NUREG–1150
Vol. 2

Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants

Appendices A, B, and C

Final Report

U.S. Nuclear Regulatory Commission

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ABSTRACT

This report summarizes an assessment of the risks from severe accidents in five commercial nuclear power plants in the United States. These risks are measured in a number of ways, including: the estimated frequencies of core damage accidents from internally initiated accidents and externally initiated accidents for two of the plants; the performance of containment structures under severe accident loadings; the potential magnitude of radionuclide releases and offsite consequences of such accidents; and the overall risk (the product of accident frequencies and consequences). Supporting this summary report are a large number of reports written under contract to NRC that provide the detailed discussion of the methods used and results obtained in these risk studies.

This report was first published in February 1987 as a draft for public comment. Extensive peer review and public comment were received. As a result, both the underlying technical analyses and the report itself were substantially changed. A second version of the report was published in June 1989 as a draft for peer review. Two peer reviews of the second version were performed. One was sponsored by NRC; its results are published as the NRC report NUREG–1420. A second was sponsored by the American Nuclear Society (ANS); its report has also been completed and is available from the ANS. The comments by both groups were generally positive and recommended that a final version of the report be published as soon as practical and without performing any major reanalysis. With this direction, the NRC proceeded to generate this final version of the report.

Volume 2 of this report contains three appendices, providing greater detail on the methods used, an example risk calculation, and more detailed discussion of particular technical issues found important in the risk studies.

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A.1 Introduction and Overview

A.1.1 Introduction

This appendix provides an overview of the NUREG-1150 risk analysis process, describing the different steps in the calculational process and the interrelationships among steps. This summary has been written for a reader familiar with risk analysis but does not discuss the subtleties and complexities of the methods used to perform the various analysis steps. The reader seeking a more comprehensive discussion is directed to References A.1 and A.2.

The analysis methods used in NUREG-1150 were selected or developed to satisfy some special objectives of the project. In particular, the following were important considerations in the selection of methods:

- The need to perform quantitative uncertainty analyses (considering both data and modeling uncertainties) as part of the calculations;
- The need to make explicit use of the data base of severe accident experimental and calculational information generated by NRC's contractors and the nuclear industry, which resulted in the development of more detailed accident progression analysis models and the use of formal methods for eliciting expert judgment;
- The ability to readily assess the impact of postulated modifications to the studied plants;
- The ability to calculate and display intermediate results and a detailed breakdown of the risk results, providing traceability throughout the computations; and
- Computational practicality.

The selection of the methods also benefited from experience obtained in conducting the analyses presented in the first draft version of NUREG-1150 (Ref. A.3) and supporting contractor reports (Refs. A.4, A.5, and A.6), and the reviews of these reports (Refs. A.7, A.8, and A.9).

The remainder of this appendix discusses the individual steps in the NUREG-1150 risk analysis process. Section A.1.2 provides an overview of the process, while Sections A.2 through A.8 describe individual steps in greater detail. Section A.2 contains a separate discussion of the methods used in the accident frequency analysis of internal events for the Surry, Sequoyah, Peach Bottom, and Grand Gulf plants; the internal-event analysis for the Zion plant; and the external-event analysis for the Surry and Peach Bottom plants. Since the accident progression, source term, and offsite consequence analysis methods did not significantly differ among the plants or for internal and external events, the discussions in Sections A.3 through A.8 are applicable to all five plants and for both internally and externally initiated accidents.

As noted above, the risk analyses of NUREG-1150 included the performance of quantitative uncertainty analysis, considering both data and modeling uncertainties. Section A.6 discusses how this uncertainty analysis was introduced and applied in the NUREG-1150 risk analyses. The methods by which expert judgments were obtained for use in the risk analyses are discussed in Section A.7.

The remaining sections of this appendix have been extracted from the contractor reports underlying NUREG-1150. Some editorial modifications have been made to improve the flow of the text.

A.1.2 Overview of Risk Analysis Process*

The risk analyses performed in NUREG-1150 have five principal steps (as shown in Fig. A.1): (1) accident frequency (systems) analysis; (2) accident progression, containment loadings, and structural response analysis; (3) radioactive material transport (source term) analysis; and (4) offsite consequence analysis. A fifth analysis part, risk calculation, combines and analyzes the information from the previous four steps.

The transfer of information between analysis steps is critical; thus, three interfaces are illustrated in Figure A.2. Each distinct continuous line that can be followed from the left of the illustration to the box marked

*This section adapted, with editorial modification, from Chapter 2 of Reference A.2.

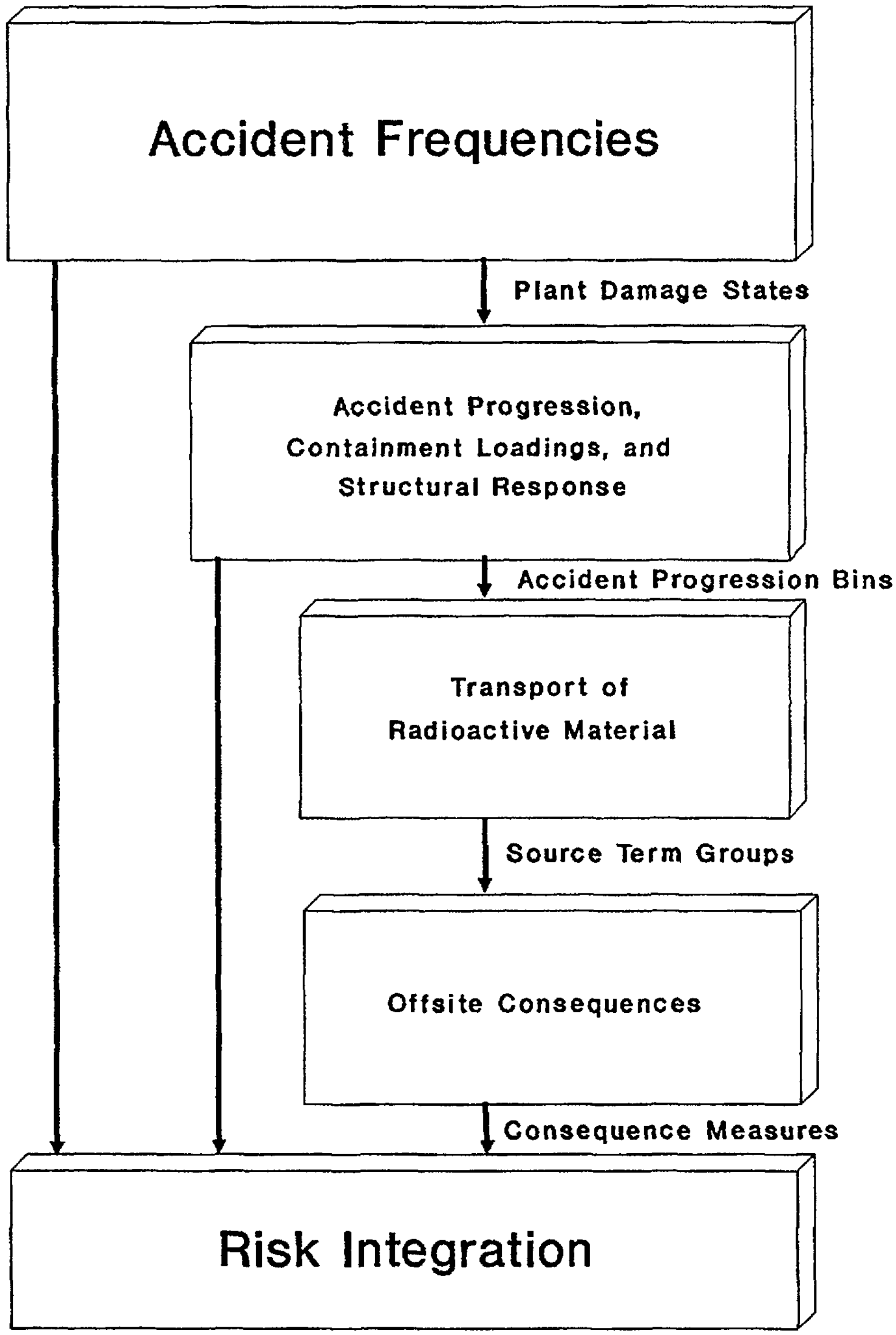


Figure A.1 Principal steps in NUREG-1150 risk analysis process.

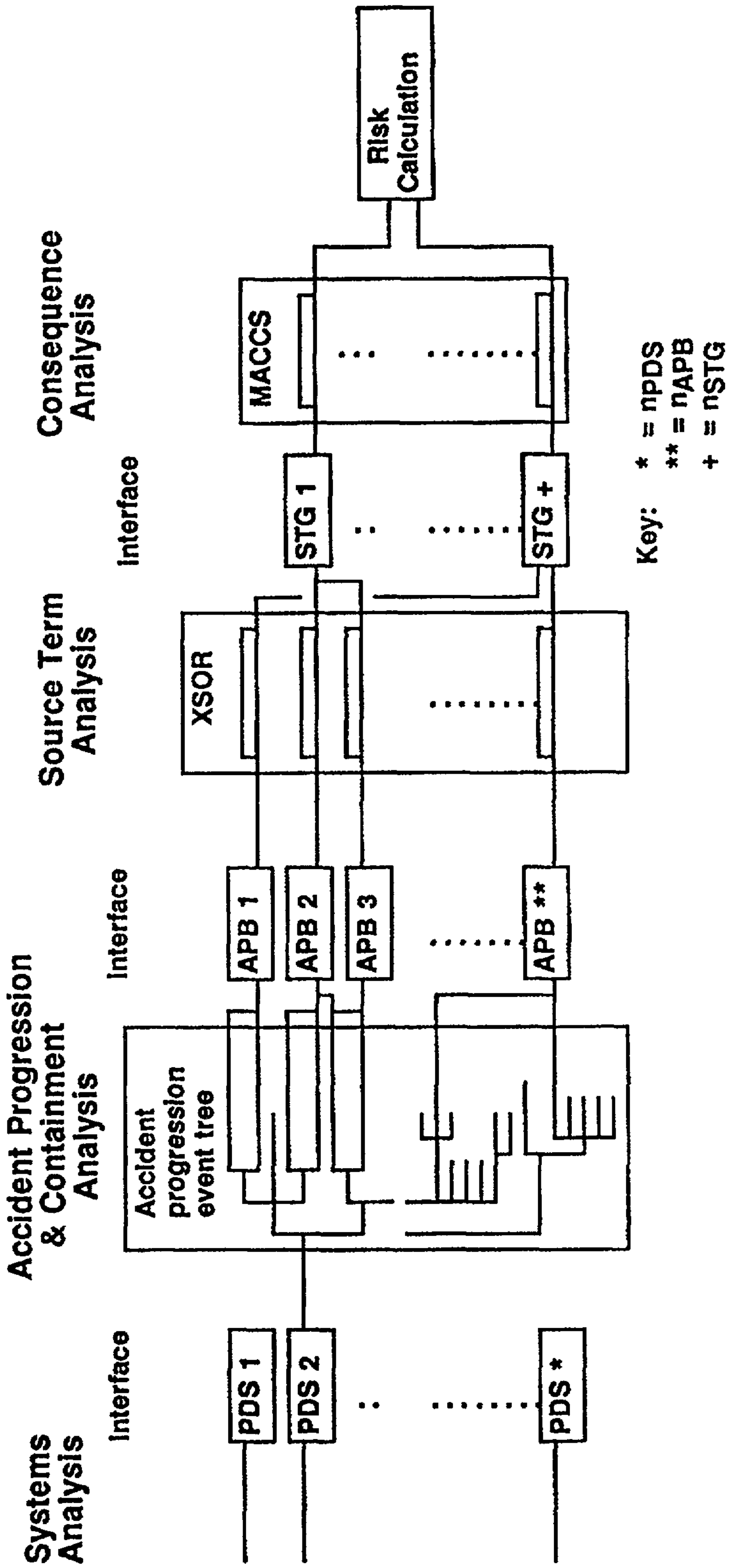


Figure A.2 Interfaces between risk analysis steps.

"Risk Calculation" corresponds to a distinct group of accidents with a particular set of characteristics in each analysis step. Each of the analysis steps produces results that are useful for understanding the plant's response to that stage or aspect of the accident, and each part also provides an ingredient necessary to the calculation of overall risk.

Each of the analysis steps is supported by a variety of information sources and supporting analyses. An ideal study might use comprehensive mechanistic models to calculate the entire sequence of events leading to core damage, release of radioactive material, and exposure to the public for each possible accident. However, a large variety of accidents will be possible because there are a variety of initiating events and because "random" events occurring during the accident can change the progress of the accident. It is presently neither practical (too many possible accidents to follow) nor possible (mechanistic models do not exist for many parts of the process) to conduct such a study. As such, PRAs have relied on the use of a variety of simple models and calculational tools to substitute where integrated mechanistic calculations were not available. Some of the tools assemble results from several existing mechanistic calculations to yield a more comprehensive result. Other models provide simplified mechanistic models with as much of the detailed analysis as possible but which are able to efficiently calculate results for the wide range of conditions needed to examine the set of possible accidents.

The accident frequency analyses identify the combination of events that can lead to core damage and estimate their frequency of occurrences. Potential accident initiating events (including external events for two plants) were examined and grouped according to the subsequent system response required. Once these groups were established, accident sequence event trees were developed that detailed the relationships among systems required to respond to the initiating event in terms of potential system successes and failures. The front-line systems in the event trees, and the related support systems, were modeled with fault trees or Boolean logic expressions as required. The core damage sequence analysis was accomplished by appropriate Boolean reduction of the fault trees in the system combinations (the accident sequences) specified by the event trees. This Boolean reduction provides the logical combinations of failures (the cut sets) that can lead to core damage. Once the important failure events are identified, probabilities are assigned to each event and the accident sequence frequencies are quantified. The accident sequence cut sets are then regrouped into plant damage states in which all cut sets are expected to result in a similar accident progression. Variations in these frequencies are explicitly considered in an uncertainty analysis using a structured Monte Carlo approach.

The NUREG-1150 accident frequency analyses have the following products:

- The total core damage frequency from internal events and, where estimated, for external events;
- The definitions and estimated frequencies of plant damage states; and
- The definitions and estimated frequencies of accident sequences.

Importance measures, including risk reduction, risk increase, and uncertainty measures, have also been assessed in NUREG-1150 accident frequency analyses.

The accident progression, containment loadings, and structural response analysis investigated the physical processes affecting the core after an initiating event occurs. In addition, this part of the analysis tracked the impact of the accident progression on the containment building. The principal tool used in NUREG-1150 for delineating and characterizing the possible scenarios in this study was the accident progression event tree. The event tree is a computational tool used to assemble a large variety of analysis results and data to yield a comprehensive result (in terms of the characteristics of alternative failure modes of the containment building and related probabilities) for each of the many accidents. The event tree is particularly suited for the study of processes that are not completely understood, permitting the study of alternative phenomenological models. The output of the accident progression event tree (APET) was a listing of numerous different outcomes of the accident progression. As illustrated in Figure A.2, these outcomes were grouped into accident progression bins (APBs) that, analogous to plant damage states, allow the collection of outcomes into groups that are similar in terms of the characteristics that are important to the next stage of the analysis, in this case source term estimation. Once the APET is constructed, the probabilities of the paths through the APET were evaluated by a computational tool, EVNTRE (Ref. A.10). EVNTRE also performs the function of grouping similar outcomes into bins. The accidents that are grouped into a single bin are similar enough in terms of timing, energy, and other characteristics that a single source term estimate suffices for estimating the radiological impact of any of the individual accidents within that bin.

The qualitative product of the containment loadings analysis is a set of accident progression bins. Each bin consists of a set of event tree outcomes (with associated probabilities) that have a similar effect on the subsequent portion of the risk analysis, analysis of radioactive material transport. Quantitatively, the product consists of a matrix of conditional failure probabilities, with one probability for each combination of plant damage state and accident progression bin. These probabilities are in the form of probability distributions, reflecting the uncertainties in accident processes.

The next step in the risk calculation was the source term analysis. Once again a relatively simple model was developed to allow consideration of alternative inputs and the assembly of information from many sources. In this study, a plant-specific model was developed for each of the plants, with the suffix SOR built into the code name (shown as XSOR in Fig. A.2) (Ref. A.11). For example, SURSOR is the source term model for the Surry plant. The results of the source term analysis were release fractions for groups of chemically similar radionuclides for each accident progression bin. As with the previous analyses, a large number of results were calculated, too many for direct transfer to the next part. The interface in this case is accomplished through the calculation of "partitioned" source term groups. The large number of XSOR results are assessed and grouped in terms of their important parameters (i.e., early health threat potential and latent health threat potential) and by similarity of accident progression as it affects warning times to the surrounding population.

The product of this step in the NUREG-1150 risk analysis was the estimate of the radioactive release of a set of source team groups, each with an associated energy content, time, and duration of release.

The offsite consequence analysis in this study was performed with the MACCS (MELCOR Accident Consequence Code System) computer code, Version 1.5 (Ref. A.12). This code has been developed as a replacement for the CRAC2 code (Ref. A.13), which had previously been used by the NRC and others to estimate consequences for nuclear power plant risk analyses and other studies. The MACCS calculations were performed for each of the partitioned source terms defined in the previous step.

The product of this part of the analysis is a set of offsite consequence measures for each source term group. For NUREG-1150, the specific consequence measures discussed include early fatalities, latent cancer fatalities, population dose (within 50 miles and total), and two measures for comparison with NRC's safety goals (average individual early fatality probability within 1 mile and average individual latent fatality probability within 10 miles) (Ref. A.14).

The final stage of the risk analysis was the assembly of the outputs of the first four steps into an expression of risk. As shown in Figure A.2, the calculation of risk can be written in terms of the outputs of the individual steps in the analyses:

$$\mathrm{Risk}_{ln} = \Sigma_h \Sigma_i \Sigma_j \Sigma_k \, f_n(\mathrm{IE}_h) \, P_n(\mathrm{IE}_h \rightarrow \mathrm{PDS}_i) \, P_n(\mathrm{PDS}_i \rightarrow \mathrm{APB}_j) \, P_n(\mathrm{APB}_j \rightarrow \mathrm{STG}_k) \, C_{lk}$$

where:

Risk_{ln} = Risk of consequence measure l for observation n (consequences/year);

$f_n(\mathrm{IE}_h)$ = Frequency (per year) of initiating event h for observation n;

$P_n(\mathrm{IE}_h \rightarrow \mathrm{PDS}_i)$ = Conditional probability that initiating event h will lead to plant damage state i for observation n;

$P_n(\mathrm{PDS}_i \rightarrow \mathrm{APB}_j)$ = Conditional probability that PDS_i will lead to accident progression bin j for observation n;

$P_n(\mathrm{APB}_j \rightarrow \mathrm{STG}_k)$ = Conditional probability that accident progression bin j will lead to source term group k for observation n; and

C_{lk} = Expected value of consequence measure l conditional on the occurrence of source term group k.

In considering this equation, the reader should note that the frequency and probabilities noted are in the form of distributions, rather than single-valued. A specialized Monte Carlo (Latin hypercube sampling) technique is used to generate these distributions (Ref. A.15). As discussed in Section A.5, however, the consequence values used were expected values, reflecting variability in meteorology only.

Because of the large information-handling requirements of all these analysis steps, computer codes have been used to manipulate the data. Figure A.3 illustrates the computer codes used in the risk assembly process in this study. The purpose of each of these codes will be discussed in the following sections.

A.2 Accident Frequency Analysis Methods

A.2.1 Internal-Event Methods for Surry, Sequoyah, Peach Bottom, and Grand Gulf*

The accident frequency analysis for the Surry, Sequoyah, Peach Bottom, and Grand Gulf plants consisted of 10 principal tasks. These are illustrated in Figure A.4. This section briefly discusses each major task and the interrelationships among tasks. These tasks are discussed in greater detail in Reference A.1.

The principal steps in the accident frequency analysis of the Surry, Sequoyah, Peach Bottom, and Grand Gulf plants were:

- Plant familiarization analysis,
- Accident sequence initiating event analysis,
- Accident sequence event tree analysis,
- Systems analysis,
- Dependent and subtle failure analysis,
- Human reliability analysis,
- Data base analysis,
- Accident sequence quantification analysis,
- Plant damage state analysis, and
- Uncertainty analysis.

Each of these steps will be discussed below.

Plant Familiarization Analysis

The initial task of this analysis was to develop familiarity with the plant, forming the foundation for the development of plant models in subsequent tasks. Information was assembled using such sources as the Final Safety Analysis Report, piping and instrumentation diagrams, technical specifications, operating procedures, and maintenance records, as well as a plant site visit to inspect the facility and clarify and gather information from plant personnel. One week was spent in the initial plant visit. Regular contact was maintained with the plant staff throughout the course of the study. The analyses discussed in NUREG-1150 reflect each plant's status as of approximately March 1988.

At the conclusion of the initial plant visit, much of the information required to perform the remaining tasks had been collected and discussed in some detail with utility personnel so that the analysis team was familiar with the design and operation of the plant. Subsequent plant contacts were used to verify the information obtained and to identify plant changes that occurred during the analysis.

Accident Sequence Initiating Event Analysis

The next task was to identify potentially important initiating events and determine the plant systems required to respond to these events. Initiating events of importance were generally those that led to a need

*This section extracted, with editorial modification, from Chapter 1 of Reference A.1.

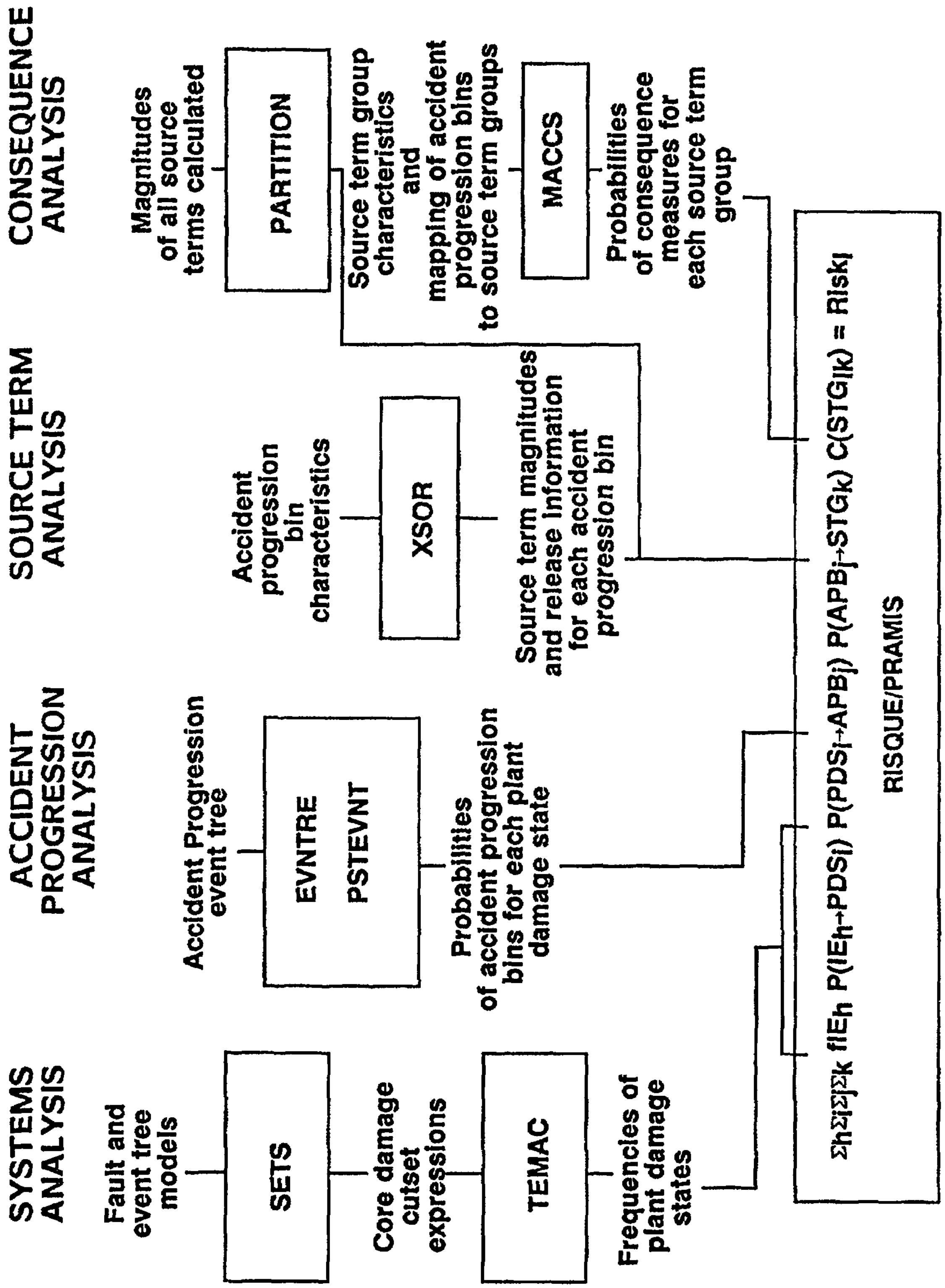


Figure A.3 Models used in calculation of risk.

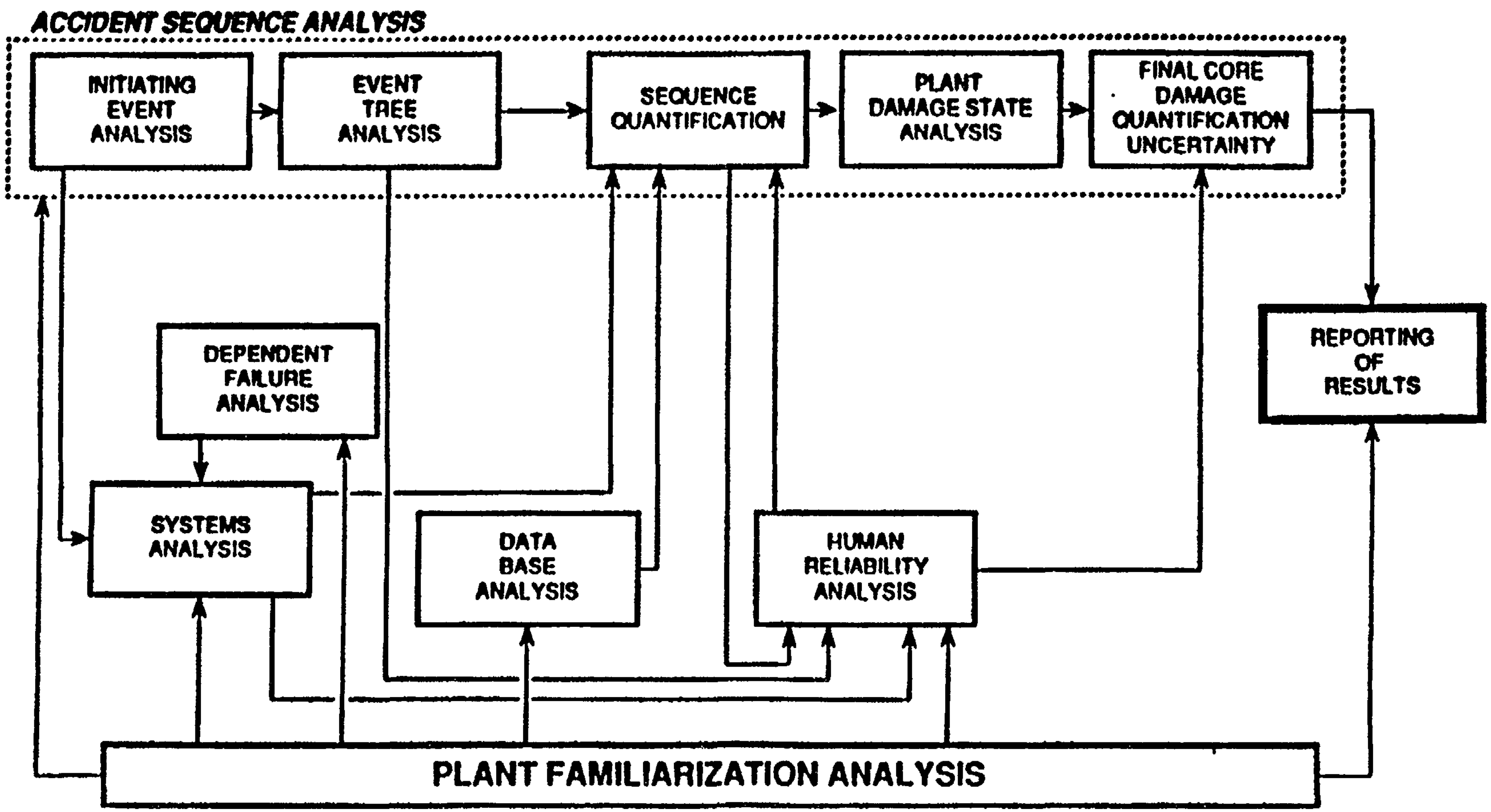


Figure A.4 Steps in accident frequency analysis of Surry, Sequoyah, Peach Bottom, and Grand Gulf.

for plant trip and removal of decay heat by plant safety systems. The analysis explicitly included initiating events due to failures in support systems, such as ac power or component cooling water. This analysis included several steps:

- Identification of initiating events to be included in the analysis by review of previous PRAs and plant data, including review of unusual or unique events that might have affected the specific plant;
- Screening of initiating events on frequency of occurrence (and elimination from further consideration events of very low frequency);*
- Identification of functions required to successfully prevent core damage by review of plant design and operational information;
- Identification of the "front-line" systems (e.g., emergency core cooling systems) performing the above functions by review of plant design and operational information;
- Identification of the support systems (e.g., ac power, component cooling water) necessary for operation of the front-line systems by review of plant design and operational information;
- Delineation of success criteria for each front-line system responding to each initiating event by review of available data and performance of additional calculations (e.g., as described in Ref. A.16); and
- Grouping of initiating events, based on similarity of system response.

At the conclusion of this task, the number and type of event trees to be constructed and the systems to be modeled had been identified. Thus, the scope of the modeling effort in subsequent tasks was defined.

Accident Sequence Event Tree Analysis

In this task, accident sequences leading to core damage were defined by constructing event trees for each initiating event group. In general, separate event trees were constructed for each group.

System event trees that included the systems responding to each initiating event group as defined in the accident sequence initiating event analysis were constructed. The event tree structure reflected system interrelationships and aspects of accident phenomenology that determined whether or not the sequences led to core damage. Phenomenological information, such as containment failure effects that potentially impact core cooling or other systems, was obtained from the staff involved in the accident progression and containment loadings analysis.

At the conclusion of this task, models that identified all those accident sequences to be assessed in the accident sequence quantification analysis task had been constructed.

Systems Analysis

In order to estimate accident sequence frequencies, the success and failure probabilities must be determined for each question (or "top event") on the system event trees. Thus, the important contributors to failure of each system must be identified, modeled, and quantified. Although the event tree questions were usually phrased in terms of system success, the fault tree top events were formulated in terms of system failure. With this transformation in mind, fault trees were constructed that reflected the success criteria specified in the three previous tasks. Each success criterion was transformed into a failure criterion that was developed for all the front-line systems included in the event trees. If these front-line systems depended on support systems, such as electric power or service water, then models were also developed for those systems. In a subsequent task, the support system trees were merged with the respective front-line system fault trees to describe the ways, including support system faults, that the undesired event may occur. Thus support system dependencies were included systematically and automatically in the quantification process.

*The reader is cautioned that the screening analysis performed and the degree of system modeling detail performed in this study were based on the designs of each of the plants. Thus, it should not be inferred that such assessments necessarily apply to other plants.

The majority of the models in this study were detailed fault trees. These were supplemented with a few simplified fault trees, Boolean equations, or black box models (event probabilities or failure rates), based on guidelines that considered such things as the relative importance of the system, complexity of the system, dominant failure modes, availability of data, etc. Selection of the level of modeling detail for each system was one of the most important steps in the analysis and did, to a great extent, determine the amount of effort required to complete the accident frequency analysis. All the front-line fluid systems required detailed fault trees, as did a few critical support systems. The outputs of this task were models for each event found in the event trees.

This task interfaced with the human reliability, dependent and subtle failure, and data base analyses. Human errors associated with test and maintenance activities and certain responses to and recovery from accident situations were modeled directly in the fault trees. Dependent and subtle failures as a result of system interdependencies and component common-cause failures were also directly modeled. The fault trees were developed to a level of detail consistent with the data base used for quantifying failure probabilities.

Dependent and Subtle Failure Analysis

Nuclear power plants are sufficiently complex that dependent and subtle failures can be of significant importance in estimating the core damage frequency. Failures that are buried in the depths of the design and operation of the plant are often not easily identifiable. Dependent and subtle failures were categorized separately because they are very distinct types of failures.

The dependent failures included:

- Direct functional dependencies that involve initiators, support systems, and shared equipment; and
- Common-cause faults involving failures that can affect multiple components.

The subtle failures included:

- Peculiar or unusual interactions of system design and interfaces, or system component operation; and
- Subtle interactions identified in previous studies and PRAs or by PRA experts.

The dependent failures were identified in the accident sequence analysis. When the subtle failures were identified, they were added to the sequence event trees or fault trees, as appropriate. In rare cases, such events were modeled by changes to failure data or the cut-set expressions.

Human Reliability Analysis

This task involved the analysis of two types of potential human errors: (1) pre-accident errors, including miscalibrations of equipment or failure to restore equipment to operability following test and maintenance, and (2) post-accident errors, including failure to diagnose and respond appropriately to accidents. In the evaluation of pre-accident faults, calibration, test, and maintenance procedures and practices were reviewed for each front-line and support system. The evaluation included the identification of components improperly calibrated or left in an inoperable state following test or maintenance activities. For post-accident faults, procedures expected to be followed in responding to accidents modeled in the event trees were identified and reviewed for possible sources of human errors that could have affected the operability or function of responding systems. In order to support eventual sequence quantification, estimates were produced for human error rates. In generating these estimates, screening values were sometimes used for initial calculations. For most of the human errors expected to be significant in the analysis, nominal human error probabilities were evaluated using modified THERP techniques (Ref. A.17) and plant-specific characteristics. For the boiling water reactor (BWR) plants in NUREG–1150, a detailed human reliability analysis (HRA) was performed on the post-accident human faults for the anticipated transient without scram (ATWS) sequences (Ref. A.18).

Data Base Analysis

This task involved the development of a data base for quantifying initiating event frequencies and basic event probabilities (other than human errors) that appeared in the models. A generic data base representing typical initiating event frequencies as well as plant component failure rates and their uncertainties was developed. Data for the plant being analyzed, however, may have differed significantly from industrywide data. In this task, the operating history of the plant (if available) was reviewed to develop plant-specific initiating event frequencies and to determine whether any plant components had unusually high or low failure rates. Test and maintenance practices and plant experiences were also reviewed to determine the frequency and duration of these activities. This information was used to supplement the generic data base.

Accident Sequence Quantification Analysis

The models from each previous step were integrated into the accident sequence quantification analysis task to calculate accident sequence frequencies. This was an iterative task performed at various times during the analysis. For example, the analyst first estimated partial sequence frequencies, sometimes conservatively. If the resulting frequency of the accident sequence, considering only some of the failures involved, was below a specified cutoff value, the sequence was dropped from further consideration. However, if the frequency of the partial accident sequence was above the cutoff value, the sequence was fully developed and recovery actions applied where appropriate using the SETS code (Ref. A.19).

Plant Damage State Analysis

Plant damage state analysis provides the information necessary to initiate an accident progression analysis in a Level 2 PRA (discussed in Section A.3). The plant damage state definitions provide the status of plant systems at the onset of core damage. These definitions include descriptions of the status of core cooling systems, containment systems, and support systems in sufficient detail to describe the state of the plant for the accident progression analysis. The development of plant damage state definitions was accomplished by adding additional questions to the end of the accident sequence event trees. However, in many cases it was not necessary to actually draw the plant damage state event tree, but rather, the questions could be dealt with in a matrix format (see Section 11 of Ref. A.1).

The questions that defined the plant damage states were selected during an iterative process with the accident progression analysis staff. During the actual analysis, the accident sequence cut sets were regrouped into plant damage states, based on the particular failures in the cut sets and the answers to the selected questions. Some accident sequences contained cut sets that contributed to several different plant damage states. Similarly, there were cases where several different accident sequences could have contributed cut sets to the same plant damage state.

Once the new plant damage state cut-set groups were formed, they were quantified in the same manner as the accident sequences, in that point estimates (using mean values) were generated and an uncertainty analysis performed (as discussed below).

Uncertainty Analysis

With the NUREG-1150 objective of assessing the uncertainties in severe accident frequencies and risks, the single-valued estimates of accident sequence and plant damage state frequencies were supplemented with quantitative uncertainty analysis. Both parameter value (data) and modeling uncertainties were included in the analysis, which involved several steps:

- Preparation of probability distributions for the set of basic events in the logic models;
- Elicitation of expert judgment (from expert panels and project staff) for those issues or parameters for which insufficient information was available to readily prepare an uncertainty distribution;
- Determination of the correlation between parameters in the logic models;
- Input of the logic models and probability distributions, including correlation factors, to a computerized analysis package (Ref. A.20) to perform the Monte Carlo sampling and importance calculations; and
- Performance of additional sensitivity studies on certain key issues.

This analysis produced a frequency distribution from which mean, median, and 5th and 95th percentile values were obtained. The underlying logic models were also analyzed to rank the basic events according to their contribution to core damage frequency (using risk-reduction and risk-increase importance measures) and the uncertainty in this frequency.

A.2.2 Internal-Event Methods for Zion*

The analysis of the Zion Nuclear Plant Unit 1 for NUREG–1150 (Ref. A.21) used the large event tree, small fault tree approach originally used in the Zion Probabilistic Safety Study (ZPSS) (Ref. A.22). Because of the existence of the ZPSS, it was determined that an accident frequency analysis of the Zion plant could be included in NUREG–1150 at a greatly reduced level of effort and cost. To achieve this, many aspects of the probabilistic risk analysis process developed in the ZPSS were carried over into the NUREG–1150 analysis.

The principal steps of the methods used in the analysis of Zion included:

- Identification of initiating events,
- Plant response modeling (including systems analysis),
- Human reliability analysis (including recovery),
- Data analysis,
- Quantification, and
- Sensitivity/uncertainty analyses.

Each of these steps is discussed in more detail in the following sections.

Identification of Initiating Events—Zion

The initiating event categories for which plant response models were developed were determined in the ZPSS and were used directly in the NUREG–1150 analysis with only minimal changes. The ZPSS used a number of sources of information to establish these initiating event categories, including:

- Zion plant operating records,
- Zion plant design features and safety analyses,
- Previous probabilistic risk analyses, and
- General industry experience.

In addition to these resources, the ZPSS analysis team developed a "Master Logic Diagram" to organize their thought processes and to structure the information. Figure A.5 shows the high-level Master Logic Diagram developed for the Zion Probabilistic Safety Study. Level I in the diagram represents the undesired event for which the risk analysis is being conducted, i.e., an offsite release of radioactive material. Level II answers the question: "How can a release to the environment occur?" Level III shows that a release of radioactive material requires simultaneous core damage and containment failure. Level IV answers the question: "How can core damage occur?" After several more levels of "how can" questions, the diagram arrives at a set of potential initiating events.

The ZPSS listed 59 internal initiating events that were assigned to the first 13 initiating event categories shown in Figure A.5. The NUREG–1150 analysis was able to reduce the number of initiating event categories by combining several that had the same plant response. For example, the loss of steam inside and outside the containment was collapsed into loss of steam. The result was 11 initiating event categories for the NUREG–1150 analysis.

Plant Response Modeling—Zion

The plant response modeling for the NUREG–1150 analysis was based on the ZPSS work and consists of three parts. The first part is event tree modeling. The ZPSS developed 14 event tree models, one for each

*This section extracted, with editorial modification, from Reference A.21.

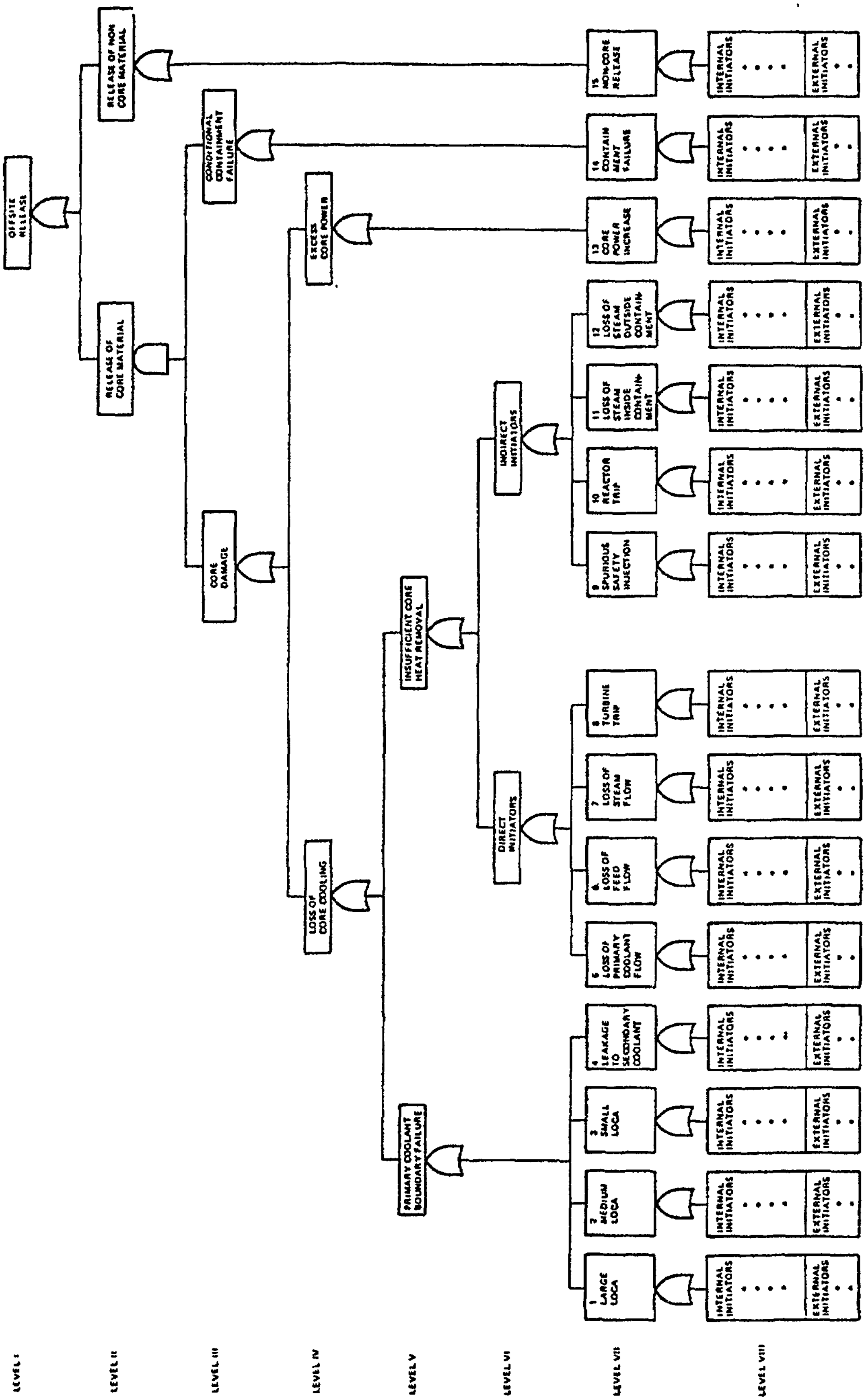


Figure A.5 Zion Probabilistic Safety Study master logic diagram.

of the initiating event categories and one for the failure of reactor trip condition (anticipated transient without scram). This last event tree is actually a subtree or extension to a number of the main event trees but was separated out to easily quantify the frequency of ATWS.

The ZPSS event trees were the basis for the NUREG–1150 event trees. Modifications were made to each of the original event trees to reflect the latest understanding of the intersystem dependencies. Many of the changes from the ZPSS to the NUREG–1150 analysis were based on the review of the ZPSS performed by Sandia National Laboratories under contract to the NRC staff (Ref. A.23) and comments on the draft version of this work (Ref. A.4).

The second part of the plant response model was the development of electric power support states. The ZPSS analysis of the Zion electric power system and the dependencies of other plant systems on electric power resulted in the identification of eight unique electric power states. Each power state defined a combination of successful and failed power sources. Each electric power state had a unique impact on the set of systems included in the event tree top events.

The final part of the plant response modeling was the analysis of the systems that provide the safety and support functions defined by the event tree top events. From the top event definitions and success criteria and the electric power states, a set of boundary conditions for each system analysis was developed. The number of unique boundary conditions determined the number of conditional split fractions that had to be modeled.

A conditional split fraction is the system availability given a specific set of conditions such as the initiating event, the electric power state, and the operational status of other required support systems. For instance, for the auxiliary feedwater system, seven conditional split fractions were needed. One (conditional split fraction "L11"), for example, was used for transients and loss-of-coolant accidents (LOCAs) with all power available.

The NUREG–1150 analysis for Zion made extensive use of the system analyses in the ZPSS. After verification of the current plant configuration, most conditional split fractions used in the NUREG–1150 analysis came directly from the ZPSS. In some cases, new conditional split fractions had to be developed to accommodate event tree model changes. These included several for the component cooling water system, the service water system, and the high-pressure injection system, among others. For the most part, the new conditional split fractions were able to be constructed from pieces of system analyses existing in the ZPSS.

Human Reliability Analysis—Zion

The human reliability analysis identified the human actions of operation, maintenance, and recovery that should be considered in the probabilistic risk analysis process. It also determined the human error rates to be used in the quantification of these actions. The NUREG–1150 analysis included human action involving: pre-initiator testing and maintenance actions; accident procedure actions; and recovery actions.

Pre-initiator testing and maintenance actions included the types of human errors that could render a portion of the plant unavailable to respond to an initiating event. Examples of these errors were improper restoration of a system after testing and miscalibration of instrument channels.

Accident procedure actions are required for the plant to fully respond to an initiating event. These actions were generally called out in the emergency operating procedures. Examples of these human actions were establishing feed-and-bleed cooling, switching from the injection mode of emergency core cooling to containment sump recirculation, and depressurizing below the steam generator safety valve setpoints during a steam generator tube rupture.

Recovery actions may or may not be called out in the emergency operating procedures. These actions are taken in response to the failure of an expected function. Examples of these types of actions included recovering ac electrical power, manually starting a pump that should have received an auto-start signal, and refilling the refueling water storage tank in the event of emergency core cooling system recirculation failure.

Pre-initiator testing and maintenance actions were usually incorporated into the system models since most of them impacted only a single system. Accident procedure actions were typically included at the event tree level as a top event because they were an expected portion of the plant/operator response to the initiating event. These actions may have been included in the system models if they impacted only a single system. Recovery actions were included either in the event trees or the system models or applied to the sequence models after processing of the plant response models.

Pre-initiating event testing and maintenance errors were included in the system models and were taken directly from the ZPSS. The accident procedure errors were also taken from the ZPSS after verification that the emergency procedures and plant operating philosophies had not changed significantly from the time of the ZPSS. Recovery actions were developed specifically for the NUREG-1150 analysis and were applied to specific system models and to specific accident sequences as appropriate.

Data Analysis—Zion

The ZPSS performed an extensive analysis of plant-specific data to determine the failure rates and demand failure probabilities for all the basic events used in the models. The plant data collected included component failure data, test frequencies and results, component service hours, and maintenance frequencies and durations.

This information was combined with generic failure data from sources such as Reactor Safety Study (Ref. A.24), IEEE-500 (Ref. A.25), and others by a single-stage or two-stage Bayesian update analysis. The generic data were reviewed and screened for applicability before being used as a prior distribution in the Bayesian updating process.

The NUREG-1150 analysis reviewed the plant operating history and determined that no significant changes had occurred that would invalidate any portion of the ZPSS data analysis. This was confirmed in discussions with the licensee. Therefore, the data used in the NUREG-1150 analysis were taken directly from the ZPSS.

Quantification—Zion

For the NUREG-1150 analysis, the event tree models and the conditional split fraction values were input and processed using computer codes designed specifically for manipulation of large event tree, small fault tree models with support system states (i.e., the models used in the ZPSS and other PRAs)(e.g., Ref. A.26). Approximately 16,000 accident sequences were quantified. Each event tree was analyzed eight times, once for each electric power state. For each analysis, the appropriate conditional split fractions were assigned to the top events. The results were single-valued estimate accident sequence frequencies.

The accident sequences with a single-valued estimate frequency less than 1E-9 per year were not processed any further and were dropped. Recovery actions pertaining to specific situations were applied to the appropriate remaining sequences. Again, any sequences that fell below the 1E-9 cutoff were dropped.

The remaining accident sequences were assigned to plant damage states (PDSs). The PDS frequencies were determined by summing the frequencies of all the sequences in a given PDS.

Sensitivity/Uncertainty Analyses—Zion

For purposes of sensitivity and uncertainty analyses, the accident sequences with a single-valued estimate frequency greater than or equal to 1E-9 per reactor year were loaded into IRRAS 2.0 (Ref. A.27), a fault tree/event tree generation and analysis model developed for NRC. Six issues were identified for which sensitivity/uncertainty evaluations were desired. These issues were determined by examining the results of the single-valued estimate quantification.

For each of these issues, an expression of the uncertainty was developed. These expressions were used in combination with uncertainties in failure data in a specialized Monte Carlo analysis method (Latin hypercube sampling) (Ref. A.15) to generate a sample of 150 observations. These observations were propagated through the system and sequence models using IRRAS 2.0 to generate 150 frequencies for each sequence and plant damage state. From these, probability distributions for individual plant damage states and total core damage frequency were determined. This information was then passed on to the accident progression and risk analysis portions of the Zion study.

A.2.3 External-Event Methods for Surry and Peach Bottom*

Seismic Accident Frequency Analysis Methods

A nuclear power plant is designed to ensure the survival of buildings and emergency safety systems in earthquakes less than one of a specific magnitude (the "safe shutdown" earthquake). In contrast, the analysis of seismic risk requires consideration of the range of possible earthquakes, including those of magnitudes less than and greater than the safe shutdown earthquake. Seismic risk is obtained by combining the frequencies of the spectrum of possible earthquakes, their potential (and very uncertain) effects on equipment and structures within the plant under study, and the subsequent effects on core and containment building integrity. In considering this, it should be noted that during an earthquake, all parts of the plant are excited simultaneously. Thus, during an earthquake, redundant safety system components experience highly correlated base motion, and there is a high likelihood that multiple redundant components would be damaged if one is damaged. Hence, the "planned-for" redundancy of equipment could be compromised. This common-cause failure mechanism represents a potentially significant risk to nuclear power plants during earthquakes.

The seismic accident frequency analysis method used in NUREG–1150 for the analysis of the Surry and Peach Bottom plants is based, in part, on the results of two earlier NRC-sponsored programs. The first was the Seismic Safety Margins Research Program (SSMRP) (Ref. A.29). In the SSMRP, a detailed seismic risk analysis method was developed. This program culminated in a detailed evaluation of the seismic core damage frequency of the Zion nuclear power station (Ref. A.30). In this evaluation, an attempt was made to accurately compute the responses of walls and floor slabs in the Zion structures, movements in the important piping systems, accelerations of all important valves, and the spectral accelerations at each safety system component (pump, electrical bus, motor control center, etc.). Correlation between the responses of all components was computed from the detailed dynamic response calculations. The important safety and auxiliary systems functions were analyzed, and fault trees were developed that traced failure down to the individual component level. Event trees related the system failures to accident sequences and radioactive release modes. Using these detailed models and calculations, it was possible to evaluate the frequency of core damage from seismic events at Zion and to determine quantitatively the risk importance of the components, initiating events, and accident sequences.

The second NRC program used in the NUREG–1150 analyses was the Eastern Seismic Hazard Characterization Program (Ref. A.31), which performed a detailed earthquake hazard assessment of nuclear power plant sites east of the Rocky Mountains. Results of these two programs formed the basis for a number of simplifications used in the seismic method reported here.

There are seven steps required for calculating the frequency of seismically initiated core damage accidents in a nuclear power plant:

- Determination of the local earthquake hazard (hazard curve and site spectra);
- Identification of accident sequences for the plant that lead to the potential for release of radioactive material (initiating events and event trees);
- Determination of failure modes for the plant safety and support systems (fault trees);
- Determination of the responses (accelerations or forces) of all structures and components (for each earthquake level);
- Determination of fragilities (probabilistic failure criteria) for the important structures and components;

*This section extracted, with editorial modification, from Part 3 of Reference A.28.

- Computation of the frequency of core damage using the information from the first five steps; and
- Estimation of the uncertainty in the core damage frequencies.

Work performed in each of these steps is summarized below.

Determination of Local Earthquake Hazard

The seismic analyses in this report made use of two data sources on the frequency of earthquakes of various intensities at the specific plant site (the seismic "hazard curve" for that site): the Eastern United States Seismic Hazard Characterization Program, funded by the NRC at Lawrence Livermore National Laboratory (LLNL) (Ref. A.31); and the Seismic Hazard Methodology for the Central and Eastern United States Program, sponsored by the Electric Power Research Institute (EPRI) (Ref. A.32). In both the LLNL and EPRI programs, seismic hazard curves were developed for all U.S. commercial power plant sites east of the Rocky Mountains, using expert panels to interpret available data. The NRC staff presently considers both program results to be equally valid (Ref. A.33). For this reason, two sets of seismic results are provided in this report. Section C.11 of Appendix C discusses the analysis of seismic hazards in more detail.

Identification of Accident Sequences

The scope of the NUREG-1150 seismic analysis includes loss-of-coolant accidents (LOCAs) (including vessel rupture and pipe ruptures of a spectrum of sizes) and transient events. Two types of transient events were considered: those in which the power conversion system (PCS) is initially available (denoted type T3 transients) and those in which the PCS is failed as a direct consequence of the initiating event (denoted type T1 transients). The event trees developed in the internal-event analyses are used. For the seismic analysis, the reactor vessel rupture and large LOCA event frequencies were based on a Monte Carlo analysis of steam generator and reactor coolant pump support failures. The frequency of Type T1 transients is based on the frequency of loss of offsite power (LOSP). This is the dominant cause of this type of transient (for plants such as those studied in NUREG-1150 in which LOSP causes loss of main feedwater). Given an earthquake of reasonable size, it is assumed that a type T3 transient occurs with a probability of unity.

Determination of Failure Modes

The internal-event fault trees were used in the seismic analysis with some modification to include basic events for seismic failure modes and to resolve the trees for pertinent cut sets to be included in the probabilistic calculations. Probabilistic culling was used in the resolution of these trees in such a way as to ensure that important correlated failure modes were not lost.

Determination of Fragilities

Component seismic fragilities were obtained both from a generic fragility data base and from plant-specific fragilities developed for components identified during the plant walkdown.

The generic data base of fragility functions for seismically induced failures was originally developed as part of the SSMRP (Ref. A.29). Fragility functions for the generic categories were developed based on a combination of experimental data, design analysis reports, and an extensive expert opinion survey. The experimental data used in developing fragility curves were obtained from the results of component manufacturers' qualification tests, independent testing laboratory failure data, and data obtained from the extensive U.S. Corps of Engineers SAFEGUARD Subsystem Hardness Assurance Program (Ref. A.34). These data were statistically combined with the expert opinion survey data to produce fragility curves for each of the generic component categories.

Detailed structural fragility analyses were performed for all important safety-related structures at the NUREG-1150 plants. In addition, an analysis of liquefaction for the underlying soils was performed. These were included directly into the accident frequency analysis.

Determination of Responses

Building and component seismic responses were estimated from peak ground accelerations at several probability intervals on the hazard curve. Three basic aspects of seismic response—best estimates, variability, and correlation—were generated. Results from the SSMRP Zion analysis (Ref. A.30) and other methods studies (Ref. A.35) formed the basis for assigning scaling, variability, and correlation of responses.

In each case, computer code calculations (using the SHAKE code (Ref. A.36)) were performed to assess the effect of the local soil column (if any) on the surface peak ground acceleration and soil-structure interactions. This permitted an evaluation of the effects of nonhomogeneous underlying soil conditions that could have strongly affected the building responses.

Fixed base mass-spring (eigen-system) models were either obtained from the plant's architect/engineer or were developed from the plant drawings. Using these models, the floor slab accelerations were calculated using the CLASSI computer code (Ref. A.37). This code uses a fixed-base eigen-system model of the structure and input-specified frequency-dependent soil impedances and computes the structural response (as well as variation in structural response if desired). Variability in responses (floor and spectral accelerations) was assigned based on results of the SSMRP.

Correlation between component failures was explicitly included in the analysis. In computing the correlation between component failures (in order to quantify the cut sets), it was necessary to consider correlations both in the responses and in the fragilities of each component. Inasmuch as there are no data as yet on correlation between fragilities, the fragility correlations between like components were taken as zero, and the possible effect of such correlation quantified in a sensitivity study. The correlation between responses is assigned according to a set of rules.

Computation of Frequency of Core Damage

Given the input from the five steps above, the SETS computer code (Ref. A.19) was used to calculate required outputs (probabilities of failure, core damage frequency, etc.).

Estimation of Uncertainty

Using Monte Carlo techniques, frequency distributions of individual parameters in the seismic analysis were combined to yield frequency distributions of accident sequences, plant damage states, and total core damage.

Fire Accident Frequency Analysis Methods

Nuclear power plants are designed to be able to safely shut down in the presence of a spectrum of possible fires throughout the plant (Ref. A.38). Nonetheless, some plant areas contain cabling for multiple trains of core cooling equipment. Fires in such areas (and in some cases in conjunction with random equipment failures not caused by a fire) can lead to accident sequences with relatively important frequencies. For this reason, the core damage frequency from fire-initiated accidents was assessed for two power plants (Surry and Peach Bottom).

The principal steps in the simplified fire accident frequency analysis method used in NUREG–1150 were as follows:

- Initial plant visit,
- Screening of potential fire locations, and
- Accident sequence quantification.

Each of these steps is summarized below.

Initial Plant Visit

Based on the internal-event and seismic analyses, the general location of cables and components of the principal plant systems had previously been developed. A plant visit was then made to provide the analysis staff with a means of seeing the physical arrangements in each of these areas. The analyst had a fire zone checklist that would aid the screening analysis and the quantification step.

The second purpose of the initial plant visit was to confirm with plant personnel that the documentation being used was in fact the best available information and to get clarification about any questions that might have arisen in a review of the documentation. As part of this, a thorough review of firefighting procedures was conducted.

Screening of Potential Fire Locations

It was necessary to select important fire locations within the power plant under study that have the greatest potential for producing accident sequences of high frequency or risk.

The screening analysis was comprised of:

- Identification of relevant fire zones

  A thorough review of the plant Appendix R (Ref. A.38) submittal was conducted to permit the division of this plant into fire zones. A fire zone can be defined as a plant area surrounded by a 3-hour-rated barrier or its equivalent. From this complete plant model, fire zones were screened from further analysis if it could be shown that neither safety-related equipment nor its associated power or control cabling was located within them.

- Screening of fire zones on probable fire-induced initiating events

  Fire zones where the overall fire occurrence frequency is less than 1E–6 per year were eliminated from further consideration. Also, certain fire-induced initiating events such as loss of offsite power could be eliminated if a particular fire zone contained none of its cabling. Therefore, even if a fire zone could not be screened as a whole, certain of the fire-induced initiators that might be postulated to occur within this zone could be eliminated.

- Screening of fire zones on both order and frequency of cut sets

  Cut sets containing random failure combinations with frequencies less than 1E–4 were eliminated from further consideration. In this step, cut sets with multiple fire zone combinations were addressed. Any cut set containing three or more fire zone combinations was screened from further consideration. These scenarios would imply the simultaneous failure of two or more 3-hour-rated fire barriers and therefore were considered probabilistically insignificant. Cut sets containing only two fire zones were eliminated on the following three criteria:

  - If there was no adjacency between the two areas;
  - If there was an adjacency, it contains no penetrations; and
  - On probability, with barrier failure probability set to 0.1.

- Analysis of each fire zone remaining to numerically evaluate and to cull on probability

  The remaining cut sets were now resolved with fire-zone-specific fire initiating event frequencies and then screened on a frequency criteria of 1E–8 per year.

Accident Sequence Quantification

After the screening analysis has eliminated all but the probabilistically significant fire zones, quantification of dominant cut sets was completed as follows:

- Determination of the temperature response in each fire zone

  The modified COMPBRN III code (Ref. A.39) was used to calculate time to damage of all critical cabling and components within a fire zone.

- Computation of component fire fragilities

  For those modeled components in the COMPBRN analysis, damageability temperatures were assigned based on fire test experience.

- Assessment of the probability of barrier failure for all remaining combinations of fire zones

The remaining cut sets that contained two fire zones had barrier failure probabilities calculated. Those cut sets that were below 1E–8 per year were eliminated from further consideration.

- Performance of recovery analyses

  In a manner like that of the internal-event recovery analysis, recovery of random failures was applied on a cut-set by cut-set basis. For sequences less than 24 hours in duration, only one recovery action was allowed. If more than one recovery action was possible for any of these given cut sets, a consistent hierarchy of which recovery action to apply was used. In sequences of greater than 24 hours, two recovery actions were allowed. The only modifications to recovery probabilities were found in areas where a fire had to first be extinguished and then the area desmoked prior to the occurrence of a local action.

This quantification was performed using specialized Monte Carlo techniques (Latin hypercube sampling) (Ref. A.15) so that individual parameter frequency distributions can be combined into frequency distributions of accident sequences, plant damage states, and total core damage frequency.

Bounding Analysis of Other External Events

Bounding analyses were performed for NUREG–1150 for those external events that were judged to potentially contribute to the estimated plant risk. Those events that were considered included extreme winds and tornadoes, turbine missiles, internal and external flooding, and aircraft impacts.

Conservative probabilistic models were used in these bounding analyses to integrate the randomness and uncertainty associated with event loads and plant responses and capacities. Clearly, if the mean initiating event frequency resulting from a conservative model was predicted to be low (e.g., less than 1E–6), the external event could be eliminated from further consideration. Using this logic, the bounding analyses identified those external events that needed to be studied in more detail as part of the risk analysis. In the case of both Peach Bottom and Surry, none of these "other external events" was found to be a potentially significant contributor to core damage frequency.

A.2.4 Products of Accident Frequency Analysis

The results of the accident frequency analyses discussed in this section can be displayed in a variety of ways. The specific products shown in NUREG–1150 are described as follows:

- The total core damage frequency for internal events and, where estimated, for external events

  For Part II of NUREG–1150 (plant-specific results), a histogram-type plot was used to represent the distribution of total core damage frequency as shown on the right side of Figure A.6. This histogram displays the fraction of Latin hypercube sampling (LHS) observations falling within each interval.* Four measures of the probability distribution are identified:

  - Mean,
  - Median,
  - 5th percentile value, and
  - 95th percentile value.

  A second display of accident frequency results is used in Part III of NUREG–1150, where results for all five plants are displayed together. This figure provides a summary of these four specific measures in a simple graphical form (shown on the left side of Fig. A.6).

  For those plants in which both internal and external events have been analyzed (Surry and Peach Bottom), the core damage frequency results are provided separately for the two classes of accident initiators.

*Care should be taken in using these histograms to estimate probability density functions. These histogram plots were developed such that the heights of the individual rectangles were not adjusted so that the rectangular areas represented probabilities. The shape of a corresponding density function may be very different from that of the histogram. The histograms represent the probability distribution of the logarithm of the core damage frequency.

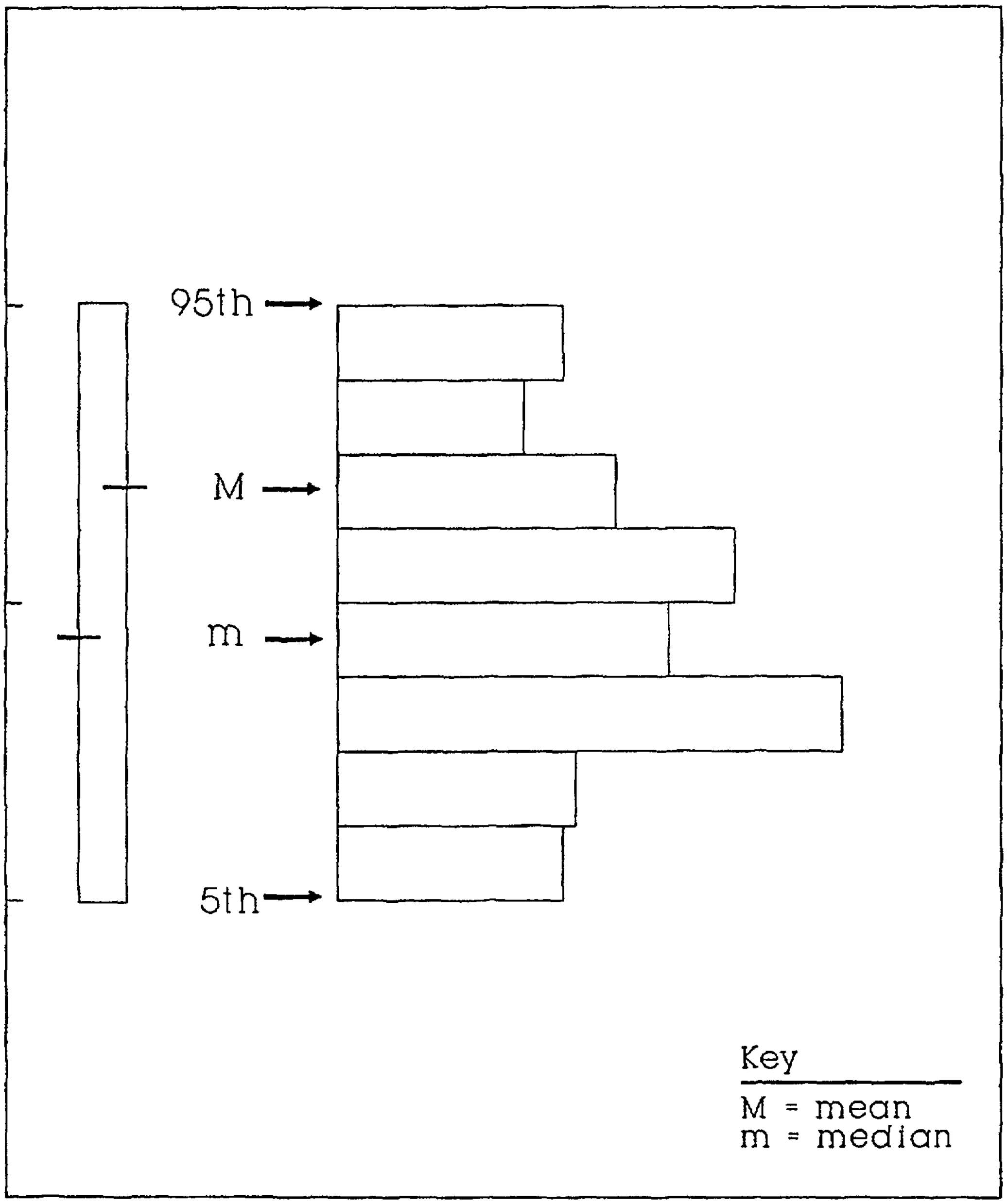


Figure A.6 Example display of core damage frequency distribution.

- The definitions and estimated frequencies of plant damage states

  The total core damage frequency estimates described above are the result of the summation of the frequencies of various types of accidents. For this summary report, the total core damage frequency has been divided into the contributions of specific plant damage states:*

  - Station blackouts, in which all ac power (coming from offsite and from emergency sources in the plant) is lost;
  - Transient events with failure of the reactor protection system (ATWS events);
  - Other transient events;
  - Loss-of-coolant accidents (LOCAs) resulting from pipe ruptures, reactor coolant pump seal failures, and failed relief valves occurring within the containment building; and
  - LOCAs that bypass the containment building (steam generator tube ruptures and other "interfacing-system LOCAs").

Figure A.7 provides an example display of mean plant damage state frequencies used in NUREG–1150.

In addition to these quantitative displays, the results of the accident frequency analyses also can be discussed with respect to the qualitative perspectives obtained. In NUREG–1150, qualitative perspectives are provided in two levels:

- *Important plant characteristics*. The discussion of important plant characteristics focuses on general system design and operational aspects of the plant. Perspectives are thus provided on, for example, the design and operation of the emergency diesel generators or the capability for the feed and bleed mode of emergency core cooling.
- *Important individual events*. One typical product of a PRA is a set of "importance measures." Such measures are used to assess the relative importance of individual items (such as the failure rates of individual plant components or the uncertainties in such failure rates) to the total core damage frequency. While a variety of measures exists, two are discussed (qualitatively) in NUREG–1150. The first importance measure (risk reduction) shows the effect of significant reductions in the frequencies of individual plant component failures or plant events (e.g., loss of offsite power, specific human errors) on the total core damage frequency. In effect, this measure shows how to most effectively reduce core damage frequency by reductions in the frequencies of these individual events. The second importance measure (uncertainty reduction) discussed in NUREG–1150 indicates the relative contribution of the uncertainty in key probability distributions to the uncertainty in total core damage frequency. In effect, this measure shows how most effectively to reduce the uncertainty in core damage frequency. A third importance measure, risk increase, is discussed in the contractor reports underlying NUREG–1150.

As illustrated in Figure A.3, the results of this analysis are the first and second inputs to the risk calculations, $F(IE_h)$, the frequency of initiating event h, and $P(IE_h \rightarrow PDS_i)$, the conditional probability of plant damage state i, given initiating event h.

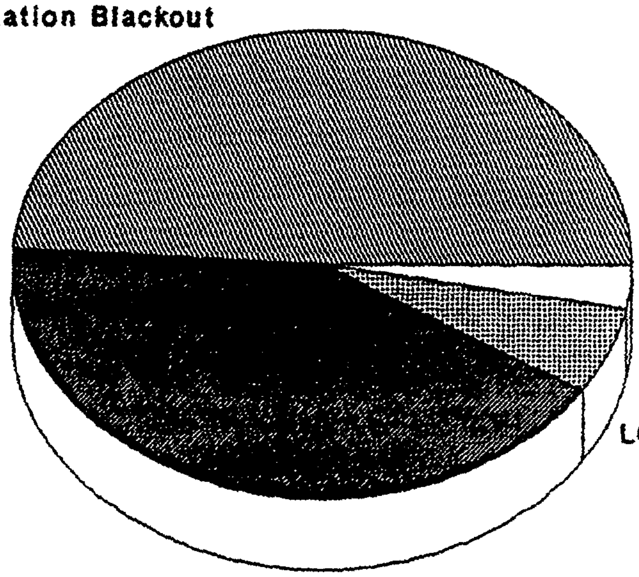
A.3 Accident Progression, Containment Loadings, and Structural Response Analysis**

A.3.1 Introduction

The purpose of the accident progression, containment loadings, and structural response analysis is to track the physical progression of the accident from the initiating event until it is concluded that no additional release of radioactive material from the containment building will occur. Thus, the core damage process is studied in the reactor vessel, as the vessel is breached, and outside the vessel. At the same time, the analysis tracks the impact of the accident progression on the containment building structure, with particular focus on the threat to containment integrity posed by pressure loadings or other physical processes.

*A more detailed set of plant damage states is provided in the supporting contractor reports.

**This section extracted, with editorial modification, from Chapter 2 of Reference A.2.



Total Mean Core Damage Frequency: 4.5E-6

Figure A.7 Example display of mean plant damage state frequencies.

The requirements of an ideal accident progression analysis would be knowledge, probably in the form of the results of mechanistic calculations from validated computer codes, of the characteristics of the set of possible accident progressions resulting from individual plant damage states defined in the previous analysis step. More than one accident progression can result from each plant damage state since random events (hydrogen detonations, for example) occurring during the accident progression can alter the course of the accident. Given the frequency of the plant damage state and the probabilities of the random events, one could determine the outcomes and frequencies of the set of possible accidents.

Knowledge of the characteristics of all possible accidents resulting from each plant damage state is clearly not available with current technology. A large number of mechanistic codes that can predict some aspects of the accident progression are available. For example, MELPROG (Ref. A.40) and CONTAIN (Ref. A.41) can be used to track in-vessel and containment events, respectively, for very explicit accident progressions. Less detailed but more comprehensive codes, such as the Source Term Code Package (STCP) (Ref. A.42), MAAP (Ref. A.43), and, more recently, MELCOR (Ref. A.44), have been developed to predict generalized characteristics of more aspects of the accident in an integrated fashion. While these codes are very useful for developing a detailed understanding of accident phenomena and how the different phenomena interact, they do not meet the constraints imposed by a PRA; i.e., the ability to analyze a very wide range of scenarios with diverse boundary conditions in a timely and cost-efficient manner. In addition, the number of code calculations necessary to investigate uncertainty and sensitivity to inputs, models, and assumptions would be prohibitively expensive. Further, these codes have not been fully validated against experiments. Thus, codes developed by different groups (for example, NRC and industry contractors) frequently include contradictory models and give different results for given sets of accident boundary conditions. Finally, these codes also do not contain models of all phenomena that may determine the progression of the accident.

The information that was available with which to conduct the accident progression analysis for NUREG-1150 consisted of the diverse body of research results from about 10 years of severe accident research within the reactor safety community. This included a large variety of severe accident computer code calculations, other mechanistic analyses, and experimental results. Much of the information represented basic understanding of some important phenomena. Because of the expense of developing and running large integrated codes, less information was in the form of integrated accident progression analyses. That which was available was usually confined to analyses of a few types of accident sequences. All existing codes were recognized to have some limitations in their abilities to mechanistically model severe accidents.

Many new calculations were conducted specifically for NUREG-1150. For example, new CONTAIN code calculations were performed to assess pressure loadings on the containment and sensitivity of the loading calculations to various phenomenological assumptions (Ref. A.45). Most of the new calculations are described in the contractor reports supporting NUREG-1150. In particular, Reference A.46 contains a complete listing and description of the new supporting calculations. For the most part, the new calculations were intended to fill the largest gaps in the present state of knowledge of accident progression for the most important accidents.

Given this state of information, the NUREG-1150 accident progression analysis was performed in a series of steps, including:

- Development of accident progression event trees,
- Structural analyses,
- Probabilistic quantification of event tree issues, and
- Grouping of event tree outcomes.

Each of these steps is discussed below.

A.3.2 Development of Accident Progression Event Trees

The NUREG-1150 accident progression analyses were conducted using plant-specific event trees, called accident progression event trees (APETs). The APETs consist of a series of questions about physical phenomena affecting the progression of the accident. A typical question would be "What is the pressure rise in the containment building at reactor vessel breach?" A complete listing of the questions that make up the accident progression event tree for each plant studied in NUREG–1150 can be found in References A.47 through A.51. Typically, the event trees for each plant consisted of about 100 questions; each question could have multiple outcomes or branches.

The NUREG–1150 APETs were general enough to efficiently calculate the impact of changes in phenomenological models on the accident progression in order to study the effect of uncertainties among these models. This generality added complexity to the analysis since, with the ability to consider different models, some paths through the tree, which would be forbidden for a specific model, had to be included when a variety of models was considered. The multiplicity of possible accident progression results caused by the consideration of multiple models for some of the accident phenomena was amplified at each additional stage of the accident progression since, in addition to creating more possible outcomes, a wider range in boundary conditions at the subsequent events was made possible. Because of the flexibility and generality of the APETs, basic principles, such as hydrogen mass conservation, steam mass conservation, etc., were incorporated into the event trees in order to automatically eliminate pathways for which the principles are violated. This was accomplished with parameters, such as hydrogen concentrations in various compartments, passed along in the tree as each accident pathway was evaluated. At some questions in the tree, the parameters were manipulated using computer subroutines. The branch taken in each question could depend on the values of such calculated parameters. The consistency of phenomenological treatment throughout each accident was also ensured by allowing questions to depend on the branches or parameters taken in previous questions.

Figure A.8 schematically illustrates the APETs used in this study. The first section of the tree (about 20 percent of the total number of questions) was used to automatically define the input conditions associated with the individual plant damage state (PDS). Thus, if one of the characteristics of a PDS was the pressure in the reactor vessel at the onset of core damage, a question was included to set the initial condition according to that variable. The next part of the tree was then devoted to determining whether or not the accident was terminated before failure of the reactor vessel. Questions pertinent to the recovery of cooling and coolability of the core were asked in this part of the tree. The next section of the tree continued the examination of the accident progression in the reactor vessel. As illustrated in Figure A.8, there were two principal areas of investigation for this part of the analysis: in-vessel phenomena that determined the radioactive release characteristics; and events that impacted the potential for containment loadings. The example in Figure A.8 shows the phenomena associated with the release of hydrogen during the in-vessel phase of accident progression and the resultant escape of that hydrogen into the containment building.

The next stage illustrated in Figure A.8 continues the examination of the accident during, and immediately after, reactor vessel breach. This included the continued core meltdown in the vessel and the simultaneous loading and response of the containment building. A good example for this stage of the APET analysis is an examination of the coolability of the debris once out of the reactor vessel, followed by questions concerning the loading of the containment as a result of core-concrete interactions.

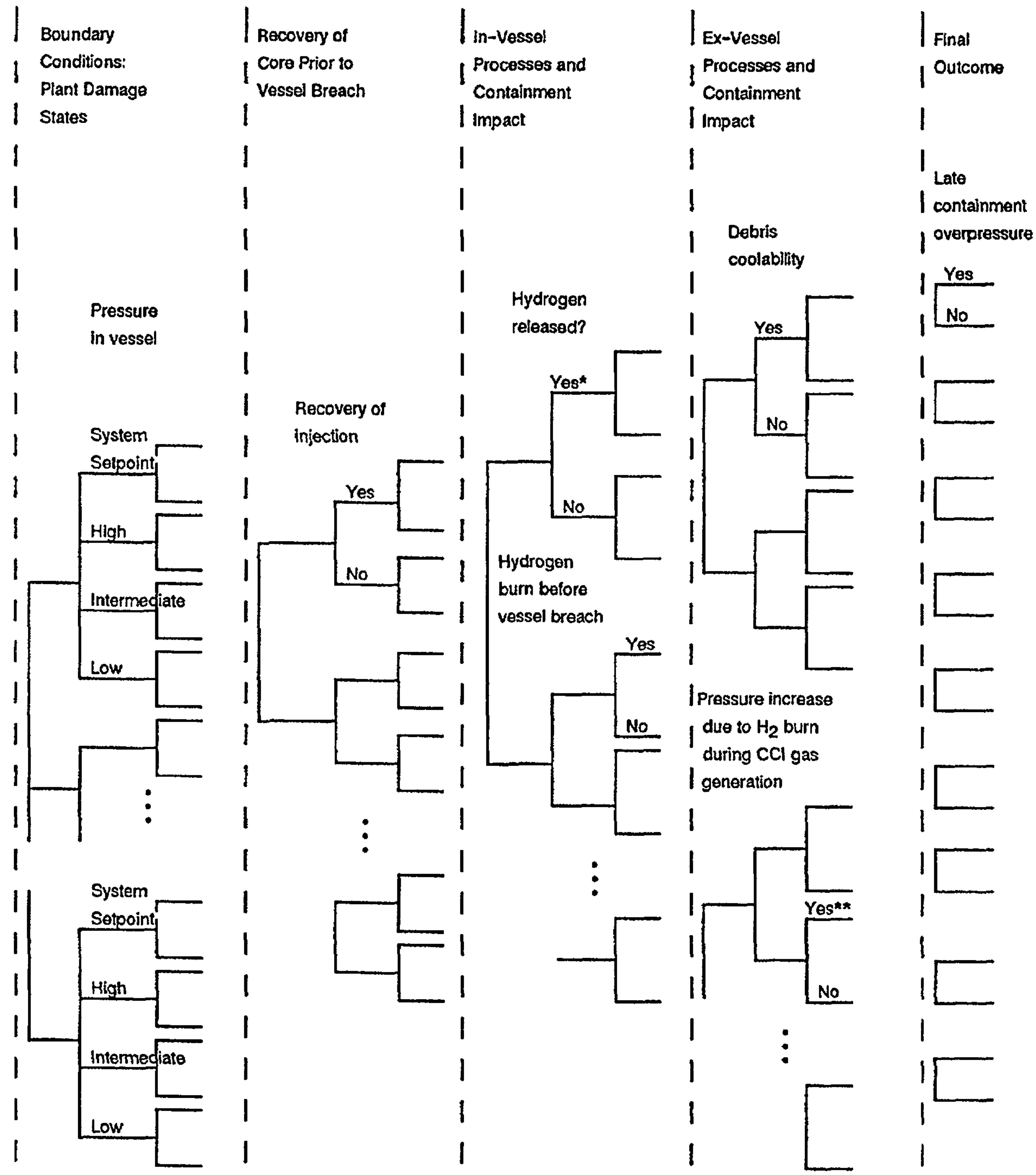
The final stage of the illustrated APET is related to the final status of containment building integrity. Long-term overpressurization, threats from combustion events, and similar questions were asked for this stage of the accident progression. For convenience, some questions that summarized the status of the containment at specific times during the accident were also included.

Throughout the progression of a severe accident, operator intervention to recover systems has the potential to mitigate the accident's impact. Such actions were considered in the APET analysis, using the same rules as those used in the accident frequency analysis.

The previous explanation has delineated the general flow of the accident progression event tree. What is not immediately apparent in this summary is the degree to which dependencies could be taken into account.

An example of the dependency treatment is a series of questions that relate to hydrogen combustion. The outcomes of the event tree questions that ask whether hydrogen deflagration occurs sometime after vessel breach and what is the resulting pressure load from the burn are highly dependent on previous questions. The individual values for the probability of ignition and the pressure rise were dependent on:

- Previous hydrogen burn questions (the amount consumed in each previous burn was tracked, and the concentration at the later time was calculated consistent with all previous hydrogen events);



*Amount of hydrogen released is sampled from continuous probability distribution

**Pressure increase is calculated from user function

Figure A.8 Schematic of accident progression event tree.

- Questions concerning the steam loading to determine whether the atmosphere was steam inert; and
- Questions concerning the availability of power, which influenced the probability of ignition.

In turn, these questions all had further dependencies on each other and on other questions. For example, the steam loading questions were dependent on the power and equipment availability since heat removal system operation would impact the steam concentration.

A.3.3 Structural Analyses

The NUREG-1150 APETs explicitly incorporate consideration of the structural response of containment buildings, including a building's ultimate strength, failure locations, and failure modes. Use was made of available detailed structural analyses (e.g., Ref. A.52) and results of recent experimental programs (e.g., Ref. A.53). The judgments of experts were used to interpret the available information and develop the required input (probability distributions) for the APET (see Section A.7 for discussion of the use of expert judgment).

A.3.4 Probabilistic Quantification of APETs

In general, phenomenological models were not directly substituted into the event trees (in the form of subroutines) at each question. Rather, the results of the model calculations were entered into the trees through the assigned branching probabilities, the dependencies of the questions on previous questions (the "case structure"), and/or tables of values that were used to determine parameters passed or manipulated by the event tree. Some questions in the trees, such as those concerning the operability of equipment and availability of power, were assigned probability distributions derived from data analogous to the process in the accident frequency analysis. Timing of key events was identified through a review of available code calculations and other relevant studies in the literature. The process of assigning values to the branch point probabilities, creating the case structure, writing the user functions, and supplying parameter values or tables is referred to as "quantification" of the tree.

Once an accident progression event tree, with its list of questions (their branches and their case structure), its subroutines, and its parameter tables, had been constructed by an analyst, it was evaluated using the computer code EVNTRE (Ref. A.10). EVNTRE can automatically track the different kinds of dependencies associated with the accident progression issues. This code was also built with specific capabilities for analyzing and investigating the tree as it was being built, allowing close scrutiny of the development of a complex model. For each plant damage state, EVNTRE evaluates the outcomes of the set of subsequent accident progressions predicted by the APET and their probabilities.

A.3.5 Grouping of Event Tree Outcomes

EVNTRE groups paths through the tree into accident progression bins. PSTEVNT (Ref. A.54) is a "rebinner" computer code that further groups the initial set of bins produced by EVNTRE.* To meet the needs of the subsequent source term analysis, the APET results are grouped into "accident progression bins."

The accident progression bins were defined through interactions between the accident progression analysts and the source term analysts. Characteristics of the bins include, for example, timing of release events, size and location of containment failure, and availability of equipment and processes that remove radioactive material. As such, the bins are relatively insensitive to many of the individual questions in the tree as they focus on the ultimate outcomes, and through the use of these bins, the paths through the tree were greatly reduced in terms of the number of unique outcomes.

A.3.6 Products of Accident Progression Analysis

The qualitative product of the accident progression, containment loadings, and structural response analysis is a set of accident progression bins. Each bin consists of a set of event tree outcomes (with associated probabilities) that have a similar effect on the subsequent portion of the risk analysis, analysis of radioactive material transport. As such, the accident progression bins are analogous to the plant damage states described in Section A.2.4.

*EVNTRE groupings can be chosen to illustrate the importance of a specific aspect of accident phenomenology, system performance, or operator performance, as long as that aspect is a distinct part of the APET.

Quantitatively, the product consists of a matrix of conditional failure probabilities, with one probability for each combination of plant damage state and accident progression bin. These probabilities are in the form of probability distributions, reflecting the uncertainties in accident processes.

In NUREG–1150, products of the accident progression analysis are shown in the following ways:

- The distribution of the probability of early containment failure* for each plant damage state (as shown in Fig. A.9).

  Measures of this distribution provided include:

  - Mean,
  - Median,
  - 5th percentile value, and
  - 95th percentile value.

- The mean probability of each accident progression bin for each plant damage state (as shown in Fig. A.10).

As illustrated in Figure A.3, the result of this process is the third input to the risk calculation, $P(PDS_i \rightarrow APB_j)$, the conditional probability of accident progression bin j given plant damage state i.

A.4 Radioactive Material Transport (Source Term) Analysis**

A.4.1 Introduction

The third part of the NUREG–1150 risk analyses is the estimation of the extent of radioactive material transport and release into the environment and the conditions of the release (timing and energy). As described above, the interface between this and the previous step (the interface being the accident progression bin) is defined to efficiently transfer the important information, while maintaining a manageable set of calculations.

The principal steps in the source term analyses were:

- Development of parametric models of material transport,
- Development of values or probability distributions for parameters in the models, and
- Grouping of radioactive releases.

Each of these steps will be discussed below.

A.4.2 Development of Parametric Models

As noted previously, in a risk analysis it is not practical to analyze every projected accident in detail with a mechanistic computer code. The method used for this part of the risk analysis was designed to be efficient enough to calculate source terms for thousands of accident progression bins and flexible enough to allow for incorporation of phenomenological uncertainties into the analysis.

For the NUREG–1150 risk analyses, parametric models were developed that allowed the calculation of source terms for a wide range of projected accidents. While the basic parametric equation for the models was largely the same for all five plants studied, it was customized to reflect plant-specific features and

*In this report, early containment failure includes failures occurring before or within a few minutes of reactor vessel breach for pressurized water reactors and those failures occurring before or within 2 hours of vessel breach for boiling water reactors. Containment bypass failures are categorized separately from early failures.

**This section adapted, with editorial modification, from Chapter 2 of Reference A.2.

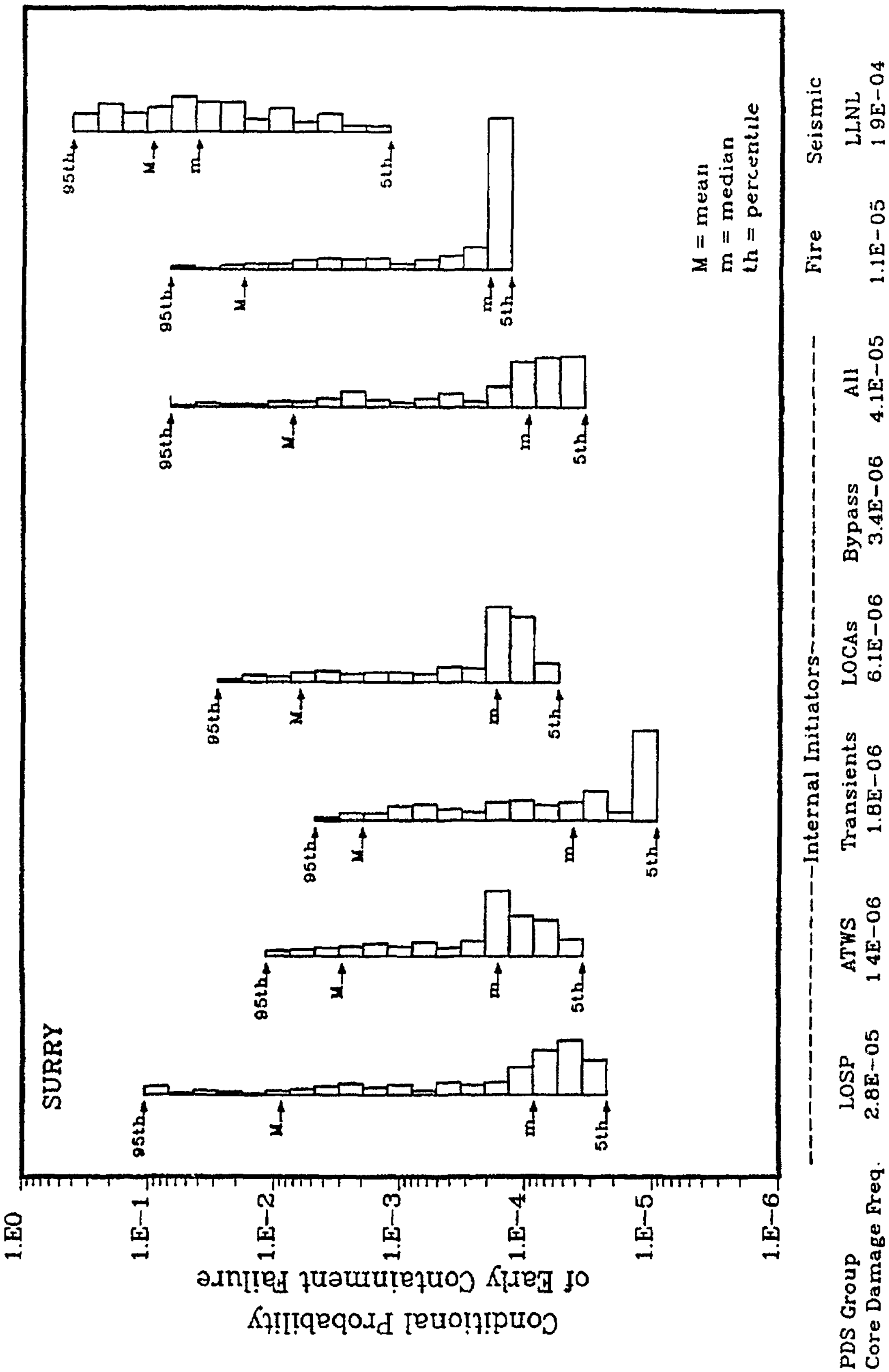


Figure A.9 Example display of early containment failure probability distribution.

SUMMARY ACCIDENT PROGRESSION BIN GROUP	LOSP (2.8E−05)	ATWS (1.4E−06)	Transients (1.8E−06)	LOCAs (6.1E−06)	Bypass (3.4E−06)	All (4.1E−05)	Fire (1.1E−05)	Seismic LLNL (1.9E−04)
VB, alpha, early CF	0.003	0.003		0.005		0.003	0.005	0.006
VB > 200 psi, early CF	0.005		0.001	0.001		0.004	0.013	0.008
VB, < 200 psi, early CF								0.082
VB, BMT or late CL	0.079	0.046	0.013	0.055		0.059	0.292	0.280
Bypass	0.003	0.078	0.007		1.000	0.122		0.001
VB, No CF	0.310	0.523	0.217	0.586		0.346	0.690	0.435
No VB	0.599	0.350	0.762	0.352		0.466		0.189

SUMMARY PDS GROUP (Mean Core Damage Frequency); LOSP, ATWS, Transients, LOCAs, Bypass and All are Internal Initiators.

Key: BMT = Basemat Melt−Through
CF = Containment Failure
CL = Containment Leak
VB = Vessel Breach

SURRY

Figure A.10 Example display of mean accident progression bin conditional probabilities.

conditions that could impact the source term estimates. As noted in Figure A.3, the codes that manipulate these parametric equations are called XSOR, where the X refers to a plant-specific abbreviation; for example, the code for Peach Bottom is PBSOR (Ref. A.11).

The parametric equations do not contain any chemistry or physics (except mass conservation) but describe the source terms as the product of release fractions and transmission factors at successive stages in the accident progression for a variety of release pathways, a variety of projected accidents, and nine classes of radionuclides. (To allow a manageable calculation, the radionuclides were treated in terms of radionuclide groups that have similar properties, the same nine groups that are defined in the Source Term Code Package (Ref. A.42)). Figure A.11 illustrates some of the release pathways and release fractions included in the model. The release is broken up into constituent parts (release fractions and transmission factors) in order to allow the input of a range of uncertainty within each part and to allow different components of the release to occur at different times.

The basic parametric equations are of the form

$$ST_i(i) + ST_h(i) + ST_e(i) + ST_l(i) + \text{Special Terms},$$

where (i) represents the radionuclide group, ST_i(i) represents releases from the fuel that occur in-vessel, ST_h(i) represents releases from the fuel that occur during high-pressure melt ejection, ST_e(i) represents releases from the fuel when the fuel is out of the vessel, primarily during core-concrete interactions, and ST_l(i) represents releases from the fuel that occur in-vessel but that plate out in the reactor coolant system (RCS) before the RCS integrity is lost and are released later. An example of a "Special Term" is an expression for releases from the plant for a bypass accident. The individual terms on the right hand side of the equation above represent different radionuclide release pathways and are represented as products of release fractions and transmission factors. For example, the expression for ST_i(i) for PWRs is given by

$$ST_i(i) = FCOR(i)*(FISG(i)*FOSG(i) + (1 - FISG(i))*FVES(i)*FCONV/DFE)$$

where FCOR(i) is the fraction of initial inventory of nuclide group i released from the fuel in-vessel, FISG(i) is the fraction of material released from the core in-vessel that enters the steam generators, FOSG(i) is the fraction of material entering the steam generators that leaves the steam generators and enters the environment, FVES(i) is the fraction of material entering the RCS that is released from the RCS, FCONV(i) is the fraction of the material released from the vessel that would be released from the containment in the absence of special decontamination mechanisms such as sprays that are included in DFE, and DFE is the decontamination factor to be applied to release from the vessel. The expression for BWRs is simpler because the terms related to the steam generators can be omitted. Similar expressions exist for ST_e(i), ST_h(i), and ST_l(i).

The parametric equation allows for uncertainty in the release fractions and for the effects of important boundary conditions, such as timing or temperature history to be included in the source term calculation. Any parameter in the equation can be represented by a probability distribution (this distribution can be sampled in the Monte Carlo analysis). All parameters (FVES(i) FISG(i), etc.) can be made to vary with accident progression bin characteristics, such as high pressure in the vessel. The accident progression bin characteristics are passed from the previous part of the risk analysis.

The expression for ST_e(i) is associated with the core-concrete interaction releases. The impact of containment conditions such as the availability of overlaying water or the operability of sprays is included in the expression for ST_e(i). In addition, the timing and mode of containment failure or leakage is considered in order to calculate a release from the containment to the environment.

Late revolatilization from the vessel and late release of iodine from water pools are included in the expression for ST_l(i). These secondary sources of radionuclides that were removed in earlier processes are kept track of in a consistent manner and made available for release at a later time.

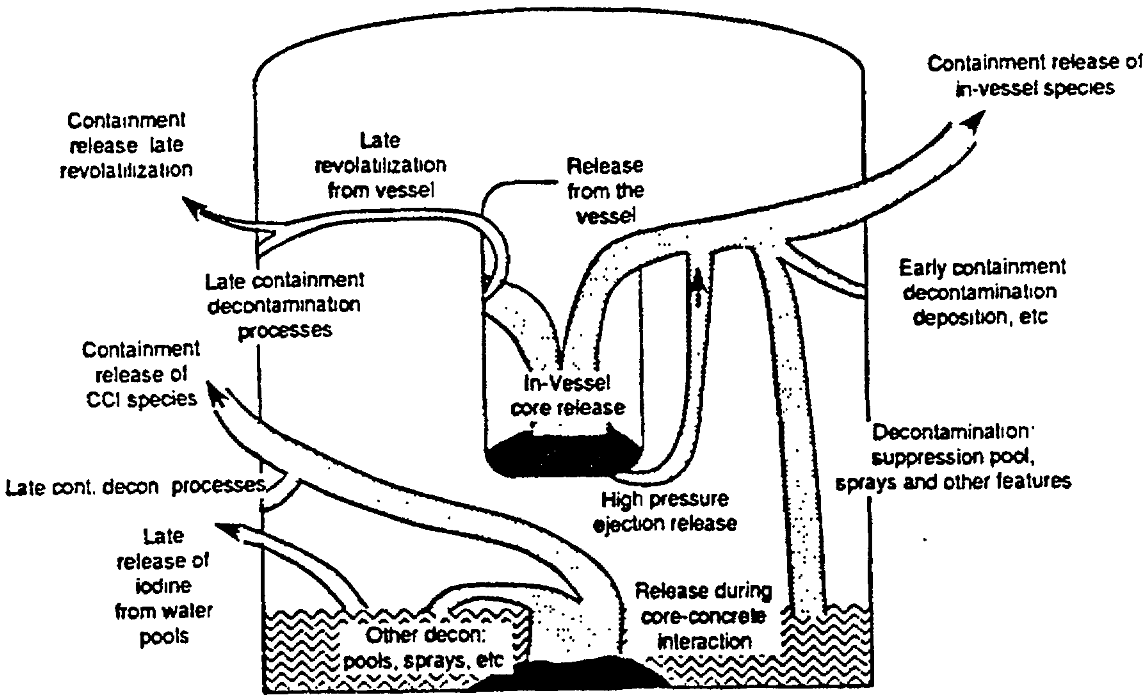


Figure A.11 Simplified schematic of source term (XSOR) algorithm.

A.4.3 Development of Values or Probability Distributions

Given the parametric equations used to define the source terms, it was necessary to define basic parameters. None of the parameters was internally calculated; the values must be specified by the user or chosen from a distribution of values by a sampling algorithm. Initially, the equations and the parameters for the equations were developed through detailed examination of the results of Source Term Code Package (STCP) analyses of selected accidents, performed specifically for the NUREG-1150 study (Refs. A.55 and A.56). Subsequent incorporation of calculations and experimental data from a variety of sources (e.g., STCP (Ref. A.42), CONTAIN (Ref. A.41), MELCOR (Ref. A.44), and other computer codes) has led to models that more broadly reflect the range of source term information available in the reactor safety research community.

With the NUREG-1150 objective of the performance of quantitative uncertainty analysis, data on the more important parameters were constructed in the form of probability distributions. Such distributions were developed using expert judgment to interpret the available data or calculations. For a few parameters that were judged of lesser importance or not considered as uncertain, single-valued estimates were used in the XSOR models. These estimates were derived from STCP and other calculations, adjusted as needed for the boundary conditions associated with the accident progression bins.

A.4.4 Grouping of Radioactive Releases

The source term calculations performed with the XSOR codes have a one-to-one correspondence with the accident progression bins. With the large number of bins used in the detailed risk analyses and the consideration of parameter uncertainties, a large number of source term calculations was required. This number of calculations was too great to be directly used in the next step in the risk analysis, the offsite consequence analysis. Therefore, the tens of thousands of source terms were grouped into about 50 groups. The source terms were grouped according to their potential for causing early fatalities, their potential for causing latent cancer fatalities, and the warning time associated with them. This grouping was accomplished with the PARTITION code (Ref. A.57). Reference A.57 explains in more detail how the early fatality and latent cancer fatality potentials and the warning times were calculated. Each source term group was represented by an average source term, where the averaging was weighted by the frequency of occurrence of the accident progression bin giving rise to that source term and where each (Monte Carlo) calculation for the uncertainty analysis was weighted equally. Characteristics such as the energy of release were not used to group the source terms, although each group was represented by an average energy of release.

A.4.5 Products of Source Term Analysis

The product of this step in the NUREG-1150 risk analysis process is the estimate of the radioactive release magnitude (in the form of a probability distribution), with associated energy content, time, and duration of release, for each of the specified source term groups.

In NUREG-1150, radioactive release magnitudes are displayed in the following ways:

- Distribution of release magnitudes for each of the nine isotopic groups for selected accident progression bins (as shown in Fig. A.12); and
- Frequency distribution (in the form of complementary cumulative distribution functions) of radioactive releases of iodine, cesium, strontium, and lanthanum (as shown in Fig. A.13).

The results of the source term analysis are the fourth input to the risk calculation, $P(APB_j \rightarrow STG_k)$, the conditional probability that accident progression bin j will lead to source term group k.

A.5 Offsite Consequence Analysis

A.5.1 Introduction

The severe reactor accident radioactive releases described in the preceding section are of concern because of their potential for impacts in the surrounding environment and population. The impacts of radioactive

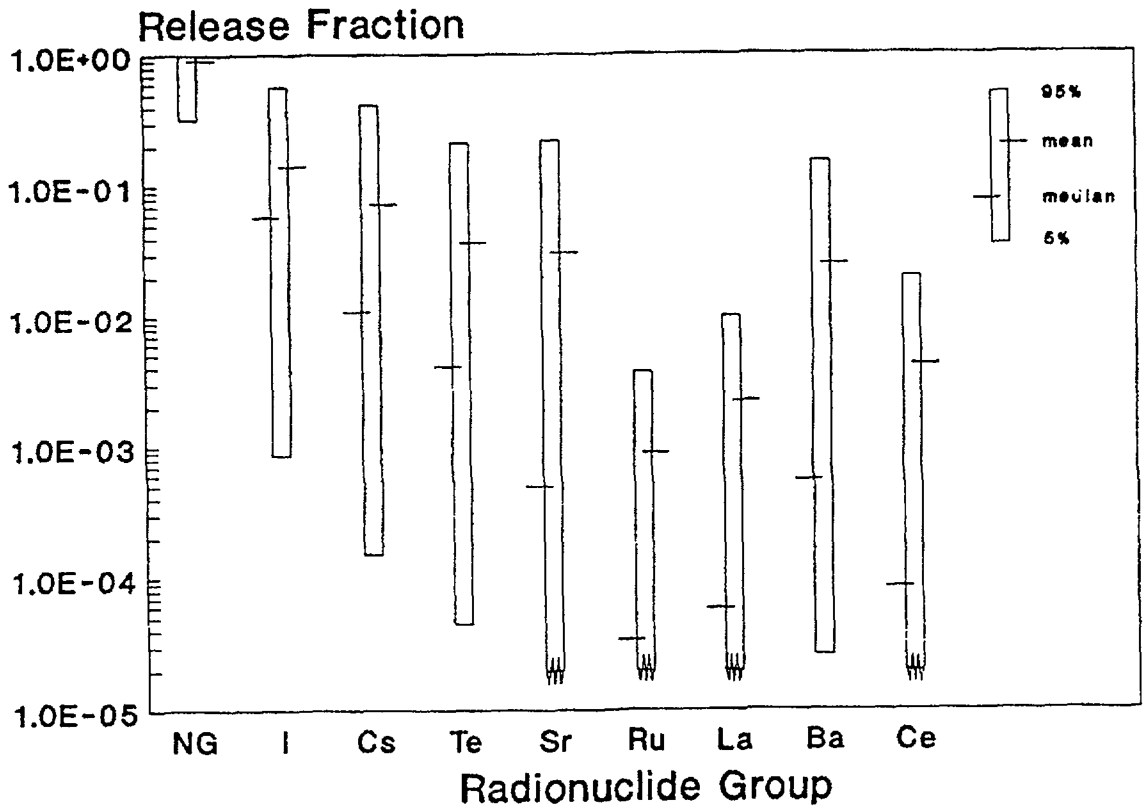


Figure A.12 Example display of radioactive release distributions for selected accident progression bin.

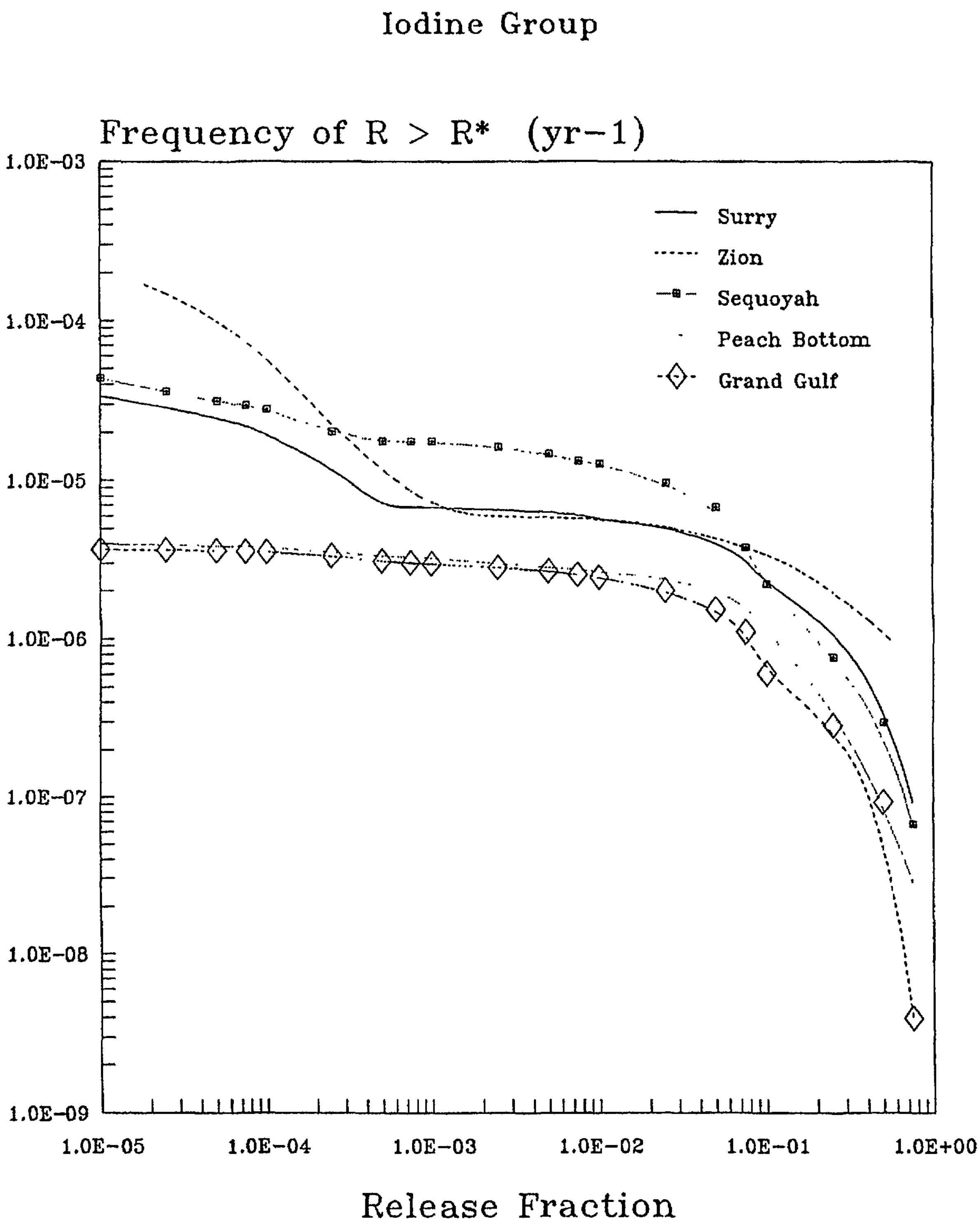


Figure A.13 Example display of source term complementary cumulative distribution function.

releases to the atmosphere from such accidents can manifest themselves in a variety of ways, such as early and delayed health effects, loss of habitability of areas close to the power plant, and economic losses. The fourth step in the NUREG-1150 risk analyses is the estimation of these offsite consequences, given the radioactive releases generated in the previous step of the analysis.

The principal steps in the offsite consequence analysis are:

- Assessment of pre-accident inventories of radioactive material;
- Analysis of the downwind transport, dispersion, and deposition of the radioactive materials released from the plant;
- Analysis of the radiation doses received by the exposed populations via direct (cloudshine, inhalation, groundshine, and deposition on skin) and indirect (ingestion) pathways;
- Analysis of the mitigation of these doses by emergency response actions (evacuation, sheltering, and relocation of people), interdiction of milk and crops, and decontamination or interdiction of land and buildings; and
- Calculation of the health effects of the release, including:
  - Number of early fatalities and early injuries expected to occur within 1 year of the accident, and the latent cancer fatalities expected to occur over the lifetimes of the exposed individuals;
  - The total population dose received by the people living within specific distances (e.g., 50 miles) of the plant; and
  - Other specified measures of offsite health effect consequences (e.g., the number of early fatalities in the population living within 1 mile of the reactor site boundary).

Each of these steps will be discussed in the following sections.

The NUREG-1150 offsite consequence calculations were performed with Version 1.5 of the MACCS (MELCOR Accident Consequence Code System) computer code (Ref. A.12).

A.5.2 Assessment of Pre-Accident Inventories

The radionuclide core inventories were calculated using the SANDIA-ORIGEN code (Ref. A.58). For PWRs, a 3412 megawatt (MW) (thermal) Westinghouse PWR was used, assuming an annual refueling cycle and an 80 percent capacity factor. The core contains 89.1 metric tons of uranium (MTU), is initially enriched to 3.3 percent U-235, and is used in a 3-year cycle, with one-third of the core being replaced each year. The specific power is 38.3 MW/MTU, which gives the burnups at the end of a 3-year cycle at 11,183 megawatt-days (MWD)/MTU, 22,366 MWD/MTU, and 33,550 MWD/MTU for each of the three regions of the core.

For BWRs, a 3578 MWT General Electric BWR-6 was used, assuming an annual refueling cycle and an 80 percent capacity factor. The core contains 136.7 MTU and has initial enrichments of 2.66 percent and 2.83 percent U-235. The 2.66 percent fuel is used for both the 3-year cycle and the 4-year cycle, while the 2.83 percent is used only for the 4-year cycle. The fuel on 4-year cycles operates at roughly average power for the first three years and is then divided into two batches for the fourth year: half going to the core center (near average power) and half going to the periphery (about half of the average power). This complex fuel management plan yields five different types of discharged spent fuel. The inventory at the end of annual refueling is then a blend of different types since the code performed the actual calculation on a per fuel assembly basis.

The core inventory of each specified plant studied was calculated by multiplying the standard PWR or BWR core inventory calculated above by the ratio of plant power level to the power level of the standard plant.

For these risk analyses, nine groups were used to represent 60 radionuclides considered to be of most importance to offsite consequences: noble gases, iodine, cesium, tellurium, strontium, ruthenium, cerium, barium, and lanthanum.

A.5.3 Transport, Dispersion, and Deposition of Radioactive Material

The MACCS code uses an empirical straightline Gaussion model for calculations of transport and dispersion of the plume that would be formed by the radioactive material released from the plant. These calculations use the sequence of successive hourly meteorological data of the reactor site for several days beginning at the release (Ref. A.12). MACCS also calculates the rise of the plume vertically while it is transported downwind if the radionuclide release is accompanied by thermal energy. Actual occurrence and the height of the plume rise would depend on the thermal release rate and the ambient meteorological conditions at the time of the release (Ref. A.59). Depletion of the plume by radioactive decay and dry and wet deposition processes during transport are taken into account. Radioactive contamination of the ground in the wake of the plume passage due to the dry and wet deposition processes is also calculated. These calculations are performed up to a very large distance, namely, 1,000 miles, from the reactor. Beyond the distance of 500 miles from the reactor, a special artifice of calculation is used to gradually deplete the plume of its remaining radionuclide content in particulate form and deposit it on the ground. The purpose of doing this is to provide a nearly complete accounting of the radionuclides released in particulate form from the plant. The impact of relatively small quantities of the noble gases (which do not deposit) leaving the 1,000-mile region is considered to be negligible. For this reason the 1,000-mile circular region is recognized as the entire impacted site region for this study.

The consequences for a given release of radioactive material would be different if the release occurred at different times of the year and under different ambient weather conditions. Consequences would also be different for different wind directions during the accident due to variations with direction in the population distribution, land use, and agricultural practice and productivity of the site region. As such, the MACCS code provides probability distributions of the consequence estimates arising from the statistical variability of seasonal and meteorological conditions during the accident. The models generally accomplish this by repeating the calculations for many weather sequences (each beginning with the release of the radioactive material) which are statistically sampled from the historical hourly meteorological data of the reactor site for 1 full year. The product of the probability of a weather sequence and the probability of wind blowing toward a direction sector of the compass provides the probability for the estimate of the magnitude of each consequence measure for this weather sequence and direction sector combination. Computer models employed in the past and present NRC studies use about 1,500 to 2,500 weather sequence and direction sector combinations. This produces a like number of magnitude and probability pairs for each consequence measure analyzed. Collectively, these pairs for a consequence measure provide a large data base to generate its meteorology-based probability distribution.

A.5.4 Calculation of Doses

MACCS calculates the radiological doses to the population resulting from several exposure pathways using a set of dose conversion factors described in References A.60 through A.62. During the early phase, which begins at the time of the radionuclide release and lasts about a week, the exposure pathways are the external radiation from the passing radioactive cloud (plume), contaminated ground, and radiation from the radionuclides deposited on the skin, and internal radiation from inhalation of radionuclides from the cloud and resuspended radionuclides deposited on the ground. Following the early phase, the long-term (chronic) exposure pathways are external radiation from the contaminated ground and internal radiation from ingestion of (1) foods (milk and crops) directly contaminated during plume passage, (2) foods grown on contaminated soil, and (3) contaminated water, and from inhalation of resuspended radionuclides.

A.5.5 Mitigation of Doses by Emergency Response Actions

In the event of a large atmospheric release of radionuclides in a severe reactor accident, a variety of emergency response and long-term countermeasures would be undertaken on behalf of the public to mitigate the consequences of the accident. The emergency response measures to reduce the doses from the early exposure pathways include evacuation or sheltering (followed by relocation) of the people in the areas relatively close to the plant site and relocation of people from highly contaminated areas farther away from the site. The long-term countermeasures include decontamination of land and property to make them usable, or temporary or permanent interdiction (condemnation) of highly contaminated land, property, and foods that cannot be effectively or economically decontaminated. These response measures are associated with expenses and losses that contribute to the offsite economic cost of the accident.

The analysis of offsite consequences for this study included a "base case" and several sets of alternative emergency response actions. For the base case, it was assumed that 99.5 percent of the population within the 10-mile emergency planning zone (EPZ) participated in an evacuation. This set of people was assumed to move away from the plant site at a speed estimated from the plant licensee's emergency plan, after an initial delay (to permit communication of the need to evacuate) also estimated from the licensee's plan. It was also assumed that the 0.5 percent of the population that did not participate in the initial evacuation was relocated within 12 to 24 hours after plume passage, based on the measured concentrations of radioactive material in the surrounding area and the comparison of projected doses with proposed Environmental Protection Agency (EPA) guidelines (Ref. A.63). Similar relocation assumptions were made for the population outside the 10-mile planning zone.

Several alternative emergency response assumptions were also analyzed in this study's offsite consequence and risk analyses. These included:

- Evacuation of 100 percent of the population within the 10-mile emergency planning zone;
- Indoor sheltering of 100 percent of the population within the EPZ (during plume passage) followed by rapid subsequent relocation after plume passage;
- Evacuation of 100 percent of the population in the first 5 miles of the planning zone, and sheltering followed by fast relocation of the population in the second 5 miles of the EPZ; and
- In lieu of evacuation or sheltering, only relocation from the EPZ within 12 to 24 hours after plume passage, using relocation criteria described above.

In each of these alternatives, the region outside the 10-mile zone was subject to a common assumption that relocation was performed based on comparisons of projected doses with EPA guidelines (as discussed above).

A.5.6 Health Effects Modeling

The potential early health effects of radioactive releases are fatalities and morbidities (injuries) occurring within about a year in the population that would receive acute and high radiological doses from the early exposure pathways. The potential delayed health effects are fatal and nonfatal cancers that may occur in the exposed population after varying periods of latency and continuing for many years; and various types of genetic effects that may occur in the succeeding generations stemming from radiological exposures of the parents. Both early and chronic exposure pathways would contribute to the latent health effects.

The early fatality models currently implemented in MACCS are based on information provided in Reference A.64. Three body organs are used in the early fatality calculations: red marrow, lung, and lower large intestine (LLI). The organ-specific early fatality threshold doses used are 150 rems, 500 rems, and 750 rems, and LD_{50} used are 400 rems, 1,000 rems, and 1,500 rems to the red marrow, lung, and LLI, respectively. The models incorporate the reduced effectiveness of inhalation dose protraction in causing early fatality and the benefits of medical treatment.

The early injury models implemented in MACCS are also threshold models and are similar to those described in Reference A.64. The candidate organs used for the current analysis are the stomach, lungs, skin, and thyroid.

The latent fatal and nonfatal cancer models implemented in MACCS are the same as described in Reference A.64, which are based on those of the BEIR III report (Ref. A.65). These models are nonthreshold and linear-quadratic types. However, only a linear model was used for latent cancer fatalities from the chronic exposure pathways since the quadratic term was small compared to the linear term because of low individual doses from these pathways. The specific organs used were red marrow (for leukemia), bone, breast, lung, thyroid, LLI, and others (based on the LLI dose representing the dose to the other organs).

Population exposure has been treated as a nonthreshold measure; truncation at low individual radiation dose levels was not performed.

A.5.7 Products of Offsite Consequence Analysis

The product of this part of the analysis is a set of offsite consequence measures for each source term group. For NUREG-1150, the specific consequence measures discussed include early fatalities, latent cancer fatalities, total population dose (within 50 miles and the entire site region), and two measures for comparison with NRC's safety goals, average individual early fatality risk within 1 mile and average individual latent fatality risk within 10 miles. In NUREG-1150, results of the offsite consequence analysis are displayed in the form of complementary cumulative distribution functions (CCDFs), as shown in Figure A.14.

The schedule for completing the risk analyses of this report did not permit the performance of uncertainty analyses for parameters of the offsite consequence analysis although variability due to annual variations in meteorological conditions is included.

The reader seeking extensive discussion of the methods used is directed to Part 7 of Reference A.46 and to Reference A.12, which discusses the computer used to perform the offsite consequence analysis (i.e., the MELCOR Accident Consequence Code System (MACCS), Version 1.5).

Through the use of the MACCS code, the fifth part of the risk calculation was developed: C_{lk}, the mean consequence (representing the meteorologically based statistical variability) for measure l given the source term group k.

A.6 Characterization and Combination of Uncertainties*

An important characteristic of the probabilistic risk analyses conducted in support of this report is that they have explicitly included an estimation of the uncertainties in the calculations of core damage frequency and risk that exist because of incomplete understanding of reactor systems and severe accident phenomena.

There are four steps in the performance of uncertainty analyses. Briefly, these are:

- *Scope of Uncertainty Analyses.* Important sources of uncertainty exist in all four stages of the risk analysis. In this study, the total number of parameters that could be varied to produce an estimate of the uncertainty in risk was large, and it was somewhat limited by the computer capacity required to execute the uncertainty analyses. Therefore, only the most important sources of uncertainty were included. Some understanding of which uncertainties would be most important to risk was obtained from previous PRAs, discussion with phenomenologists, and limited sensitivity analyses. Subjective probability distributions for parameters for which the uncertainties were estimated to be large and important to risk and for which there were no widely accepted data or analyses were generated by expert panels. Those issues for which expert panels generated probability distributions are listed in Table A.1.

- *Definition of Specific Uncertainties.* In order for uncertainties in accident phenomena to be included in this study's probabilistic risk analyses, they had to be expressed in terms of uncertainties in the parameters that were used in the study. Each section of the risk analysis was conducted at a slightly different level of detail. However, each analysis part (except for offsite consequence analysis, which was not included in the uncertainty analysis) did not calculate the characteristics of the accidents in as much detail as would a mechanistic and detailed computer code. Thus, the uncertain input parameters used in this study are "high level" or summary parameters. The relationships between fundamental physical parameters and the summary parameters of the risk analysis parts are not always clear; this lack of understanding leads to what is referred to in this study as modeling uncertainties. In addition, the values of some important physical or chemical parameters are not known and lead to uncertainties in the summary parameters. These uncertainties were referred to as data uncertainties. Both types of uncertainties were included in the study and no consistent effort was made to differentiate between the effects of the two types of uncertainties.

  As noted above, parameters were chosen to be included in the uncertainty analysis if they were estimated to be large and important to risk and if there were no widely accepted data or analysis.

*This section adapted, with editorial modification, from Section 2 of Reference A.2.

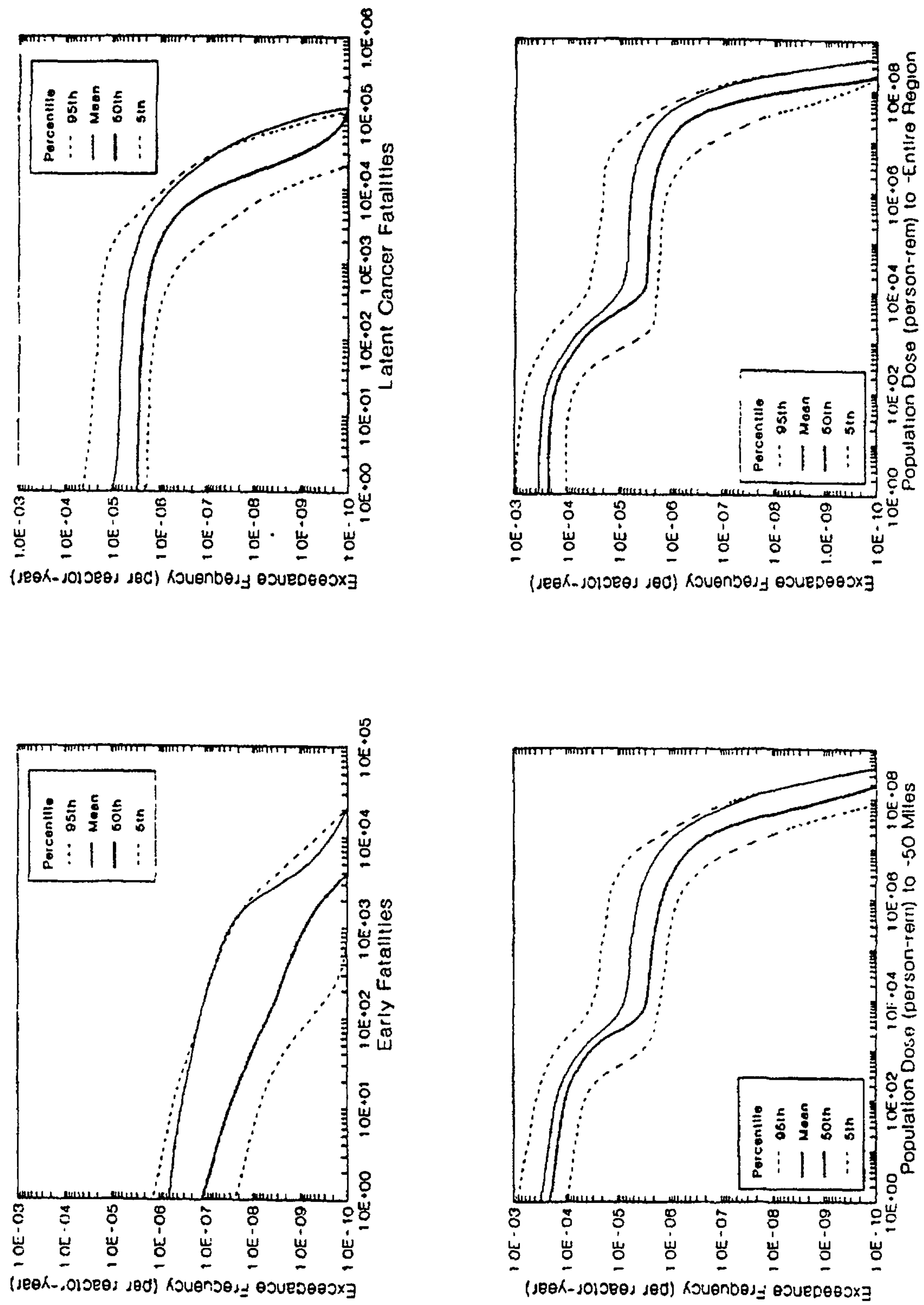


Figure A.14 Example display of offsite consequences complementary cumulative distribution function.

Table A.1 Issues considered by expert panels.

- **Accident Frequency Analysis Panel**

  Failure probabilities for check valves in the quantification of interfacing-system LOCA frequencies (PWRs)

  Physical effects of containment structural or vent failures on core cooling equipment (BWRs)

  Innovative recovery actions in long-term accident sequences (PWRs and BWRs)

  Pipe rupture frequency in component cooling water system (Zion)

  Use of high-pressure service water system as source for drywell sprays (Peach Bottom)

- **Reactor Coolant Pump Seal Performance Panel**

  Frequency and size of reactor coolant pump seal failures (PWRs)

- **In-Vessel Accident Progression Panel**

  Probability of temperature-induced reactor coolant system hot leg failure (PWRs)

  Probability of temperature-induced steam generator tube failure (PWRs)

  Magnitude of in-vessel hydrogen generation (PWRs and BWRs)

  Mode of temperature-induced reactor vessel bottom head failure (PWRs and BWRs)

- **Containment Loadings Panel**

  Containment pressure increase at reactor vessel breach (PWRs and BWRs)

  Probability and pressure of hydrogen combustion before reactor vessel breach (Sequoyah and Grand Gulf)

  Probability and effects of hydrogen combustion in reactor building (Peach Bottom)

- **Molten Core-Containment Interactions Panel**

  Drywell shell meltthrough (Peach Bottom)

  Pedestal erosion from core-concrete interaction (Grand Gulf)

- **Containment Structural Performance Panel**

  Static containment failure pressure and mode (PWRs and BWRs)

  Probability of ice condenser failure due to hydrogen detonation (Sequoyah)

  Strength of reactor building (Peach Bottom)

  Probability of drywell and containment failure due to hydrogen detonation (Grand Gulf)

  Pedestal strength during concrete erosion (Grand Gulf)

- **Source Term Expert Panel**

  In-vessel retention and release of radioactive material (PWRs and BWRs)

  Revolatization of radioactive material from the reactor vessel and reactor coolant system (early and late) (PWRs and BWRs)

  Radioactive releases during high-pressure melt ejection/direct containment heating (PWRs and BWRs)

  Radioactive releases during core-concrete interaction (PWRs and BWRs)

  Retention and release from containment of core-concrete interaction radioactive releases (PWRs and BWRs)

  Ice condenser decontamination factor (Sequoyah)

  Reactor building decontamination factor (Grand Gulf)

  Late sources of iodine (Grand Gulf)

- *Development of Probability Distributions.* Probability distributions for input parameters were developed by a number of methods. As stated previously, distributions for the input parameters having the highest uncertainties and believed to be of the largest importance to risk were determined by panels of experts. The experts used a wide variety of techniques to generate probability distributions, including reliance on detailed code calculations, extrapolation of existing experimental and accident data to postulated conditions during the accident, and complex logic networks. Probability distributions were obtained from the expert panels using formalized procedures designed to minimize bias and maximize accuracy and scrutability of the experts' results. These procedures are described in more detail in Section A.7. Probability distributions for parameters believed to be of less importance to risk were generated by analysts on the project staff or by phenomenologists from several different national laboratories using techniques like those employed with the expert panels. This list of issues assigned probability distributions for the Surry plant is provided in Section C.1 of Appendix C. Similar lists for the other plants are provided in References A.48 thorugh A.51.

- *Combination of Uncertainties.* A specialized Monte Carlo method, Latin hypercube sampling (Ref. A.15), was used to sample the probability distributions defined for the many input parameters. The sample observations were propagated through the constituent analyses to produce probability distributions for core damage frequency and risk. Monte Carlo methods produce results that can be analyzed with a variety of techniques, such as regression analysis. Such methods can treat distributions with wide ranges and can incorporate correlations between variables. Latin hypercube sampling provides for a more efficient sampling technique than straightforward Monte Carlo sampling while retaining the benefits of Monte Carlo techniques. It has been shown to be an effective technique when compared to other, more costly, methods (Ref. A.66). Since many of the probability distributions used in the risk analyses are subjective distributions, the composite probability distributions for core damage frequency and risk must also be considered subjective.

As stated in Section A.1.2, the results of the risk analysis and its constituent analyses are subjective probability distributions for the quantities in the following equation:

$$\mathrm{Risk}_{ln} = \Sigma_h \Sigma_i \Sigma_j \Sigma_k\ f_n(\mathrm{IE}_h)\ P_n(\mathrm{IE}_h \rightarrow \mathrm{PDS}_i)\ P_n(\mathrm{PDS}_i \rightarrow \mathrm{APB}_j)\ P_n(\mathrm{APB}_j \rightarrow \mathrm{STG}_k)\ C_{lk}$$

where:

Risk_{ln} = Risk of consequence measure l for observation n (consequences/year);

$f_n(\mathrm{IE}_h)$ = Frequency (per year) of initiating event h for observation n;

$P_n(\mathrm{IE}_h \rightarrow \mathrm{PDS}_i)$ = Conditional probability that initiating event h will lead to plant damage state i for observation n;

$P_n(\mathrm{PDS}_i \rightarrow \mathrm{APB}_j)$ = Conditional probability that PDS_i will lead to accident progression bin j for observation n;

$P_n(\mathrm{APB}_j \rightarrow \mathrm{STG}_k)$ = Conditional probability that accident progression bin j will lead to source term group k for observation n; and

C_{lk} = Expected value of consequence measure l conditional on the occurrence of source term group k.

With Latin hypercube sampling, the probability distributions are estimated with a limited number (about 200) of calculations of risk, each calculation being equally likely. That is, for the uncertainty analysis about 200 values of Risk_{ln} are generated. Risk_{ln} can then be described in a number of ways, such as a histogram describing the distribution of Risk_{ln} values, the average (mean) value of risk, etc. Explanations for the tables and figures in this document that show the results of the risk analysis and its constituent analyses are provided in Section A.9.

Detailed discussion of the NUREG–1150 uncertainty analysis methods is provided in Reference A.2.

A.7 Elicitation of Experts*

The risk analysis of severe reactor accidents inherently involves the consideration of parameters for which little or no experiential data exist. Expert judgment was needed to supplement and interpret the available data on these issues. The elicitation of experts on key issues was performed using a formal set of procedures, discussed in greater detail in Reference A.2. The principal steps of this process are shown in Figure A.15. Briefly, these steps are:

- *Selection of Issues*. As stated in Section A.6, the total number of uncertain parameters that could be included in the core damage frequency and risk uncertainty analyses was somewhat limited. The parameters considered were restricted to those with the largest uncertainties, expected to be the most important to risk, and for which widely accepted data were not available. In addition, the number of parameters that could be determined by expert panels was further restricted by time and resource limitations. The parameters that were determined by expert panels are, in the vernacular of this project, referred to as "issues." An initial list of issues was chosen from the important uncertain parameters by the plant analyst, based on results from the first draft NUREG-1150 analyses (Ref. A.3). The list was further modified by the expert panels.

- *Selection of Experts*. Seven panels of experts were assembled to consider the principal issues in the accident frequency analyses (two panels), accident progression and containment loading analyses (three panels), containment structural response analyses (one panel), and source term analyses (one panel). The experts were selected on the basis of their recognized expertise in the issue areas, such as demonstrated by their publications in refereed journals. Representatives from the nuclear industry, the NRC and its contractors, and academia were assigned to each panel to ensure a balance of "perspectives." Diversity of perspectives has been viewed by some (e.g., Refs. A.67 and A.68) as allowing the problem to be considered from more viewpoints and thus leading to better quality answers. The panels contained from 3 to 10 experts.

- *Training in Elicitation Methods*. Both the experts and analysis team members received training from specialists in decision analysis. The team members were trained in elicitation methods so that they would be proficient and consistent in their elicitations. The experts' training included an introduction to the elicitation and analysis methods, to the psychological aspects of probability estimation (e.g., the tendency to be overly confident in the estimation of probabilities), and to probability estimation. The purpose of this training was to better enable the experts to transform their knowledge and judgments into the form of probability distributions and to avoid particular psychological biases such as overconfidence. Additionally, the experts were given practice in assigning probabilities to sample questions with known answers (almanac questions). Studies such as those discussed in Reference A.69 have shown that feedback on outcomes can reduce some of the biases affecting judgmental accuracy.

- *Presentation and Review of Issues*. Presentations were made to each panel on the set of issues to be considered, the definition of each issue, and relevant data on each issue. Other parameters considered by the analysis staff to be of somewhat lesser importance were also described to the experts. The purposes of these presentations were to permit the panel to add or drop issues depending on their judgments as to their importance; to provide a specific definition of each issue chosen and the sets of associated boundary conditions imposed by other issue definitions; and to obtain information from additional data sources known to the experts.

  In addition, written descriptions of the issues were provided to the experts by the analysis staff. The descriptions provided the same information as provided in the presentations, in addition to reference lists of relevant technical material, relevant plant data, detailed descriptions of the types of accidents of most importance, and the context of the issue within the total analysis. The written descriptions also included suggestions of how the issues could be decomposed into their parts using logic trees. The issues were to be decomposed because the decomposition of problems has been shown to ease the cognitive burden of considering complex problems and to improve the accuracy of judgments (Ref. A.70).

*This section adapted, with editorial modification, from Section 2 of Reference A.2.

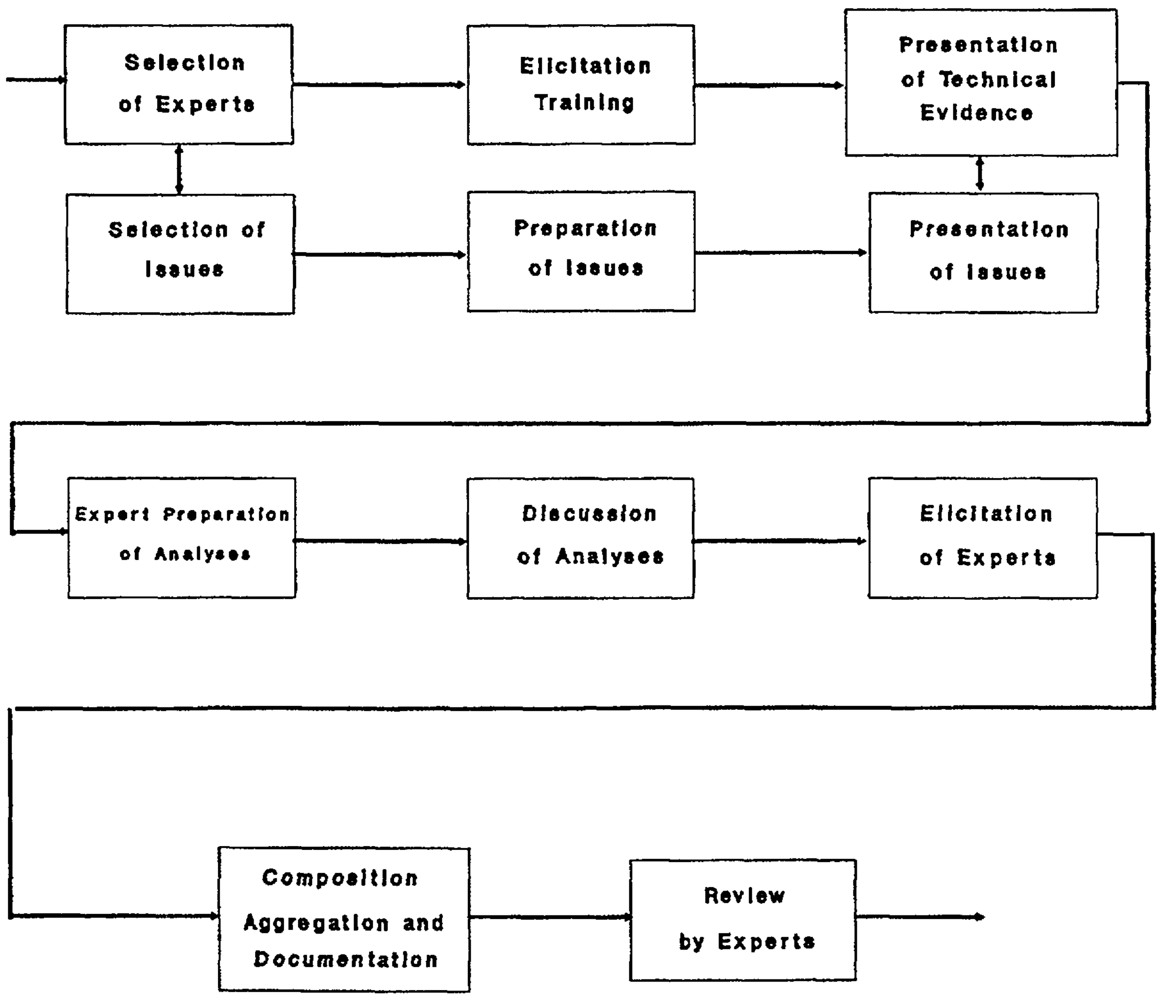


Figure A.15 Principal steps in expert elicitation process.

For the initial meeting, researchers, plant representatives, and interested parties were invited to present their perspectives on the issues to the experts. Frequently, these presentations took several days.

- *Preparation of Expert Analyses.* After the initial meeting in which the issues were presented, the experts were given time to prepare their analyses of the issues. This time ranged from 1 to 4 months. The experts were encouraged to use this time to investigate alternative methods for decomposing the issues, to search for additional sources of information on the issues, and to conduct calculations. During this period, several panels met to exchange information and ideas concerning the issues. During some of these meetings, expert panels were briefed by the project staff on the results from other expert panels in order to provide the most current data.
- *Expert Review and Discussion.* After the expert panels had prepared their analyses, a final meeting was held in which each expert discussed the methods he/she used to analyze the issue. These discussions frequently led to modifications of the preliminary judgments of individual experts. However, the experts' actual judgments were not discussed in the meeting because group dynamics can cause people to unconsciously alter their judgments in the desire to conform (Ref. A.71).
- *Elicitation of Experts.* Following the panel discussions, each expert's judgments were elicited. These elicitations were performed privately, typically with an individual expert, an analysis staff member trained in elicitation techniques, and an analysis staff member familiar with the technical subject. With few exceptions, the elicitations were done with one expert at a time so that they could be performed in depth and so that an expert's judgments would not be adversely influenced by other experts. Initial documentation of the expert's judgments and supporting reasoning were obtained in these sessions.
- *Composition and Aggregation of Judgments.* Following the elicitation, the analysis staff composed probability distributions for each expert's judgments. The individual judgments were then aggregated to provide a single composite judgment for each issue. Each expert was weighted equally in the aggregation because this simple method has been found in many studies (e.g., Ref. A.72) to perform the best.
- *Review by Experts.* Each expert's probability distribution and associated documentation developed by the analysis staff was reviewed by that expert. This review ensured that potential misunderstandings were identified and corrected and that the issue documentation properly reflected the judgments of the expert.

Detailed documentation of the expert elicitations is provided in References A.46 and A.73.

A.8 Calculation of Risk*

A.8.1 Methods for Calculation of Risk

The constituent parts of the risk calculation have been described in previous sections. As illustrated in Figure A.3, a number of computer codes were used to generate a variety of intermediate information. This information is then processed by an additional code, RISQUE, to calculate risk. RISQUE is a matrix manipulation code. As illustrated in Figure A.16 and explained in Section A.1.2, the elements of the risk calculation can be represented in a vector/matrix format.

The initiating event frequencies f(IE) constitute a vector of n_{IE} dimensions, where n_{IE} is the number of initiating events. The plant damage state frequencies f(PDS) constitute a vector of n_{PDS} dimension, where n_{PDS} is derived from f(IE) by multiplying it by the n_{IE} by n_{PDS} matrix [P(IE → PDS)]. $P(IE_h \rightarrow PDS_i)$ is the conditional probability that initiating event h will result in plant damage state i. In the detailed analyses underlying this study, there are approximately 20 plant damage states. The f(PDS) vector is a product of the accident frequency analysis.

Similarly, to obtain the accident progression bin frequencies, the plant damage state vector is multiplied by the accident progression tree output matrix [P(PDS → APB)]. The [P(PDS → APB)] matrix is the principal product of the accident progression analysis. This n_{PDS} by n_{APB} matrix represents the conditional

*This section adapted, with editorial modification, from Section 2 of Reference A.2.

Systems Analysis	Accident Progression Analysis	Source Term Analysis	Consequence Analysis	Risk Results

LHS#1

$$\underset{f(PDS)}{\begin{bmatrix} f_1 \;\cdots\cdots\; f_{n_{PDS}} \end{bmatrix}} \times \underset{P(PDS \rightarrow APB)}{\begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{APB}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{PDS}1} & & \cdots\cdots P_{n_{PDS}n_{APB}} \end{bmatrix}} \times \underset{P(APB \rightarrow STG)}{\begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{STG}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{APB}1} & & \cdots\cdots P_{n_{APB}n_{STG}} \end{bmatrix}} \times \underset{C(STG)}{\begin{bmatrix} C_{11} & C_{12} & \cdots\cdots C_{1n_C} \\ C_{21} & & \\ \vdots & & \\ C_{n_{STG}1} & & \cdots\cdots P_{n_{STG}n_C} \end{bmatrix}} = \underset{RISK}{\begin{bmatrix} RISK_1 \;\cdots\cdots\; RISK_{n_C} \end{bmatrix}}$$

LHS#2

$$\underset{f(PDS)}{\begin{bmatrix} f_1 \;\cdots\cdots\; f_{n_{PDS}} \end{bmatrix}} \times \underset{P(PDS \rightarrow APB)}{\begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{APB}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{PDS}1} & & \cdots\cdots P_{n_{PDS}n_{APB}} \end{bmatrix}} \times \underset{P(APB \rightarrow STG)}{\begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{STG}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{APB}1} & & \cdots\cdots P_{n_{APB}n_{STG}} \end{bmatrix}} \times \underset{C(STG)}{\begin{bmatrix} C_{11} & C_{12} & \cdots\cdots C_{1n_C} \\ C_{21} & & \\ \vdots & & \\ C_{n_{STG}1} & & \cdots\cdots P_{n_{STG}n_C} \end{bmatrix}} = \underset{RISK}{\begin{bmatrix} RISK_1 \;\cdots\cdots\; RISK_{n_C} \end{bmatrix}}$$

$$\vdots$$

LHS#n_{LHS}

$$\underset{f(PDS)}{\begin{bmatrix} f_1 \;\cdots\cdots\; f_{n_{PDS}} \end{bmatrix}} \times \underset{P(PDS \rightarrow APB)}{\begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{APB}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{PDS}1} & & \cdots\cdots P_{n_{PDS}n_{APB}} \end{bmatrix}} \times \underset{P(APB \rightarrow STG)}{\begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{STG}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{APB}1} & & \cdots\cdots P_{n_{APB}n_{STG}} \end{bmatrix}} \times \underset{C(STG)}{\begin{bmatrix} C_{11} & C_{12} & \cdots\cdots C_{1n_C} \\ C_{21} & & \\ \vdots & & \\ C_{n_{STG}1} & & \cdots\cdots P_{n_{STG}n_C} \end{bmatrix}} = \underset{RISK}{\begin{bmatrix} RISK_1 \;\cdots\cdots\; RISK_{n_C} \end{bmatrix}}$$

Figure A.16 Matrix formulation of risk analysis calculation.

probability that an accident grouped in plant damage state l will result in an accident grouped in the jth accident progression bin. In the detailed analyses underlying this study, there are between a few hundred and a few thousand accident progression bins (n_{APB} = 1000) depending on the plant.

The result of the previous calculation is multiplied by a third matrix that represents the outcome of the source term and partitioning analyses [P(APB $\rightarrow$ STG)]. This n_{APB} by n_{STG} matrix represents the conditional probability that an accident progression bin j will be assigned to source term group k. There are approximately 50 source term groups (n_{STG} = 50). This yields a vector f(STG) of frequencies of the source term groups.

The final element of the risk calculation is a matrix representing the consequences for each of the source term groups C. The n_{STG} by n_C matrix is the product of the consequence analysis, where n_C represents the number of consequence measures. For this study, eight consequence measures were calculated (n_C = 8). Risk is the product of the frequency vector for the source term groups f(STG) and the consequence matrix C. Risk is an eight-component vector, for the eight consequence measures, and represents consequences averaged over the source term groups.

There are n_{LHS} sets of vectors and matrices described above, one for each sample member. Each sample member represents a unique set of values for each uncertainty issue and is equally likely. Since consequence uncertainty was not included in LHS sampling, only one consequence matrix C is required; the last term in Figure A.16 is the same for each and every sample member.

The matrix manipulations described above were carried out using the RISQUE code. The risk calculation is a fairly straightforward process, but the number of numerical manipulations is large, since the risk vector must be calculated n_{LHS} times, where n_{LHS} is 150 for the Zion calculation, 200 for the Surry, Sequoyah, and Peach Bottom calculations, and 250 for the Grand Gulf calculation. Results form a distribution in risk values that represent the uncertainty associated with the issues.

The Monte Carlo-based techniques are amenable to statistical examination to provide insights concerning the result. Descriptive statistics such as central measures, variance, and range can be calculated. The relative importance of the issues to uncertainty in risk can be determined through examination of the results with statistical techniques such as regression analysis. The individual observations can also be examined. For example, if the final distribution contains some results that are quite different from all the others (say five observations an order of magnitude higher in consequences than any other observations), the individual five sample members can be examined as separate complete risk analyses to determine the important effects causing the overall result.

One of the key developments in this program is the automation of the risk assembly process. The most significant advantage of this methods package is the ability to recalculate an entire risk result very efficiently, even given major changes in the constituent analyses. The manipulation of these models in sensitivity studies allows efficient, focused examination of particular issues and significant ability for examining changes in the plants or in the analysis.

The objectives of the program included not only calculations and conclusions concerning the risk results, but also intermediate results were quite important. Each of the analysis steps resulted in intermediate outputs. The intermediate outputs were examined by analysts to ensure the correctness of each step. The nomenclature and representation of the results described in this section are used consistently throughout the documentation of both the methods and the results for a specific plant. The same intermediate results are illustrated for each facility, and the terminology used to describe those results is consistent with that developed here.

A.8.2 Products of Risk Calculation

The risk analyses performed in the NUREG-1150 project can be displayed in a variety of ways. The specific products shown in NUREG-1150 are described in the following sections, with similar products provided for early fatality risk, latent cancer fatality risk, average individual early fatality risk within 1 mile (for comparison with NRC safety goals (Ref. A.14)), average individual latent cancer fatality risk within 10 miles of the site boundary (for safety goal comparison), population dose risk within 50 miles, and population dose risk within the entire region.

- The total risk from internal events and, where estimated, for external events

  Reflecting the uncertain nature of risk results, such results can be displayed using a probability distribution. For Part II of NUREG–1150 (plant-specific results), a histogram is used to represent this probability distribution (like that shown on the right side of Fig. A.6). Four measures of the probability distribution are identified in NUREG–1150:

  - Mean,
  - Median,
  - 5th percentile, and
  - 95th percentile.

  A second display of risk results is used in Part III of this report, where results for all five plants are displayed together. This rectangular display (shown on the left side of Fig. A.6) provides a summary of these four specific measures in a simple graphical form.

- Contributions of plant damage states and accident progression bins to mean risk

  The risk results generated in the NUREG–1150 project can be studied to determine the relative contribution of individual plant damage states and accident progression bins to the mean risk. An example display of the results of such a study is shown in Figure A.17.

A.9 Additional Explanation of Some Figures, Tables, and Terms

A.9.1 Additional Explanation of Some Figures and Tables

Most of the results presented in this report are generalized or summary results. They are similar to the intermediate results described in Section A.8.1. However, the groupings of postulated accidents that take place at the end of each constituent part of the risk calculation are more general in this document than in the contractor reports and than described in Section A.8.1. For example, in reporting the results for the Surry power plant, only five (summary) plant damage states are used, rather than the nine plant damage states described in the supporting documents. The descriptions of the results at both levels of detail are consistent with each other, and one can derive the more generalized results presented in this document from those presented in the supporting documents. Details of this derivation are presented in the supporting documents.

Since a Latin hypercube sample of size n_{LHS} is being used for the risk analyses, there are n_{LHS} values of the generalized frequency vectors f(IE), f(PDS), f(APB), f(STG), and RISK. (PDS, APB, and STG refer to the generalized groupings of projected accidents used in this report.) Due to the nature of Latin hypercube sampling, each of these observations has probability equal to $1/n_{LHS}$. Thus, the mean value of the ith element of the vector f(PDS), (i.e., $f(PDS_i)$) is given by

$$f(PDS_i)_{mean} = \Sigma_n f(PDS_i)_n / n_{LHS}$$

where $f(PDS_i)_n$ is the frequency of the generalized plant damage state i for Latin hypercube member n. Further, individual analysis results for the n_{LHS} sample elements can be ordered from the smallest to the largest and then used to estimate desired quantiles (i.e., 5th, median, and 95th), where the 'q'th quantile is the value of the variable that is greater than or equal to the 'q' of the observed results. Median is the commonly used term for the 50th quantile.

The n_{LHS} values of $f(PDS_i)$ can also be used to construct estimated probability density functions for $f(PDS_i)$. The estimated density function is constructed by discretizing the range of values of $f(PDS_i)$ into a number of equal intervals. The estimated density function over each of these intervals is the fraction of Latin hypercube members with values that fall within that interval. In Figure A.18, P_m is an estimate of the probability that $f(PDS_i)$ will fall in interval I_m. However, because most of the histograms/density plots presented in NUREG–1150 span several orders of magnitude, the plots are provided on a logarithmic scale. Thus, the corresponding histogram/density functions presented are for the logarithm of the variable under consideration. In these cases, the histogram/density functions represent the probability that the

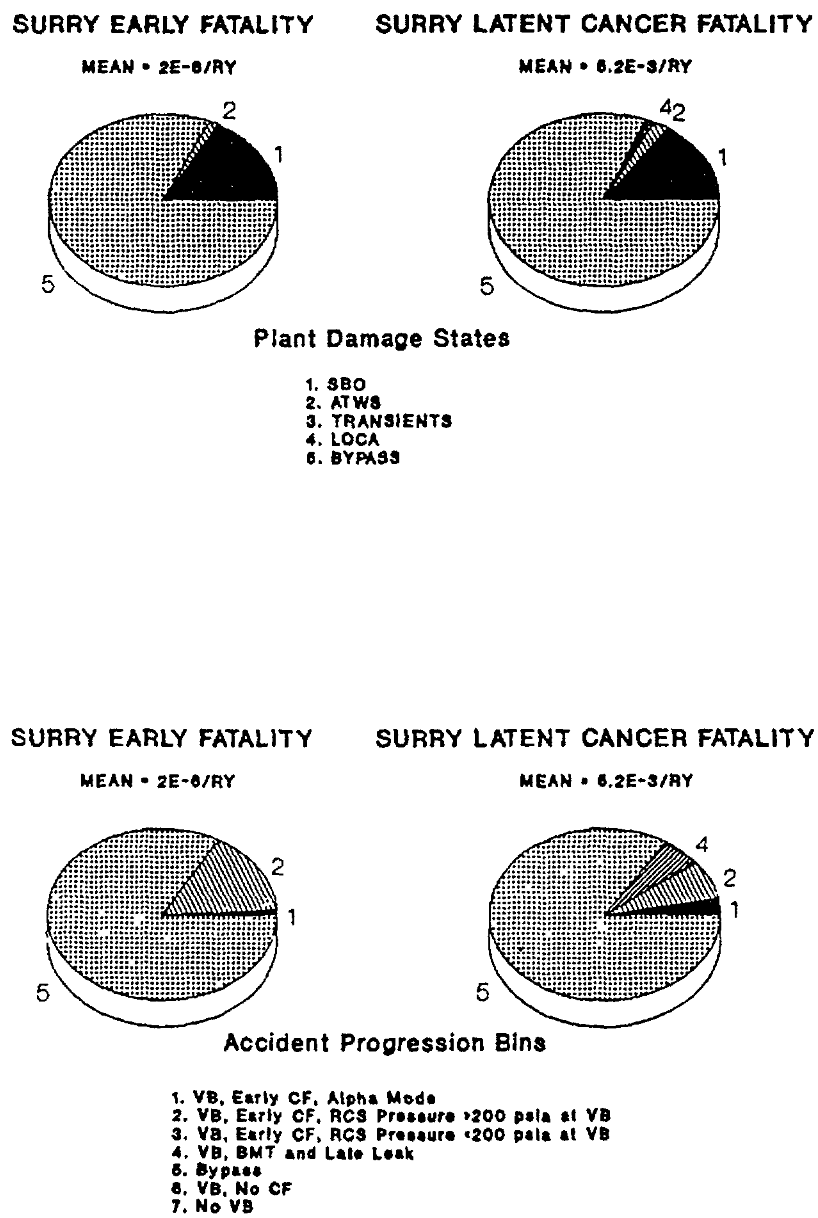


Figure A.17 Example display of relative contributions to mean risk.

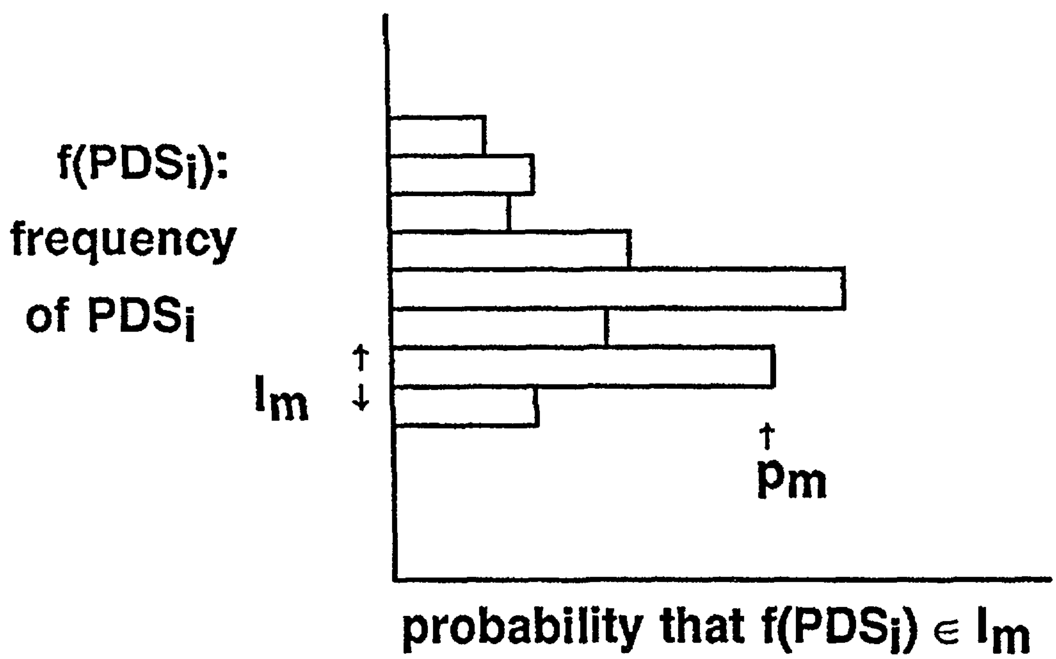


Figure A.18 Probability that $f(PDS_i)$ will fall in interval I_m.

logarithm of the variable falls in various intervals. Whether a density function is for a variable or its logarithm can be recognized by the scale used on the axis corresponding to the variable.

Explanation of Figure A.6: Figure A.6 represents an estimated probability density function, as explained above, for the total core damage frequency. The total core damage frequency for a single observation is related to the vector $f(PDS_h)$ by

$$\text{total core damage frequency} = \text{TCDF} = \Sigma_i\ f(PDS_i).$$

Total core damage frequency is calculated for each observation and used to estimate a core damage histogram as described above.

Explanation of Figure A.7: Figure A.7 shows the mean value of the total core damage frequency, where the mean is over all the Latin hypercube sample members, as explained above. The fractional contributions indicated by sections of the pie charts are the ratios of the mean values of the frequencies of the summary plant damage states $f(PDS_i)$ to the mean value of the total core damage frequency.

Explanation of Figure A.10: Figure A.10 is a table of mean transition probabilities (the mean taken over all Latin hypercube members) of the matrix $(P(PDS \rightarrow APB))$, using summary plant damage states and summary accident progression bins. The summary plant damage states and accident progression bins are described in the figure and the figure key.

Explanation of Figures A.13 and A.14: The results of the risk analyses are also used in the construction of complementary cumulative distribution functions (CCDFs). Examples of mean CCDFs appear in Figures A.13 and A.14. The CCDFs in Figure A.13 are for source term magnitude. The CCDFs in Figure A.14 are for consequence results and incorporate both stochastic weather variation and variation/uncertainty in accident initiation, progression, and source term characteristics. In figures of this type, the value on the ordinate (y-axis) gives the frequency at which the corresponding value on the abscissa (x-axis) is exceeded. A discussion of the construction of the CCDFs is provided in Appendix B.

A.9.2 Explanation of Some Terms

An *uncertain variable* (often called a *random variable* in statistical texts) can take on any of several possible values, but it is impossible to predict which value will be observed in any given trial. The possible specific values are called *realizations* of the uncertain variable. Although there is no precise knowledge which realization will occur, there is a rule that tells which of the possible realizations is most likely; in fact, the rule quantifies the likelihood of each possible realization. The rule is called a probability distribution. For any possible realization, the *probability distribution* tells the probability of that value occurring.

There is controversy about the meaning of the probability distribution. The two principal interpretations are the *frequentist* and the *subjective* approaches. The frequentist orientation defines the probability as the frequency of obtaining the specific value in a very long number of independent trials. For example, if the uncertain variable took the value x1 500 times out of 1000 trials, then the probability attached to the value x1 is 0.50. The subjective approach defines the probability as an individual's degree of belief in the likelihood of obtaining the specific value. The subjective probability can be defined as the odds that an individual would be equally willing to give or take on a bet that the uncertain variable would have the specific value. For example, if an individual will accept even money odds that the uncertain variable will have the value x1 and is equally willing to take either side of the bet, then his probability for the value x1 is 0.50.

For many variables, the probability distribution for their realizations is unknown or the laws of nature affecting the probability distribution are imperfectly understood. However, an expert might understand which laws could apply and have an opinion as to which law is more likely. If the expert combines his knowledge of the known parts of the situation with his opinions about the relevant unknown parts, he can develop a personal estimate of the probability distribution. This is a *subjective probability distribution* (SPD). It is subjective because it varies from one expert to another. SPDs are manipulated by precisely the same rules as probability distributions developed from a frequentist approach.

If, in a group of experts who are representative of the possible pool of experts, each expert produces a subjective probability distribution, the distributions of the group members can be *aggregated* or combined in such a way that the aggregate distribution can be generalized to the entire pool of possible experts.* The most important uncertain variables of this study were developed by groups of experts and so aggregated.

There is an important difference in interpretation between subjective probability distributions and data-based probability distributions. The latter represent the probability that a specific value *will* occur on a given trial. The SPD expresses a degree of belief that the value *might* occur. The distribution can be considered a distribution of belief rather than of knowledge. It must not be supposed that any value will be realized with the probability indicated by the SPD, nor even that an occurrence must be contained within the experts' aggregated range. However, although experts are sometimes wrong, the aggregated opinions of experts should be superior to the opinions of non-experts.

Most of the variables in this study are actually continuous and have an infinite number of possible realizations. Almost all uncertain variables have a minimum possible value and a maximum possible value; the distance between the two is the *range* of the uncertain variable. The probability that the uncertain variable will take on just one value out of an infinite number of possible values within the range is zero. However, it is possible to speak of the *density* of probability about any specific value. The rule that describes the density of probability over the range of the variable is the *probability density function* (PDF). It is the probability that a realization will occur within the neighborhood of each value, divided by the width of the neighborhood. The integral of the PDF over the range is 1.0; this says that any realization must be within the range. The integral of the PDF between the minimum value of the range and any specific point in the range is the probability that the next realization will have a value less than or equal to the specific point. If the integral is carried out for every point in the range, the resulting function is the *cumulative distribution function* (CDF) or *cumulative probability distribution* (CPD). The CDF was used to characterize the uncertainty in each of the sampled variables considered in this study but does not generally appear in this report.

The *complementary cumulative distribution function* (CCDF) is closely related to the CDF. It is the probability that the "true" realization will be greater than any specific point in the range. The CCDF is simply 1.0 minus the CDF at every point. The CCDF is used in some instances in this report.

The PDF is difficult to compute accurately from a limited sample of data. However, the PDF can be approximated by the *frequency histogram*. This is the number of observations falling in each finite interval of the range. If the intervals are suitably chosen, the frequency histogram can be a good approximation of the PDF. Frequency histograms are often used in this report.

Initiating events are characterized by their frequency—the number of times such events can be expected to occur per year. As long as the frequency is substantially less than 1.0, this is equivalent to the probability of the event occurring in any given year. Succeeding events are characterized by their *conditional probability*. The conditional probability of B given A is the probability that B will occur if A has already occurred. The characterization of succeeding events can also be thought of as a *relative frequency*, that is, their frequency relative to the frequency of the preceding event. The methods for manipulation of chains of conditional probabilities are well known.

Additional information on statistics and probability can be found in References A.74 through A.78.

*This is so because (absent any other information about the population) the sample mean is the best estimate of the population mean, and the population mean (absent any special information about individuals in the population) is the best estimate of the responses of any member of the population.

REFERENCES FOR APPENDIX A

A.1 D. M. Ericson, Jr., (Ed.) et al., "Analysis of Core Damage Frequency: Internal Events Methodology," Sandia National Laboratories, NUREG/CR-4550, Vol. 1, Revision 1, SAND86-2084, January 1990.

A.2 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.*

A.3 U.S. Nuclear Regulatory Commission (USNRC), "Reactor Risk Reference Document," NUREG-1150, Vols. 1-3, Draft for Comment, February 1987.

A.4 F. T. Harper et al., "Analysis of Core Damage Frequency from Internal Events: Methodology Guidelines," Sandia National Laboratories, NUREG/CR-4550, Vol. 1, SAND86-2084, September 1987.

A.5 A. S. Benjamin et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Surry Power Station, Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 1, Draft for Comment, SAND86-1309, February 1987.

A.6 A. S. Benjamin et al., "Containment Event Analysis for Postulated Severe Accidents: Surry Nuclear Power Station, Unit 1," Sandia National Laboratories, NUREG/CR-4700, Vol. 1, Draft for Comment, SAND86-1135, February 1987.

A.7 H. J. C. Kouts et al., "Methodology for Uncertainty Estimation in NUREG-1150 (Draft): Conclusions of a Review Panel," Brookhaven National Laboratory, NUREG/CR-5000, BNL-NUREG-52119, December 1987.

A.8 W. E. Kastenberg et al., "Findings of the Peer Review Panel on the Draft Reactor Risk Reference Document, NUREG-1150," Lawrence Livermore National Laboratory, NUREG/CR-5113, UCID-21346, May 1988.

A.9 L. LeSage et al., "Initial Report of the Special Committee on Reactor Risk Reference Document (NUREG-1150)," American Nuclear Society, April 1988.

A.10 J. M. Griesmeyer and L. N. Smith, "A Reference Manual for the Event Progression Analysis Code (EVNTRE)," Sandia National Laboratories, NUREG/CR-5174, SAND88-1607, September 1989.

A.11 H-N Jow et al., "XSOR Codes User's Manual," Sandia National Laboratories, NUREG/CR-5360, SAND89-0943, to be published.*

A.12 D. I. Chanin, H-N Jow, J. A. Rollstin et al., "MELCOR Accident Consequence Code System (MACCS)," Sandia National Laboratories, NUREG/CR-4691, Vols. 1-3, SAND86-1562, February 1990.

A.13 L. T. Ritchie et al., "CRAC2 Model Description," Sandia National Laboratories, NUREG/CR-2552, SAND82-0342, April 1984.

A.14 USNRC, "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement," *Federal Register*, Vol. 51, p. 30028, August 21, 1986.

A.15 M. D. McKay et al., "A Comparison of Three Methods for Selecting Values in Input Variables in the Analysis of Output from a Computer Code," *Technometrics* 21(2), 1979.

A.16 M. T. Drouin et al., "Analysis of Core Damage Frequency: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 6, Revision 1, SAND86-2084, September 1989.

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

A.17 A. D. Swain III, "Accident Sequence Evaluation Program—Human Reliability Analysis Procedure," Sandia National Laboratories, NUREG/CR-4772, SAND86-1996, February 1987.

A.18 W. J. Luckas, Jr., "A Human Reliability Analysis for the ATWS Accident Sequence with MSIV Closure at the Peach Bottom Atomic Power Station," Brookhaven National Laboratory, May 1986.

A.19 D. W. Stack, "A SETS User's Manual for Accident Sequence Analysis," Sandia National Laboratories, NUREG/CR-3547, SAND83-2238, January 1984.

A.20 R. L. Iman and M. J. Shortencarier, "A User's Guide for the Top Event Matrix Analysis Code (TEMAC)," Sandia National Laboratories, NUREG/CR-4598, SAND86-0960, August 1986.

A.21 M. B. Sattison and K. W. Hall, "Analysis of Core Damage Frequency: Zion Unit 1," Idaho National Engineering Laboratory, NUREG/CR-4550, Vol. 7, Revision 1, EGG-2554, May 1990.

A.22 Commonwealth Edison Company of Chicago, "Zion Probabilistic Safety Study," September 1981.

A.23 D. L. Berry et al., "Review and Evaluation of the Zion Probabilistic Safety Study: Plant Analysis," Sandia National Laboratories, NUREG/CR-3300, SAND83-1118, May 1984.

A.24 USNRC, "Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

A.25 Institute of Electrical and Electronics Engineers, Inc. (IEEE), "IEEE Guide to the Collection and Presentation of Electrical, Electronic, Sensing Component, and Mechanical Equipment Reliability Data for Nuclear-Power Generating Stations," IEEE Standard 500-1984, 1983.

A.26 Houston Lighting and Power Company, "South Texas Project Probabilistic Safety Assessment," April 1989.

A.27 K. D. Russell et al., "Integrated Reliability and Risk Analysis System (IRRAS) Version 2.0 User's Guide," Idaho National Engineering Laboratory, NUREG/CR-5111, EGG-2535, June 1990.

A.28 R. C. Bertucio and J. A. Julius, "Analysis of Core Damage Frequency: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Revision 1, SAND86-2084, April 1990.

A.29 G. E. Cummings, "Summary Report on the Seismic Safety Margins Research Program," Lawrence Livermore National Laboratory, NUREG/CR-4431, UCID-20549, January 1986.

A.30 M. P. Bohn et al., "Application of the SSMRP Methodology to the Seismic Risk at the Zion Nuclear Power Plant," Lawrence Livermore National Laboratory, NUREG/CR-3428, UCRL-53483, January 1984.

A.31 D. L. Bernreuter et al., "Seismic Hazard Characterization of 69 Nuclear Power Sites East of the Rocky Mountains," Lawrence Livermore National Laboratory, NUREG/CR-5250, Vols. 1-8, UCID-21517, January 1989.

A.32 Seismicity Owners Group and Electric Power Research Institute, "Seismic Hazard Methodology for the Central and Eastern United States," EPRI NP-4726, July 1986.

A.33 J. E. Richardson, USNRC, letter to R. A. Thomas, Seismicity Owners Group, "Safety Evaluation Review of the SOG/EPRI Topical Report Titled 'Seismic Hazard Methodology for the Central and Eastern United States,' " dated September 20, 1988.

A.34 U.S. Army Corps of Engineers, Huntsville Division, SAFEGUARD Shock Test Program, HNDDSP-72-151-ED-R, Vol. 1, 1973.

A.35 M. P. Bohn and J. A. Lambright, "Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150," Sandia National Laboratories, NUREG/CR-4840, SAND88-3102, November 1990.

A.36 P. B. Schnabel, J. Lysmer, and H. B. Seed, "SHAKE—A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites," Earthquake Engineering Research Center, University of California at Berkeley, EERC 72-12, 1972.

A.37 H. L. Wong and J. E. Luco, "Soil-Structure Interaction: A Linear Continuum Mechanics Approach (CLASSI)," Department of Civil Engineering, University of Southern California, CE79-03, 1980.

A.38 Code of Federal Regulations, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to Part 50, "Domestic Licensing of Production and Utilization Facilities," of Title 10, "Energy."

A.39 V. Ho et al., "COMPBRN III—A Computer Code for Modeling Compartment Fires," University of California at Los Angeles, UCLA-ENG-8524, November 1985.

A.40 S. S. Dosanjh, "MELPROG-PWR/MOD1: A Two-Dimensional, Mechanistic Code for Analysis of Reactor Core Melt Progression and Vessel Attack Under Severe Accident Conditions," Sandia National Laboratories, NUREG/CR-5193, SAND88-1824, May 1989.

A.41 K. D. Bergeron et al., "User's Manual for CONTAIN 1.0, A Computer Code for Severe Reactor Accident Containment Analysis," Sandia National Laboratories, NUREG/CR-4085, SAND84-1204, July 1985.

A.42 J. A. Gieseke et al., "Source Term Code Package: A User's Guide," Battelle Columbus Division, NUREG/CR-4587, BMI-2138, July 1986.

A.43 Fauske and Associates, Inc., "MAAP Modular Accident Analysis Program User's Manual, " Vols. I and II, IDCOR Technical Report 16.2-3, February 1987.

A.44 R. M. Summers et al., "MELCOR In-Vessel Modeling," *Proceedings of the Fifteenth Water Reactor Safety Information Meeting* (Gaithersburg, MD), NUREG/CP-0091, Feburary 1988.

A.45 N. K. Tutu et al., "Estimation of Containment Loading Due to Direct Containment Heating for the Zion Plant," Brookhaven National Laboratory, NUREG/CR-5282, BNL-NUREG-52181, to be published.*

A.46 F. T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

A.47 R. J. Breeding et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, October 1990.

A.48 A. C. Payne, Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft Revision 1, SAND86-1309, to be published.*

A.49 J. J. Gregory et al., "Evaluation of Severe Accident Risks: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, December 1990.

A.50 T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 6, Draft Revision 1, SAND86-1309, to be published.*

A.51 C. K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

A.52 K. Mokhtarian et al., "MARK I Containment Severe Accident Analysis for the MARK I Owners' Group," CBI NA-CON, Inc., April 1987.

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

A.53 D. B. Clauss, "Comparison of Analytical Predictions and Experimental Results for a 1:8-Scale Steel Containment Model Pressurized to Failure," Sandia National Laboratories, NUREG/CR-4209, SAND85-0679, September 1985.

A.54 S. J. Higgins, "A User's Manual for the Postprocessing Program PSTEVNT," Sandia National Laboratories, NUREG/CR-5380, SAND88-2988, November 1989.

A.55 R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios," Battelle Columbus Division, NUREG/CR-4624, Vols. 1-5, BMI-2139, July 1986.

A.56 R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios: Supplemental Calculations," Battelle Columbus Division, NUREG/CR-4624, Vol. 6, BMI-2139, August 1990.

A.57 R. L. Iman et al., "PARTITION: A Program for Defining the Source Term/Consequence Analysis Interface in the NUREG-1150 Probabilistic Risk Assessments," Sandia National Laboratories, NUREG/CR-5253, SAND88-2940, May 1990.

A.58 D. Bennett, "SANDIA-ORIGEN User's Manual," Sandia National Laboratories, NUREG/CR-0987, SAND79-0299, December 1979.

A.59 G. A. Briggs, "Plume Rise Prediction," *Proceedings of Workshop: Lectures on Air Pollution and Environmental Analysis*, American Meteorological Society, 1975.

A.60 D. C. Kocher, "Dose Rate Conversion Factors for External Exposure to Photons and Electrons," Oak Ridge National Laboratory, NUREG/CR-1918, ORNL/NUREG-79, August 1981.

A.61 International Commission on Radiological Protection, "Recommendations of ICRP," Publication 26, *Annals of ICRP*, Vol. l, No. 3, 1977.

A.62 International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," Publication 30, *Annals of ICRP*, Vol. 2, Nos. 3 and 4, 1978.

A.63 U. S. Environmental Protection Agency, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," Office of Radiation Programs, Draft, 1989.

A.64 J. S. Evans et al., "Health Effects Model for Nuclear Power Plant Accident Consequence Analysis," Harvard University, NUREG/CR-4214, SAND85-7185, August 1985.

A.65 U.S. National Research Council, National Academy of Sciences, Committee on the Biological Effects of Ionizing Radiation, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980," National Academy Press, 1980.

A.66 R. L. Iman and J. C. Helton, "A Comparison of Uncertainty and Sensitivity Analysis Techniques for Computer Models," Sandia National Laboratories, NUREG/CR-3904, SAND84-1461, May 1985.

A.67 P. A. Seaver, "Assessments of Group Preferences and Group Uncertainty for Decision Making," University of Southern California, Social Sciences Research Institute, 1976.

A.68 J. M. Booker and M. A. Meyer, "Sources and Effects of Interexpert Correlation: An Empirical Study," *IEEE Transactions on Systems, Man, and Cybernetics*, Vol. 18, No. 1, pp. 135-142, 1988.

A.69 S. Lictenstein et al., "Calibration of Probabilities: The State of the Art to 1980," *Judgment Under Uncertainty: Heuristics and Biases*, Cambridge University Press, 1982.

A.70 J. S. Armstrong et al., "Use of the Decomposition Principle in Making Judgments," *Organizational Behavior and Human Performance*, 14: 257-263, 1975.

A.71 I. C. Janis, *Victims of Group Think: A Psychological Study of Foreign Policy Decisions and Fiascoes,* Houghton Mifflin, Boston, MA.

A.72 H. F. Martz et al., "Eliciting and Aggregating Subjective Judgment—Some Experimental Results," *Proceedings of the 1984 Statistical Symposium on National Energy Issues* (Seattle, WA), NUREG/CP-0063, July 1985.

A.73 T. A. Wheeler et al., "Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation," Sandia National Laboratories, NUREG/CR-4550, Vol. 2, SAND86-2084, April 1989.

A.74 I. J. Good, "Axioms of Probability," *Encyclopedia of Statistical Sciences*, Vol. 1, pp. 169-176, Wiley, New York, 1983.

A.75 E. Sverdrup, "Frequency Interpretation in Probability and Statistical Inference," *Encyclopedia of Statistical Sciences*, Vol. 3, pp. 225-231, Wiley, New York, 1983.

A.76 T. L. Fine, "Foundation of Probability," *Encyclopedia of Statistical Sciences*, Vol. 3, pp. 175-184, Wiley, New York, 1983.

A.77 R. V. Hogg and A. T. Craig, *Introduction to Mathematical Statistics*, 3rd Ed., Macmillan, New York, 1970.

A.78 V. Barnett, *Comparative Statistical Inference*, 2nd Ed., Wiley, New York, 1982.