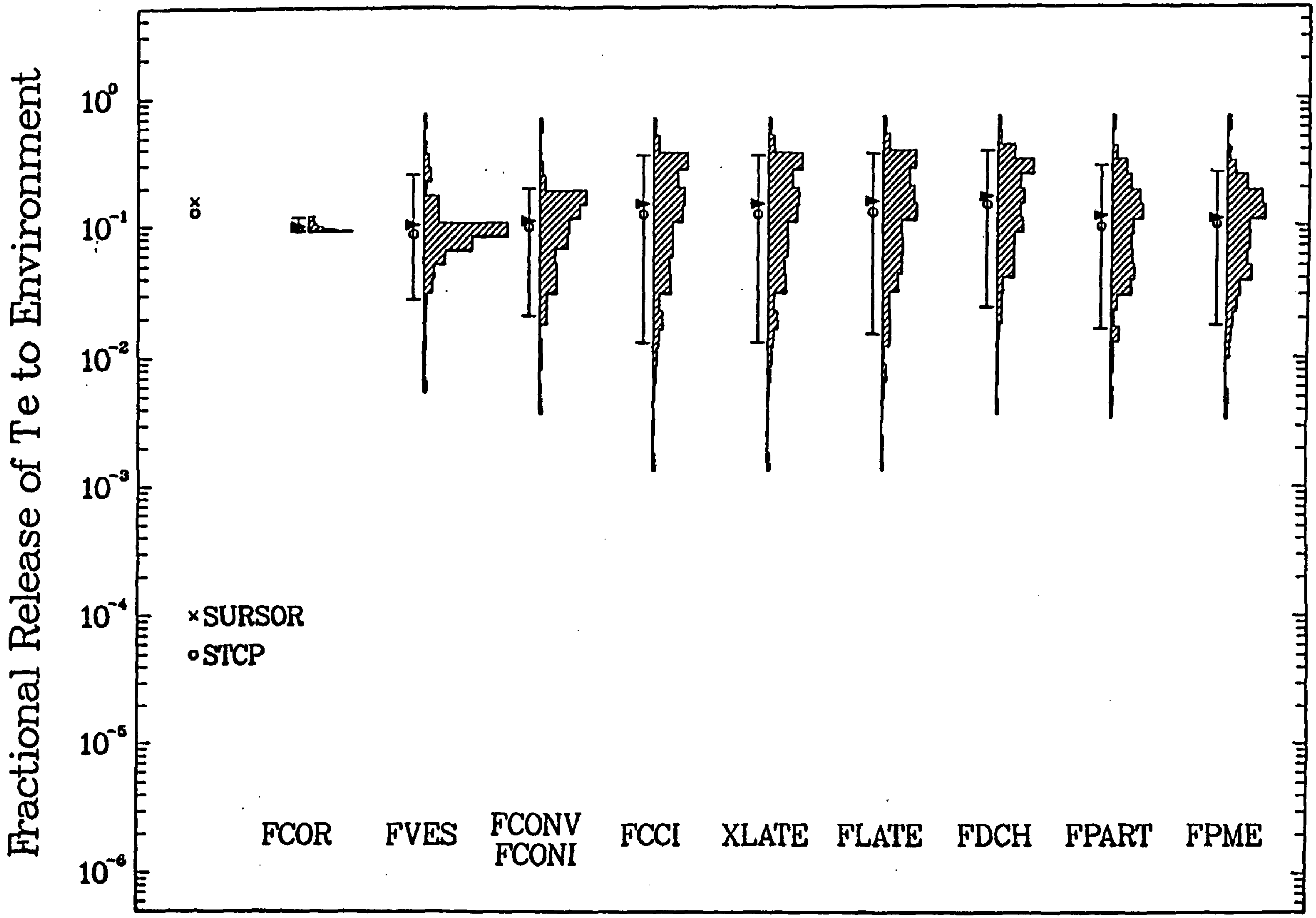


Figure 3.3-3 Fractional Release of Tellurium to the Environment

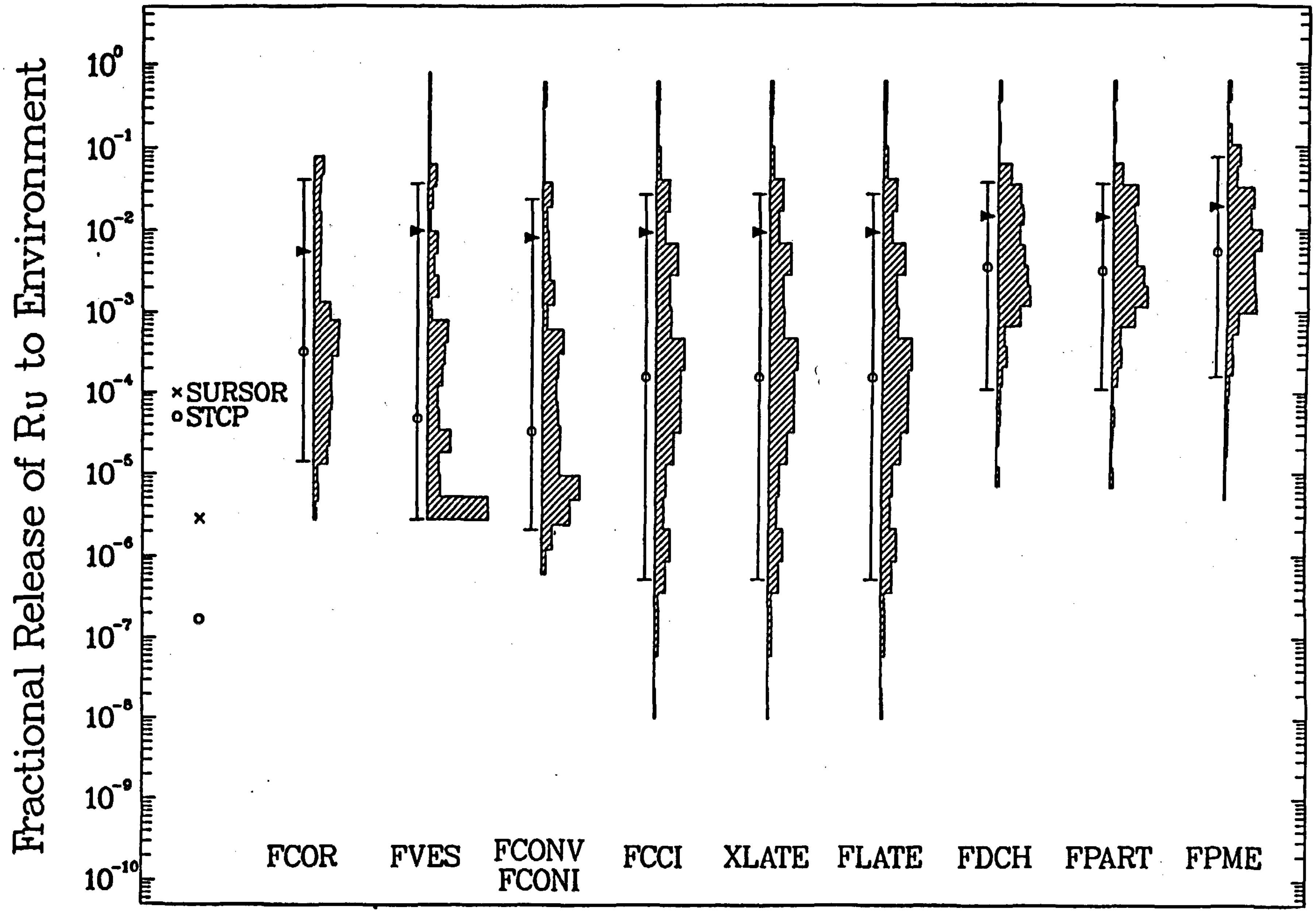


Figure 3.3-4 Fractional Release of Rubidium to the Environment

Another form of the information generated from the source term analysis is the complementary cumulative density function (CCDF), showing the exceedance frequency of certain magnitude of radioactivity release. Figures 3.3-5 and 3.3-6, for example, are CCDFs for Iodine and Strontium, one from volatile groups, and the other from refractory groups, respectively. These CCDFs can be used in comparison of the release fraction and the corresponding frequency with some predetermined criteria. For example, the mean frequency of an Iodine release fraction equal to or greater than 0.2 os 2.6E-6 per reactor-year. For Strontium, the release fraction corresponding to the same mean frequency is 0.006.

3.3.3 Comparison of ZISOR with STCP Results

Four Zion sequences were analyzed by BCD using the STCP.[4] The definition, together with its corresponding bin definition for each of the four sequences is summarized in Table 3.3-6.

To compare the ZISOR results with the STCP results, two sets of ZISOR calculations were made. In one set of calculations, only point value estimations were used. In this calculation, the numerical value of the parameter representing the phenomena not modeled by the STCP was set equal to zero. In the second set of calculations, a source term distribution was produced by covarying 12 input variables in 150 LHS samples. The results of these two calculation sets and the STCP results are displayed in Figures 3.3-7 through 3.3-10. The source term distributions shown in these figures are histograms of the results of 150 LHS samples.

The STCP calculations for all nuclides, except Iodine and Cesium, is fairly close to the median of the distribution obtained by ZISOR. It should be remembered that, in the present analysis, ZISOR includes revaporization processes in the calculation of Iodine and Cesium releases to the environment, hence these distributions are much higher than the STCP calculations.

3.4 Partitioning of the Source Terms for Consequence Analysis

The accident progression analysis and the subsequent source term analysis resulted in the generation of a very large number source terms. It is not computationally possible to perform a calculation with the MACCS consequence model[6] for each of these source terms. Therefore, the interface between the source term analysis and the consequence analysis is formed by grouping this large number of source terms into a much smaller number of source term groups. These groups are defined so that the source terms within them have similar properties and a frequency-weighted mean source term is determined for each group. Then, a single MACCS calculation is performed for each mean source term. This grouping of the source terms is performed with the PARTITION program,[7] and the process is referred to as "partitioning the source terms" or just "partitioning."

The partitioning process involves the following steps: definition of an early health effect weight (EH) for each source term, definition of a chronic health effect weight (CH) for each source term, subdivision (partitioning) of the source terms on the basis of EH and CH, a further subdivision on the basis of evacuation timing, and calculation of frequency-weighted mean source terms. The

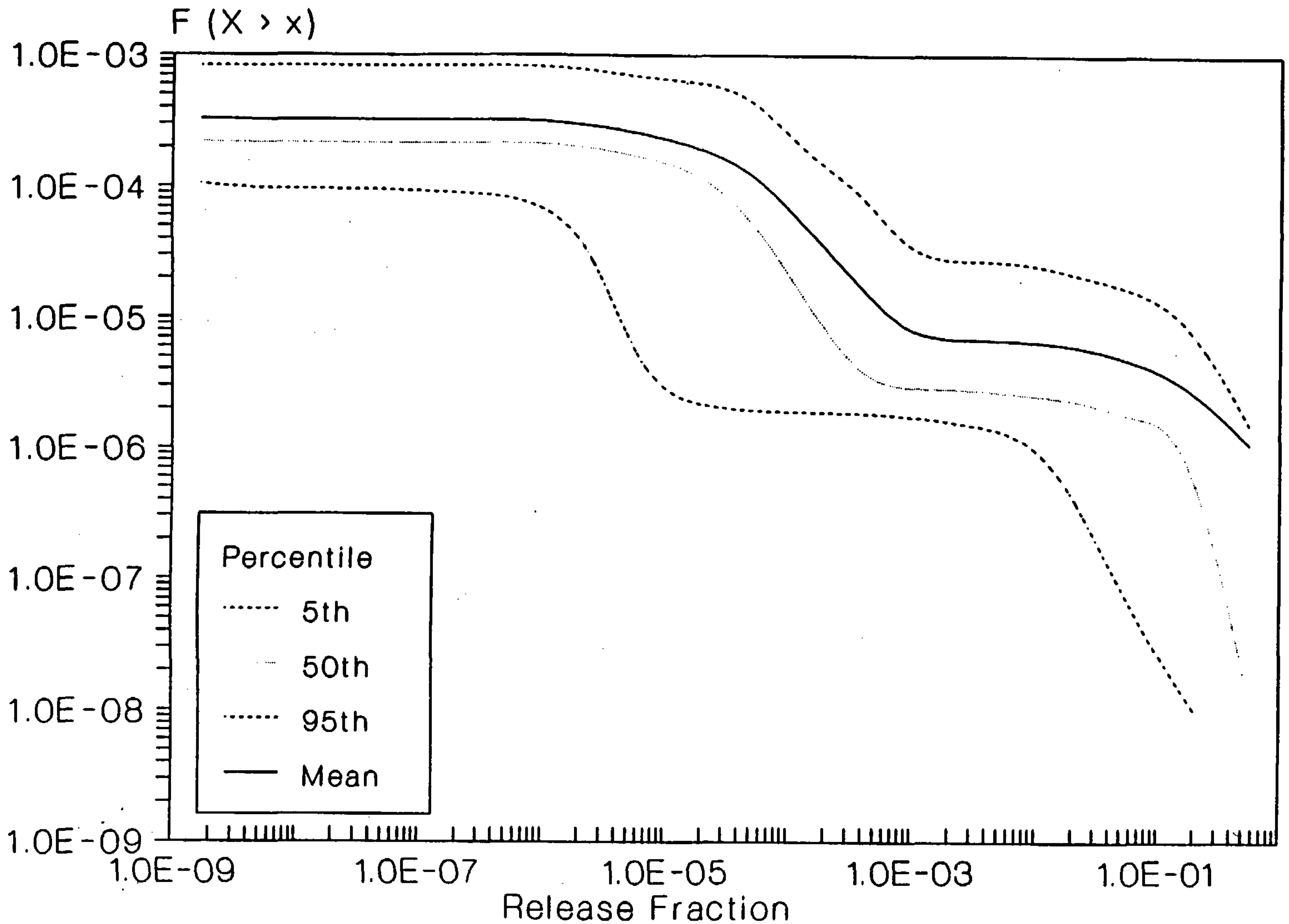


Figure 3.3-5 CCDF For Iodine Release

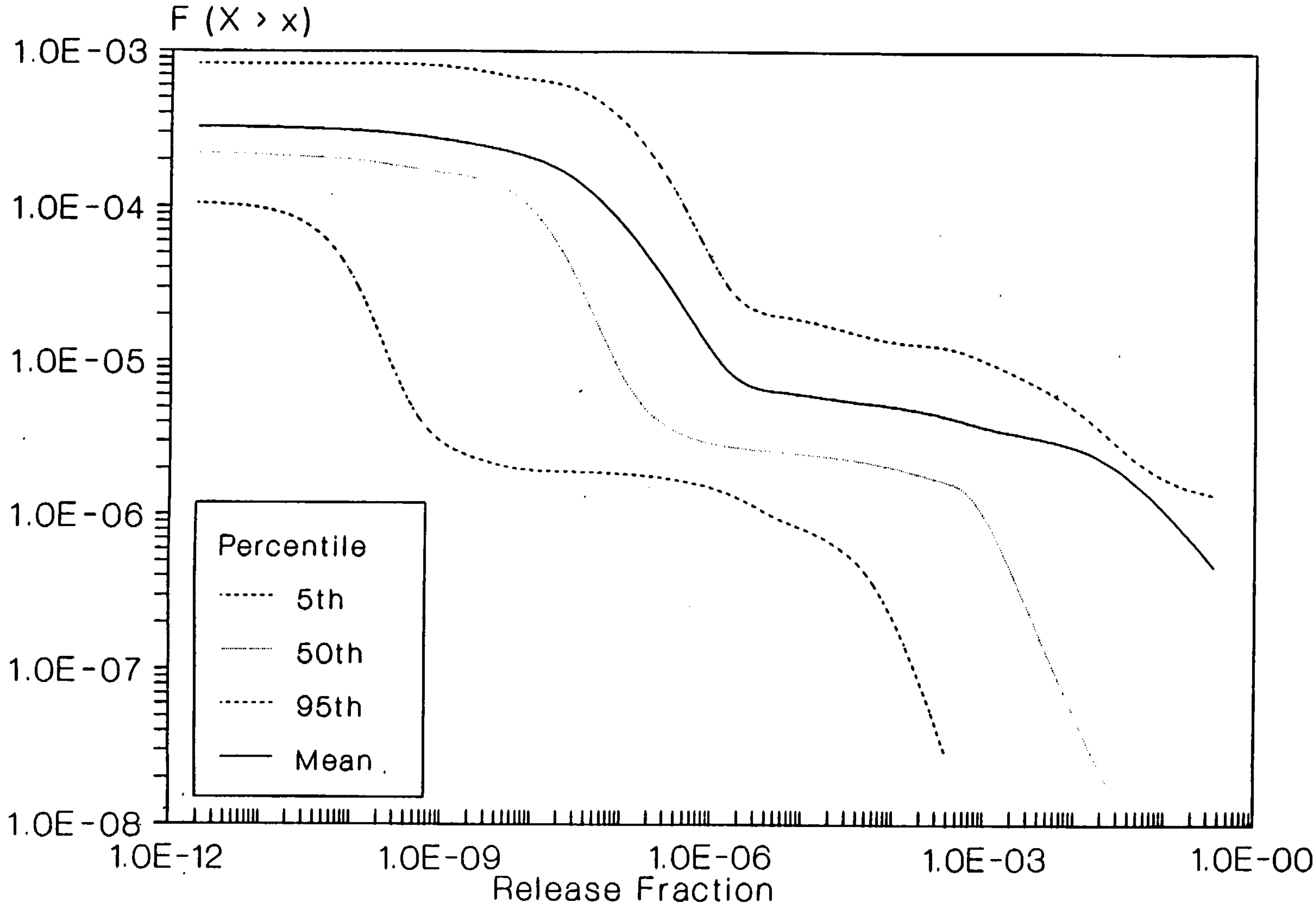


Figure 3.3-6 CCDF for Strontium Release

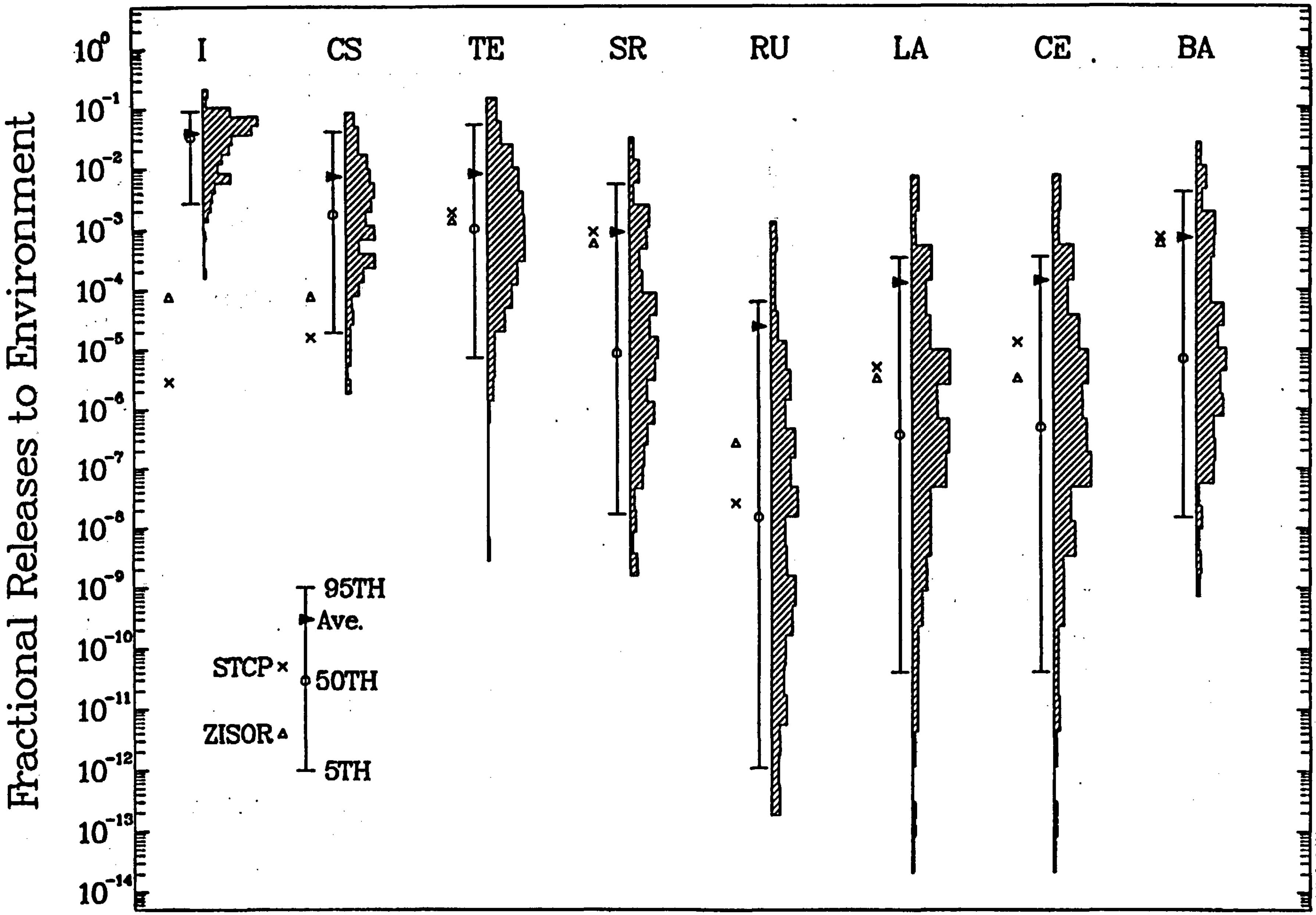


Figure 3.3-7 Comparison to the STCP Results: S_2DCr (Containment Fails Late, Rupture)

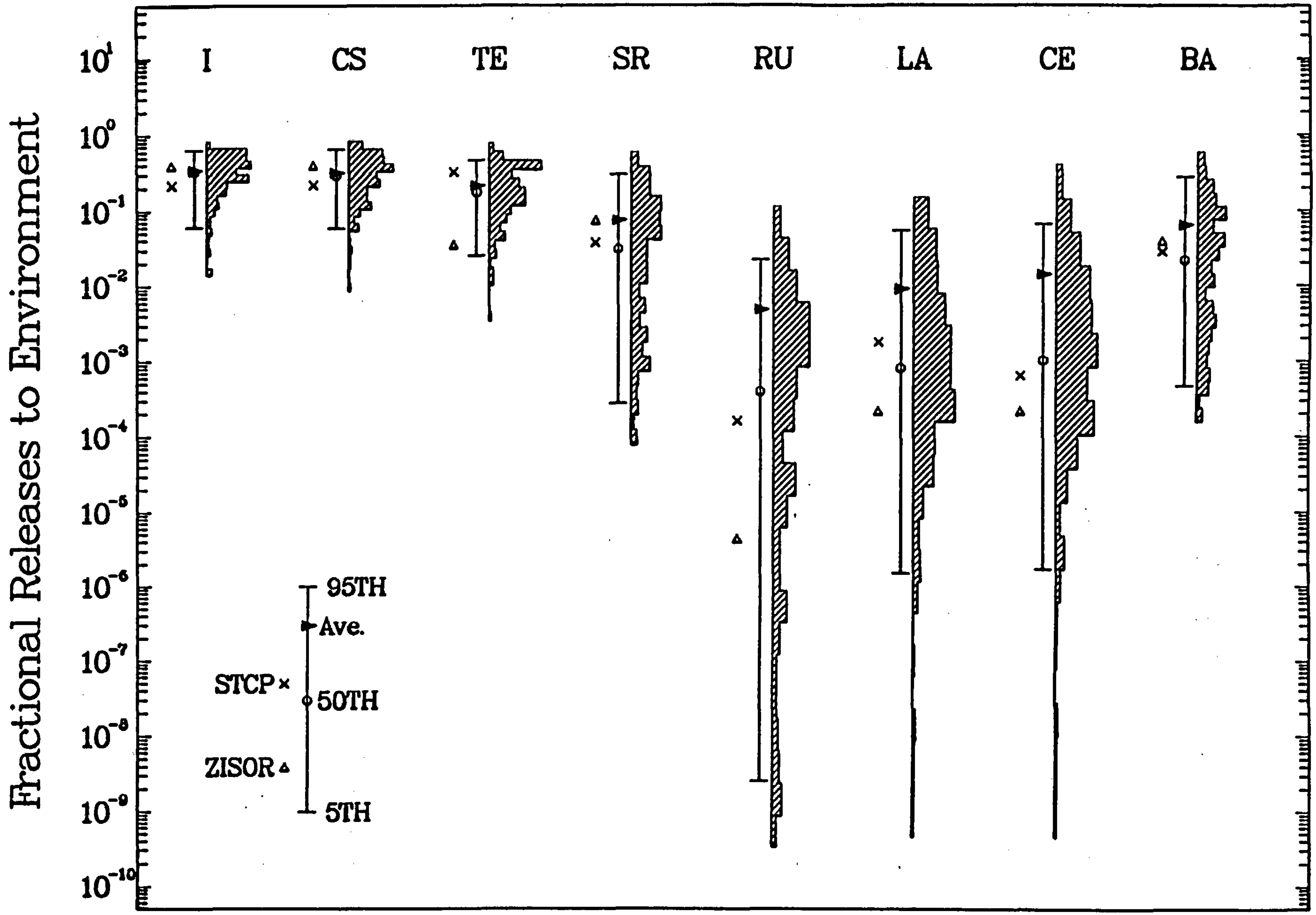


Figure 3.3-8 Comparison to the STCP Results: S_2DCF (Containment Fails at VB, Rupture)

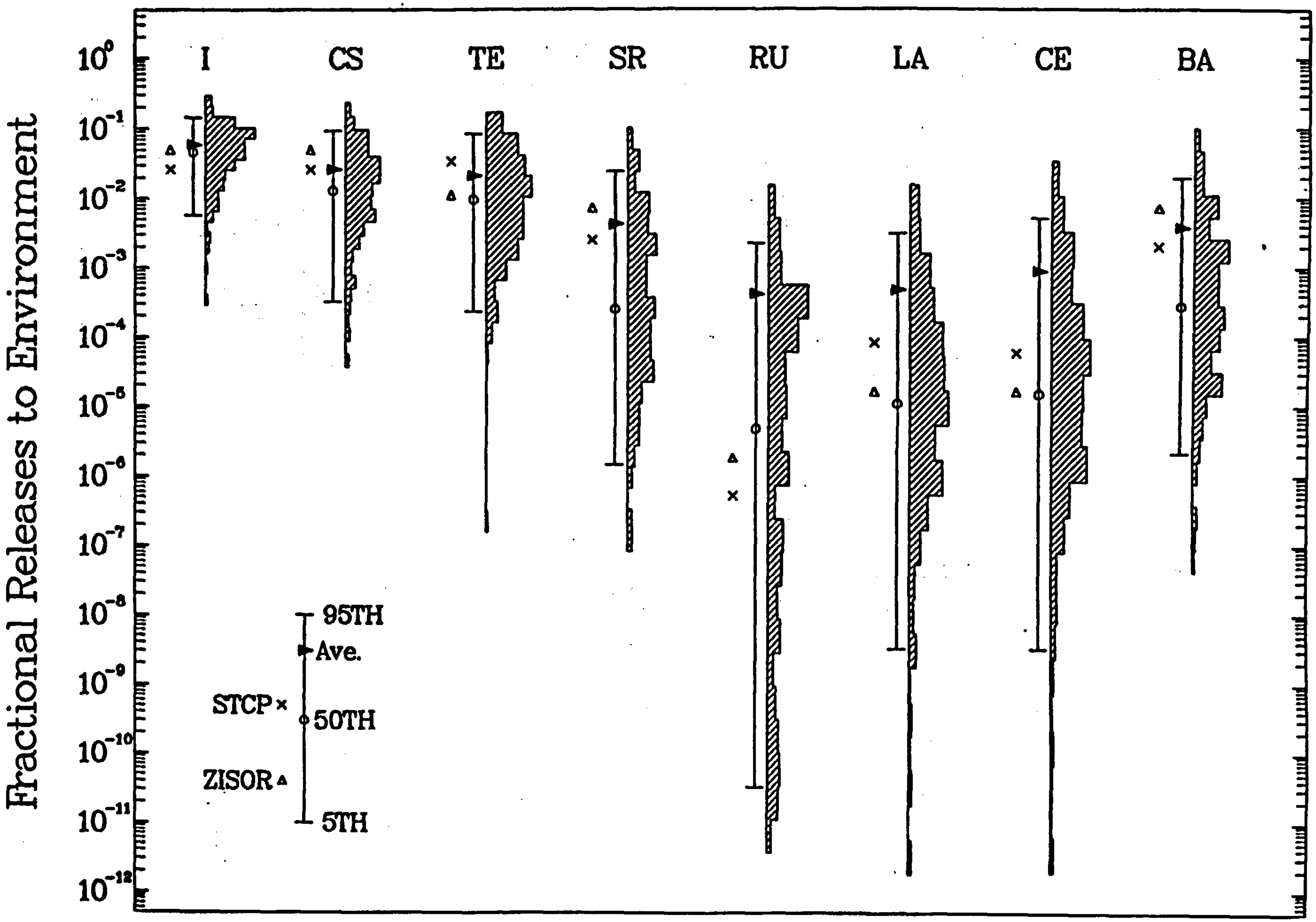


Figure 3.3-9 Comparison to the STCP Results: S_2DCF (Containment Fails Late, Rupture)

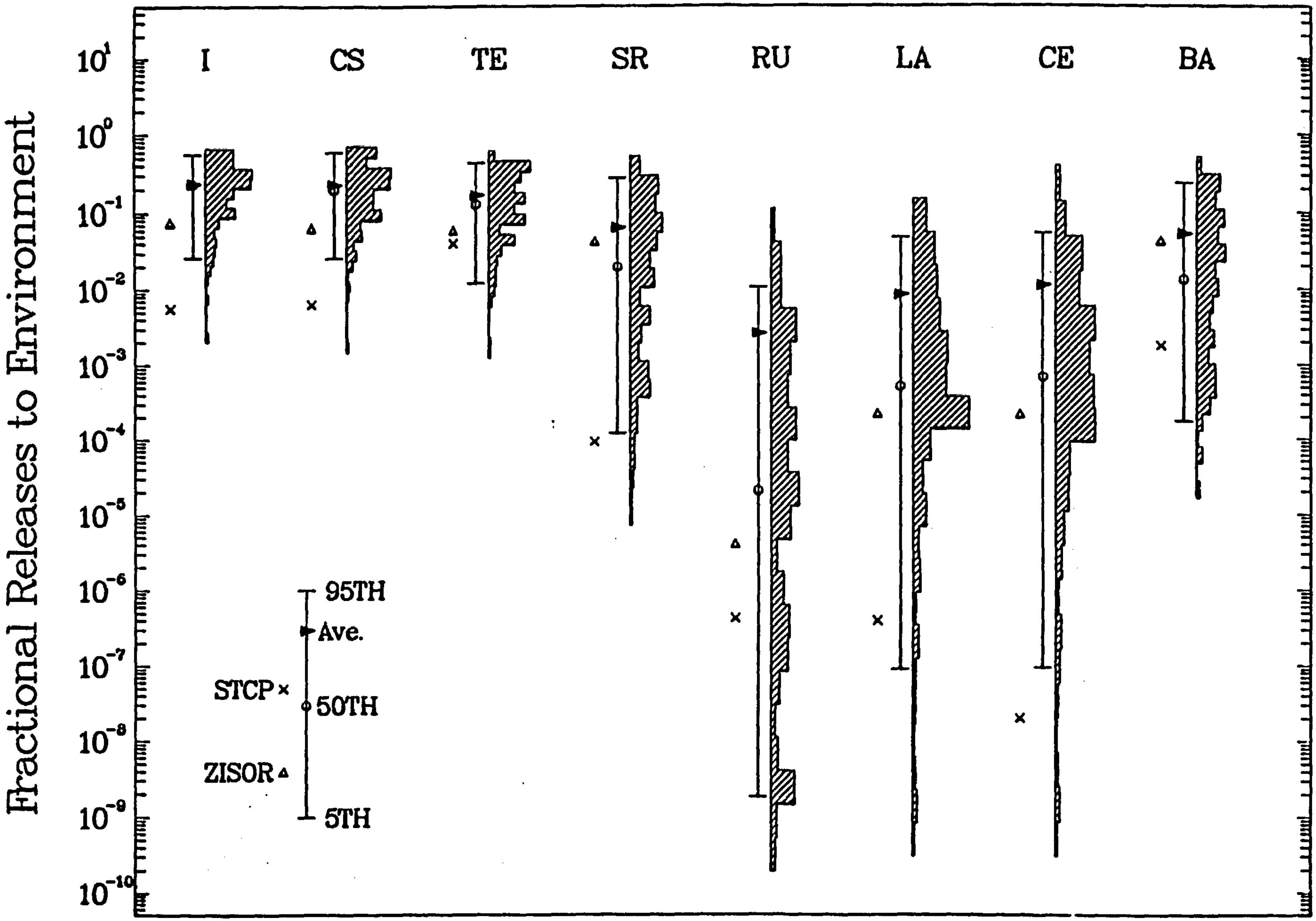


Figure 3.3-10 Comparison to the STCP Results: TMLU (Containment Fails at VB, Rupture)

Table 3.3-6

Sequences Analyzed by STCP

Sequence	Definition	Bin in NUREG-1150
S_2DCR	LOCA due to the RCP seals failure with the simultaneous failures of containment cooling and ECCSs. Late overpressure containment failure.	EBDBBCABDBAB
S_2DCF1	LOCA by the failure of the RCP seals. The ECCSs, containment sprays, and fan coolers are inoperable. Containment failure at vessel breach.	DHEBBCABDBAB
S_2DCF2	LOCA by the failure of the RCP seals. The ECCSs, containment sprays, and fan coolers are inoperable. Late containment failure.	EHBBBCABDBAB
TMLU	Transient with coincident failure of the ECCSs. Containment cooling available, early containment failure due to overpressure.	DAEABCABDBAA

partitioning process is described in detail in NUREG/CR-4551, Vol. 1, and the user's manual for the PARTITION program.[7] This section describes the details of the partitioning process for source terms generated in the source term analysis for internal initiators.

The early health effect weight EH is based on converting the radionuclide release associated with a source term into an equivalent I-131 release and then estimating the number of early fatalities that would result from this equivalent I-131 release. This estimated number of early fatalities is the early health effect weight EH. The chronic health effect weight CH is based on an assumed linear relationship between cancer fatalities due to a radionuclide and the amount of that radionuclide released. Specifically, a site-specific MACCS calculation is performed for a fixed release for each of the 60 radionuclides included in the NUREG-1150 consequence calculations. The results of these calculations and the assumed linear relationship between the amount released and cancer fatalities for each radionuclide are then used to estimate the total number of chronic fatalities associated with a source term. This estimated number of chronic fatalities is the chronic health effect weight CH.

The site-specific MACCS calculations that underlie the early and chronic health effect weights were performed with very conservative assumptions with respect to the energy and timing of the releases and also with respect to the emergency responses taken. As a result, these weights should be regarded as a measure of the potential of a source term to cause early and chronic fatalities rather than as an estimate of the fatalities that would actually result from a source term.

The partitioning process treats the cases for $EH>0$ and $CH>0$ and for $EH=0$ and $CH>0$ separately. The case for $EH>0$ and $CH>0$ is treated first by PARTITION. Figure 3.4-1 shows a plot of the pairs (CH, EH) for the source terms for which both EH and CH are nonzero. The partitioning process is based on laying a grid on the (CH, EH) space shown in Figure 3.4-1 and then pooling cells that have either a small frequency or contain a small number of source terms. Specifically, the grid is selected so that the ratio between the maximum and minimum value for CH in any cell and also the ratio between the maximum and minimum value for EH in any cell will be less than a specified value. The result of placing the selected grid on the (CH, EH) space is also shown in Figure 3.4-1. The source term groups were then subdivided based on the differences in the amount of time available for actual evacuation, i.e., time difference between the warning time (T_w) and the first radioactivity release (T_i).

For the first subgroup, the maximum warning interval, $T_i - T_w$, was selected to be 1 h, for the second subgroup it was 2 h, whilst the third subgroup contained all releases with a warning interval larger than 2 h. These break points were selected on the basis of the evacuation delay time, which for the Zion site was estimated by the NRC project staff to be 2.3 h. Thus, in almost all the spacial grid points, for the first subgroup evacuation is expected to start after the first radioactive release, for the second subgroup at or about the time of release, and for the last subgroup before the first release has occurred.

Tables 3.4-1 and 3.4-2 show the mapping of all the source terms after the partitioning process. Since the data base included in PARTITION/BNL was Zion-specific but derived from calculations performed with the earlier MACCS Version

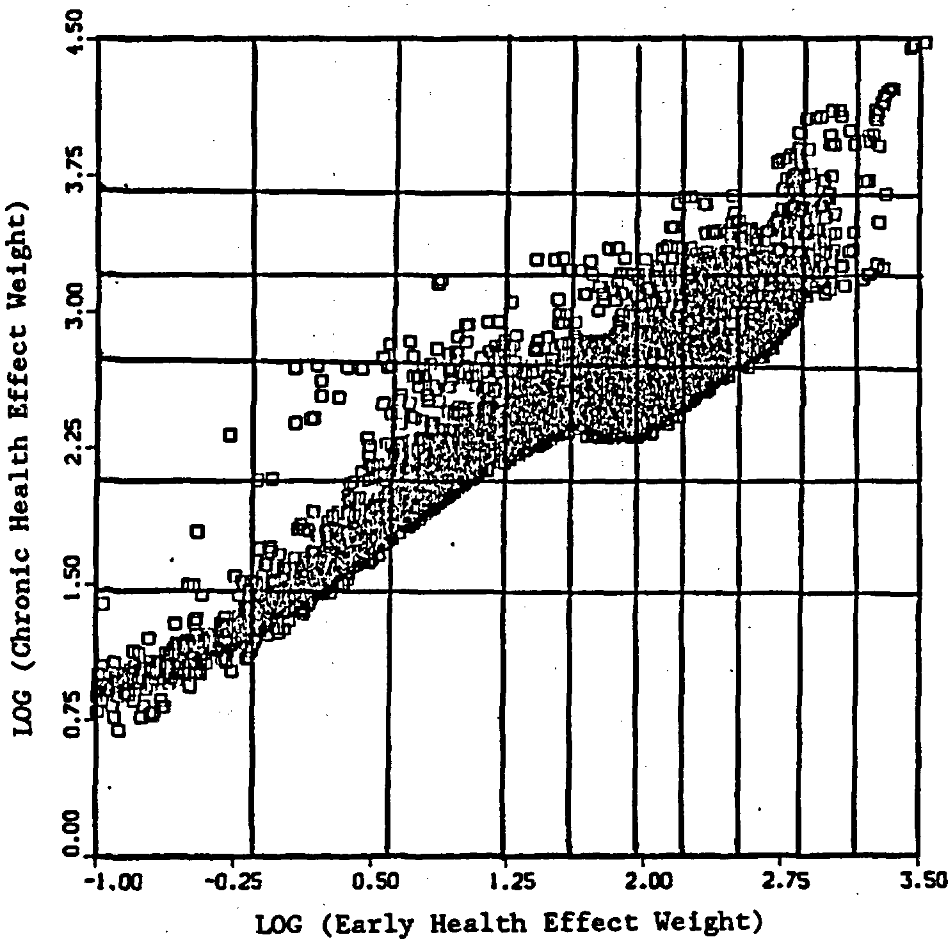


Figure 3.4-1 Distribution of Nonzero Early and Chronic Health Effects Weights for Internal Initiators

Table 3.4-1 Distribution of Source Terms with Non-Zero Early Fatality and Chronic Fatality Weights

							- 1
	68	49	17				-.776
		24	330				-.327
			539				.121
			604	188			.569
				763			1.018
				140			1.466
				592	677		1.915
					213	329	2.363
						199	2.811
						53	3.260
							3.484
.703	1.01	1.64	2.26	2.89	3.51	4.14	

Table 3.4-2 Distribution of Source Terms with Zero Early Fatality Weight and Non-Zero Chronic Fatality Weight

516	604	170
1	2	3

1.4, with consequences extended to 500 miles, Table 3.4-3 shows a comparison of latent cancer fatalities projected by PARTITION/BNL with the ones calculated by MACCS Version 1.5 for mean cell source terms. The agreement is fairly close.

Table 3.4-4 shows the composition of each PARTITION group and relative contribution from various containment failure bins for the PARTITION groups. The last three groups (Groups 301, 302, and 303) represent no containment failures or BMT with a small fraction of excessive containment leakage. There are two distinctive features in this table: one is that four containment bins, i.e., the alpha-mode failure, containment isolation failure, bypass, and SGTR, appear almost in every group. Another is that the relative contribution of these four bins are dependent on their relative frequency of occurrence. The former feature is due to the sampling scheme, or the type of distribution for each containment failure. These four failure bins have continuous distributions, differing from others which have discrete distributions. The discrete distribution is mainly due to the zero-one sampling scheme adopted for this study.

High consequence groups (Groups 106 to 179) are either from the alpha mode failure bin or the SGTR bin. Lower consequence groups (Groups 1 to 105), on the other hand, mainly consist of isolation failure or leakage bins. Even though the source terms from those bins such as catastrophic rupture, and rupture, have high source terms, the contributions are relatively small because of their relatively low frequency of occurrence.

The mean source terms for each of the partition groups is shown in Table 3.4-5. In the consequence analysis, a full MACCS calculation is performed for the mean source terms in Table 3.4-5 (Chapter 4). Table 3.4-5 provides information on fractional release of various fission product groups, elevation of release, energy of release, and timing information for each source term group. In addition, the mean frequency of each group is provided with the group identification number and the population of individual source terms within the groups.

3.5 References

1. H-N. Jow, W.B. Murfin, and J.D. Johnson, "XSOR Codes User's Manual," NUREG/CR-5360, SAND89-0943, to be published.*

2. E. G. Bergeron et al., "Evaluation of Severe Accident Risks: Methodology," NUREG/CR-4551, Vol. 1, to be published.*

3. F. Harper, et al., Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," NUREG/CR-4551, SAND86-1309, Vol. 2, Part 4, June 1992.

4. R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios," NUREG/CR-4624, BMI-2139, Vol. 6, August 1990.

5. B. J. Garrick et al., "Seabrook Station Probabilistic Safety Assessment," PLG-0300, 1983.

* Available in the NRC Public Document Room, 2120 L Street, NW., Washington DC.

Table 3.4-3
Comparison of PARTITION/BNL Predictions for
Latent Cancer Fatalities with MACCS 1.5 Calculations

Source Term Groups	PARTITION/BNL Prediction	MACCS 1.5 Calculations (to 500 miles)
1 - 30	25	49
31 - 60	80	69
61 - 90	250	408
91 - 120	1000	1540
121 - 150	4000	3670
151 - 180	15800	9910

Table 3.4-4
Composition of Groups by Bins

Group No.	Alpha	ECR	ER	ES	EL	Is. Leak	LR	LS	LL	BMT	NoCF	SGTR	V
1	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.59E-03	9.96E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.73E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E+00	0.00E+00
31	0.00E+00	0.00E+00	0.00E+00	5.03E-02	5.93E-04	9.44E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.30E-03	0.00E+00
33	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.58E-01	2.42E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
61	0.00E+00	0.00E+00	0.00E+00	1.52E-01	0.00E+00	8.37E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.08E-02	0.00E+00
64	1.24E-01	0.00E+00	3.46E-04	0.00E+00	3.04E-02	8.35E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.30E-03	8.29E-04
65	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.49E-03	1.44E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.97E-01	0.00E+00
66	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
67	1.96E-01	1.15E-03	1.62E-04	3.43E-02	3.84E-02	6.95E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.16E-02	3.09E-03
68	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.72E-02	2.29E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.80E-01	0.00E+00
69	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.36E-03	9.96E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
70	3.00E-01	3.86E-04	1.31E-04	1.21E-04	4.43E-01	2.41E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.41E-02	1.52E-03
71	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.40E-04	7.45E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.92E-01	0.00E+00
72	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.09E-01	6.91E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
100	2.15E-01	0.00E+00	0.00E+00	0.00E+00	6.63E-05	7.23E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.90E-02	3.18E-03
101	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.02E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.90E-01	0.00E+00
103	6.10E-01	6.76E-05	0.00E+00	0.00E+00	8.26E-02	2.57E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.51E-02	5.63E-03
104	2.26E-03	0.00E+00	0.00E+00	0.00E+00	3.32E-03	5.89E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.89E-01	0.00E+00
105	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.00E-01	5.00E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
106	6.89E-01	0.00E+00	0.00E+00	0.00E+00	6.59E-03	7.28E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.20E-01	1.12E-02
107	1.64E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.28E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.92E-01	0.00E+00
136	8.84E-01	2.42E-05	0.00E+00	4.32E-05	8.20E-03	1.68E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.35E-02	2.73E-02
137	5.15E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.88E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.90E-01	0.00E+00
139	9.24E-01	1.55E-03	0.00E+00	0.00E+00	3.11E-02	2.27E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.49E-02	5.61E-03
140	5.85E-03	4.81E-04	0.00E+00	0.00E+00	6.48E-03	7.50E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.80E-01	0.00E+00
142	9.67E-01	2.08E-04	0.00E+00	0.00E+00	6.83E-05	1.00E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.18E-02	1.05E-02
143	1.15E-02	5.91E-05	0.00E+00	0.00E+00	0.00E+00	5.58E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.83E-01	0.00E+00
172	5.20E-01	6.16E-02	0.00E+00	0.00E+00	3.60E-03	4.89E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.32E-01	1.34E-01
173	7.81E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.89E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.87E-01	0.00E+00
175	9.48E-01	6.80E-03	0.00E+00	0.00E+00	3.32E-04	5.10E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.96E-02	1.99E-02
176	2.74E-03	6.95E-02	0.00E+00	0.00E+00	4.49E-02	4.52E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.78E-01	0.00E+00

Table 3.4-4 (Continued)
Composition of Groups by Bins

Group No.	Alpha	ECR	ER	ES	EL	Is. Leak	LR	LS	LL	BMT	NoCF	SGTR	V
178	9.05E-01	0.00E+00	0.00E+00	0.00E+00	8.06E-05	1.75E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.92E-02	1.86E-02
179	6.52E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.67E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.30E-01	0.00E+00
301	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.49E-02	9.15E-01	0.00E+00	0.00E+00
302	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.83E-01	6.17E-01	0.00E+00	0.00E+00
303	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.34E-03	4.98E-03	0.00E+00	0.00E+00	4.68E-05	9.79E-01	0.00E+00	1.42E-02	0.00E+00

Table 3.4-5
Zion Input for MACCS Calculations

Group No.	Popn.	Frequency	TW (sec)	T1 T2 (sec)	DT1 DT2 (sec)	Elevation (m)	E1 E2 (W)	NG1 NG2	I1 I2	Cs1 Cs2	Te1 Te2	Sr1 Sr2	Ru1 Ru2	La1 La2	Ce1 Ce2	Ba1 Ba2
1	58	4.1E-08	1.4E+04	1.8E+04	1.8E+03	10.0	3.1E+05	7.3E-01	2.4E-04	1.7E-04	3.2E-05	1.3E-06	3.9E-07	5.8E-08	1.8E-07	1.9E-06
				2.0E+04	3.6E+04		1.6E+06	2.7E-01	8.4E-04	4.4E-05	5.1E-05	5.3E-05	1.5E-06	1.0E-05	1.1E-05	4.5E-05
2	18	2.4E-08	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	3.8E-01	2.6E-03	1.5E-03	1.3E-04	1.5E-06	9.3E-08	1.7E-08	2.0E-08	4.0E-06
				5.6E+04	2.2E+04		1.7E+04	8.9E-04	8.5E-05	2.2E-09	7.0E-08	6.5E-09	1.4E-14	1.5E-10	1.8E-10	5.0E-09
31	46	7.0E-08	1.4E+04	1.8E+04	1.8E+03	10.0	3.5E+05	9.3E-01	4.8E-04	3.5E-04	1.4E-04	2.0E-05	4.7E-06	9.7E-07	3.4E-06	2.6E-05
				2.0E+04	3.6E+04		1.6E+06	5.3E-02	2.1E-03	1.0E-04	2.7E-05	4.4E-06	9.9E-12	9.9E-08	1.5E-07	3.5E-06
33	4	3.6E-10	9.4E+03	3.8E+04	1.8E+03	4.3	1.9E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				4.0E+04	3.6E+04		1.9E+05	8.9E-01	3.6E-03	4.4E-05	1.2E-05	2.2E-07	1.2E-07	6.3E-09	1.0E-08	3.7E-07
61	19	2.3E-08	1.4E+04	1.8E+04	1.8E+03	10.0	3.4E+05	9.3E-01	5.5E-04	3.7E-04	1.0E-04	1.9E-06	1.1E-06	6.2E-08	1.2E-07	3.1E-06
				2.0E+04	3.6E+04		1.6E+06	6.7E-02	3.0E-03	3.8E-05	1.8E-05	3.9E-08	1.4E-13	4.3E-10	5.0E-10	3.8E-08
64	320	2.4E-07	1.4E+04	1.8E+04	1.8E+03	10.0	3.6E+05	8.8E-01	1.7E-03	1.2E-03	3.2E-04	5.0E-05	1.1E-05	3.3E-06	1.4E-05	5.7E-05
				2.0E+04	3.6E+04		1.6E+06	1.1E-01	4.7E-03	4.8E-04	2.3E-04	3.6E-05	2.2E-08	2.4E-06	2.8E-06	2.8E-05
65	33	7.5E-08	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	6.0E-01	1.1E-02	7.9E-03	1.7E-03	6.6E-05	3.3E-05	3.2E-06	9.4E-06	1.0E-04
				5.6E+04	2.2E+04		1.7E+04	9.3E-04	3.1E-04	4.1E-08	4.3E-07	9.1E-08	7.3E-11	1.2E-08	1.3E-08	7.5E-08
66	21	2.0E-07	1.4E+04	4.3E+04	1.8E+03	10.0	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				4.5E+04	3.6E+04		1.9E+05	1.0E+00	7.2E-03	1.1E-03	1.0E-04	8.7E-07	2.0E-08	7.1E-08	7.6E-08	6.9E-07
67	454	3.7E-07	1.4E+04	1.8E+04	1.7E+03	10.0	5.2E+05	9.0E-01	3.7E-03	3.0E-03	1.1E-03	1.0E-04	2.8E-05	7.2E-06	3.1E-05	1.2E-04
				1.9E+04	3.6E+04		1.6E+06	8.1E-02	1.6E-02	3.3E-03	7.3E-04	2.3E-04	1.3E-07	1.0E-05	1.2E-05	1.7E-04
68	105	1.6E-07	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	7.3E-01	2.6E-02	2.0E-02	5.1E-03	3.2E-04	1.0E-04	1.4E-05	4.7E-05	4.1E-04
				5.6E+04	2.2E+04		1.7E+04	5.3E-04	5.8E-04	4.5E-07	5.2E-06	1.5E-06	4.5E-11	5.3E-08	5.4E-08	1.0E-06

Table 3.4-5 (Continued)
Zion Input for MACCS Calculations

Group No.	Popn.	Frequency	TW (sec)	T1 T2 (sec)	DT1 DT2 (sec)	Elevation (m)	E1 E2 (W)	NG1 NG2	I1 I2	Cs1 Cs2	Te1 Te2	Sr1 Sr2	Ru1 Ru2	La1 La2	Ce1 Ce2	Ba1 Ba2
69	43	9.5E-09	6.9E+03	3.6E+04	1.8E+03	10.0	1.9E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.8E+04	3.6E+04		1.9E+05	1.0E+00	2.8E-02	2.8E-03	7.3E-03	1.4E-04	2.9E-06	8.6E-06	8.8E-06	1.1E-04
70	514	1.1E-06	1.4E+04	1.8E+04	1.6E+03	10.0	4.4E+05	9.3E-01	5.4E-03	4.6E-03	1.1E-03	1.1E-04	3.1E-05	5.4E-06	1.7E-05	1.2E-04
				2.0E+04	3.6E+04		1.6E+06	5.6E-02	3.1E-02	4.2E-03	4.4E-03	2.6E-03	2.1E-06	9.5E-05	9.5E-05	1.5E-03
71	112	1.0E-07	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	7.6E-01	5.2E-02	4.0E-02	8.7E-03	1.6E-04	7.3E-05	5.5E-06	1.2E-05	2.9E-04
				5.6E+04	2.2E+04		1.7E+04	9.9E-04	3.3E-04	1.1E-06	1.3E-05	3.7E-06	2.3E-08	4.8E-07	5.0E-07	3.0E-06
72	45	4.0E-09	7.5E+03	3.6E+04	1.8E+03	6.8	1.9E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.8E+04	3.6E+04		1.9E+05	1.0E+00	5.0E-02	7.5E-03	2.6E-02	3.8E-04	4.3E-06	2.3E-05	2.4E-05	2.8E-04
100	161	9.7E-08	1.4E+04	1.8E+04	1.7E+03	10.0	5.0E+05	9.1E-01	1.8E-02	1.5E-02	4.7E-03	4.4E-04	1.4E-04	2.2E-05	7.7E-05	5.3E-04
				2.0E+04	3.6E+04		1.6E+06	6.5E-02	3.7E-02	7.3E-03	9.4E-03	7.9E-04	7.0E-07	4.6E-05	5.1E-05	5.8E-04
101	44	3.8E-08	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	6.9E-01	7.3E-02	5.6E-02	1.3E-02	4.4E-04	2.0E-04	1.9E-05	5.2E-05	7.0E-04
				5.6E+04	2.2E+04		1.7E+04	8.9E-04	6.4E-04	1.6E-06	3.9E-05	8.3E-06	6.8E-09	4.9E-07	5.6E-07	6.4E-06
103	646	6.5E-07	1.4E+04	1.8E+04	1.4E+03	9.9	7.7E+05	8.5E-01	1.8E-02	1.6E-02	4.2E-03	3.1E-04	1.3E-04	2.8E-05	6.2E-05	3.9E-04
				1.9E+04	3.5E+04		1.5E+06	1.3E-01	7.2E-02	3.6E-02	1.6E-02	3.7E-03	4.2E-04	5.7E-04	8.4E-04	3.2E-03
104	159	2.2E-07	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	7.8E-01	1.3E-01	9.7E-02	2.4E-02	1.6E-03	5.7E-04	7.5E-05	2.4E-04	2.4E-03
				5.6E+04	2.2E+04		1.7E+04	1.7E-03	4.8E-04	7.8E-05	1.6E-04	4.7E-05	1.3E-07	1.8E-06	1.8E-06	2.8E-05
105	2	2.5E-11	1.4E+04	4.3E+04	1.8E+03	5.0	1.9E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				4.5E+04	3.6E+04		1.9E+05	1.0E+00	1.3E-01	1.2E-01	6.1E-02	3.2E-03	2.3E-09	8.3E-05	8.5E-05	1.8E-03

Table 3.4-5 (Continued)
Zion Input for MACCS Calculations

Group No.	Popn.	Frequency	TW (sec)	T1 T2 (sec)	DT1 DT2 (sec)	Elevation (m)	E1 E2 (W)	NG1 NG2	I1 I2	Cs1 Cs2	Te1 Te2	Sr1 Sr2	Ru1 Ru2	La1 La2	Ce1 Ce2	Ba1 Ba2
106	108	2.9E-08	1.4E+04	1.8E+04	1.4E+03	9.9	8.5E+05	8.6E-01	2.8E-02	2.4E-02	1.7E-02	3.6E-03	9.6E-04	3.1E-04	1.9E-03	3.9E-03
				1.9E+04	3.5E+04		1.5E+06	8.1E-02	8.6E-02	6.8E-02	6.5E-02	8.8E-03	5.2E-07	2.0E-04	2.0E-04	5.0E-03
107	38	4.2E-08	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	7.7E-01	1.8E-01	1.5E-01	3.9E-02	2.0E-03	7.3E-04	9.5E-05	2.8E-04	3.2E-03
				5.6E+04	2.2E+04		1.7E+04	7.7E-04	4.3E-04	8.8E-05	7.4E-05	5.0E-06	1.3E-10	2.0E-07	2.1E-07	3.0E-06
136	483	4.7E-07	1.3E+04	1.7E+04	1.3E+03	9.5	9.2E+05	7.4E-01	4.6E-02	4.3E-02	1.2E-02	9.4E-04	3.4E-04	5.7E-05	1.7E-04	1.2E-03
				1.9E+04	3.4E+04		1.4E+06	2.1E-01	9.4E-02	8.7E-02	8.7E-02	1.6E-02	2.1E-05	1.4E-03	1.6E-03	1.2E-02
137	194	2.3E-07	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	8.6E-01	2.5E-01	2.1E-01	6.1E-02	5.4E-03	1.9E-03	2.4E-04	7.9E-04	7.0E-03
				5.6E+04	2.2E+04		1.7E+04	7.6E-04	5.7E-04	2.1E-04	2.5E-04	3.4E-05	4.5E-08	3.5E-06	3.6E-06	2.6E-05
139	441	8.1E-07	1.4E+04	1.7E+04	1.0E+03	9.9	8.3E+05	8.9E-01	5.9E-02	4.7E-02	2.6E-02	1.9E-02	4.1E-03	2.0E-03	1.3E-02	1.9E-02
				1.9E+04	3.6E+04		1.6E+06	1.1E-01	1.5E-01	1.3E-01	1.1E-01	3.0E-02	4.4E-05	3.8E-03	3.8E-03	2.4E-02
140	164	1.4E-07	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	9.3E-01	4.0E-01	3.6E-01	1.3E-01	2.1E-02	4.7E-03	8.9E-04	3.2E-03	2.4E-02
				5.6E+04	2.2E+04		1.7E+04	1.6E-03	4.2E-04	4.4E-05	9.6E-04	1.6E-04	4.1E-07	1.3E-05	1.4E-05	1.2E-04
142	156	2.9E-07	1.4E+04	1.8E+04	9.8E+02	9.8	7.6E+05	2.6E-01	1.7E-02	1.6E-02	3.1E-03	2.6E-04	8.8E-05	1.8E-05	5.4E-05	3.2E-04
				1.9E+04	3.6E+04		1.6E+06	7.4E-01	3.1E-01	3.2E-01	2.7E-01	1.3E-01	4.4E-04	2.0E-02	2.0E-02	1.1E-01
143	56	6.0E-08	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	9.5E-01	5.8E-01	5.4E-01	2.2E-01	6.4E-03	4.7E-03	3.0E-04	7.1E-04	1.1E-02
				5.6E+04	2.2E+04		1.7E+04	4.5E-03	4.4E-04	2.8E-04	3.1E-03	1.4E-03	6.2E-06	2.2E-04	2.3E-04	1.2E-03
172	281	4.7E-08	7.5E+03	1.2E+04	2.1E+03	8.3	5.5E+06	8.3E-01	3.5E-01	3.2E-01	1.4E-01	4.9E-02	1.0E-02	3.6E-03	1.7E-02	5.2E-02
				1.5E+04	3.0E+04		9.9E+05	1.0E-01	3.0E-02	2.1E-02	1.1E-01	3.9E-02	2.3E-04	5.4E-03	5.6E-03	3.2E-02
173	51	4.9E-08	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	1.0E+00	6.2E-01	5.9E-01	4.0E-01	1.5E-01	2.7E-02	8.5E-03	3.6E-02	1.7E-01

Table 3.4-5 (Continued)
Zion Input for MACCS Calculations

Group No.	Popn.	Frequency	TW (sec)	T1 T2 (sec)	DT1 DT2 (sec)	Elevation (m)	E1 E2 (W)	NG1 NG2	I1 I2	Cs1 Cs2	Te1 Te2	Sr1 Sr2	Ru1 Ru2	La1 La2	Ce1 Ce2	Ba1 Ba2
				5.6E+04	2.2E+04		1.7E+04	3.5E-04	2.7E-04	9.0E-05	1.6E-03	8.8E-04	4.1E-06	1.4E-04	1.4E-04	7.4E-04
175	149	3.2E-07	1.4E+04	1.8E+04	9.9E+02	9.7	9.4E+05	2.7E-01	4.0E-02	3.9E-02	1.7E-02	5.7E-03	1.2E-03	3.5E-04	1.3E-03	6.2E-03
				1.9E+04	3.5E+04		1.5E+06	7.3E-01	3.7E-01	3.7E-01	3.9E-01	3.7E-01	6.3E-02	1.3E-01	1.8E-01	3.7E-01
176	51	4.7E-08	3.6E+04	5.1E+04	5.2E+03	10.0	1.1E+06	1.0E+00	7.5E-01	7.3E-01	5.7E-01	2.9E-01	4.3E-02	1.5E-02	6.5E-02	3.0E-01
				5.6E+04	2.2E+04		1.7E+04	8.4E-05	1.3E-03	2.3E-05	2.4E-04	3.3E-04	1.9E-05	1.2E-04	1.3E-04	2.9E-04
178	36	1.2E-08	5.7E+03	9.6E+03	1.3E+03	9.5	5.9E+06	7.3E-01	4.4E-01	4.3E-01	8.0E-02	4.8E-02	9.1E-03	4.8E-03	3.0E-02	4.9E-02
				1.1E+04	3.4E+04		1.4E+06	2.5E-01	4.3E-02	2.6E-02	3.7E-01	3.5E-01	5.9E-02	9.2E-02	1.6E-01	3.4E-01
179	17	2.6E-08	3.6E+04	5.1E+04	5.4E+03	10.0	1.0E+06	1.0E+00	6.4E-01	6.3E-01	5.3E-01	6.0E-01	9.9E-02	5.9E-02	3.4E-01	6.0E-01
				5.6E+04	2.2E+04		1.7E+04	0.0E+00	1.8E-03	1.0E-03	1.4E-03	2.4E-09	1.4E-09	1.8E-05	5.7E-06	3.2E-09
301	530	1.3E-04	1.4E+04	8.6E+04	1.8E+03	0.0	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				8.8E+04	3.6E+04		0.0E+00	4.4E-03	6.6E-06	1.0E-07	6.8E-08	1.6E-08	4.1E-10	1.7E-09	2.1E-09	1.4E-08
302	607	1.3E-04	1.4E+04	8.6E+04	1.8E+03	0.0	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				8.8E+04	3.6E+04		0.0E+00	4.2E-03	7.2E-05	2.8E-07	1.8E-07	6.1E-08	2.5E-09	8.5E-09	1.0E-08	5.4E-08
303	173	6.2E-06	1.5E+04	8.6E+04	1.9E+03	0.2	1.7E+04	1.3E-02	4.9E-06	3.4E-06	7.4E-07	1.1E-08	6.7E-09	2.9E-10	6.1E-10	1.7E-08
				8.7E+04	3.6E+04		7.0E+03	6.9E-03	3.2E-04	7.8E-07	4.6E-07	2.0E-07	9.6E-09	3.5E-08	4.0E-08	1.7E-07

6. D. I. Chanin, J. L. Sprung, and L. T. Ritchie, "MELCOR Accident Consequence Code System (MACCS)," NUREG/CR-4991, SAND86-1562, Sandia National Laboratories, February 1990.

7. R.L. Iman, J.C. Helton, and J.D. Johnson, "A User's Guide for PARTITION: A Program for Defining the Source Term/Consequence Analysis Interfaces in the NUREG-1150 Probabilistic Risk Assessments," NUREG/CR-5253, SAND88-2940, May 1990.

4. CONSEQUENCE ANALYSIS

Offsite consequences were calculated with MACCS[1,2,3] for each of the source term groups defined in the partitioning process. This code has been in use for some time and will not be described in any detail. Although the variables thought to be the largest contributors to the uncertainty in risk were sampled from distributions in the accident frequency analysis, the accident progression analysis, and the source term analysis, there was no analogous treatment of uncertainties in the consequence analysis. Variability in the weather was fully accounted for, but the uncertainty in other parameters such as the dry deposition velocity or the evacuation rate was not considered.

The MACCS model underwent several updates during the course of the analysis. All inconsistencies identified in the calculations were resolved, such as weather sampling, or bypassed, such as the multiple source term option, with the exception of the calculation of indirect dose through food ingestion. Revision of the population dose from food ingestion and the corresponding revision of cancer estimates in NUREG-1150 will be undertaken later.[4] A post-processor code developed by SNL and modified by INEL to accommodate the specific Zion results was used to extract the CCDF data.

4.1 Description of the Consequence Analysis

Offsite consequences were calculated with MACCS for each of the source term ground defined in the partitioning process. MACCS tracks the dispersion of the radioactive material in the atmosphere from the plant and computes its deposition on the ground. MACCS then calculates the effects of this radioactivity on the population and the environment. Doses and the ensuing health effects from 60 radionuclides are computed for the following pathways: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, inhalation of resuspended ground contamination, ingestion of contaminated water and ingestion of contaminated food.

MACCS treats atmospheric dispersion by the use of multiple, straight-line Gaussian plumes. Each plume can have a different direction, duration, and initial radionuclide concentration. Cross-wind dispersion is treated by a multi-step function. Dry and wet deposition are treated as independent processes. The weather variability is treated by means of a stratified sampling process.

For early exposure, the following pathways are considered: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, and inhalation of resuspended ground contamination. Skin deposition and inhalation of resuspended ground contamination have generally not been considered in previous consequence models. For the long-term exposure, MACCS considers the following four pathways: groundshine, inhalation of resuspended ground contamination, ingestion of contaminated water and ingestion of contaminated food. The direct exposure pathways groundshine, and inhalation of resuspended ground contamination, produce doses in the population living in the area surrounding the plant. The indirect exposure pathways, ingestion of contaminated water and food, produce doses in those who ingest food or water emanating from the area around the accident site. The contamination of water bodies is estimated for the washoff of land-deposited material as well as direct deposition. The food pathway model includes direct deposition onto crop and uptake from the soil.

Both short-term and long-term mitigative measures are modeled in MACCS. Short-term actions include evacuation, sheltering and emergency relocation out of the emergency planning zone (EPZ). Long-term actions include later relocation and restrictions on land use and crop disposition. Relocation and land decontamination, interdiction, and condemnation are based on projected long-term doses from groundshine and inhalation of resuspended radioactivity. The disposal of agricultural products is based on the yearly doses induced by consumption of these products, and the removal of farmland from crop production is based on ground contamination criteria.

The health effects models link the dose received by an organ to predicted morbidity or mortality. The models used in MACCS calculate both short-term and long-term effects for a number of organs.

The MACCS consequence model calculates a large number of different consequence measures. Results for the consequence measures described in Table 4.1-1 are given in this report. For the analyses performed for NUREG-1150, 99.5% of the population evacuates and 0.5% of the population does not evacuate and continues normal activity. Details of the methods used to incorporate the consequence results for the source term groups into the integrated risk analysis are given in Volume 1 of this report.

Table 4.1-1

Definition of Consequence Analysis Results

Variable	Definition
Early fatalities	Number of fatalities occurring within 1 year of the accident (early exposure).
Early injuries	Number of individuals experience prodromal vomiting from early exposure.
Total latent cancer fatalities (within 50 miles)	Number of latent cancer fatalities due to both initial and long term/chronic exposure.
Total latent cancer fatalities (entire region)	Number of latent cancer fatalities due to both initial and long term/chronic exposure.
Population dose (within 50 miles)	Population dose, expressed in effective dose equivalents for wholebody exposure (rem), due to early and chronic exposure pathways within 50 mi of the reactor. Due to the nature of the chronic pathways models, the actual exposure due to food and water consumption may take place beyond 50 mi from the reactor.

Table 4.1-1 (Continued)

Variable	Definition
Population dose (entire region)	Population dose, expressed in effective dose equivalents for wholebody exposure (rem), due to early and chronic exposure pathways for the entire region
Total cost (within 50 mile)	Cost of emergency response actions and long-term protective actions
Total cost (entire region)	Cost of emergency response actions and long-term protective action
Individual early fatality risk	Probability that an individual 1 mi from the exclusion zone will be an early fatality. For a given weather sequence, this result is obtained by dividing the region around the reactor into 48 uniform (i.e., 7.5 degree) sectors and then calculating the probability that an individual in each of these sectors 1 mi from the site boundary will be an early fatality. The reported risk value is the arithmetic mean of the 48 individual probabilities. In the calculation, the wind is assumed to blow in the direction of only 1 of the 48 sectors, with dispersion possibly spreading the plume into other sectors.
Individual latent cancer risk within 10 miles	Cancer fatality risk within 10 mi of the plant (i.e., Σ (cf/pop)p, where cf is the number of cancer fatalities due to direct exposure in the resident population, pop is the population size, p is the weather condition probability, and the summation is over all weather conditions).
Probability of a Large Release	A large release has not yet been defined by the NRC staff. The probability of one or more early fatalities within 1 mi of the exclusion boundary has been used as a surrogate for a large release for the purposes of the current analysis (i.e., Σ (ef/pop)p, where ef is the number of early fatalities, pop is the population size, p is the weather condition probability, and the summation is over all weather conditions).

4.2 MACCS Input for Zion

The values of most MACCS input parameters (e.g., aerosol dry deposition velocity, health effects model parameter values, food pathway transfer factors) do not depend on site characteristics. For those parameters that do depend on site characteristics (e.g., evacuation speed, shielding factors, farmland usage), the methods used to calculate the parameters are essentially the same for all sites. Because the methods used to develop input parameter values for the MACCS NUREG-1150 analyses and the parameter values developed using those methods are documented in Volume 2, Part 7 of this report, only a small portion of the MACCS input is presented here.

Table 4.2-1 lists the MACCS input parameters that have strong site dependencies and presents the values of these parameters used in the MACCS calculations for the Zion site. The evacuation delay period begins when general emergency conditions occur and ends when the general public starts to evacuate; non-farm wealth includes personal, business, and public property; and the farmland fractions do not add to one because not all farmland is under cultivation.

Table 4.2-1

Site Specific Input Data for Zion MACCS Calculations

Parameter	
Reactor Power Level (MWt)	3250
Containment Height (m)	50
Containment Width (m)	40
Exclusion Zone Distance (km)	0.4
Evacuation Delay (h)	2.3
Evacuation Speed (m/s)	1.1
Farmland Fractions by Crop Categories	
Pasture	0.045
Stored Forage	0.11
Grains	0.26
Green Leafy Vegetables	0.0004
Legumes and Seeds	0.13
Roots and Tubers	0.002
Other Food Crops	0.001
Non-Farm Wealth ($/person)	76,000
Farm Wealth	
Value ($/hectare)	2897
Fraction in Improvements	0.49

In addition to the site specific data presented in Table 4.2-1, the Zion MACCS calculations used one year of meteorological data from the Zion site and regional population data developed from the 1980 census tapes. The following table gives the population within certain distances of the plant as summarized from the MACCS demographic input.

Distance from Plant (km)	(miles)	Population
1.6	1.0	3,090
4.8	3.0	22,268
16.1	10.0	246,871
48.3	30.0	1,451,585
160.9	100.0	11,908,546
563.3	350.0	48,772,266
1609.3	1000.0	185,744,925

Table 4.2-2 lists the shielding factors for the Zion consequence analysis. The MACCS code considers three different portions of the population or cohorts during the emergency phase of an accident. The appropriate shielding factors are applied according to the response of the people to the declared emergency. The "evacuate" and "take shelter" shielding factors apply only during the emergency phase of the consequence calculation. The normal activity shielding factors apply to all those who are not actively evacuating or taking shelter. Thus, the normal activity shielding factors apply to individuals before they begin evacuating or taking shelter, to individuals who choose not to evacuate or take shelter, and to everyone outside the EPZ. Furthermore, the normal activity shielding factors are used for all exposure calculations after the emergency phase of the accident, that is, for the chronic exposure computations.

For accidents initiated by internal events the three cohorts treated by MACCS are: (1) those who evacuate; (2) those who continue normal activities; and (3) those who take shelter. Exposure to each cohort is calculated using the shielding factors shown in Table 4.2-2. The risk results are based on the judgment[5] that 99.5% of the population in the emergency response zone would evacuate and the other 0.5% would continue normal activities.

Table 4.2-2

Shielding Factors Used for Zion MACCS Calculations

	Population Response		
Radiation Pathway	Evacuate	Normal Activity	Take Shelter
Cloudshine	1.0	0.75	0.50
Groundshine	0.5	0.33	0.10
Inhalation	1.0	0.41	0.33
Skin	1.0	0.41	0.33

4.3 Results of MACCS Consequence Calculations

The results given in this section are conditional on the occurrence of a release. That is, given that a release takes place, with release fractions and other characteristics as defined by one of the source term groups, then the consequences reported in this section are calculated. The tables and figures in this section contain no information about the frequency with which these consequences may be expected. Information about the frequencies of consequences of various magnitudes is contained in the risk results (Chapter 5).

4.3.1 Results of Internal Initiators

The integration of the NUREG-1150 probabilistic risk assessments uses the results of the MACCS consequence calculations. A single mean (over weather variation) result is reported for each consequence measure. This produces a nSTG x nC matrix of mean consequence measures, where nSTG is the number of source term groups and nC is the number of consequence measures under consideration. For internal initiators at Zion, nSTG = 36 and nC = 11. The resultant 36 x 11 matrix of mean consequence measures is shown in Table 4.3-1 for "base case" emergency response measures (i.e., that 0.5% of the population would not participate in an emergency). The source terms that give rise to these mean consequence measures are given in Table 3.4-5. The mean consequence measures in Table 4.3-1 are used in the calculation of the mean risk results for internal initiators at Zion. An early fatality consequence value less than 1.0 may be interpreted as the probability of obtaining one fatality. The population dose is the effective dose equivalent to the whole body for the population in the region indicated.

Early health effects are sensitive not only to the magnitude of the release but also to the warning interval (time from warning to the first plume release). For example, results for Source Terms number 178 and 179 can be compared. They had been identified as having approximately the same radiological potential by the PARTITION code, but early fatalities for the first have been calculated to be 420, for the second 34.9. This difference is explained because the first source term represents mostly early Containment Failures (refer to Table 3.4-4), with a warning interval of about 1 h (refer to Table 3.4-5), while the second is a SGTR with a secondary side stuck-open relief valve (refer again to Table 3.4-4), with a warning interval of about 4 h (also given in Table 3.4-5). The calculation for Source Term 179 was repeated with a warning interval of 1 h, resulting in about 650 early fatalities.

Latent health effects present a more complicated picture, as shown in the breakdown of percent of total exposure shown in Tables 4.3-2 and 4.3.3. The contribution from early exposure ranges from 4.5 to 78.1%, but is not dominantly dependent on source term magnitude or warning interval. This, however, can be partly explained by the decrease of ingestion exposure contribution from milk and crops with increasing releases, where the water ingestion becomes increasingly dominant, reaching 100% in the case of the largest source terms. In these tables, the contribution to exposure from decontamination activities is not included.

Calculations were performed for four emergency responses to show the impact on mean consequences:

- Total evacuation to 16 km (10 mi) (refer to Table 4.3-4)

- Normal or Hot-Spot relocation only. A normal relocation at 1 day takes place if the projected dose to the Whole Body in 1 wk will exceed 25 rem. Hot-Spot relocation at 1/2 day takes place if the projected dose to the Whole Body in 1 week will exceed 50 rem. The time reference is from plume arrival (refer to Table 4.3-5)

- Sheltering to 16 km (10 mi). Shelter duration was prescribed by the NRC to be based on the passage of the second release plume, to avoid exiting under a plume. For Zion this ranged from approximately 14 h to 24 h from the arrival of the first plume (refer to Table 4.3-6)

- Evacuation within 8 km (5 mi) and sheltering between 8 and 16 km (5 and 10 mi). This was only performed for the Zion analysis (refer to Table 4.3-7)

Beyond 16 km (10 mi) in all cases normal and Hot-Spot relocations were applied. These calculations provide an indication of the sensitivity of the risk estimates to various emergency offsite response assumptions. The number of people participating in an evacuation is uncertain. For the current analyses it was determined[5] that 0.5% of the population would not participate in an emergency. Thus the base case is defined as a combination of the first two sensitivity cases, with 99.5% of the people participating in the evacuation and 0.5% continuing normal activities. The CCDFs were accordingly combined.

4.3.2 Sensitivity Analysis Results

Two additional sensitivity studies were performed for the Zion plant:

- Evacuation within 8 km (5 mi) and sheltering between 8 and 16 km (5 and 10 mi) was repeated, with the dose criterion for the long term phase relocation (Chronic) changed from 4 rem in 5 yr to 25 rem in 30 yr (refer to Tables 4.3-8 to 4.3-10)

- Dose versus distance was calculated for four source terms judged by BNL to be representative of Early High, Early Low, Late High, and Late Low releases. These source terms are respectively number 178, 1, 105 and 33 in Table 3.4-5. Six responses were considered:

  Evacuation 1 h prior to first release.
  Evacuation at release.
  Evacuation 1 h after first release.
  Continue normal activity.
  Shelter in basement.
  Shelter in large buildings.

The results for these calculations are shown in Tables 4.3-11 to 4.3-14. For normal activity and sheltering, a 6 h exposure after arrival of the first plume was assumed, after which no further dose was accumulated. For the evacuation cases, all people within 16 km (10 mi) evacuated to 32 km (20 mi) at 1.1 m/s (3.6 ft/s), after which no further dose was accumulated.

Overall results are not very sensitive to evacuation strategies, especially in the case of early containment failure releases, in which people travel with the plume in case of evacuation and may be exposed to larger doses if stationary or sheltered. Evacuation is more effective than relocation alone or sheltering in reducing early consequences in the case of late SGTR releases with secondary side stuck-open valve, such as Source Term 105.

Table 4.3-8 shows the same overall (early and chronic combined) results for the sensitivity to the dose criterion for long term phase relocation. Since early contributions are identical to the corresponding case (evacuation to 8 km (5 mi), shelter to 16 km (10 mi), the differences are due to the chronic models. Tables 4.3-9 and 4.3-10 show late exposure consequences alone for the base case and this sensitivity, respectively. The increase in the dose criterion increases the doses by at most a factor of two, and sometimes as little as 10%, especially in the low releases.

Tables 4.3-11 through 4.3-14 show the results of the dose versus distance calculations. In all cases, the two plume segments were required to move in the same direction, so that the code could calculate centerline dose. This increases the centerline dose estimates. The tables show the probabilities (expressed as percentages) that either a 200 rem or a 50 rem acute Red Marrow dose will be exceeded at several distances from the reactor as a function of the four source terms and the six emergency responses. Evacuation at 1 h before release was only calculated for the Early High Source Term. An examination of the tables shows that for the other three cases this response would lead to a probability of zero, except possibly for 50 rem at 1.6 km (2 mi) for the Early Low Source Term. For the High Source Terms the mean Red Marrow dose at several distances is given, whereas for the Low Source Terms the probability of exceeding 5 rem is provided. For the Early High Source Term, the probability of exceeding 200 rem dose is significant for all responses except evacuation 1 h before release. Beyond 8 km (5 mi) sheltering and evacuation at release become effective, and at 16 km (10 mi) all responses are effective. For the other source terms, all responses are effective in preventing 200 rem doses. For the Early High Source Term all responses (with the exception of sheltering) continue to have a probability of a 50 rem to a distance of about 16 km (10 mi). For the other three releases beyond 4.8 km (3 mi) all responses prevent 50 rem acute Red Marrow doses.

4.4 References

1. D. I. Chanin, J. L. Sprung, L. T. Ritchie, and H.-N. Jow, "MELCOR Accident Consequence Code System (MACCS): User's Guide," NUREG/CR-4691, SAND86-1562, Vol. 1, Sandia National Laboratories, February 1990.

2. H.-N. Jow, J. L. Sprung, J. A. Rollstin, and D. I. Chanin, "MELCOR Accident Consequence Code System (MACCS): Mode Description," NUREG/CR-4691, SAND86-1562, Vol. 2, Sandia National Laboratories, February 1990.

3. J. A. Rollstin, D. I. Chanin, and H.-N. Jow, "MELCOR Accident Consequence Code System (MACCS): Programmer's Reference Manual," NUREG/CR-4691, SAND86-1562, Vol. 3, Sandia National Laboratories, February 1990.

4. USNRC Memorandum for Mark A. Cunningham from Sarbes Acharya, May 26, 1989.

5. USNRC Memorandum from Sarbes Acharya to Joseph Murphy, April 4, 1989, "Assumption on Non-Evacuating Population Fraction for NUREG-1150 Offsite Consequece/Risk Analysis."

Table 4.3-1 Consequences by Release Group
Base Case

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
1	2.29E-01	8.66E+00	4.86E+01	3.45E+01	2.39E-05	8.72E-05	9.90E-03	1.69E+05	2.53E+05	1.90E+07	1.90E+07
2	0.00E+00	0.00E+00	1.39E+02	6.31E+01	0.00E+00	4.70E-05	0.00E+00	3.76E+05	8.07E+05	1.20E+08	1.20E+08
31	5.18E-01	1.30E+01	5.53E+01	3.99E+01	4.83E-05	1.03E-04	4.20E-02	1.93E+05	2.89E+05	2.20E+07	2.00E+07
33	0.00E+00	5.00E-05	1.10E+01	5.51E+00	0.00E+00	5.74E-06	0.00E+00	3.66E+04	7.98E+04	1.10E+07	8.30E+06
61	4.19E-01	1.16E+01	8.33E+01	5.31E+01	4.05E-05	1.14E-04	4.10E-02	2.76E+05	4.56E+05	3.00E+07	2.70E+07
64	3.39E-01	1.07E+01	2.06E+02	1.07E+02	3.36E-05	1.54E-04	3.90E-02	6.06E+05	1.17E+06	1.60E+08	1.50E+08
65	0.00E+00	1.00E-03	5.44E+02	2.33E+02	2.60E-06	8.55E-05	5.00E-06	1.40E+06	3.15E+06	7.60E+08	5.70E+08
66	0.00E+00	1.50E-04	1.03E+02	4.80E+01	0.00E+00	5.04E-05	0.00E+00	2.90E+05	6.20E+05	8.90E+07	8.20E+07
67	2.50E-01	1.19E+01	4.75E+02	2.29E+02	2.66E-05	2.12E-04	4.00E-02	1.35E+06	2.77E+06	6.30E+08	4.90E+08
68	4.00E-04	9.00E-03	1.10E+03	4.51E+02	7.55E-08	1.31E-04	9.50E-05	2.72E+06	6.39E+06	2.30E+09	1.50E+09
69	2.00E-04	9.50E-03	2.48E+02	1.12E+02	4.01E-08	7.31E-05	4.50E-05	6.93E+05	1.50E+06	2.80E+08	2.30E+08
70	2.82E-01	1.27E+01	7.64E+02	3.59E+02	3.38E-05	2.38E-04	4.70E-02	2.18E+06	4.57E+06	2.50E+09	8.70E+08
71	4.05E-03	3.30E-02	1.57E+03	6.49E+02	8.05E-07	1.45E-04	3.75E-04	3.91E+06	9.18E+06	5.30E+09	3.30E+09
72	7.65E-03	5.05E-02	5.49E+02	2.56E+02	1.53E-06	1.12E-04	3.85E-04	1.56E+06	3.29E+06	7.60E+08	5.10E+08
100	6.08E-01	1.62E+01	1.18E+03	5.65E+02	8.58E-05	3.71E-04	4.85E-02	3.38E+06	6.97E+06	3.00E+09	1.70E+09
101	1.10E-02	6.10E-02	1.92E+03	7.31E+02	2.21E-06	1.44E-04	6.20E-04	4.42E+06	1.13E+07	7.70E+09	4.90E+09
103	3.76E-01	1.44E+01	2.09E+03	9.55E+02	5.61E-05	4.48E-04	4.90E-02	5.80E+06	1.25E+07	9.50E+09	4.60E+09
104	4.05E-02	1.49E-01	2.60E+03	8.96E+02	7.90E-06	1.44E-04	1.16E-03	5.46E+06	1.54E+07	1.40E+10	7.60E+09
105	6.10E-02	2.10E-01	3.28E+03	1.08E+03	1.22E-05	1.56E-04	1.28E-03	6.49E+06	1.92E+07	1.70E+10	8.70E+09
106	3.27E+00	3.14E+01	3.48E+03	1.51E+03	5.26E-04	1.27E-03	9.92E-02	8.47E+06	2.01E+07	2.00E+10	8.50E+09
107	8.15E-02	2.55E-01	3.30E+03	1.03E+03	1.53E-05	1.36E-04	1.37E-03	6.27E+06	1.95E+07	1.90E+10	1.00E+10
136	1.00E+00	1.95E+01	3.93E+03	1.49E+03	1.47E-04	6.21E-04	9.14E-02	9.00E+06	2.32E+07	2.80E+10	1.10E+10
137	1.52E-01	4.23E-01	3.96E+03	1.17E+03	2.60E-05	1.74E-04	1.53E-03	7.19E+06	2.36E+07	2.60E+10	1.20E+10
139	3.90E+01	8.36E+01	5.90E+03	2.87E+03	3.57E-03	4.96E-03	2.56E-01	1.34E+07	3.20E+07	4.70E+10	1.60E+10
140	3.61E-01	9.05E-01	5.41E+03	1.47E+03	4.79E-05	2.62E-04	1.76E-03	8.98E+06	3.23E+07	4.30E+10	1.70E+10
142	3.37E+00	1.05E+01	6.83E+03	2.52E+03	6.22E-04	1.64E-03	9.47E-02	1.41E+07	4.10E+07	7.10E+10	1.90E+10
143	5.75E-01	1.62E+00	6.69E+03	1.78E+03	6.15E-05	2.85E-04	1.84E-03	1.09E+07	3.97E+07	5.10E+10	2.00E+10

Table 4.3-1 (Continued)

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
172	1.34E+02	3.90E+02	8.29E+03	3.96E+03	6.09E-03	7.19E-03	2.83E-01	1.90E+07	4.59E+07	4.50E+10	1.50E+10
173	1.18E+00	5.85E+00	8.08E+03	3.00E+03	8.05E-05	1.03E-03	2.05E-03	1.55E+07	4.67E+07	7.70E+10	2.30E+10
175	1.43E+02	9.31E+01	1.46E+04	8.13E+03	1.48E-02	8.37E-03	3.69E-01	3.24E+07	7.43E+07	1.10E+11	2.70E+10
176	1.99E+00	1.85E+01	1.00E+04	4.22E+03	1.04E-04	1.77E-03	1.24E-02	1.94E+07	5.53E+07	9.60E+10	2.60E+10
178	4.20E+02	6.51E+02	1.71E+04	1.03E+04	1.90E-02	1.56E-02	4.36E-01	4.11E+07	8.56E+07	1.30E+11	3.10E+10
179	3.49E+01	6.59E+01	2.00E+04	1.25E+04	1.96E-03	6.25E-03	1.18E-01	3.93E+07	8.44E+07	1.20E+11	2.80E+10
301	0.00E+00	0.00E+00	3.01E-02	2.00E-02	0.00E+00	1.78E-08	0.00E+00	1.27E+02	2.43E+02	3.00E+06	3.00E+06
302	0.00E+00	0.00E+00	1.50E-01	8.02E-02	0.00E+00	7.52E-08	0.00E+00	6.18E+02	1.26E+03	3.10E+06	3.10E+06
303	0.00E+00	0.00E+00	9.51E-01	5.11E-01	0.00E+00	5.96E-07	0.00E+00	3.59E+03	7.29E+03	3.40E+06	3.40E+06

Table 4.3-2 Relative Long-Term Population Dose
Direct and Ingestion Exposure Pathways (0-50 Miles)

		% of Total Exposure		% of Chronic Exposure		% of Ingestion Exposure				
Source Term	Total Exposure	Early Exposure	Chronic Exposure	Direct Exposure	Ingestion Exposure	Milk Direct	Crop Direct	Milk Root	Crop Root	Water Ingestio
1	1.69E+05	50.75	49.25	86.64	13.24	10.36	65.09	3.89	18.18	2.06
2	3.76E+05	4.54	95.46	93.59	6.04	4.29	75.58	4.65	9.22	6.68
31	1.93E+05	43.19	56.81	87.76	11.56	5.37	78.24	4.27	9.53	1.61
33	3.66E+04	42.33	57.67	78.67	20.47	9.17	71.99	12.55	5.53	0.86
61	2.76E+05	46.30	53.70	88.64	11.59	5.53	79.74	5.54	6.67	2.57
64	6.06E+05	24.98	75.02	93.85	5.74	3.60	68.72	5.47	15.47	6.63
65	1.40E+06	5.41	94.59	97.73	2.01	1.40	48.68	6.08	14.98	28.83
66	2.90E+05	9.70	90.30	91.22	8.55	2.92	77.68	6.07	9.20	4.13
67	1.35E+06	16.26	83.74	96.72	2.61	2.38	52.52	5.66	21.10	18.27
68	2.72E+06	6.87	93.13	97.63	1.21	1.43	17.39	4.46	13.52	63.19
69	6.93E+05	19.46	80.54	95.15	4.15	1.16	59.31	8.10	21.04	10.48
70	2.18E+06	18.20	81.80	97.58	1.30	2.35	29.77	3.71	21.17	42.99
71	3.91E+06	8.55	91.45	97.21	1.35	0.81	6.35	3.61	9.07	80.08
72	1.56E+06	19.78	80.22	97.60	2.30	0.83	50.17	5.96	20.56	22.65
100	3.38E+06	16.38	83.62	97.48	1.38	0.98	34.62	3.01	12.96	48.52
101	4.42E+06	10.76	89.24	96.46	1.56	0.48	3.83	2.10	5.95	87.70
103	5.80E+06	15.34	84.66	97.34	1.23	0.75	17.80	1.56	6.87	72.91
104	5.46E+06	15.20	84.80	95.03	2.22	0.41	3.87	0.76	3.06	91.55
105	6.49E+06	13.08	86.92	95.92	1.99	0.23	4.21	0.95	3.41	91.07
106	8.47E+06	24.64	75.36	96.33	1.47	0.37	3.34	0.55	3.28	92.42
107	6.27E+06	18.53	81.47	93.15	2.95	0.02	1.91	0.41	1.34	96.69
136	9.00E+06	23.57	76.43	95.49	1.86	0.30	2.24	0.42	2.16	95.31
137	7.19E+06	23.27	76.73	91.12	3.80	0.30	2.19	0.20	0.79	96.19
139	1.34E+07	49.87	50.13	91.64	2.67	0.11	0.30	0.11	0.87	98.88
140	8.98E+06	31.90	68.10	86.60	5.88	0.09	0.18	0.07	0.62	99.17
142	1.41E+07	43.92	56.08	86.42	4.34	0.01	0.48	0.10	0.37	98.83
143	1.09E+07	33.48	66.52	84.69	7.23	0.03	0.08	0.06	0.46	99.24
172	1.90E+07	64.71	35.29	84.92	5.48	0.06	2.45	0.41	0.36	96.52
173	1.55E+07	51.23	48.77	78.81	8.61	0.02	0.05	0.03	0.33	99.54
175	3.24E+07	70.58	29.42	72.51	4.93	0.01	0.02	0.02	0.12	99.79
176	1.94E+07	58.26	41.74	74.14	10.55	0.03	0.03	0.02	0.26	99.65
178	4.11E+07	76.88	23.12	69.08	6.09	0.00	0.00	0.00	0.02	100.00
179	3.93E+07	78.12	21.88	65.81	10.87	0.07	1.32	0.06	0.11	98.50
301	1.27E+02	26.72	73.28	48.50	51.50	48.33	41.04	5.23	5.10	0.19
302	6.18E+02	32.11	67.89	44.86	55.14	57.63	30.13	8.22	4.07	0.11
303	3.59E+03	25.09	74.91	64.93	34.72	38.30	49.40	7.63	4.53	0.41

Table 4.3-3 Relative Long-Term Population Dose
Direct and Ingestion Exposure Pathways (Entire Region)

		% of Total Exposure		% of Chronic Exposure		% of Ingestion Exposure				
Source Term	Total Exposure	Early Exposure	Chronic Exposure	Direct Exposure	Ingestion Exposure	Milk Direct	Crop Direct	Milk Root	Crop Root	Water Ingestion
1	2.53E+05	38.27	61.73	56.65	43.11	21.47	58.36	3.46	16.43	0.46
2	8.07E+05	3.21	96.79	58.26	41.61	17.45	74.15	2.66	5.05	0.59
31	2.89E+05	30.82	69.18	57.44	42.56	21.46	66.75	3.36	7.54	0.48
33	7.98E+04	35.58	64.42	40.66	59.34	51.48	37.38	7.70	3.14	0.16
61	4.56E+05	33.21	66.79	56.60	43.40	25.13	65.65	3.73	4.70	0.45
64	1.17E+06	15.44	84.56	58.70	41.08	17.40	70.24	3.32	8.42	0.58
65	3.15E+06	3.41	96.59	62.83	37.17	11.15	77.17	3.29	7.12	0.90
66	6.20E+05	7.71	92.29	57.22	42.61	17.60	72.31	3.90	5.50	0.52
67	2.77E+06	9.63	90.37	61.76	37.87	12.91	71.75	3.79	10.49	0.77
68	6.39E+06	4.08	95.92	65.74	33.77	8.55	77.29	3.68	8.94	1.25
69	1.50E+06	13.43	86.57	58.08	41.77	12.82	72.01	5.05	9.48	0.61
70	4.57E+06	11.18	88.82	63.44	36.02	12.39	57.31	4.34	25.15	0.99
71	9.18E+06	5.07	94.93	75.00	24.43	9.81	69.95	5.63	12.25	2.40
72	3.29E+06	13.80	86.20	63.96	35.69	9.58	72.38	5.24	11.98	0.88
100	6.97E+06	9.76	90.24	67.96	31.62	8.20	73.25	4.19	13.11	1.26
101	1.13E+07	5.88	94.12	75.09	24.53	8.50	70.00	5.69	13.08	2.77
103	1.25E+07	9.03	90.97	73.25	26.27	9.34	66.16	4.68	17.95	1.97
104	1.54E+07	7.56	92.44	75.35	23.66	9.52	65.77	5.71	15.15	3.75
105	1.92E+07	6.56	93.44	72.22	26.94	9.42	67.84	5.09	14.89	2.87
106	2.01E+07	13.35	86.65	75.58	23.55	10.86	58.27	4.69	23.41	2.96
107	1.95E+07	8.45	91.55	77.09	21.62	10.18	62.02	6.38	16.43	5.01
136	2.32E+07	12.32	87.68	75.86	23.30	11.59	57.08	4.78	23.04	3.59
137	2.36E+07	10.18	89.82	77.83	20.52	10.30	60.46	5.79	17.40	6.21
139	3.20E+07	26.60	73.40	76.60	21.79	12.09	47.66	4.43	30.66	4.98
140	3.23E+07	13.30	86.70	77.14	20.89	11.74	56.58	4.70	18.63	8.21
142	4.10E+07	22.89	77.11	77.53	19.46	10.99	45.53	4.10	31.06	8.16
143	3.97E+07	13.83	86.17	75.15	22.43	12.49	58.02	5.36	15.12	9.04
172	4.59E+07	33.73	66.27	77.82	19.46	14.06	54.20	3.60	19.22	8.84
173	4.67E+07	27.23	72.77	82.06	13.94	9.47	34.18	4.60	32.91	19.07
175	7.43E+07	48.83	51.17	76.84	15.26	9.90	35.86	4.12	37.76	12.47
176	5.53E+07	33.79	66.21	82.83	12.10	8.65	25.45	4.12	34.23	27.48
178	8.56E+07	53.28	46.72	75.56	16.81	8.93	36.94	3.80	36.65	13.75
179	8.44E+07	58.77	41.23	79.31	11.90	6.26	18.70	3.60	37.68	33.82
301	2.43E+02	32.07	67.93	33.21	66.67	55.18	34.18	5.39	5.43	0.12
302	1.26E+03	24.80	75.20	23.72	76.28	69.17	20.38	6.97	3.40	0.05
303	7.29E+03	19.86	80.14	37.01	63.15	56.81	33.42	6.04	3.47	0.14

Table 4.3-4 Consequences by Release Group
Total Evacuation

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
1	2.30E-01	8.70E+00	4.87E+01	3.46E+01	2.39E-05	8.74E-05	9.90E-03	1.69E+05	2.53E+05	1.90E+07	1.90E+07
2	0.00E+00	0.00E+00	1.39E+02	6.31E+01	0.00E+00	4.70E-05	0.00E+00	3.76E+05	8.07E+05	1.20E+08	1.20E+08
31	5.20E-01	1.30E+01	5.54E+01	4.00E+01	4.84E-05	1.03E-04	4.20E-02	1.93E+05	2.89E+05	2.20E+07	2.00E+07
33	0.00E+00	0.00E+00	1.10E+01	5.50E+00	0.00E+00	5.69E-06	0.00E+00	3.65E+04	7.97E+04	1.10E+07	8.30E+06
61	4.20E-01	1.16E+01	8.34E+01	5.32E+01	4.05E-05	1.14E-04	4.10E-02	2.76E+05	4.56E+05	3.00E+07	2.70E+07
64	3.40E-01	1.07E+01	2.06E+02	1.07E+02	3.36E-05	1.54E-04	3.90E-02	6.06E+05	1.17E+06	1.60E+08	1.50E+08
65	0.00E+00	0.00E+00	5.44E+02	2.33E+02	0.00E+00	8.53E-05	0.00E+00	1.40E+06	3.15E+06	7.60E+08	5.70E+08
66	0.00E+00	0.00E+00	1.03E+02	4.80E+01	0.00E+00	5.03E-05	0.00E+00	2.90E+05	6.20E+05	8.90E+07	8.20E+07
67	2.50E-01	1.19E+01	4.75E+02	2.29E+02	2.65E-05	2.12E-04	4.00E-02	1.35E+06	2.77E+06	6.30E+08	4.90E+08
68	0.00E+00	0.00E+00	1.10E+03	4.51E+02	0.00E+00	1.31E-04	0.00E+00	2.72E+06	6.39E+06	2.30E+09	1.50E+09
69	0.00E+00	0.00E+00	2.48E+02	1.12E+02	0.00E+00	7.27E-05	0.00E+00	6.92E+05	1.50E+06	2.80E+08	2.30E+08
70	2.80E-01	1.27E+01	7.64E+02	3.59E+02	3.34E-05	2.38E-04	4.70E-02	2.18E+06	4.57E+06	2.50E+09	8.70E+08
71	0.00E+00	0.00E+00	1.57E+03	6.49E+02	0.00E+00	1.44E-04	0.00E+00	3.91E+06	9.18E+06	5.30E+09	3.30E+09
72	0.00E+00	0.00E+00	5.49E+02	2.56E+02	0.00E+00	1.11E-04	0.00E+00	1.56E+06	3.29E+06	7.60E+08	5.10E+08
100	6.00E-01	1.62E+01	1.18E+03	5.65E+02	8.40E-05	3.71E-04	4.80E-02	3.38E+06	6.97E+06	3.00E+09	1.70E+09
101	0.00E+00	0.00E+00	1.92E+03	7.31E+02	0.00E+00	1.43E-04	0.00E+00	4.42E+06	1.13E+07	7.70E+09	4.90E+09
103	3.60E-01	1.44E+01	2.09E+03	9.55E+02	5.29E-05	4.47E-04	4.80E-02	5.80E+06	1.25E+07	9.50E+09	4.60E+09
104	0.00E+00	0.00E+00	2.60E+03	8.96E+02	0.00E+00	1.43E-04	0.00E+00	5.46E+06	1.54E+07	1.40E+10	7.60E+09
105	0.00E+00	0.00E+00	3.28E+03	1.08E+03	0.00E+00	1.54E-04	0.00E+00	6.49E+06	1.92E+07	1.70E+10	8.70E+09
106	3.20E+00	3.12E+01	3.48E+03	1.51E+03	5.12E-04	1.27E-03	9.80E-02	8.47E+06	2.01E+07	2.00E+10	8.50E+09
107	0.00E+00	0.00E+00	3.30E+03	1.03E+03	0.00E+00	1.34E-04	0.00E+00	6.27E+06	1.95E+07	1.90E+10	1.00E+10
136	9.00E-01	1.92E+01	3.93E+03	1.49E+03	1.27E-04	6.19E-04	9.00E-02	8.99E+06	2.32E+07	2.80E+10	1.10E+10
137	0.00E+00	0.00E+00	3.96E+03	1.17E+03	0.00E+00	1.71E-04	0.00E+00	7.19E+06	2.36E+07	2.60E+10	1.20E+10
139	3.89E+01	8.32E+01	5.90E+03	2.87E+03	3.54E-03	4.97E-03	2.55E-01	1.34E+07	3.20E+07	4.70E+10	1.60E+10
140	0.00E+00	0.00E+00	5.41E+03	1.47E+03	0.00E+00	2.57E-04	0.00E+00	8.97E+06	3.23E+07	4.30E+10	1.70E+10
142	2.60E+00	8.90E+00	6.83E+03	2.52E+03	5.24E-04	1.63E-03	9.30E-02	1.40E+07	4.09E+07	7.10E+10	1.90E+10
143	0.00E+00	1.50E-01	6.69E+03	1.78E+03	0.00E+00	2.79E-04	0.00E+00	1.09E+07	3.97E+07	5.10E+10	2.00E+10

Table 4.3-4 (Continued)

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
172	1.34E+02	3.90E+02	8.30E+03	3.97E+03	6.04E-03	7.21E-03	2.83E-01	1.90E+07	4.59E+07	4.50E+10	1.50E+10
173	0.00E+00	3.00E+00	8.07E+03	2.99E+03	0.00E+00	1.01E-03	0.00E+00	1.55E+07	4.67E+07	7.70E+10	2.30E+10
175	1.40E+02	8.90E+01	1.46E+04	8.12E+03	1.47E-02	8.33E-03	3.68E-01	3.24E+07	7.43E+07	1.10E+11	2.70E+10
176	6.00E-02	1.40E+01	9.99E+03	4.21E+03	1.18E-05	1.74E-03	1.00E-02	1.94E+07	5.53E+07	9.60E+10	2.60E+10
178	4.18E+02	6.49E+02	1.71E+04	1.03E+04	1.89E-02	1.56E-02	4.36E-01	4.11E+07	8.56E+07	1.30E+11	3.10E+10
179	3.01E+01	5.88E+01	2.00E+04	1.25E+04	1.85E-03	6.18E-03	1.16E-01	3.92E+07	8.43E+07	1.20E+11	2.80E+10
301	0.00E+00	0.00E+00	3.00E-02	2.00E-02	0.00E+00	1.77E-08	0.00E+00	1.27E+02	2.43E+02	3.00E+06	3.00E+06
302	0.00E+00	0.00E+00	1.50E-01	8.00E-02	0.00E+00	7.45E-08	0.00E+00	6.17E+02	1.26E+03	3.10E+06	3.10E+06
303	0.00E+00	0.00E+00	9.50E-01	5.10E-01	0.00E+00	5.93E-07	0.00E+00	3.58E+03	7.28E+03	3.40E+06	3.40E+06

Table 4.3-5 Consequences by Release Group
Relocation Only

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
1	8.00E-02	2.00E-01	3.74E+01	2.33E+01	1.55E-05	4.17E-05	9.60E-03	1.29E+05	2.12E+05	1.90E+07	1.90E+07
2	0.00E+00	0.00E+00	1.41E+02	6.47E+01	0.00E+00	5.44E-05	0.00E+00	3.88E+05	8.19E+05	1.20E+08	1.20E+08
31	1.60E-01	3.60E+00	4.21E+01	2.66E+01	3.10E-05	4.83E-05	3.30E-02	1.48E+05	2.43E+05	2.20E+07	2.00E+07
33	0.00E+00	1.00E-02	1.35E+01	7.90E+00	0.00E+00	1.53E-05	0.00E+00	5.46E+04	9.78E+04	1.10E+07	8.30E+06
61	1.50E-01	3.50E+00	7.10E+01	4.08E+01	3.09E-05	6.14E-05	3.30E-02	2.34E+05	4.14E+05	3.00E+07	2.70E+07
64	1.60E-01	3.60E+00	1.95E+02	9.65E+01	3.17E-05	1.10E-04	3.70E-02	5.72E+05	1.14E+06	1.60E+08	1.50E+08
65	0.00E+00	2.00E-01	5.51E+02	2.40E+02	5.20E-04	1.15E-04	1.00E-03	1.44E+06	3.20E+06	7.60E+08	5.70E+08
66	0.00E+00	3.00E-02	1.07E+02	5.20E+01	0.00E+00	6.67E-05	0.00E+00	3.22E+05	6.47E+05	8.90E+07	8.20E+07
67	2.30E-01	4.70E+00	4.67E+02	2.22E+02	4.57E-05	1.83E-04	4.60E-02	1.35E+06	2.76E+06	6.30E+08	4.90E+08
68	8.00E-02	1.80E+00	1.11E+03	4.68E+02	1.51E-05	1.96E-04	1.90E-02	2.82E+06	6.49E+06	2.30E+09	1.50E+09
69	4.00E-02	1.90E+00	2.66E+02	1.29E+02	8.02E-06	1.43E-04	9.00E-03	8.15E+05	1.62E+06	2.80E+08	2.30E+08
70	6.00E-01	7.20E+00	7.63E+02	3.58E+02	1.08E-04	2.34E-04	5.00E-02	2.21E+06	4.60E+06	2.50E+09	8.70E+08
71	8.10E-01	6.60E+00	1.59E+03	6.74E+02	1.61E-04	2.45E-04	7.50E-02	4.08E+06	9.35E+06	5.30E+09	3.30E+09
72	1.53E+00	1.01E+01	5.85E+02	2.92E+02	3.06E-04	2.55E-04	7.70E-02	1.78E+06	3.51E+06	7.60E+08	5.10E+08
100	2.20E+00	1.40E+01	1.17E+03	6.28E+02	4.39E-04	3.22E-04	1.47E-01	3.38E+06	6.96E+06	3.00E+09	1.70E+09
101	2.20E+00	1.22E+01	1.96E+03	7.65E+02	4.41E-04	2.83E-04	1.24E-01	4.65E+06	1.15E+07	7.70E+09	4.90E+09
103	3.50E+00	2.18E+01	2.16E+03	9.81E+02	6.85E-04	5.48E-04	2.40E-01	5.92E+06	1.27E+07	9.50E+09	4.60E+09
104	8.10E+00	2.97E+01	2.67E+03	9.51E+02	1.58E-03	3.87E-04	2.31E-01	5.83E+06	1.57E+07	1.40E+10	7.60E+09
105	1.22E+01	4.19E+01	3.36E+03	1.17E+03	2.43E-03	5.24E-04	2.55E-01	7.03E+06	1.98E+07	1.70E+10	8.70E+09
106	1.80E+01	6.70E+01	3.38E+03	1.41E+03	3.38E-03	7.23E-04	3.40E-01	8.28E+06	2.00E+07	2.00E+10	8.50E+09
107	1.63E+01	5.10E+01	3.38E+03	1.11E+03	3.05E-03	4.73E-04	2.73E-01	6.79E+06	2.01E+07	1.90E+10	1.00E+10
136	2.17E+01	8.34E+01	4.01E+03	1.57E+03	4.19E-03	9.54E-04	3.70E-01	9.25E+06	2.35E+07	2.80E+10	1.10E+10
137	3.04E+01	8.46E+01	4.09E+03	1.30E+03	5.20E-03	6.93E-04	3.06E-01	7.95E+06	2.44E+07	2.60E+10	1.20E+10
139	6.00E+01	1.65E+02	5.33E+03	2.31E+03	1.01E-02	2.67E-03	4.33E-01	1.23E+07	3.10E+07	4.70E+10	1.60E+10
140	7.22E+01	1.81E+02	5.67E+03	1.73E+03	9.57E-03	1.29E-03	3.51E-01	1.04E+07	3.37E+07	4.30E+10	1.70E+10
142	1.57E+02	3.31E+02	7.40E+03	3.08E+03	2.02E-02	3.85E-03	4.41E-01	1.65E+07	4.33E+07	7.10E+10	1.90E+10

Table 4.3-5 (Continued)

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
143	1.15E+02	2.94E+02	6.97E+03	2.06E+03	1.23E-02	1.44E-03	3.67E-01	1.27E+07	4.15E+07	5.10E+10	2.00E+10
172	1.52E+02	4.31E+02	6.64E+03	2.81E+03	1.58E-02	3.40E-03	3.59E-01	1.45E+07	3.77E+07	4.50E+10	1.50E+10
173	2.36E+02	5.72E+02	9.03E+03	3.95E+03	1.61E-02	4.90E-03	4.09E-01	1.99E+07	5.11E+07	7.70E+10	2.30E+10
175	7.05E+02	9.04E+02	1.66E+04	9.49E+03	4.09E-02	1.63E-02	5.00E-01	4.51E+07	8.70E+07	1.10E+11	2.70E+10
176	3.86E+02	9.10E+02	1.14E+04	5.59E+03	1.84E-02	7.32E-03	4.83E-01	2.62E+07	6.21E+07	9.60E+10	2.60E+10
178	7.21E+02	1.10E+03	1.76E+04	1.07E+04	4.44E-02	1.74E-02	5.22E-01	4.24E+07	9.07E+07	1.30E+11	3.10E+10
179	9.95E+02	1.48E+03	2.33E+04	1.59E+04	2.31E-02	1.98E-02	5.00E-01	6.11E+07	1.06E+08	1.20E+11	2.80E+10
301	0.00E+00	0.00E+00	4.00E-02	2.00E-02	0.00E+00	3.81E-08	0.00E+00	1.65E+02	2.81E+02	3.00E+06	3.00E+06
302	0.00E+00	0.00E+00	1.80E-01	1.10E-01	0.00E+00	2.06E-07	0.00E+00	8.93E+02	1.54E+03	3.10E+06	3.10E+06
303	0.00E+00	0.00E+00	1.11E+00	6.70E-01	0.00E+00	1.22E-06	0.00E+00	4.90E+03	8.60E+03	3.40E+06	3.40E+06

Table 4.3-6 Consequences by Release Group

Sheltering

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
1	6.00E-03	1.90E-01	3.50E+01	2.09E+01	1.28E-06	3.20E-05	9.60E-03	1.18E+05	2.01E+05	1.90E+07	1.90E+07
2	0.00E+00	0.00E+00	1.40E+02	6.36E+01	0.00E+00	4.98E-05	0.00E+00	3.81E+05	8.12E+05	1.20E+08	1.20E+08
31	2.00E-02	1.20E+00	3.90E+01	2.35E+01	3.11E-06	3.56E-05	3.30E-02	1.34E+05	2.29E+05	2.20E+07	2.00E+07
33	0.00E+00	0.00E+00	1.22E+01	6.60E+00	0.00E+00	1.01E-05	0.00E+00	4.58E+04	8.90E+04	1.10E+07	8.30E+06
61	1.00E-02	1.10E+00	6.76E+01	3.73E+01	2.64E-06	4.98E-05	3.30E-02	2.18E+05	3.98E+05	3.00E+07	2.70E+07
64	1.00E-02	1.10E+00	1.90E+02	9.17E+01	2.62E-06	9.08E-05	3.70E-02	5.48E+05	1.12E+06	1.60E+08	1.50E+08
65	0.00E+00	0.00E+00	5.46E+02	2.35E+02	0.00E+00	9.53E-05	1.00E-03	1.42E+06	3.17E+06	7.60E+08	5.70E+08
66	0.00E+00	0.00E+00	1.05E+02	4.97E+01	0.00E+00	5.74E-05	0.00E+00	3.06E+05	6.31E+05	8.90E+07	8.20E+07
67	1.00E-02	1.30E+00	4.58E+02	2.13E+02	2.96E-06	1.46E-04	4.60E-02	1.30E+06	2.71E+06	6.30E+08	4.90E+08
68	0.00E+00	1.00E-01	1.10E+03	4.58E+02	2.55E-08	1.58E-04	1.90E-02	2.76E+06	6.43E+06	2.30E+09	1.50E+09
69	0.00E+00	3.00E-01	2.56E+02	1.20E+02	7.43E-08	1.04E-04	9.00E-03	7.55E+05	1.56E+06	2.80E+08	2.30E+08
70	3.00E-02	1.90E+00	7.48E+02	3.43E+02	6.44E-06	1.75E-04	5.00E-02	2.12E+06	4.52E+06	2.50E+09	8.70E+08
71	1.00E-02	7.90E-01	1.58E+03	6.61E+02	2.35E-06	1.91E-04	0.00E+00	4.00E+06	9.27E+06	5.30E+09	3.30E+09
72	1.00E-01	2.70E+00	5.67E+02	2.74E+02	2.02E-05	1.84E-04	0.00E+00	1.68E+06	3.41E+06	7.60E+08	5.10E+08
100	2.10E-01	4.00E+00	1.15E+03	6.07E+02	4.27E-05	2.38E-04	4.50E-02	3.26E+06	6.85E+06	3.00E+09	1.70E+09
101	9.00E-02	1.80E+00	1.94E+03	7.48E+02	1.75E-05	2.14E-04	0.00E+00	4.55E+06	1.14E+07	7.70E+09	4.90E+09
103	3.40E-01	5.60E+00	2.13E+03	9.47E+02	6.70E-05	4.13E-04	4.80E-02	5.76E+06	1.25E+07	9.50E+09	4.60E+09
104	8.00E-01	6.30E+00	2.64E+03	9.26E+02	1.67E-04	2.85E-04	2.31E-01	5.68E+06	1.56E+07	1.40E+10	7.60E+09
105	3.30E+00	1.50E+01	3.33E+03	1.14E+03	6.50E-04	3.91E-04	2.55E-01	6.83E+06	1.96E+07	1.70E+10	8.70E+10
106	3.40E+00	1.95E+01	3.32E+03	1.35E+03	6.81E-04	5.11E-04	9.80E-02	8.02E+06	1.97E+07	2.00E+10	8.50E+09
107	2.50E+00	1.25E+01	3.35E+03	1.08E+03	4.97E-04	3.40E-04	2.70E-01	6.59E+06	1.99E+07	1.90E+10	1.00E+10
136	4.40E+00	2.56E+01	3.95E+03	1.51E+03	8.80E-04	7.04E-04	1.00E-01	8.94E+06	2.32E+07	2.80E+10	1.10E+10
137	6.40E+00	2.35E+01	4.05E+03	1.26E+03	1.26E-03	5.05E-04	2.00E-01	7.66E+06	2.41E+07	2.60E+10	1.20E+10
139	2.55E+01	6.12E+01	5.19E+03	2.18E+03	4.68E-03	2.13E-03	2.00E-01	1.17E+07	3.04E+07	4.70E+10	1.60E+10
140	2.11E+01	5.60E+01	5.59E+03	1.65E+03	3.82E-03	9.76E-04	2.00E-01	9.86E+06	3.32E+07	4.30E+10	1.70E+10
142	7.92E+01	1.46E+02	7.24E+03	2.93E+03	1.27E-02	3.29E-03	3.00E-01	1.56E+07	4.24E+07	7.10E+10	1.90E+10
143	3.38E+01	9.03E+01	6.88E+03	1.97E+03	5.70E-03	1.05E-03	2.00E-01	1.20E+07	4.08E+07	5.10E+10	2.00E+10

Table 4.3-6 (Continued)

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
172	5.39E+01	1.41E+02	6.46E+03	2.63E+03	8.59E-03	2.83E-03	1.50E-01	1.36E+07	3.68E+07	4.50E+10	1.50E+10
173	1.03E+02	1.83E+02	8.83E+03	3.75E+03	1.19E-02	4.07E-03	3.00E-01	1.85E+07	4.97E+07	7.70E+10	2.30E+10
175	5.00E+02	4.44E+02	1.61E+04	8.97E+03	3.61E-02	1.42E-02	4.00E-01	4.12E+07	8.31E+07	1.10E+11	2.70E+10
176	1.74E+02	2.97E+02	1.11E+04	5.34E+03	1.47E-02	6.30E-03	3.00E-01	2.41E+07	6.01E+07	9.60E+10	2.60E+10
178	4.78E+02	4.88E+02	1.70E+04	1.01E+04	3.76E-02	1.49E-02	4.00E-01	4.24E+07	8.68E+07	1.30E+11	3.10E+10
179	6.21E+02	5.61E+02	2.29E+04	1.54E+04	2.06E-02	1.80E-02	4.00E-01	5.54E+07	1.01E+08	1.20E+11	2.80E+10
301	0.00E+00	0.00E+00	3.00E-02	1.00E-02	0.00E+00	2.08E-08	0.00E+00	1.33E+02	2.49E+02	3.00E+06	3.00E+06
302	0.00E+00	0.00E+00	1.50E-01	8.00E-02	0.00E+00	8.95E-08	0.00E+00	6.57E+02	1.30E+03	3.10E+06	3.10E+06
303	0.00E+00	0.00E+00	9.80E-01	5.40E-01	0.00E+00	6.87E-07	0.00E+00	3.82E+03	7.51E+03	3.40E+06	3.40E+06

Table 4.3-7 Consequences by Release Group
Evacuation to 5 Mi, Sheltering 5-10 Mi

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
1	1.20E-01	6.10E+00	4.04E+01	2.52E+01	1.25E-05	4.78E-05	9.90E-03	1.34E+05	2.26E+05	3.20E+07	3.20E+07
2	0.00E+00	0.00E+00	1.40E+02	6.33E+01	0.00E+00	4.80E-05	0.00E+00	3.78E+05	8.09E+05	1.20E+08	1.20E+08
31	4.40E-01	1.12E+01	6.67E+01	3.99E+01	4.18E-05	7.03E-05	4.20E-02	2.14E+05	3.73E+05	4.10E+07	3.90E+07
33	0.00E+00	0.00E+00	1.14E+01	5.80E+00	0.00E+00	7.06E-06	0.00E+00	3.94E+04	8.26E+04	1.50E+07	1.20E+07
61	4.20E-01	1.11E+01	6.25E+01	3.77E+01	4.08E-05	6.64E-05	4.10E-02	2.02E+05	3.53E+05	3.40E+07	3.10E+07
64	3.50E-01	1.03E+01	1.77E+02	8.93E+01	3.50E-05	1.09E-04	3.90E-02	5.16E+05	1.03E+06	1.70E+08	1.60E+08
65	0.00E+00	0.00E+00	5.45E+02	2.34E+02	0.00E+00	8.91E-05	0.00E+00	1.40E+06	3.16E+06	7.60E+08	5.80E+08
66	0.00E+00	0.00E+00	1.04E+02	4.83E+01	0.00E+00	5.24E-05	0.00E+00	2.95E+05	6.20E+05	9.30E+07	8.50E+07
67	2.30E-01	1.08E+01	5.04E+02	2.34E+02	2.65E-05	1.68E-04	4.00E-02	1.40E+06	2.95E+06	6.70E+08	5.20E+08
68	0.00E+00	0.00E+00	1.10E+03	4.53E+02	0.00E+00	1.40E-04	0.00E+00	2.73E+06	6.41E+06	2.40E+09	1.50E+09
69	0.00E+00	0.00E+00	2.50E+02	1.14E+02	0.00E+00	8.09E-05	0.00E+00	7.10E+05	1.52E+06	2.90E+08	2.40E+08
70	5.90E-01	1.36E+01	6.80E+02	3.18E+02	5.84E-05	1.87E-04	4.70E-01	1.96E+06	4.13E+06	2.50E+09	8.70E+08
71	0.00E+00	0.00E+00	1.57E+03	6.35E+02	0.00E+00	1.60E-04	0.00E+00	3.95E+06	9.22E+06	5.20E+09	3.30E+09
72	0.00E+00	0.00E+00	5.53E+02	2.60E+02	0.00E+00	1.29E-04	0.00E+00	1.60E+06	3.32E+06	7.70E+08	5.10E+08
100	6.00E-01	1.61E+01	1.34E+03	6.12E+02	8.13E-05	2.78E-04	4.80E-02	3.69E+06	7.88E+06	3.00E+09	1.70E+09
101	0.00E+00	0.00E+00	1.93E+03	7.37E+02	0.00E+00	1.66E-04	0.00E+00	4.47E+06	1.13E+07	7.70E+09	4.90E+09
103	4.90E-01	1.52E+01	2.31E+03	1.01E+03	6.58E-05	3.62E-04	4.80E-02	6.13E+06	1.38E+07	9.50E+09	4.60E+09
104	0.00E+00	0.00E+00	2.61E+03	9.07E+02	0.00E+00	1.88E-04	0.00E+00	5.54E+06	1.55E+07	1.40E+10	7.60E+09
105	0.00E+00	0.00E+00	3.29E+03	1.09E+03	0.00E+00	2.07E-04	0.00E+00	6.59E+06	1.93E+07	1.70E+10	8.70E+09
106	2.30E+00	2.41E+01	3.30E+03	1.36E+03	3.18E-04	7.08E-04	9.80E-02	8.00E+06	1.95E+07	2.10E+10	8.50E+09
107	0.00E+00	0.00E+00	3.32E+03	1.04E+03	0.00E+00	2.01E-04	0.00E+00	6.39E+06	1.97E+07	1.90E+10	1.00E+10
136	1.40E+00	2.19E+01	3.86E+03	1.47E+03	2.19E-04	5.62E-04	9.00E-02	8.83E+06	2.30E+07	2.80E+10	1.10E+10
137	0.00E+00	0.00E+00	3.99E+03	1.20E+03	0.00E+00	2.84E-04	0.00E+00	7.38E+06	2.38E+07	2.60E+10	1.20E+10
139	3.85E+01	6.36E+01	5.24E+03	2.22E+03	3.54E-03	2.31E-03	2.55E-01	1.19E+07	3.05E+07	4.70E+10	1.60E+10
140	0.00E+00	2.00E-02	5.48E+03	1.55E+03	0.00E+00	5.44E-04	0.00E+00	9.33E+06	3.27E+07	4.30E+10	1.70E+10
142	2.50E+00	9.50E+00	6.99E+03	2.65E+03	4.92E-04	2.18E-03	9.30E-02	1.44E+07	4.13E+07	7.10E+10	1.90E+10
143	0.00E+00	9.40E-01	6.77E+03	1.86E+03	0.00E+00	6.00E-04	0.00E+00	1.13E+07	4.01E+07	5.10E+10	2.00E+10

Table 4.3-7 (Continued)

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
172	1.45E+02	2.59E+02	6.05E+03	2.39E+03	6.14E-03	2.61E-03	2.50E-01	1.28E+07	3.52E+07	4.50E+10	1.50E+10
173	1.00E-02	1.01E+01	8.50E+03	3.42E+03	0.00E+00	2.76E-03	0.00E+00	1.67E+07	4.79E+07	7.70E+10	2.30E+10
175	1.45E+02	1.04E+02	1.56E+04	9.03E+03	1.47E-02	1.22E-02	4.00E-01	3.45E+07	7.64E+07	1.10E+11	2.70E+10
176	7.40E-01	4.07E+01	1.07E+04	4.96E+03	1.18E-05	4.74E-03	1.00E-02	2.13E+07	5.72E+07	9.60E+10	2.60E+10
178	2.87E+02	3.48E+02	1.65E+04	9.56E+03	1.29E-02	1.29E-02	2.00E-01	3.80E+07	8.26E+07	1.30E+11	3.10E+10
179	1.53E+02	1.61E+02	2.25E+04	1.50E+04	1.85E-03	1.64E-02	2.00E-01	4.67E+07	9.08E+07	1.20E+11	2.80E+10
301	0.00E+00	0.00E+00	3.00E-02	2.00E-02	0.00E+00	1.85E-08	0.00E+00	1.29E+02	2.44E+02	6.70E+06	6.70E+06
302	0.00E+00	0.00E+00	1.50E-01	8.00E-02	0.00E+00	7.80E-08	0.00E+00	6.39E+02	1.31E+03	6.70E+06	6.70E+06
303	0.00E+00	0.00E+00	1.03E+00	5.60E-01	0.00E+00	6.85E-07	0.00E+00	3.90E+03	7.75E+03	7.00E+06	7.00E+06

Table 4.3-8 Consequences by Release Group
25 Rem/30 Yrs (Early & Chronic);
Evacuation to 5 Mi, Sheltering 5-10 Mi

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
1	1.20E-01	6.10E+00	4.35E+01	2.71E+01	1.25E-05	5.36E-05	9.90E-03	1.47E+05	2.48E+05	3.20E+07	3.20E+07
2	0.00E+00	0.00E+00	1.52E+02	7.54E+01	0.00E+00	9.41E-05	0.00E+00	4.50E+05	8.81E+05	1.20E+08	1.20E+08
31	4.40E-01	1.12E+01	9.15E+01	5.34E+01	4.18E-05	9.38E-05	4.20E-02	2.14E+05	4.17E+05	4.10E+07	3.90E+07
33	0.00E+00	0.00E+00	1.17E+01	6.10E+00	0.00E+00	8.41E-06	0.00E+00	4.13E+04	8.46E+04	1.50E+07	1.20E+07
61	4.20E-01	1.11E+01	5.45E+01	3.51E+01	4.08E-05	7.05E-05	4.10E-02	1.86E+05	3.08E+05	3.40E+07	3.10E+07
64	3.50E-01	1.03E+01	1.72E+02	9.49E+01	3.50E-05	1.58E-04	3.90E-02	5.51E+05	1.01E+06	1.70E+08	1.60E+08
65	0.00E+00	0.00E+00	6.06E+02	2.94E+02	0.00E+00	1.79E-04	0.00E+00	1.75E+06	3.52E+06	7.60E+08	5.80E+08
66	0.00E+00	0.00E+00	1.11E+02	5.60E+01	0.00E+00	8.36E-05	0.00E+00	3.41E+05	6.65E+05	9.30E+07	8.50E+07
67	2.30E-01	1.08E+01	5.92E+02	2.97E+02	2.65E-05	3.24E-04	4.00E-02	1.78E+06	3.47E+06	6.70E+08	5.20E+08
68	0.00E+00	0.00E+00	1.27E+03	6.16E+02	0.00E+00	2.52E-04	0.00E+00	3.70E+06	7.42E+06	2.40E+09	1.50E+09
69	0.00E+00	0.00E+00	2.69E+02	1.34E+02	0.00E+00	1.53E-04	0.00E+00	8.28E+05	1.64E+06	2.90E+08	2.40E+08
70	5.90E-01	1.36E+01	6.68E+02	3.50E+02	5.84E-05	3.52E-04	4.70E-01	2.18E+06	4.12E+06	2.50E+09	8.70E+08
71	0.00E+00	0.00E+00	1.98E+03	1.03E+03	0.00E+00	3.42E-04	0.00E+00	6.35E+06	1.17E+07	5.20E+09	3.30E+09
72	0.00E+00	0.00E+00	5.91E+02	2.99E+02	0.00E+00	2.32E-04	0.00E+00	1.84E+06	3.56E+06	7.70E+08	5.10E+08
100	6.00E-01	1.61E+01	1.69E+03	7.77E+02	8.13E-05	4.82E-04	4.80E-02	5.11E+06	9.90E+06	3.00E+09	1.70E+09
101	0.00E+00	0.00E+00	2.55E+03	1.32E+03	0.00E+00	3.88E-04	0.00E+00	7.96E+06	1.50E+07	7.70E+09	4.90E+09
103	4.90E-01	1.52E+01	3.05E+03	1.60E+03	6.58E-05	6.76E-04	4.80E-02	9.66E+06	1.84E+07	9.50E+09	4.60E+09
104	0.00E+00	0.00E+00	3.75E+03	1.86E+03	0.00E+00	4.55E-04	0.00E+00	1.12E+07	2.24E+07	1.40E+10	7.60E+09
105	0.00E+00	0.00E+00	4.65E+03	2.19E+03	0.00E+00	5.33E-04	0.00E+00	1.32E+07	2.74E+07	1.70E+10	8.70E+09
106	2.30E+00	2.41E+01	4.40E+03	2.39E+03	3.18E-04	1.14E-03	9.80E-02	1.43E+07	2.62E+07	2.10E+10	8.50E+09
107	0.00E+00	0.00E+00	4.94E+03	2.26E+03	0.00E+00	4.33E-04	0.00E+00	1.37E+07	2.93E+07	1.90E+10	1.00E+10
136	1.40E+00	2.19E+01	5.45E+03	2.83E+03	2.19E-04	9.37E-04	9.00E-02	1.71E+07	3.29E+07	2.80E+10	1.10E+10
137	0.00E+00	0.00E+00	5.98E+03	2.56E+03	0.00E+00	4.94E-04	0.00E+00	1.56E+07	3.56E+07	2.60E+10	1.20E+10
139	3.85E+01	6.36E+01	7.72E+03	4.11E+03	3.54E-03	2.65E-03	2.55E-01	2.38E+07	4.61E+07	4.70E+10	1.60E+10
140	0.00E+00	2.00E-02	8.18E+03	3.08E+03	0.00E+00	7.21E-04	0.00E+00	1.85E+07	4.89E+07	4.30E+10	1.70E+10
142	2.50E+00	9.50E+00	1.02E+04	4.58E+03	4.92E-04	2.43E-03	9.30E-02	2.64E+07	6.09E+07	7.10E+10	1.90E+10
143	0.00E+00	9.40E-01	1.03E+04	3.44E+03	0.00E+00	7.73E-04	0.00E+00	2.08E+07	6.13E+07	5.10E+10	2.00E+10

Table 4.3-8 (Continued)

Group No.	Early Fatalities	Early Injury	Total Latent Cancer Fatalities (1000 mi)	Total Latent Cancer Fatalities (50 mi)	Individual Early Fatality Risk (1 mi)	Individual Latent Cancer Fatality Risk (10 mi)	Probability of One Fatality	Pop. Dose (50 mi)	Pop. Dose (1000 mi)	Cost (1000 mi)	Cost (50 mi)
172	1.45E+02	2.59E+02	8.47E+03	3.55E+03	6.14E-03	2.77E-03	2.50E-01	1.99E+07	5.01E+07	4.50E+10	1.50E+10
173	1.00E-02	1.01E+01	1.24E+04	5.01E+03	0.00E+00	2.92E-03	0.00E+00	2.66E+07	7.23E+07	7.70E+10	2.30E+10
175	1.45E+02	1.04E+02	2.00E+04	1.14E+04	1.47E-02	1.25E-02	4.00E-01	4.66E+07	1.06E+08	1.10E+11	2.70E+10
176	7.40E-01	4.07E+01	1.52E+04	6.46E+03	1.18E-05	4.91E-03	1.00E-02	3.08E+07	8.54E+07	9.60E+10	2.60E+10
178	2.87E+02	3.48E+02	2.12E+04	1.13E+04	1.29E-02	1.31E-02	2.00E-01	5.06E+07	1.16E+08	1.30E+11	3.10E+10
179	1.53E+02	1.61E+02	2.77E+04	1.62E+04	1.85E-03	1.66E-02	2.00E-01	5.53E+07	1.27E+08	1.20E+11	2.80E+10
301	0.00E+00	0.00E+00	3.00E-02	2.00E-02	0.00E+00	1.85E-08	0.00E+00	1.29E+02	2.44E+02	6.70E+06	6.70E+06
302	0.00E+00	0.00E+00	1.50E-01	9.00E-02	0.00E+00	7.77E-08	0.00E+00	6.42E+02	1.33E+03	6.70E+06	6.70E+06
303	0.00E+00	0.00E+00	1.10E+00	6.00E-01	0.00E+00	7.54E-07	0.00E+00	4.16E+03	8.18E+03	7.00E+06	7.00E+06

Table 4.3-9 Consequences by Release Group
Chronic Only, 4 Rems/5 Years (Base Case)

Group No.	Total Fatalities (Entire Region)	Total Fatalities (50 Mi)	Individual Latent Cancer Fatality Risk (10 Mi)	Pop. Dose (50 Mi)	Pop. Dose (Entire Region)
1	2.45E+01	1.27E+01	1.78E-05	7.65E+04	1.46E+05
2	1.36E+02	6.07E+01	4.69E-05	3.59E+05	7.81E+05
31	2.65E+01	1.36E+01	1.89E-05	8.15E+04	1.61E+05
33	7.10E+00	3.40E+00	5.69E-06	2.11E+04	5.14E+04
61	5.44E+01	2.69E+01	3.28E-05	1.60E+05	3.22E+05
64	1.73E+02	7.82E+01	7.05E-05	4.66E+05	1.01E+06
65	5.28E+02	2.22E+02	8.31E-05	1.32E+06	3.04E+06
66	9.67E+01	4.43E+01	5.03E-05	2.62E+05	5.68E+05
67	4.25E+02	1.87E+02	1.12E-04	1.12E+06	2.48E+06
68	1.06E+03	4.25E+02	1.26E-04	2.53E+06	6.13E+06
69	2.20E+02	9.31E+01	7.27E-05	5.57E+05	1.30E+06
70	6.90E+02	2.98E+02	1.18E-04	1.81E+06	4.11E+06
71	1.50E+03	6.02E+02	1.35E-04	3.58E+06	8.72E+06
72	4.82E+02	2.10E+02	1.11E-04	1.25E+06	2.83E+06
100	1.05E+03	5.31E+02	1.44E-04	2.75E+06	6.18E+06
101	1.83E+03	6.63E+02	1.30E-04	3.95E+06	1.06E+07
103	1.94E+03	8.01E+02	1.97E-04	4.84E+06	1.13E+07
104	2.44E+03	7.72E+02	1.20E-04	4.63E+06	1.42E+07
105	3.08E+03	9.48E+02	1.54E-04	5.64E+06	1.80E+07
106	2.93E+03	1.05E+03	1.12E-04	6.32E+06	1.74E+07
107	3.06E+03	8.59E+02	1.01E-04	5.11E+06	1.79E+07
136	3.47E+03	1.14E+03	1.96E-04	6.88E+06	2.04E+07
137	3.61E+03	9.28E+02	1.17E-04	5.52E+06	2.12E+07
139	3.83E+03	1.09E+03	1.96E-04	6.70E+06	2.35E+07
140	4.75E+03	1.03E+03	1.24E-04	6.12E+06	2.80E+07
142	5.08E+03	1.27E+03	2.20E-04	7.88E+06	3.15E+07
143	5.85E+03	1.22E+03	1.35E-04	7.25E+06	3.42E+07
172	4.28E+03	9.58E+02	9.79E-05	5.77E+06	2.57E+07
173	5.57E+03	1.26E+03	1.30E-04	7.55E+06	3.40E+07
175	5.80E+03	9.17E+02	2.03E-04	9.60E+06	3.81E+07
176	5.93E+03	1.35E+03	1.20E-04	8.12E+06	3.67E+07
178	6.05E+03	1.48E+03	9.50E-05	9.56E+06	4.00E+07
179	5.28E+03	1.40E+03	1.29E-04	8.60E+06	3.48E+07
301	2.00E-02	1.00E-02	1.77E-08	9.30E+01	1.65E+02
302	1.10E-01	5.00E-02	7.48E-08	4.25E+02	9.54E+02
303	7.60E-01	4.00E-01	5.93E-07	2.62E+03	5.74E+03

Table 4.3-10 Consequences by Release Group
Chronic Only, 25 Rem/30 Years (Base Case)

Group No.	Total Fatalities (Entire Region)	Total Fatalities (50 Mi)	Individual Latent Cancer Fatality Risk (10 Mi)	Pop. Dose (50 Mi)	Pop. Dose (Entire Region)
1	2.76E+01	1.46E+01	2.36E-05	8.99E+04	1.68E+05
2	1.48E+02	7.28E+01	9.30E-05	4.31E+05	8.53E+05
31	5.13E+01	2.71E+01	4.24E-05	8.16E+04	2.05E+05
33	7.40E+00	3.70E+00	7.04E-06	2.30E+04	5.34E+04
61	4.64E+01	2.43E+01	3.69E-05	1.44E+05	2.77E+05
64	1.68E+02	8.38E+01	1.19E-04	5.01E+05	9.86E+05
65	5.89E+02	2.82E+02	1.73E-04	1.67E+06	3.40E+06
66	1.04E+02	5.20E+01	8.15E-05	3.08E+05	6.13E+05
67	5.13E+02	2.50E+02	2.68E-04	1.50E+06	3.00E+06
68	1.23E+03	5.88E+02	2.38E-04	3.50E+06	7.14E+06
69	2.39E+02	1.13E+02	1.45E-04	6.75E+05	1.42E+06
70	6.78E+02	3.30E+02	2.83E-04	2.03E+06	4.10E+06
71	1.91E+03	1.00E+03	3.17E-04	5.98E+06	1.12E+07
72	5.20E+02	2.49E+02	2.14E-04	1.49E+06	3.07E+06
100	1.40E+03	6.96E+02	3.48E-04	4.17E+06	8.20E+06
101	2.45E+03	1.25E+03	3.52E-04	7.44E+06	1.43E+07
103	2.68E+03	1.39E+03	5.11E-04	8.37E+06	1.59E+07
104	3.58E+03	1.72E+03	3.87E-04	1.03E+07	2.11E+07
105	4.44E+03	2.05E+03	4.80E-04	1.22E+07	2.61E+07
106	4.03E+03	2.08E+03	5.43E-04	1.26E+07	2.41E+07
107	4.68E+03	2.08E+03	3.33E-04	1.24E+07	2.75E+07
136	5.06E+03	2.50E+03	5.71E-04	1.51E+07	3.03E+07
137	5.60E+03	2.29E+03	3.27E-04	1.37E+07	3.30E+07
139	6.31E+03	2.98E+03	5.35E-04	1.86E+07	3.91E+07
140	7.45E+03	2.56E+03	3.01E-04	1.53E+07	4.42E+07
142	8.24E+03	3.20E+03	4.69E-04	1.99E+07	5.11E+07
143	9.40E+03	2.80E+03	3.08E-04	1.67E+07	5.54E+07
172	6.70E+03	2.12E+03	2.56E-04	1.29E+07	4.06E+07
173	9.50E+03	2.85E+03	2.88E-04	1.74E+07	5.84E+07
175	1.02E+04	3.32E+03	4.72E-04	2.17E+07	6.81E+07
176	1.04E+04	2.85E+03	2.89E-04	1.76E+07	6.49E+07
178	1.07E+04	3.20E+03	2.98E-04	2.22E+07	7.31E+07
179	1.05E+04	2.64E+03	2.79E-04	1.72E+07	7.12E+07
301	2.00E-02	1.00E-02	1.77E-08	9.30E+01	1.65E+02
302	1.10E-01	6.00E-02	7.45E-08	4.28E+02	9.78E+02
303	8.30E-01	4.40E-01	6.62E-07	2.88E+03	6.17E+03

Table 4.3-11 Relative Effectiveness of Emergency Response Actions Assuming Early Containment Failure with High Source Term

Emergency Actions	Conditional Probability (%) of Exceeding 200-Rem Acute Red Marrow Dose					Conditional Probability (%) of Exceeding 50-Rem Acute Red Marrow Dose						Mean Red Marrow Dose				
	Distance from Reactor					Distance from Reactor						Distance from Reactor				
	1 Mile	2 Miles	3 Miles	5 Miles	10 Miles	1 Mile	2 Miles	3 Miles	5 Miles	10 Miles	20 Miles	2 Miles	3 Miles	5 Miles	10 Miles	20 Miles
Continue normal activity	65	58	50	26	1	83	71	65	58	32	0	501	297	132	45.2	10.8
Basement shelter	62	48	36	8	0	78	64	60	45	16	0	271	160	70.8	24.3	5.9
Shelter in large building	33	9	0	0	0	62	50	38	12	0	0	77.2	45.8	20.5	7.15	1.77
Evacuation starting																
1 h before release											NA	50.3	37.1	21.0	4.8	NA
at release	49	40	18	2	0	56	54	51	39	7	NA	174	106	50.1	13.5	NA
1 h after release	54	48	38	17	0	70	64	55	48	20	NA	284	184	97.2	27.1	NA

*Source term 178 is given in Table 3.4-5.
**Ranges reflect the MACCS CCDFS output representation.

Table 4.3-12 Relative Effectiveness of Emergency Response Actions Assuming Early Containment Failure with Low Source Term

Emergency Actions	Conditional Probability (%) of Exceeding 200-Rem Acute Red Marrow Dose — Distance from Reactor					Conditional Probability (%) of Exceeding 50-Rem Acute Red Marrow Dose — Distance from Reactor						Conditional Probability (%) of Exceeding 5-Rem Acute Red Marrow Dose — Distance from Reactor				
	1 Mile	2 Miles	3 Miles	5 Miles	10 Miles	1 Mile	2 Miles	3 Miles	5 Miles	10 Miles	20 Miles	2 Miles	3 Miles	5 Miles	10 Miles	20 Miles
Continue normal activity	1	0	0	0	0	38	17	8	0	0	0	60	58	53	19	0
Basement shelter	0	0	0	0	0	23	8	2	0	0	0	59	57	40	11	0
Shelter in large building	0	0	0	0	0	2	0	0	0	0	0	50	35	12	0	0
Evacuation starting:																
at release	0	0	0	0	0	10	3	0	0	0	NA	57	52	31	4	NA
1 h after release	3	1	0	0	0	37	18	12	0	0	NA	61	58	50	14	NA

*Source term 001 is given in Table 3.4-5.

**Ranges reflect the MACCS CCDFS output representation.

Table 4.3-13 Relative Effectiveness of Emergency Response Actions Assuming Late Containment Failure with High Source Term

	Conditional Probability (%) of Exceeding 200-Rem Acute Red Marrow Dose					Conditional Probability (%) of Exceeding 50-Rem Acute Red Marrow Dose						Mean Red Marrow Dose				
	Distance from Reactor					Distance from Reactor						Distance from Reactor				
Emergency Actions	1 Mile	2 Miles	3 Miles	5 Miles	10 Miles	1 Mile	2 Miles	3 Miles	5 Miles	10 Miles	20 Miles	2 Miles	3 Miles	5 Miles	10 Miles	20 Miles
Continue normal activity	5	0	0	0	0	34	16	0	0	0	0	20.5	10.4	3.8	1.1	0.2
Basement shelter	0	0	0	0	0	18	4	0	0	0	0	11.2	5.7	2.1	0.6	0.1
Shelter in large building	0	0	0	0	0	0	0	0	0	0	0	3.3	1.7	0.7	0.2	0.04
Evacuation starting:																
at release	0	0	0	0	0	0	0	0	0	0	NA	1.4	0.9	0.5	0.1	NA
1 h after release	0	0	0	0	0	0	0	0	0	0	NA	4.5	2.6	1.0	0.2	NA

*Source term 105 is given in Table 3.4-5.

**Ranges reflect the MACCS CCDFS output representation.

Table 4.3-14 Relative Effectiveness of Emergency Response Actions Assuming Late Containment Failure with Low Source Term

Emergency Actions	Conditional Probability (%) of Exceeding 200-Rem Acute Red Marrow Dose					Conditional Probability (%) of Exceeding 50-Rem Acute Red Marrow Dose						Conditional Probability (%) of Exceeding 5-Rem Acute Red Marrow Dose				
	Distance from Reactor					Distance from Reactor						Distance from Reactor				
	1 Mile	2 Miles	3 Miles	5 Miles	10 Miles	1 Mile	2 Miles	3 Miles	5 Miles	10 Miles	20 Miles	2 Miles	3 Miles	5 Miles	10 Miles	20 Miles
Continue normal activity	0	0	0	0	0	0	0	0	0	0	0	13	5	0	0	0
Basement shelter	0	0	0	0	0	0	0	0	0	0	0	7	0	0	0	0
Shelter in large building	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Evacuation starting:																
at release	0	0	0	0	0	0	0	0	0	0	NA	0	0	0	0	NA
1 h after release	0	0	0	0	0	0	0	0	0	0	NA	0	0	0	0	NA

*Source term 33 is given in Table 3.4-5.

**Ranges reflect the MACCS CCDFS output representation.

5. ZION RISK RESULTS

This section gives the results of the integrated risk analysis for the Zion plant. Section 5.1 gives the risk results for internal initiators.

Risk is determined by bringing together the results of four constituent analyses: the accident frequency, accident progression, source term, and consequence analyses. The phrase integrated risk analysis is used to refer to the combined result when all four analyses are combined. The way in which these analyses contribute to risk analysis is summarized in Section 1.4 of this volume. More detail on the methods used in calculating risk can be found in Volume 1.

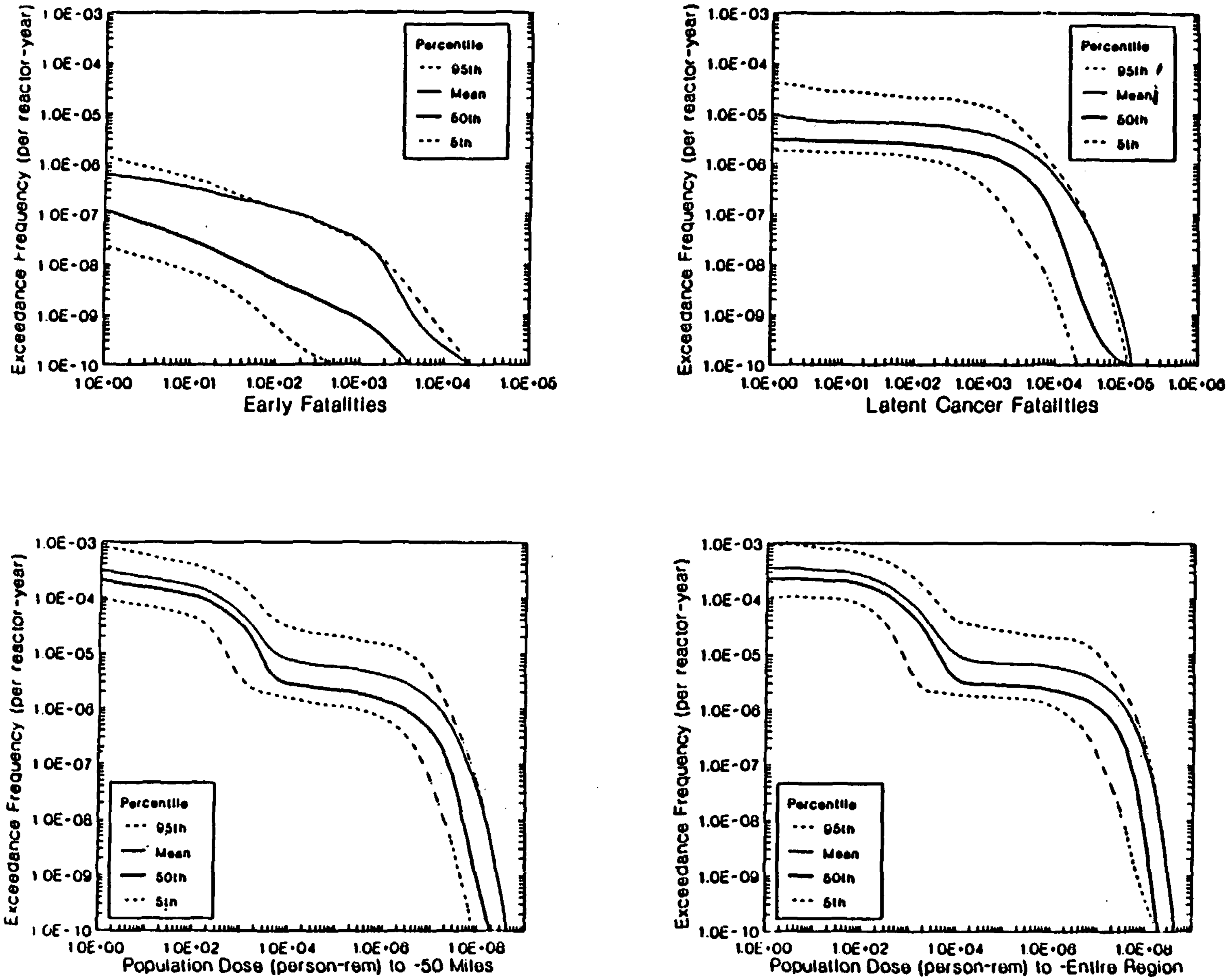
The figures in this section present only a very small portion of the total risk output available. More details are provided in Appendix D.

5.1 Results for Internal Initiators

This section describes the results of the integrated risk analysis for internal initiators at the Zion plant. Section 5.1.1 is a discussion of basic risk results for internal initiators. A second sample was run completely through the integrated risk analysis for the Surry Plant. The implications of this second sample are discussed in Section 5.1.2. Section 5.1.3 addresses the types of accidents and plant features which are important in determining the risk from internal initiators at Zion. Section 5.1.4 gives the results of the regression analysis performed to determine the important contributors to the uncertainty in risk. Accidents involving induced reactor coolant pump seal LOCAs were found to dominate the estimated core damage frequency and risk at Zion. After completion of the draft revision 1 analysis for Zion Unit 1, Commonwealth Edison made commitments to the NRC to make plant and procedure changes to address the major contributor to the core damage frequency. The impact of these changes would be a reduction in the core damage frequency of approximately 80%. With these changes seal LOCAs contribute significantly less to the core damage frequency. The Zion risk estimates reported in this volume do not reflect these changes. However, a sensitivity study was performed to assess the impact of the changes in the mean Zion risk estimates. The results of the sensitivity analysis are discussed in Section 5.1.5.

5.1.1 Risk Results

Figure 5.1-1 shows the basic results of the integrated risk analysis for internal initiators at Zion. This figure shows the complementary cumulative distribution functions (CCDFs) for early fatalities, latent cancer fatalities, population dose within 50 miles and population dose within the entire region. The CCDFs display the relationship between the frequency of the consequence and the magnitude of the consequence. As there are 150 observations in the sample for Zion, the actual risk results at the most basic level are 150 CCDFs for each consequence measure. Only four statistical measures of the 150 curves are shown in Figure 5.1-1. These measures are generated by analyzing the plots in the vertical direction. For each consequence value on the abscissa, there are 150 values of the exceedance frequency (one for each observation or sample element) and from these 150 values the mean, median, 95th percentile, and 5 percentile values are calculated. When this is done for each value of the consequence measure, the curves in Figure 5.1-1 are obtained. Thus, Figure 5.1-1 gives the relationship



Note: As discussed in Reference 1, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in risk analyses

Figure 5.1-1 Frequency Distributions of Offsite Consequence Measures at Zion - Internal Initiators between the magnitude of the consequence and the frequency at which the consequence is exceeded, as well as the variation in that relationship. The percentile curves in Figure 5.1-1 and similar figures can be read from either axis; however, the mean curve is only valid when read from the abscissa.

The curves for latent cancer fatalities in Figure 5.1-1 are relatively flat from 1 to about 500 fatalities. This means that latent cancer fatalities in this range are very unlikely. Any type of containment failure or bypass is likely to lead to more than 500 delayed fatalities. If the containment does not fail, the eventual release of the noble gases (xenon and krypton) from the containment due to design basis leakage will probably cause less than 1 latent cancer fatality.

The variation from the 5th to the 95th percentiles indicates the uncertainty in the risk estimates due to uncertainty in the basic parameters in the three sampled constituent analyses (the accident frequency, accident progression, and source term analyses). The variation along a curve in Figure 5.1-1 is indicative of the variation in risk due to different types of accidents and due to different weather conditions at the time of the accident. Thus the individual curves can be viewed as representing stochastic variability (i.e., the effects of probabilistic events in which it is possible for the accident to develop in more than one way) and the variability between curves can be seen as representing the effects of imprecisely known parameters and processes that are mostly nonstochastic. As the magnitude of the consequence measure increases, the mean curve typically approaches or exceeds the 95th percentile curve. This results when the mean is dominated by a few large observations, which often happens for large values of the consequences because only a few observations have nonzero exceedance frequencies for these large consequences.

Figure 5.1-1 shows the following mean and median exceedance frequencies for fixed values of early fatalities (EF) and latent cancer fatalities (LCF):

	Exceedance Frequency (1/R-yr)	
Consequence	Mean	Median
1 EF	6E-7	1E-7
100 EF	2E-7	5E-9
100 LCF	7E-6	3E-6
10,000 LCF	7E-7	1E-7

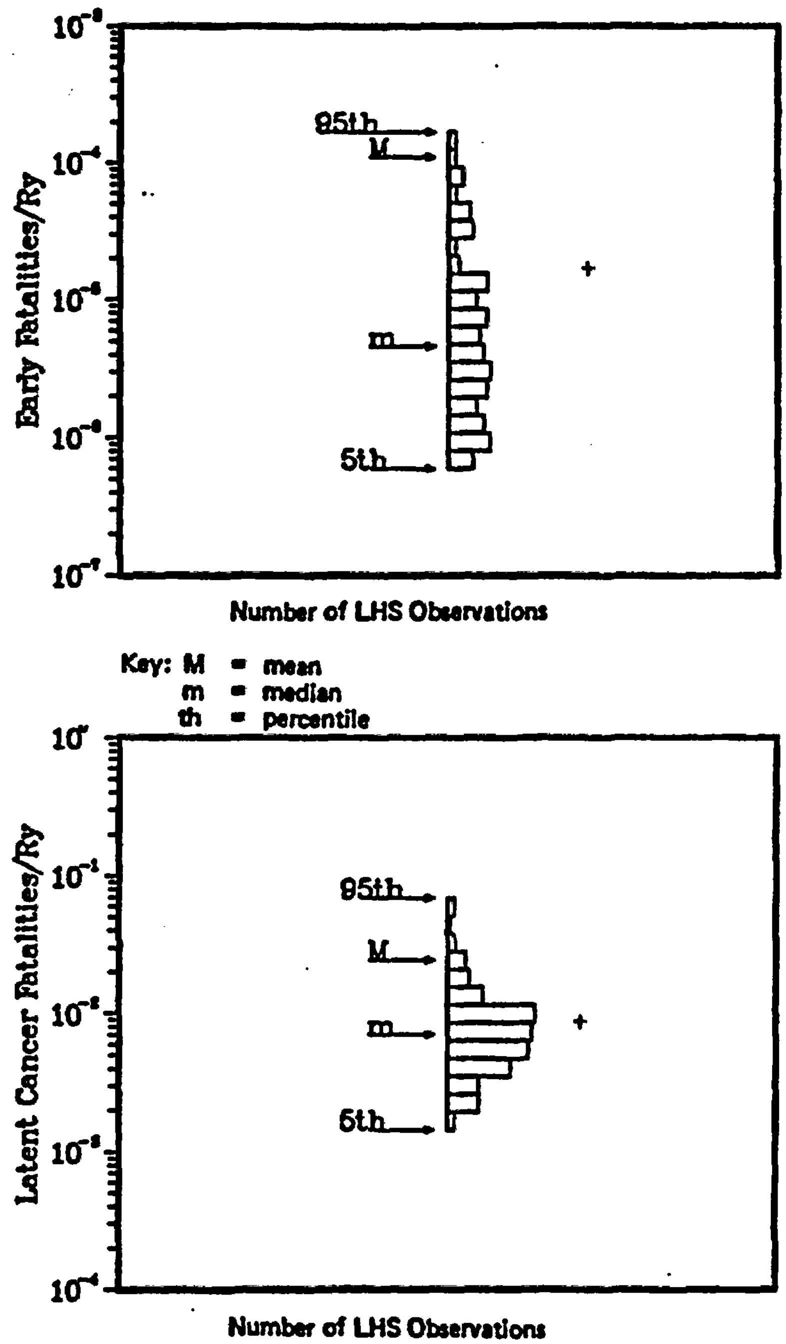
Although the latent cancer fatality values mentioned above may appear large, they must be considered in perspective; the calculated latent cancer fatalities occur throughout the entire region and over several decades. Between 400,000 to 500,000 deaths due to cancer occur every year in the U.S. The population within 350 miles of the plant is about 49 million and the population within 1000 miles of the plant is about 186 million. When spread over two or three decades, even tens of thousands of additional latent cancer fatalities are statistically indistinguishable from the general background morbidity due to malignant neoplasms in such a large population.

Although the CCDF for each observation conveys the most information about risk, a single number may be generated for each consequence measure for each observation. This value, denoted annual risk, is determined by summing the product of the frequencies and consequences for all the points that are used to construct the CCDF for each observation in the sample. The construction of annual risk has the effect of averaging over the different weather states and includes contributions from all the different types of accidents that can occur. Since the complete analysis consisted of a sample of 150 observations, there are 150 values of annual risk for each consequence measure. These 150 values may be ordered and plotted as histograms, which is done in Figures 5.1-2 to 5.1-4. The same four statistical measures utilized above are shown in these plots and are also reported in Table 5.1-1. Note that considerable information has been lost in going from the CCDFs to the histograms of annual values in Figures 5.1-2 to 5.1-4; the relationship between the size of the consequence and its frequency has been sacrificed to obtain a single value for risk for each observation.

The plots in Figures 5.1-2 to 5.1-4 show the variation in the annual risk for six consequence measures. Where the mean is close to the 95th percentile, it may be inferred that a relatively small number of observations dominate the mean value. This is more likely to occur for the early fatality consequence measures than for the latent cancer fatality or population dose consequence measures due to the threshold effect for early fatalities. In essence, these figures show the probability density functions (PDF) of the logarithms of the consequence measures. Equivalent density functions could be generated for the consequence measures themselves, but would appear quite different due to the change in scale. Another alternative, but equivalent display, for the results in Figures 5.1-2 to 5.1-4 would be to use cumulative distribution functions.

The safety goals are expressed in terms of mean individual fatality risks, which is really an individual's probability of becoming a casualty of a reactor accident in a given year. The individual early fatality risk within one mile is the frequency (per year) that a person living within one mile of the site boundary will die within a year due to the accident. The entire population within one mile is considered to obtain an average value. The individual latent cancer fatality risk within 10 miles is the frequency (per year) that a person living within 10 miles of the plant will die many years later from cancer due to radiation exposure received from the accident. The entire population within 10 miles is considered to obtain an average value. A single value for individual fatality risk for each observation is obtained by reducing the CCDF for each observation to a single value. The density distribution of these 150 values is plotted in Figure 5.1-4. Although the values are really frequencies, they are so small that they are essentially probabilities that an individual will become a casualty of a reactor accident in a given year. The plots for individual risk in Figure 5.1-4 show that both internally indicated risk distributions for Zion fall well below the safety goal.

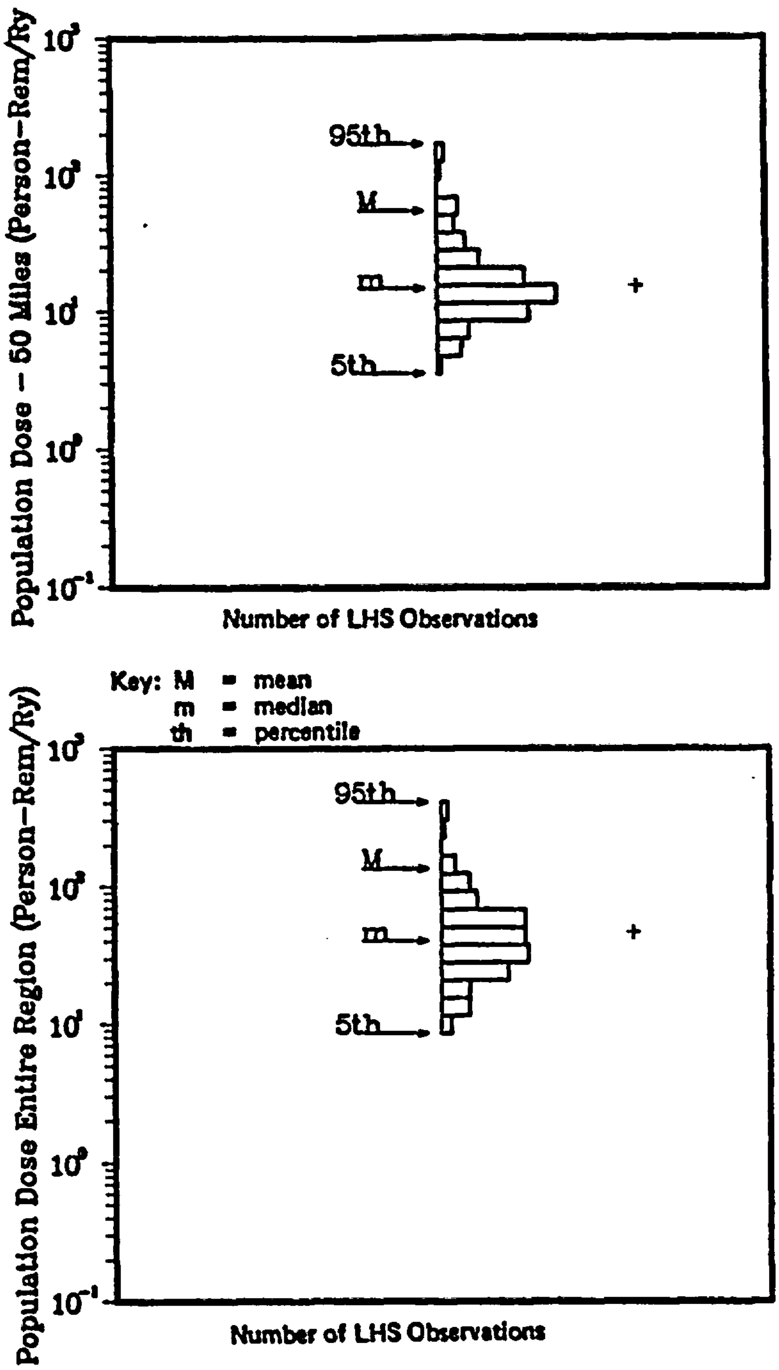
A single measure of risk for the entire sample may be obtained by taking the average value from the histograms in Figures 5.1-2 to 5.1-4. This measure of risk is commonly called mean risk, although it is actually the average of the annual risk, or the mean value of the mean risk. The mean risk values for the six consequence measures reported here are displayed in Figures 5.1-2 to 5.1-4. The important contributors to mean risk are considered in subsection 5.1.3.



Notes: As discussed in Reference 1, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

"+" shows recalculated mean value based on plant modifications discussed in text.

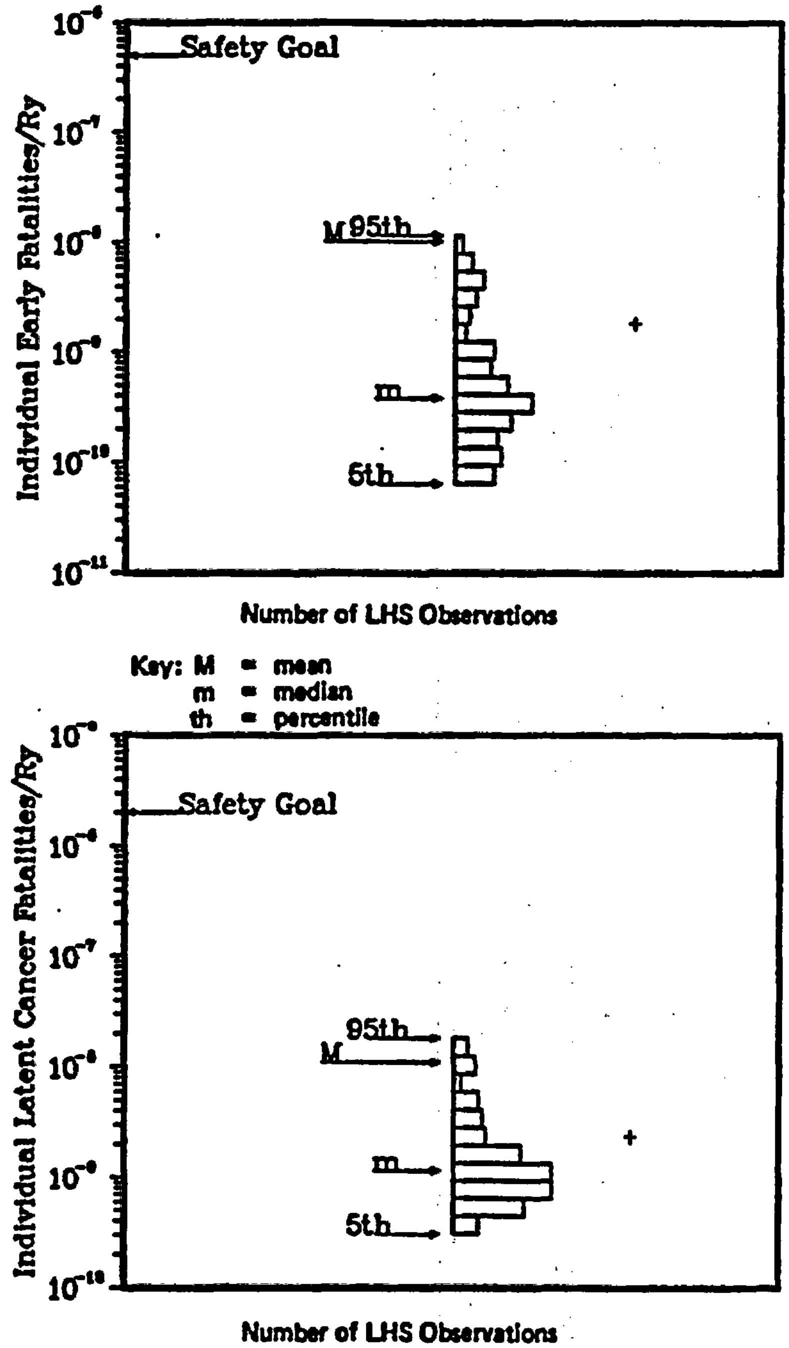
Figure 5.1-2 Early and Latent Cancer Fatality Risks at Zion Internal Initiators



Notes: As discussed in Reference 1, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

"+" shows recalculated mean value based on plant modifications discussed in text.

Figure 5.1-3 Population Dose Risks at Zion - Internal Initiators



Notes: As discussed in Reference 1, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of other health effects not studied in the risk analyses.

"+" shows recalculated mean value based on plant modifications discussed in text.

Figure 5.1-4 Individual Early and Latent Cancer Fatality Risks at Zion Internal Initiators

Table 5.1-1

Distributions for Annual Risk at Zion Due to Internal Initiators

	5th	Median	Mean	95-th
Early Fatalities (EF)	5.93E-07	4.59E-06	1.10E-04	1.16E-04
Early Injuries	9.44E-06	3.68E-05	2.12E-04	5.66E-04
LCFs (Entire Region)	1.50E-03	7.04E-03	2.44E-02	5.91E-02
LCFs (50 mi)	6.48E-04	2.49E-03	1.11E-02	2.32E-02
Individual EF Risk 0-1 mile	7.13E-11	3.78E-10	1.03E-08	8.06E-09
Individual LCF Risk 0-10 miles	3.59E-10	1.16E-09	1.10E-08	1.59E-08
Population Dose (50 mi)	3.78E+00	1.49E+02	5.47E+01	1.39E+02
Population Dose (Entire Region)	8.66E+00	4.05E+01	1.36E-02	3.52E+02

One of the main factors accounting for the low values for early fatality risk at Zion is the strength of the containment. Estimates of failure pressure have increased over the years; the median and mean values of the distribution for the failure pressure of the Zion containment for this analysis are about 150 psia. Although the pressure risk at VB now contains the contributions from direct heating of the containment, the addition of this mechanism has been more than offset by the mechanisms considered that lead to depressurization of the RCS before failure of the vessel. The RCS depressurization mechanism included are: T-I failure of the hot leg or surge line, PORVs sticking open, T-I RCP seal failure, T-I SGTR, and deliberate opening of the PORVs by the operators. Only the first three of these mechanisms are very effective, but this was sufficient to ensure that only a small fraction of the accidents that were at full system pressure at the onset of core damage were still at that pressure at vessel breach. Reducing the RCS pressure at vessel breach, of course, reduces the loads placed on the containment at vessel breach, and thus reduces the probability of CF.

Another change in the analysis which has reduced the probability of CF at vessel breach, thus the early fatality risk, is the consideration of arresting the core damage process before vessel failure and achieving a safe, stable state, as at TMI-2. Obtaining sufficient ECCI after the onset of core damage may come about through the recovery of offsite power, or the depressurization of the RCS to the point where injection by systems operating at the onset of core damage commences.

This was found to have a significant impact on the risk results for Surry (refer to Volume 3) where for a signification fraction of the time, the accidents in the most likely PDS resulted in arrest of the core damage process and no vessel breach. However, for the Zion plant the PDS that dominates the risk is LOCA induced accidents in which the potential for core damage arrest was found to be less important than for some of the other less probable PDS.

In summary, the risk of early fatalities at Zion is low because the most likely CD accidents (LOCAs) rarely lead to CF at vessel breach. However, CF at vessel breach was found to dominate early fatality risk at Zion because bypass events were found to have an even lower frequency.

Latent cancer fatality risk at Zion is also fairly low. For this risk measure CF was again found to be more important than bypass events.

5.1.2 Second Sample

To determine the reproducibility of the integrated risk analyses performed for NUREG-1150, a second sample was run through the entire integrated risk analysis for Surry. The second sample is just as valid as the first sample, and differs from the first sample only in the fact that a different random seed was used in the Latin Hypercube Sampling (LHS) program. Therefore, differences in the results between the two samples are an indication of the robustness of the analysis methods. The results of the second sample are discussed in Volume 3 of this report and show good agreement with the results of the first Surry sample. The good agreement between the results of the two samples for Surry indicates that the methods used for this integrated risk analysis are sound.

5.1.3 Contributors to Risk

There exist two distinct ways to calculate contribution to risk. To facilitate their definition, the following quantities are introduced:

rC_j = risk (units: consequences/R-yr) for consequence measure j,

rC_{ij} = value for rC_j obtained for observation i,

rC_{jk} = risk (units: consequences/R-yr) for consequence measure j due to PDS group k,

rC_{ijk} = value for rC_{jk} obtained for observation i, and

nLHS = number of observations in the Latin Hypercube Sample.

The notation used here is similar to that used in Section 1.4. The value of nLHS is 150 for Zion. The risk rC_{ij} is the jth element of the vector rC_i in Equation (1.9) of Section 1.4. The risk rC_{ijk} is the jth element of the vector rC_i when the frequencies of all the PDS groups except group k in the vector $fPDS_i$ are set to zero. The vector $fPDS_i$ is equal to the product $fIE_i\ P_i(IE \rightarrow PDS)$.

The result of the first method for computing contribution to risk is denoted the fractional contribution to mean risk and abbreviated FCMR. The contribution of PDS group k to the risk for consequence measure j, $FCMR_{jk}$, is defined as the ratio of the annual risk due to PDS group k to the total annual risk. That is, $FCMR_{jk}$ is defined by

$$FCMR_{jk} = E(\ rC_{jk}\ )\ /\ E(\ rC_j\ ),$$

where E(x) represents the annual value of x. Computationally, $FCMR_{jk}$ is found by use of the relation

$$FCMR_{jk} = [\ \Sigma\ rC_{ijk}\ /\ nLHS\ ]\ /\ [\ \Sigma\ rC_{ij}\ /\ nLHS\ ]$$

$$= \Sigma\ rC_{ijk}\ /\ \Sigma\ rC_{ij}$$

where the summations are from i = 1 to i = nLHS.

The result of the second method for computing contribution to risk is denoted the mean fractional contribution to risk and abbreviated MFCR. The contribution of PDS group k to the risk for consequence measure j, $FCMR_{jk}$, is defined as the annual value of ratio of the risk due to PDS group k to the total risk. That is:

$$MFCR_{jk} = E(\ rC_{jk}\ /\ rC_j\ ).$$

Computationally, $MFCR_{jk}$ is found by use of the relation

$$MFCR_{jk} = \Sigma\ (\ rC_{ijk}\ /\ rC_{ij}\ )\ /\ nLHS,$$

where the summation again is from i = 1 to i = nLHS.

For the FCMR the averaging over the observations is done before the ratio of group risk to total risk is formed; for MFCR the averaging over the observations is done after the ratio of group risk to total risk is formed.

Both methods of computing the contributions to risk were used for the Surry plant for both samples. The reader is referred to Volume 3 of this report for a more detailed discussion on the two approaches. It is clear from the results in Volume 3 that contributors to mean risk should only be interpreted in a broad sense. In this volume the FCMR method was used to calculate the risk contributors. The results of the calculations are summarized in Tables 5.1-2 and 5.1-3 and displayed as pie charts in Figures 5.1-5 - 5.1-8.

Figures 5.1-5 and 5.1-6 show the major plant damage state contributors to the mean early and latent cancer fatality risk estimates for Zion. It is clear from these figures that induced seal LOCAs are the major contributors to both the early and latent risk estimates. This is simply because the frequency of the LOCA PDS is higher than the other PDS frequencies and each of the PDSs have a similar potential for containment failure (refer to Figure 2.5-3) and hence offsite consequences.

Figures 5.1-7 and 5.1-8 show the major accident progression bin contributors to the mean early and latent cancer fatality risk estimates for Zion. These figures show that early containment failure bins are dominant contributors to both the early and latent risk estimates. These results are again driven by the relatively high frequency of the LOCA PDS.

Accidents involving induced reactor coolant pump seal LOCAs dominate the estimated CDF and the risk estimates at Zion. In section 5.1.5 a sensitivity study is presented which indicates the impact on CDF and risk of changes planned by Commonwealth Edison to address this class of accidents.

5.1.4 Contributors to Uncertainty

Figure 5.1-1 provides information on the frequency at which values for individual consequence measures will be exceeded. Specifically, mean, median, 5th percentile, and 95th percentile values are shown for these exceedance frequencies. Thus, Figure 5.1-1 can be viewed as presenting uncertainty analysis results for the risk at Zion due to internal initiators.

As the curves in Figure 5.1-1 show, there is significant uncertainty in the frequency at which a given consequence value will be exceeded. Due to the complexity of the underlying analysis and the concurrent variation of a large number of variables within this analysis, it is difficult to ascertain the cause of this uncertainty on the basis of a simple inspection of the results. However, numerical sensitivity analysis techniques provide a systematic way of investigating the observed variation in exceedance frequencies.

This section presents the results of using regression-based sensitivity analysis techniques to examine the variability in the consequences of internally initiated accidents at Zion.

Table 5.1-2 Major Contributors (Plant Damage Sates) to Risk at Zion - Internal Initiators

	SBO	LOCA	Trans	Bypass
Early Fatalities	9.57E-02	8.16E-01	3.18E-02	5.67E-02
Early Injuries	1.09E-01	7.80E-01	4.56E-02	6.58E-02
LCFs (Entire Region)	3.65E-02	7.27E-01	1.98E-02	2.17E-01
LCFs (50 mi)	4.00E-02	7.60E-01	2.05E-02	1.80E-01
Individual EF Risk 0 - 1 mile	5.48E-02	8.81E-01	2.17E-02	4.27E-02
Individual LCF Risk 0 - 10 miles	5.75E-02	8.42E-01	2.52E-02	7.58E-02
Pop Dose (50 mi)	3.84E-02	7.51E-01	2.10E-02	1.90E-01
Pop Dose (Entire Region)	3.56E-02	7.24E-01	1.99E-02	2.21E-01

Table 5.1-3 Major Contributors (Accident Progression Bins) to Risk at Zion - Internal Initiators

	ECF	LCF	NCF	Bypass
Early Fatalities	9.32E-01	2.68E-07	0.00E+00	6.81E-02
Early Injuries	9.15E-01	1.41E-06	0.00E+00	8.52E-02
LCFs (Entire Region)	7.79E-01	1.86E-03	7.67E-04	2.18E-01
LCFs (50 mi)	8.15E-01	2.03E-03	9.52E-04	1.82E-01
Individual EF Risk 0 - 1 mile	9.52E-01	5.75E-07	0.00E+00	4.84E-02
Individual LCF Risk 0 - 10 miles	9.14E-01	2.08E-03	8.92E-04	8.28E-02
Pop Dose (50 mi)	8.03E-01	2.71E-03	1.44E-03	1.93E-01
Pop Dose (Entire Region)	7.74E-01	2.28E-03	1.18E-03	2.23E-01

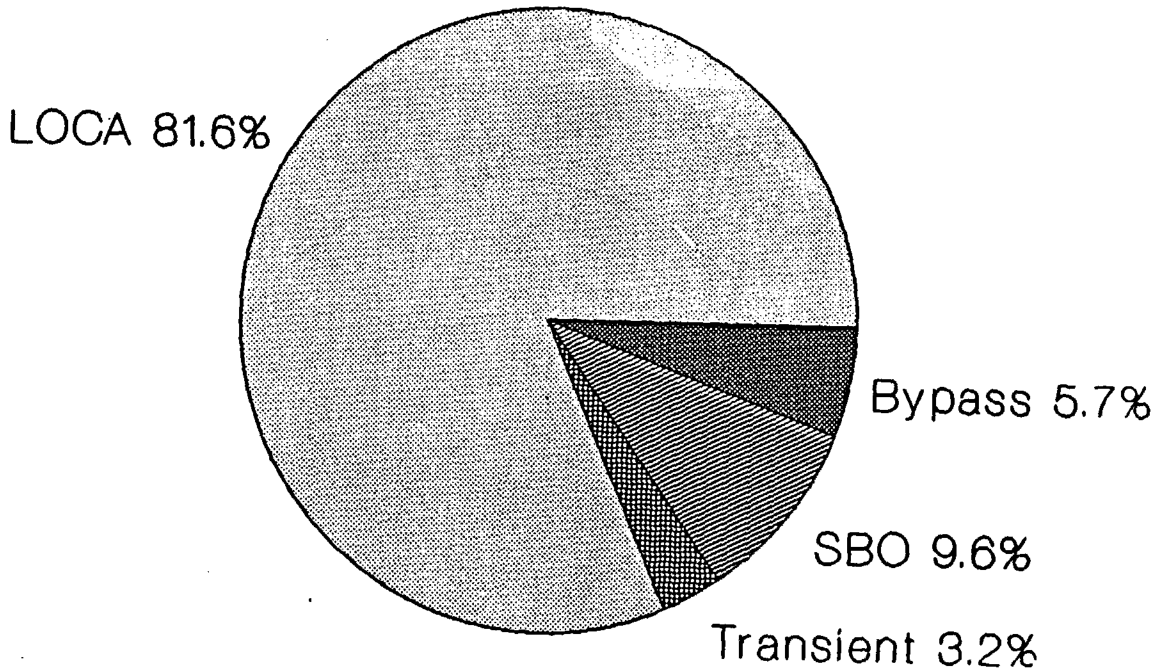


Figure 5.1-5 Major Contributors (Plant Damage States) to Mean Early Fatality Risk at Zion - Internal Initiators

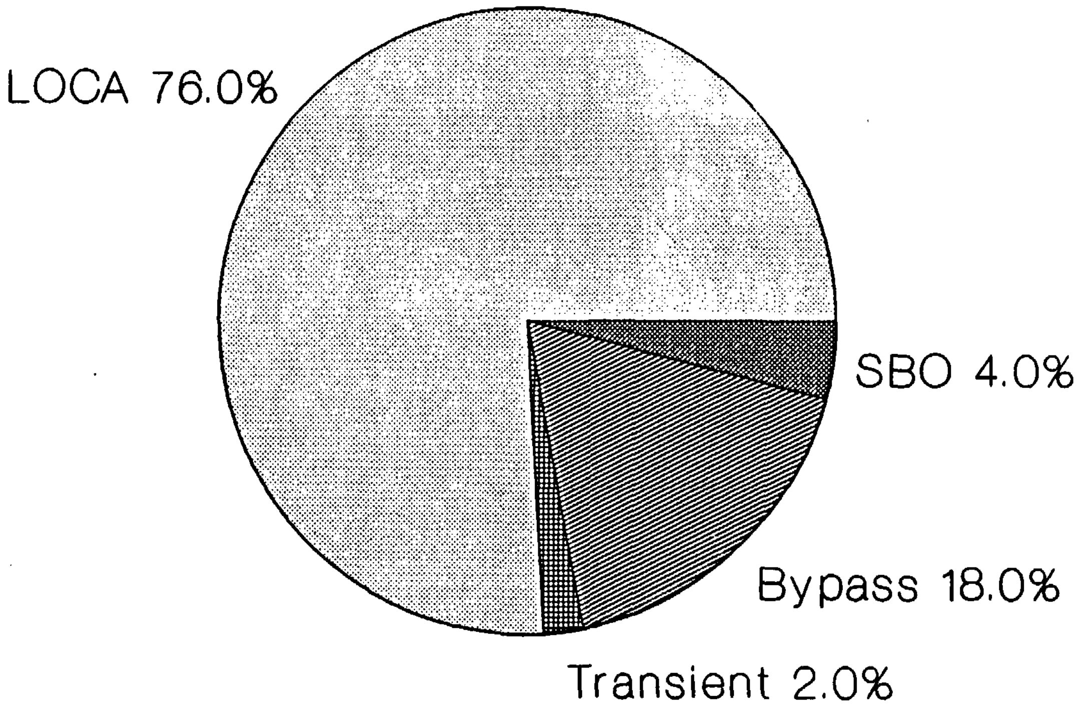


Figure 5.1-6 Major Contributors (Plant Damage States) to Mean Latent Cancer Fatality Risk (within 50 miles) at Zion - Internal Initiators

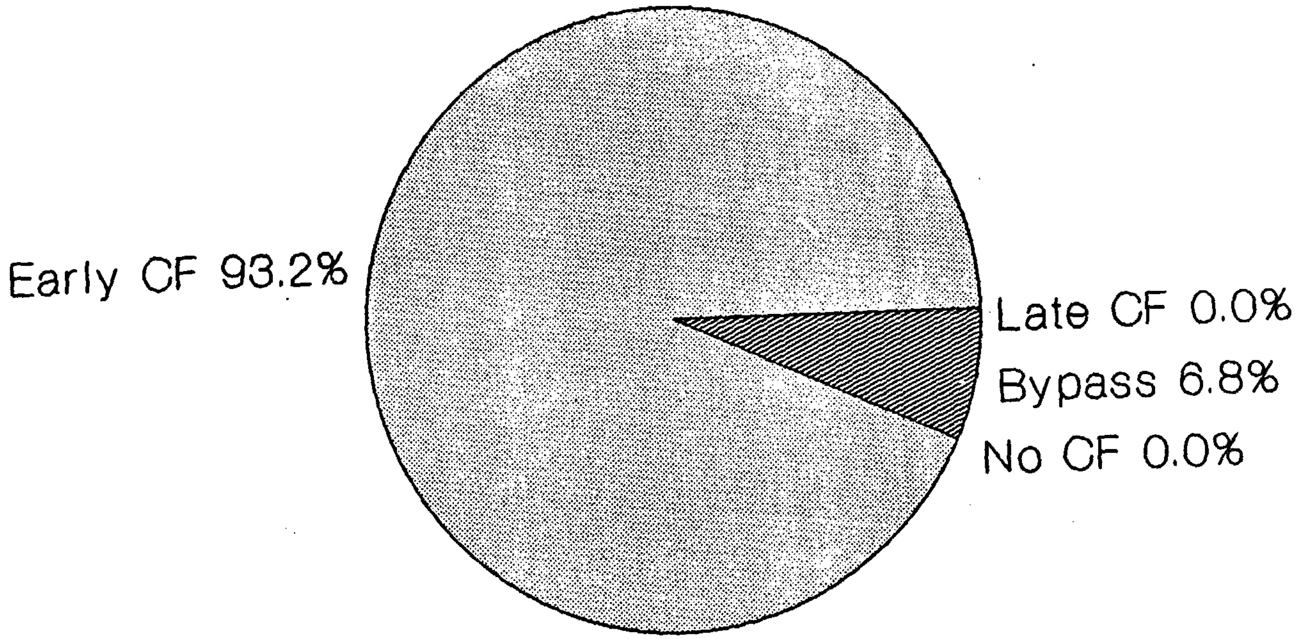


Figure 5.1-7 Major Contributors (Accident Progression Bins) to Mean Early Fatality Risk at Zion - Internal Initiators

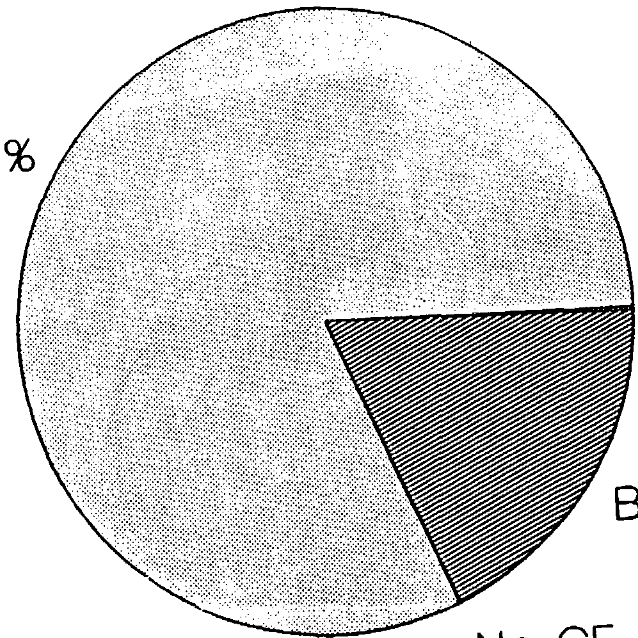


Figure 5.1-8 Major Contributors (Accident Progression Bins) to Mean Latent Cancer Fatality Risk (within 50 miles) at Zion - Internal Initiators

A regression analysis was performed using the Partial Rank Correlation Coefficient (PRCC) as a measure of importance. Table 5.1-4 summarizes the results of the regression analysis. Six variables were identified as important contributors to the uncertainties of the various risk estimations. These six variables were selected based on calculated PRCC values greater than 0.3 for several risk indices.

Two accident frequency issues were identified as important. The frequencies of these issues are directly related to the total core damage frequency. The failure of CCWS means that there is no seal cooling nor emergency coolant injection. Late AC power recovery negatively contributes to the latent cancer fatalities. One containment issue was identified as important. The frequency of in-vessel steam explosion (alpha-mode containment failure) was found to be an important contributor to the uncertainty of all the risk indices. There were three source term parameters (or source term issues) identified as being important in terms of their contribution to the risk uncertainties.

The regression analysis was repeated for four evacuation scenarios. The results are presented in Appendix D.4.

5.1.5 Impact on Risk of the Service Water and Component Cooling Water Upgrade

In April 1989, the licensee for Zion Unit 1, Commonwealth Edison Co., made commitments to the NRC to make plant and procedure changes to address the major contributor to the core damage frequency, the development of an unmitigated reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) caused by a loss of component cooling water (CCW) or service water (SW) system.[2] The status of these commitments was provided in another letter to the NRC[3] in which the status was reported as:

1. Zion Station provided an auxiliary water supply to each charging pump's oil cooler via the fire protection system (FPS). Hoses, fittings and tools are locally available at each unit's charging pump area, allowing for immediate hookup to existing taps on the oil coolers, if required. This action was completed in April 1989.

2. A formal procedure change was made to AOP 4.1, entitled "Loss of Component Cooling Water," on April 12, 1989, providing instruction to the operators to align emergency cooling to the centrifugal charging pumps. Specific instructions are included for each charging pump with a diagram of the lube oil cooling valves and piping.

3. As of May 1990, the new heat-resistant RCP seal o-rings were unavailable from Westinghouse. Therefore, during the latest unit outages, the RCP seal o-rings were not replaced. When the new o-rings are available, the existing o-rings will be changed when each pump is disassembled for routinely scheduled seal maintenance.

Table 5.1-4
Risk Importance Analysis: Partial Rank Correlation Coefficient

Variable Description	Early Fatalities	Early Injuries	Probability of One Fatality	Latent Cancer Fatalities (50 Mi)	Latent Cancer Fatalities (Entire Region)	Safety Goals: Individual Early Fatalities	Safety Goals: Societal Individual Latent Cancer Fatalities	Pop. Dose (50 Mi)
CCWS Initiating Event		0.49	0.55					
CCWS Hardware Recovery	0.36	0.58	0.67	0.40	0.45	0.37	0.49	0.45
Early Containment[1] Failure	0.45	0.56	0.62	0.60	0.62	0.50	0.56	0.62
FCOR[2]	0.42	0.42	0.36	0.56	0.56	0.46	0.48	0.55
FCONV[3]	0.40					0.41		
FISG*FOSG[4]	0.45	0.35		0.74	0.68	0.47	0.47	0.69
R^2	0.67	0.74	0.78	0.79	0.78	0.70	0.71	0.78

Notes:
1 - Early containment failure was most influenced by steam explosions and direct containment heating.
2 - Fraction of initial inventory of nuclide group release from the fuel in-vessel.
3 - Containment transport fraction for releases prior to or at vessel breach.
4 - Fraction of fuel release to the environment through the steam generator.

The scope of the sensitivity study presented in this section is to examine how the plant and procedure changes that provide an auxiliary source of cooling water to the charging pump oil coolers might impact the CDF and risk estimates. The base case Zion analysis, as documented in Reference 4, models the failure of the component cooling water (CCW) or service water (SW) systems as a challenge to the integrity of the RCP seals. The failure of CCW causes overheating of the charging pumps that supply seal injection water. CCW also provides the RCP seal cooling water. The CCW system uses the SW system as its heat sink.

The failure of both seal cooling and seal injection (via failure of the charging pumps due to overheating) places the RCP seals in jeopardy of failure. The failure model for the RCP seals states the mean probability of having a seal LOCA, given a loss of seal injection and seal cooling as 0.73. Failure of seal injection or seal cooling separately does not challenge seal integrity.

The changes made by Commonwealth Edison Co. are designed to break the common-cause failure mechanism represented by the CCW and SW systems. Since exact design, procedure, and training changes were not available, the following assumptions were made in the analysis:

1. The auxiliary water supply provided by the FPS is connected such that charging pump oil cooler heat is rejected without depending on the rest of the CCW system or any of the SW system.

2. The FPS does not depend on any support from the SW system.

3. The failure to provide an auxiliary water supply to the charging pump oil coolers is dominated by the operator actions to properly determine that such action is necessary and the proper execution of the task and not by hardware failures of the FPS.

4. The operator action to provide an alternative source of cooling water is comparable to the operator action to diagnose the need for feed and bleed cooling and manually starting the high-pressure injection system; therefore, a comparable failure probability may be used. To account for the large uncertainty in this value, an error factor of 10 is deemed appropriate. The failure probability assigned for the failure to provide an alternative source of water to the charging pump oil coolers is a lognormal distribution with a mean of 0.01 and an error factor of 10.

The base case analysis gave credit for the operators recovering certain CCW and SW failures by shedding unnecessary loads, starting standby pumps, and isolating piping sections where possible. These actions were modeled in the event trees as top event RE and specifically conditional split fractions (CSFs) RE1 and RE2. These CSFs were assigned a failure probability of 0.13.

The provision of an alternative water supply to the charging pump oil coolers was modeled as a change to the recoverability of the CCW and SW systems. Thus, RE1 and RE2 were changed from 0.13 and 0.01.

The 203 highest frequency accidents and 58 plant damage states were requantified by INEL and a new Latin hypercube uncertainty analysis was performed using the failure probability data described above. Table 5.1-5 shows the comparison of the plant damage state results between the base case and this sensitivity analysis. The change in the core damage frequency from a mean value of 3.4E-4 to 6.2E-5 per reactor year represents a decrease of about 81 percent.

Table 5.1-5 Plant Damage State Comparison

	Base Case		Sensitivity Case	
Plant Damage State	Frequency per Reactor Year	%	Frequency per Reactor Year	%
LOCAs	3.1E-4	93.2	3.9E-5	62.7
ATWS and transients	1.4E-5	4.0	1.4E-5	21.9
Station blackouts	9.3E-6	2.8	9.3E-6	15.0
Bypass	2.6E-7	--	2.6E-7	0.4
Total	3.4E-4	100.0	6.2E-5	100.0

The impact of the sensitivity analysis was a significant reduction in the mean core damage frequency, which was obtained by reducing the plant damage states involving CCW and SW induced seal LOCAs. Other plant damage states remained unchanged. Thus, in the sensitivity study, CCW and SW induced seal LOCAs contribute only 24 percent to the mean core damage frequency compared with 86 percent in the base case. This reduction in LOCAs means that other plant damage states such as bypass and station blackout (SBO) become larger contributors to the lower mean core damage frequency. The contribution of SBO accidents increased from about 2 to percent over 10 percent in the sensitivity study. Bypass accidents contributed 0.4 percent to the mean core damage frequency in the sensitivity study compared with a negligible contribution in the base case. As station blackout and bypass accidents tend to be more challenging (in terms of containment performance) than LOCAs, the risk estimates should not be reduced by as large a fraction as the mean core damage frequency. Table 5.1-6 presents new mean risk estimates based on the sensitivity study and compares them with original base case risk taken from Table 5.1-1. The results in Table 5.1-6 indicate that the risk measures did in fact decrease by smaller fractions than the mean core damage frequency. The early fatality risk decreases by about 70 percent and the latent cancer fatality risk by about 60 percent.

The new mean risk estimates were not obtained by performing a completely new uncertainty analysis as was done for the accident frequency analysis at INEL. The mean risk estimates in Table 5.1-6 were obtained by using mean risk values conditional on the occurrence of the various plant damage states. The conditional mean risk measures were simply multiplied by the new mean frequencies in Table 5.1-5 and summed to obtain the new risk estimates. This approach is not as rigorous as a complete requantification of the uncertainty analysis in which new distributions for the risk measures would have been obtained. However, the approach is straightforward and gives a reasonable estimate of how the mean risk estimates would be reduced by the changes made by Commonwealth Edison Company to eliminate the common-cause failure mechanism represented by the CCW and SW systems.

Table 5.1-6 Comparison of Mean Risk Values

Risk Measure (Per Reactor Year)	Base Case	Sensitivity Case
Early Fatalities	1.1E-4	3.0E-5
Total Fatalities (Entire Region)	2.4E-2	8.8E-3
Total Fatalities (50 mi)	1.1E-2	3.7E-3
Individual Early Fatality Risk (0 - 1 mile)	1.0E-8	3.0E-9
Individual Latent Cancer Fatality Risk (0 - 10 miles)	1.1E-8	2.8E-9
Population Dose (person-rem) (50 mi)	5.5E+1	1.8E+1
Population Dose (person-rem) (Entire Region)	1.4E+2	4.6E+1

The results of the sensitivity study given in Table 5.1-6 were also included in Figures 5.1-2 through 5.1-4 for completeness.

5.2 References

1. H. J. C. Kouts, et al., "Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)," NUREG-1420, August 1990.

2. Letter from Cordell Reed, Commonwealth Edison Co. (CECo) to T. E. Murley, NRC, dated March 13, 1989.

3. Letter from R. A. Chrzanowski, CECo, to NRC Document Control Desk, August 24, 1990.

4. M. B. Sattison and K. W. Hall, "Analysis of Core Damage Frequency: Zion Unit 1," Idaho National Engineering Laboratory, NUREG/CR-4550, Vol. 7, Rev. 1, EGG-2554, May 1990.

6. INSIGHTS AND CONCLUSIONS

Reactor Coolant Pump Seal LOCA. Accidents involving induced reactor coolant pump seal LOCAs dominate the estimated core damage frequency. After completion of the draft revision 1 analysis for Zion Unit 1, Commonwealth Edison made commitments to the NRC to make plant and procedure changes to address the major contributor to the core damage frequency. The impact of these changes would be a significant reduction in the core damage frequency. The Zion risk estimates reported in this volume do not reflect these changes. However, a sensitivity study was performed to assess the impact of the changes in the mean Zion risk estimates. The result of the sensitivity analysis was a reduction in all of the mean risk measures.

Depressurization of the RCS. Depressurization of the RCS before the vessel fails is important in reducing the loads placed upon the containment at vessel breach and in arresting core damage before VB. For accidents in which the RCS is at the PORV setpoint pressure during core degradation, the effective mechanisms for pressure reduction are temperature-induced failure of the hot leg or surge line, temperature-induced failure of the RCP seals, and the sticking open of the PORVs.

Containment Failure. If a core damage accident proceeds to the point where the lower head of the reactor vessel fails, the containment is unlikely to fail at this time. This is partially due to the depressurization of the RCS before vessel failure and partially due to the strength of the Zion containment relative to the loads expected. If the containment does fail, it is more likely to fail many hours after VB than at VB. The mode and time of failure depends upon the availability of CHR. If CHR is recovered within a day or so, basemat meltthrough is the most probably failure mode. If CHR is not recovered within days, an overpressure failure is possible.

Bypass Accidents. Bypass accidents can potentially result in a large early release and in the Zion Analysis (refer to Volume 3 of this report) were found to be important contributors to risk. However, because of the low frequency of bypass events when compared with the core damage frequency they were not found to be dominant contributors to risk at Zion.

Fission Product Releases. There is considerable uncertainty in the release fractions for all types of accidents. For most accidents, the central portions of the release fraction distributions are below most release fraction estimates made several years ago. While the upper portions of the release fraction distributions are comparable with the values of the RSS,[3] many of these distributions now extend to release fractions several orders of magnitude lower than those of the RSS.

Uncertainty in Risk. Considerable uncertainty is associated with the risk estimates produced in this analysis. The largest contributors to this uncertainty are the initiating events, early containment failure modes and the uncertainty in some of the parameters that determine the magnitude of the fission product release to the environment.

Comparison of the Safety Goals. For both distributions for individual fatality probability, the 95th percentile value for annual risk falls more than an order of magnitude below the safety goal.

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11. ABSTRACT (200 words or less)

In support of the Nuclear Regulatory Commission's (NRC's) assessment of the risk from severe accidents at commercial nuclear power plants in the U.S. reported in NUREG-1150, revised calculation of the risk to the general public from severe accidents at the Zion Power Station, Unit 1 has been completed. This power plant, located on the western shore of Lake Michigan on the outskirts of Zion, is operated by the Commonwealth Edison Company.

The emphasis in this risk analysis was not on determining a "so-called" point estimate of risk. Rather, it was to determine the distribution of risk, and to discover the uncertainties that account for the breadth of this distribution. Off-site risk initiation by events, both internal to the power station and external to the power station was assessed.

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