NUREG–1150
Vol. 1

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

Final Summary Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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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 1 of this report has three parts. Part I provides the background and objectives of the assessment and summarizes the methods used to perform the risk studies. Part II provides a summary of results obtained for each of the five plants studied. Part III provides perspectives on the results and discusses the role of this work in the larger context of the NRC staff's work.

CONTENTS

FIGURES

TABLES

Page

ACKNOWLEDGMENTS

This report is a summary of the risk analyses of five nuclear power plants performed under contract to NRC. It is the result of the tireless, creative, and professional efforts by a large number of people on the NRC staff and the staff of its contractors.

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PART I

Introduction

Summary of Methods

1. INTRODUCTION

1.1 Background

In 1975, the U.S. Nuclear Regulatory Commission (NRC) completed the first study of the probabilities and consequences of severe reactor accidents in commercial nuclear power plants—the Reactor Safety Study (RSS) (Ref. 1.1). This work for the first time used the techniques of probabilistic risk analysis (PRA) for the study of core meltdown accidents in two commercial nuclear power plants. The RSS indicated that the probabilities of such accidents were higher than previously believed but that the offsite consequences were significantly lower. The product of probability and consequence—a measure of the risk of severe accidents—was estimated to be quite low relative to other man-made and naturally occurring risks.

Following the completion of these first PRAs, the NRC initiated research programs to improve the staff's ability to assess the risks of severe accidents in light-water reactors. Development began on advanced methods for assessing the frequencies of accidents. Improved means for the collection and use of plant operational data were put into place, and advanced methods for assessing the impacts of human errors and other common-cause failures were developed. In addition, research was begun on key severe accident physical processes identified in the RSS, such as the interactions of molten core material with concrete.

In parallel, the NRC staff began to gradually introduce the use of PRA in its regulatory process. The importance to public risk of a spectrum of generic safety issues facing the staff was investigated and a list of higher priority issues developed (Ref. 1.2). Risk studies of other plant designs were begun (Ref. 1.3). However, such uses of PRA by the staff were significantly tempered by the peer review of the RSS, commonly known as the Lewis Committee report (Ref. 1.4), and the subsequent Commission policy guidance to the staff (Ref. 1.5).

The 1979 accident at Three Mile Island substantially changed the character of NRC's analysis of severe accidents and its use of PRA. Based on the comments and recommendations of both major investigations of this accident (the Kemeny and Rogovin studies (Refs. 1.6 and 1.7)), a substantial research program on severe accident phenomenology was planned and initiated (Refs. 1.8 and 1.9). This program included experimental and analytical studies of accident physical processes. Computer models were developed to simulate these processes. The Kemeny and Rogovin investigations also recommended that PRA be used more by the staff to complement its traditional, nonprobabilistic methods of analyzing nuclear plant safety. In addition, the Rogovin investigation recommended that NRC policy on severe accidents be reconsidered in two respects: the need to specifically consider more severe accidents (e.g., those involving multiple system failures) in the licensing process, and the need for probabilistic safety goals to help define the level of plant safety that was "safe enough."

By the mid-1980's, the technology for analyzing the physical processes of severe accidents had evolved to the point that a new computational model of severe accident physical processes had been developed—the Source Term Code Package—and subjected to peer review (Ref. 1.10). General procedures for performing PRAs were developed (Ref. 1.11), and a summary of PRA perspectives available at that time was published (Ref. 1.12). The Commission had developed and approved policy guidance on how severe accident risks were to be assessed by NRC (Ref. 1.13) as well as safety goals against which these risks could be measured (Ref. 1.14) and methods by which potential safety improvements could be evaluated (Ref. 1.15).

In 1988, the staff requested information on the assessment of severe accident vulnerabilities by each licensed nuclear power plant (Ref. 1.16). This "individual plant examination" could be done either with PRA or other approved means. (In response, virtually all licensees indicated that they intended to perform PRAs in their assessments.) The staff also developed its plans for integrating the reviews of these examinations with other severe accident-related activities by the staff and for coming to closure on severe accident issues on the set of operating nuclear power plants (Ref. 1.17).

One principal supporting element to the staff's severe accident closure process is the reassessment of the risks of such accidents, using the technology developed through the 1980's. This reassessment updates the first staff PRA—the Reactor Safety Study—and provides a "snapshot" (in time) of estimated plant risks in 1988 for five commercial nuclear power plants of different design. For this reassessment, the plants have been studied by teams of PRA specialists under contract to NRC (Refs. 1.18 through 1.31). This report, NUREG-1150, summarizes the results of these studies and provides perspectives on how the results may be used by the NRC staff in carrying out its safety and regulatory responsibilities.

NUREG-1150 was first issued in draft form in February 1987 for public comment. In response, 55 sets of comments were received, totaling approximately 800 pages. In addition, comments were received from three organized peer review committees, two sponsored by NRC (Refs. 1.32 and 1.33) and one by the American Nuclear Society (Ref. 1.34). Appendix D provides a summary of the principal comments (and their authors) on this first draft of NUREG-1150 and the staff's responses. A second draft version of NUREG-1150 was issued in June 1989, taking into account the comments received and reflecting improvements in methods identified in the course of performing the draft risk analyses, in the design and operation of the studied plants, and in the information base of severe accident phenomenology.

Because of the significant criticisms of the first draft of NUREG-1150, and the substantial changes made in response, the second version of the report was issued as a draft for peer review. A review committee was established under the provisions of the Federal Advisory Committee Act (Ref. 1.35). This committee reviewed the report for approximately 1 year and published its results in August 1990 (Ref. 1.36). In parallel, the American Nuclear Society-sponsored review of the report continued; its results were published in June 1990 (Ref. 1.37). Also, the NRC's Advisory Committee on Reactor Safeguards (ACRS) reviewed the analyses and provided comments (Ref. 1.38). Four sets of public comments were also received. While all committees suggested that some changes be made to the report, the comments received were, in general, positive, with all review committees recommending that the report be published in final form as soon as possible and without extensive reanalysis or changes.

This is the final version of NUREG-1150. In keeping with the review committees' recommendations, the staff has made relatively modest changes to the second draft of the report, with essentially no additional technical analysis. (Appendix E provides a summary of the comments and recommendations made by the review committees and the staff's responses. It also includes the ACRS comments in toto.)

Two other recommendations of the review committees should also be noted here. First, the ANS committee indicated that the changes made between the first and second drafts of NUREG-1150 were so substantial that the former should be considered, in effect, obsolete. The staff agrees with this comment and recommends that the analyses and results contained in the first draft no longer be used. Second, the ACRS cautioned that the results should be used only by those who have a thorough understanding of their limitations. The staff agrees with this comment as well.

1.2 Objectives

The objectives of this report are:

- To provide a current assessment of the severe accident risks of five nuclear power plants of different design, which:
  - Provides a snapshot of risks reflecting plant design and operational characteristics, related failure data, and severe accident phenomenological information available as of March 1988;
  - Updates the estimates of NRC's 1975 risk assessment, the Reactor Safety Study;
  - Includes quantitative estimates of risk uncertainty in response to a principal criticism of the Reactor Safety Study; and
  - Identifies plant-specific risk vulnerabilities for the five studied plants, supporting the development of the NRC's individual plant examination (IPE) process;
- To summarize the perspectives gained in performing these risk analyses, with respect to:
  - Issues significant to severe accident frequencies, containment performance, and risks;
  - Risk-significant uncertainties that may merit further research;
  - Comparisons with NRC's safety goals; and
  - The potential benefits of a severe accident management program in reducing accident frequencies; and
- To provide a set of PRA models and results that can support the ongoing prioritization of potential safety issues and related research.

In considering these objectives and the risk analyses in this and supporting contractor reports, it is important to consider both what NUREG–1150 is and what it is not:

- NUREG–1150 **is** a snapshot in time of severe accident risks in five specific commercial nuclear power plants. This snapshot is obtained using, in general, PRA techniques and severe accident phenomenological information of the mid-1980's, but with significant advances in certain areas. The plant analyses reflect design and operational information as of roughly March 1988.

- NUREG–1150 **is** an important resource document for the NRC staff, providing quantitative and qualitative PRA information on a set of five commercial nuclear power plants of different design with respect to important severe accident sequences, and a means for investigating where safety improvements might best be pursued, the cost-effectiveness of possible plant modifications, the importance of generic safety issues, and the sensitivity of risks to issues as they arise.

- NUREG–1150 **is** an estimate of the actual risks of the five studied plants. It is a set of modern PRAs, having the limitations of all such studies. These limitations relate to the quantitative measurement of certain types of human actions (errors of commission, heroic recovery actions); variations in the licensee's organizational/management safety commitments; failure rates of equipment, especially to common-cause effects such as maintenance, environment, design and construction errors, and aging; sabotage risks; and an incomplete understanding of the physical progression and consequences of core damage accidents.

- NUREG–1150 **is not** the sole basis for making plant-specific or generic regulatory decisions. Such decisions must be more broadly based on information on the extant set of regulatory requirements, reflecting the present level of required safety, cost-benefit studies (in some circumstances), risk analysis results (from this and other relevant PRAs), and other technical and legal considerations.

- NUREG–1150 **is not** an estimate of the risks of all commercial nuclear power plants in the United States or abroad. One of the clear perspectives from this study of severe accident risks and other such studies is that characteristics of design and operation specific to individual plants can have a substantial impact on the estimated risks.

1.3 Scope of Risk Analyses

The five risk analyses discussed in this report include the analysis of the frequency of severe accidents, the performance of containment and other mitigative systems and structures in such accidents, and the offsite consequences (health effects, property damage, etc.) of these accidents. In assessing accident frequencies, the five risk analyses consider events initiated while the reactor is at full-power operation.* For two plants, both "internal" events (e.g., random failures of plant equipment, operator errors) and "external" events (e.g., earthquakes, fires) have been considered as initiating events. For the remaining three plants, only internal events have been studied.

The five commercial nuclear power plants studied in this report are:

- Unit 1 of the Surry Power Station, a Westinghouse-designed three-loop reactor in a subatmospheric containment building, located near Williamsburg, Virginia (including the analysis of both internal and external events);**

- Unit 1 of the Zion Nuclear Plant, a Westinghouse-designed four-loop reactor in a large, dry containment building, located near Chicago, Illinois;

- Unit 1 of the Sequoyah Nuclear Power Plant, a Westinghouse-designed four-loop reactor in an ice condenser containment building, located near Chattanooga, Tennessee;

- Unit 2 of the Peach Bottom Atomic Power Station, a General Electric-designed BWR-4 reactor in a Mark I containment building, located near Lancaster, Pennsylvania (including the analysis of both internal and external events);** and

- Unit 1 of the Grand Gulf Nuclear Station, a General Electric-designed BWR-6 reactor in a Mark III containment building, located near Vicksburg, Mississippi.

*Analysis of shutdown and low-power accident risks for the Surry and Grand Gulf plants was initiated in FY 1989.

**These plants were used as models in the Reactor Safety Study.

The external-event analysis summarized in this report includes discussion of the core damage frequency and containment performance from seismically initiated accidents. The offsite consequences and risks are not provided. The reason for this limitation is related to the offsite effects of a large earthquake.

Two sets of hazard curves are used (and reported separately) in the seismic analysis. One set was prepared by Lawrence Livermore National Laboratory (Ref. 1.39) under contract to NRC. Analysis performed using these hazard curves (which have been prepared for the Surry and Peach Bottom sites and other reactor sites east of the Rocky Mountains) suggest that relatively rare but large earthquakes contribute significantly to the risk from seismic events. A second set of hazard curves was also prepared for sites east of the Rocky Mountains for the Electric Power Research Institute (Ref. 1.40). Although both projects made extensive use of expert judgment and formal methods for obtaining these judgments (as did many parts of this project, as discussed in Chapter 2), there were some important differences in methods. Nonetheless, the NRC believes that at present both methods are fundamentally sound.

A significant portion of the estimated seismic-induced core damage frequency for the Surry and Peach Bottom plants arises from large earthquakes. Should such a large earthquake occur in the Eastern United States (e.g., at the Surry or Peach Bottom site), there would likely be substantial damage to some older residential structures, commercial structures, and high hazard facilities such as dams. This could have a major societal impact over a large region, including property damage, injuries, and fatalities. The technology for assessing losses from such earthquakes is a developing one. There are several studies of this technology at this time, including work at the United States Geological Survey. There is no agreed-upon method for this purpose, although a recent report of the National Academy of Sciences (Ref. 1.41) suggests some broad guidelines. The NRC, in its promulgation of safety goals, indicated a preference for quantitative goals in the form of a ratio or percentage of nuclear risks relative to non-nuclear risks. For example, the probability of an early fatality from a nuclear power plant accident should not exceed 1/1000 of the "background" accidental death rate. The NRC intends to further investigate the methods for assessing losses from earthquakes in the vicinity of the Surry and Peach Bottom sites with a view of comparing the ratio of seismically induced reactor accident losses with the overall losses. There has been at least one study (Ref. 1.42) that suggests that the reactor accident contribution to seismic losses is very small relative to the non-nuclear losses. However, this study did not explicitly consider the two sites of interest in this report.

In contrast, because they are aimed at experts in the field of risk analysis, the contractor reports underlying this report (Refs. 1.20, 1.21, 1.27, and 1.28) present the seismic risk results in the form of a set of sensitivity analyses. These analyses consider the effects of the alternative sets of earthquake frequencies and severities noted above, as well as alternative assumptions on the performance of containment structures in large earthquakes, and the possible regional effects of earthquakes (lack of shelter, difficulty in evacuation and relocation, nonradiologically induced injuries and fatalities, etc.) on estimates of plant risk. The reader is cautioned that the results presented in the contractor reports should be used only in the broader context of the overall societal response.

1.4 Structure of NUREG-1150 and Supporting Documents

This report has three parts:

- Part I discusses the background, objectives, and methods used in this assessment of severe accident risks;
- Part II provides summary results and discussion of the individual risk studies of the five examined plants; and
- Part III provides:
  - Perspectives on the collective results of these five PRAs, organized by the principal subject areas of risk analysis: accident frequencies; accident progression, containment loadings, and structural response; transport of radioactive material; offsite consequences; and integrated risk (the product of frequencies and consequences);
  - Discussion of how the risk estimates have changed (and reasons why) for the two plants studied in both the Reactor Safety Study and this report (Surry and Peach Bottom); and

- Discussion of the role of NUREG-1150 as a resource document in the staff's assessment of severe accidents.

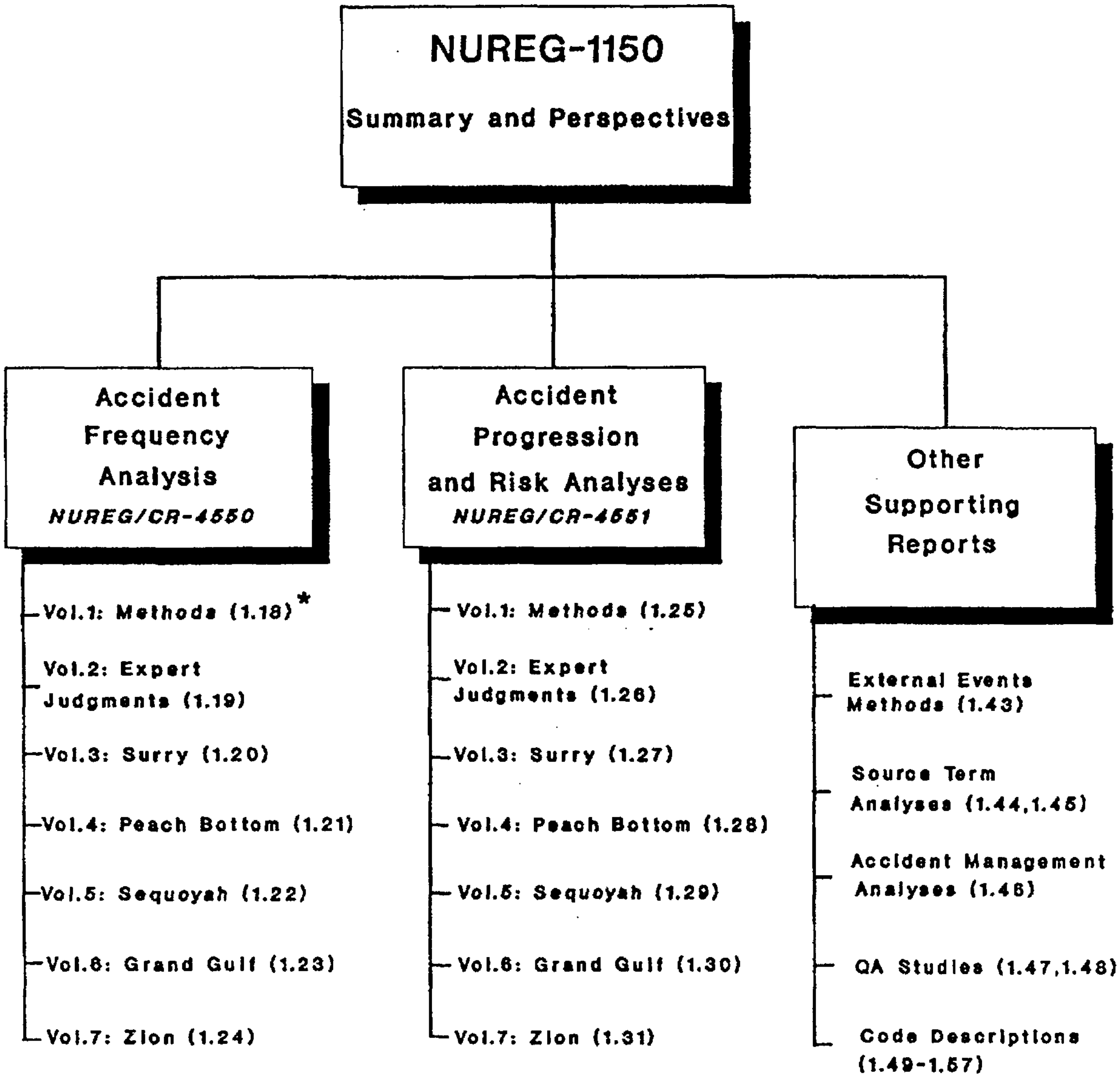
Three appendices are contained in Volume 2 of this report. Appendix A discusses in greater detail the methods used to perform the five risk analyses.* In Appendix B, an example calculation is provided to describe the flow of data through the individual elements of the NUREG-1150 risk analysis process. Appendix C provides supplemental information on key technical issues in the risk analyses. Volume 3 contains two additional appendices. As indicated previously, Appendices D and E provide summaries of comments received on the first and second versions of draft NUREG-1150, respectively, and the associated responses.

As noted above, this report provides a summary of five PRAs performed under contract to NRC. Volume 1 is written for an intended audience of people with a general familiarity with nuclear reactor safety and probabilistic risk analysis. Appendices A, B, and C are written for an intended audience of specialists in reactor safety and risk analysis.

As shown in Figure 1.1, supporting this report are a series of contractor reports providing the detailed substance of the five risk studies. These reports are written for specialists in reactor safety and PRA. The staff's principal contractors for this work have been:

- Sandia National Laboratories, Albuquerque, New Mexico;
- Brookhaven National Laboratory, Upton, New York;
- Idaho National Engineering Laboratory, Idaho Falls, Idaho;
- Battelle Memorial Institute, Columbus, Ohio; and
- Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

*The sections of Appendix A are adapted, with editorial modification, from References 1.18 and 1.25.



*See reference list at end of Chapter 1.

Figure 1.1 Reports supporting NUREG-1150.

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2. SUMMARY OF METHODS

2.1 Introduction

In many respects, the five probabilistic risk analyses (PRAs) performed in support of this report (Refs. 2.1 through 2.14) have been performed using PRA methods typical of the mid-1980's (Refs. 2.15 and 2.16). However, in certain areas, more advanced techniques have been applied. In particular, advancements have occurred in the following areas:

- The estimation of the size of the uncertainties in core damage frequency* and risk due to incomplete understanding of the systems responses, severe accident progression, containment building structural response, and in-plant radioactive material transport;
- The formal elicitation and documentation of expert judgments;**
- The more detailed definition of plant damage states, improving the efficiency of the interface between the accident frequency and accident progression analyses;
- The types of events and outcomes explicitly considered in the accident progression and containment loading analyses;
- The analysis of radioactive material releases and the integration of experimental and calculational results into this analysis;
- The use of more efficient methods for estimating the frequency of core damage accidents resulting from external events (e.g., earthquakes); and
- The application of new computer models in the analysis and integration of risk information.

The assessment of severe accident risks performed for this report can be divided into five general parts (shown in Fig. 2.1): accident frequency; accident progression, containment loading, and structural response; transport of radioactive material; offsite consequences; and integrated risk analyses. This last part combines the information from the first four parts into estimates of risk. These parts are described in Sections 2.2, 2.3, 2.4, 2.5, and 2.8, respectively. Additional discussion of each of these parts is provided in Appendix A and in substantial detail in References 2.1 and 2.8.

Because the estimation of uncertainties in core damage frequency and risk due to uncertainties in the constituent analyses is important to the overall objectives of this study, the descriptions of the constituent analyses will include discussions of uncertainties. The parts of the accident frequency analyses, the accident progression analyses, the containment building structural response analyses, and the radioactive transport analyses that are highly uncertain have been identified. In place of single "best estimates" for parameters representing these uncertain parts of the analyses, probability distributions have been developed. The methods for obtaining probability distributions for uncertain parameters (through, for the most part, the use of expert judgment) and the methods by which the probability distributions in the constituent analyses are propagated through the analyses to yield estimates of the uncertainties in core damage frequency and risk are described in Sections 2.7 and 2.6, respectively. Additional discussion of these two subjects is provided in Sections 6 and 7 of Appendix A and in detail in References 2.1 and 2.8.

The principal results obtained from the five PRAs that form the basis of this report are probability distributions. For simplicity, these distributions may be described by a number of statistical characteristics. The characteristics generally used in this report are the mean, the median, and 5th percentile and 95th percentile of the distributions. No one characteristic conveys all the information necessary to describe the distribution, and any one can be misleading. In particular, for very broad distributions (spanning several orders of magnitude), the mean can be dominated by the high value part of the distribution. If this is also a low probability part of the distribution, the estimate of the mean can exhibit a high degree of statistical variability. Conclusions based on mean values of such distributions must be carefully examined to ensure that dependencies and trends seen in the mean values apply to entire distributions. Conclusions stated in this report have not been based entirely on characteristics of mean values. In some circumstances, median values or entire distributions are used. In particular, the

*Table 2.1 provides definitions of key terms used in this report.

**Risk analyses and other technical studies routinely make use of expert judgment. It is the use of formal procedures to obtain and document these judgments that is noteworthy here.

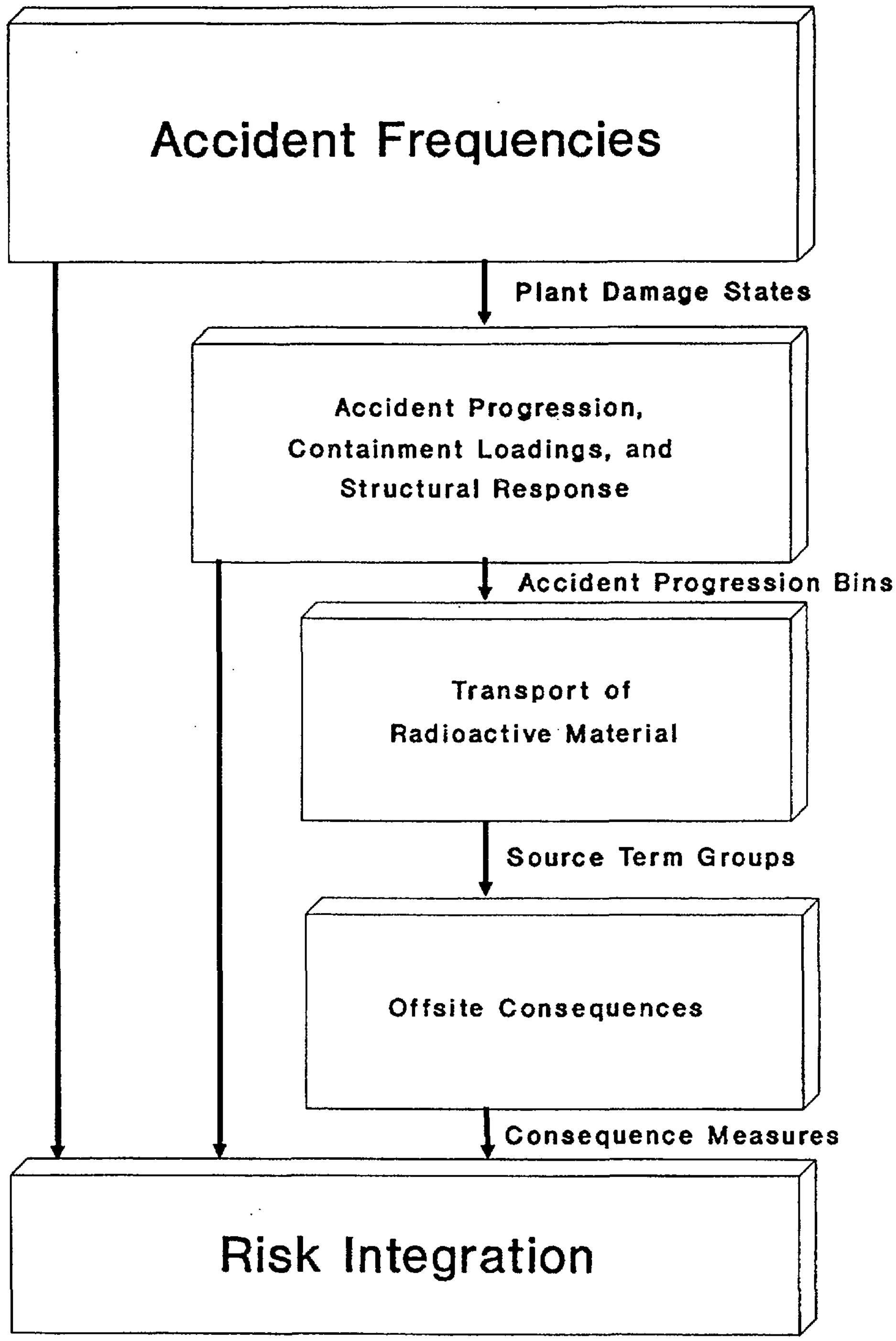


Figure 2.1 Elements of risk analysis process.

Table 2.1 Definition of some key NUREG-1150 risk analysis terms.

Core Damage Frequency: The frequency of combinations of initiating events, hardware failures, and human errors leading to core uncovery with reflooding of the core not imminently expected. For the pressurized water reactors (PWRs) discussed in this report, it was assumed that onset of core damage occurs at uncovery of the top of the active fuel (without imminent recovery). For the boiling water reactors (BWRs) discussed in this report, it was assumed that onset of core damage would occur when the water level was less than 2 feet above the bottom of the active fuel (without imminent recovery). (Ref. 2.1 discusses the reasons for the BWR/PWR differences.)

Internal Initiating Events: Initiating events (e.g., transient events requiring reactor shutdown, pipe breaks) occurring during the normal power generation of a nuclear power plant. In keeping with PRA tradition, loss of offsite power is considered an internal initiating event.

External Initiating Events: Events occurring away from the reactor site that result in initiating events in the plant. In keeping with PRA tradition, some events occurring within the plant during normal power plant operation, e.g., fires and floods initiated within the plant, are included in this category.

Plant Damage State: A group of accident sequences that has similar characteristics with respect to accident progression and containment engineered safety feature operability.*

Accident Progression Bin: A group of postulated accidents that has similar characteristics with respect to (for this summary report) the timing of containment building failure and other factors that determine the amount of radioactive material released.* These are analogous to containment failure modes used in previous PRAs.

Early Containment Failure: Those containment failures occurring before or within a few minutes of reactor vessel breach for PWRs and those failures occurring before or within 2 hours of vessel breach for BWRs. Containment bypass failures (e.g., interfacing-system loss-of-coolant accidents) are categorized separately from early failures.

Source Term: The fractions defining the portion of the radionuclide inventory in the reactor at the start of an accident that is released to the environment. Also included in the source term are the initial elevation, energy, and timing of the release.

Source Term Group: A group of releases of radioactive material that has similar characteristics with respect to the potential for causing early and latent cancer fatality consequences and warning times.

Offsite Consequences: The effects of a release of radioactive material from the power plant site, measured (for this summary report) as the number of early fatalities in the area surrounding the site and within 1 mile of the site boundary, latent cancer fatalities in the area surrounding the site and within 10 miles of the power plant, and population dose in the area surrounding the site and within 50 miles of the power plant.

Probability Density Function: The derivative of the cumulative distribution function. A function used to calculate the probability that a random variable (e.g., amount of hydrogen generated in a severe accident) will fall in a given interval. That probability is proportional to the height of the distribution function in the given interval.

Cumulative Distribution Function: The cumulative distribution function gives the probability of a parameter being less than or equal to a specified value. The *complementary cumulative distribution function* gives the probability of a parameter value being equal to or greater than a specified value.

*Groupings of this sort can be made in a variety of ways; the contractor reports underlying this report provide more detailed groups (Refs. 2.3 through 2.7 and 2.10 through 2.14).

reader is cautioned that an estimated mean may vary by about a factor of two because of sample variation. This variation can also impact the relative contribution of factors (e.g., plant damage states) to the mean (particularly small contributions).

In many risk analyses, "best estimate" analyses are performed. For these studies, many input parameters, even highly uncertain ones, are represented by single "best" values rather than probability distributions as done in this study. The resulting estimate of risk calculated with such best estimate parameter values is not simply related to the mean, median, or any other value of the distributions of risk calculated in this study.

As is implicit in Figure 2.1, the five principal risk analysis parts have clearly defined interfaces through which summary information passes to and from the constituent parts of the analysis and which provide convenient intermediate results for examination and review. Such summary information will be provided in this report; the form of the information presented will be described in the following sections.

2.2 Accident Frequency Estimation

The accident frequency estimation methods underlying this report considered accidents initiated by events occurring during the normal full-power generation* of a nuclear power plant ("internal events") and those initiated by events occurring away from the plant site ("external events"). (Historically, accidents initiated by loss of offsite power have been included in the category of internal events, while fires and floods within the plant during normal operation have been included in the category of external events. This tradition is continued in this report.) The discussion below summarizes accident frequency estimation methods first for internally initiated accidents, followed by those for externally initiated accidents.

2.2.1 Methods

2.2.1.1 Internal-Event Methods

The first part of the analysis shown in Figure 2.1 ("Accident Frequencies") represents the estimation of the frequencies of accident sequences leading to core damage. In this portion of the analysis, combinations of potential accident initiating events (e.g., a pipe break in the reactor coolant system) and system failures that could result in core damage are defined and frequencies of occurrence calculated. The methods for performing this analysis are discussed in Appendix A and in considerable detail in Reference 2.1. In summary, the basic steps in this analysis are:

- *Plant Familiarization*: In this step, information is assembled from plant documentation 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, gather further data, and clarify information with plant personnel. Regular contact is maintained with the plant personnel throughout the study to ensure that current information is used. The analyses discussed in this report reflect each plant's status as of approximately March 1988. This step of the accident frequency analysis was performed in a manner typical of recent PRAs (e.g., as described in Ref. 2.15).
- *Accident Sequence Initiating Event Analysis*: Information is assembled on the types of accident initiating events of potential interest for the specific plant. The initiating events identified include those that could result from support system failures, such as electric power or cooling water faults. Frequencies of initiating events are then assessed. In some cases, the assessed frequencies of certain events were very low; such events were not carried forward into the remaining analysis. Then, the safety functions required to prevent core damage for the individual initiating events are identified, along with specific plant systems required to perform those safety functions, the systems' success criteria (e.g., how much water flow is required from a pumping system), and related operating procedures. The initiating events are then grouped based upon the similarity of response needed from the various plant systems. This step of the analysis was performed in a manner typical of recent PRAs.
- *Accident Sequence Event Tree Analysis*: Using information from the previous step, system event trees that display the combinations of plant system failures that can result in core damage are constructed for each initiating event group. An individual path through such an event tree (an accident sequence) identifies specific combinations of system successes and failures leading to (or avoiding) core damage. As such, the event tree qualitatively identifies what systems must fail in a plant in order to cause core damage (the associated

*Accidents initiated in non-full-power operation are the subject of ongoing study for the Surry and Grand Gulf plants.

system failure probabilities are obtained in following steps). This step of the analysis was performed in a more advanced manner relative to other recent PRAs. For example, the analyses supporting this report considered a significantly greater number of systems in the event trees, including the potential effects on core damage processes from failures of containment functions and systems.

- *Systems Analysis*: In order to estimate the frequencies of accident sequences, the failure probability of each system must be obtained. The important contributors to failure of each system are defined using fault tree analysis methods. Such methods allow the analyst to identify the ways in which system failure may occur, assign failure probabilities to individual plant components (e.g., pumps or valves) and human actions related to the system's operation, and combine the failure probabilities of individual components into an overall system failure probability. This step was performed in a manner typical of that of recent PRAs. The level of detail was determined by the system's relative importance to core damage frequency, based on screening assessments and perspectives from other studies and PRAs.*

- *Dependent and Subtle Failure Analysis*: In addition to the combining of individual component failures, plant systems can fail as a result of the failure of multiple components due to a common cause. Such "dependent failures" may be separated into two types. First, there are direct functional dependencies that can lead to failure of multiple components (e.g., lack of electric power from emergency diesel generators causing failure of emergency core cooling systems). Such dependencies are incorporated directly into the fault or event trees. Second, there are dependent failures that have been experienced in plant operations due to less direct causes and often for which no direct causal relationships have been found. Various methods exist for incorporating such "miscellaneous" failures into the quantification of system fault trees. For this study, a modified "beta factor" method was used (Ref. 2.17). This step of the accident frequency analysis was performed in greater depth than that of typical recent PRAs, in that considerable effort was devoted to generating beta factors for multiple failures (i.e., more than two) using recent advances in common-cause analytical methods. In addition, a subtle failure "checklist" was developed and used. This checklist defined subtle failures found in previous PRAs.

- *Human Reliability Analysis*: As noted in previous steps, explicit consideration of human error was included in the analysis. Errors of two types were incorporated: pre-accident errors, including, for example, failure to properly return equipment to service after maintenance; and post-accident initiation errors, including failure to properly diagnose or respond to and recover from accident conditions. In order to assess failure probabilities for such events, operating procedures for the specific plant under study were obtained and reviewed. In general, the analysis of such errors was made using methods typical of recent PRAs (i.e., modifications of the "THERP" method (Ref. 2.18)) but at a somewhat reduced level of effort. An initial screening analysis was performed to focus the analysis to the potentially most important operator actions (including recovery actions), permitting some savings of effort. More detailed analyses were performed for the BWR anticipated transient without scram (ATWS) accident sequences (Refs. 2.6 and 2.19).

- *Data Base Analysis*: In general, a common data base of equipment and human failure rates and initiating event frequencies was used in the five plant risk analyses, based on operating experience in all commercial nuclear power plants (Ref. 2.1). In addition, the operating experience of each plant studied for this report was examined for relevant failure data on key systems and equipment. The "generic" data base (from all plants) was then replaced with plant-specific data (if available) for these key components in cases where the plant-specific data were significantly different. The methods used to obtain and apply plant-specific data were typical of those of recent PRAs; however, the level of effort expended was less than that generally performed because of limitations in the original analysis scope and, in some cases, because a plant's operating life had been too short to generate an adequate data base.

- *Accident Sequence Quantification Analysis*: In this step, the information from the

*The reader is cautioned that the level of analysis detail and screening assessments used for systems in this study was based on the designs of each of the plants. Thus, it should not be inferred that the results of such assessments necessarily apply to other plants.

preceding steps was assembled into an assessment of the frequencies of individual accident sequences, using the fault trees and event trees to combine probabilities of individual events. This was performed in a manner typical of recent PRAs.

- *Plant Damage State Analysis*: In order to assist the analysis of the physical processes of core damage accidents (i.e., the subsequent steps in a risk analysis), it is convenient to group the various combinations of events comprising the accident sequences into "plant damage states." These states are defined by the operability of plant systems (e.g., the availability of containment spray systems) and by certain key physical conditions in an accident (e.g., reactor coolant system pressure). The definition of the plant damage states and the associated frequencies are the principal products provided to the next step in the risk analysis, i.e., the analysis of accident progression, containment loadings, and structural response. This step was performed in a manner more advanced than most recent PRAs because of the complexity of the interface with the more detailed accident progression analysis.

- *Uncertainty Analysis and Expert Judgment*: As noted in Section 2.1, the risk analyses underlying this report include the quantitative analysis of uncertainties. This analysis was performed using the Latin hypercube sampling technique (Ref. 2.20), a specialized modification of Monte Carlo simulation techniques often used in the combination of uncertainties. The elicitation of expert judgments was necessary to develop the probability distributions for some individual parameters in this uncertainty analysis. For certain key issues in the uncertainty analysis, panels of experts were convened to discuss and help develop the needed probability distributions. The methods used for uncertainty analysis and expert judgment elicitation are discussed in Sections 2.6 and 2.7. For the accident frequency analysis, six issues were evaluated by two expert panels and probability distributions developed; these issues are shown in Table 2.2. Probability distributions were developed for many other parameters as well. Section C. 1 of Appendix C includes a listing of the set of accident frequency issues assigned distributions for the Surry plant. Similar lists for the other plants may be found in References 2.11 through 2.14.

Appendix B provides a detailed example calculation for a particular accident (a station blackout) at the Surry plant. Section B.2 of that appendix describes the analysis of the accident sequence frequency.

It should be noted that the methods used in the accident frequency analysis of the Zion plant varied from those described above. A PRA was completed for this plant by the licensee (Commonwealth Edison Company) in 1981 (Ref. 2.21). This PRA was subsequently reviewed by the NRC staff and its contractors (Ref. 2.22), with the review completed in 1985. For the Zion accident

Table 2.2 Accident frequency analysis issues evaluated 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) frequency analysis summarized in this report, this previous PRA (as modified by the 1985 staff review) was updated to reflect the plant design and operational features in place in early 1988. As such, the Zion accident frequency analysis relied substantially on the previous PRA, rather than performing a new study.

The methods used to perform the Zion accident frequency analysis are discussed in greater detail in Section A.2.2 of Appendix A and in Reference 2.7.*

2.2.1.2 External-Event Methods

The analysis of accident frequencies for the Surry and Peach Bottom plants included the consideration of accidents initiated by external events (e.g., earthquakes, floods, fires) (Refs. 2.3 and 2.4). The methods used to perform these analyses are more efficient versions of previous methods and are described in Section A.2.3 of Appendix A and in more detail in Reference 2.23.

1. **External-Event Methods: Seismic Analysis**

The seismic analysis methods performed for this study consisted of seven steps. Briefly, these are:

- *Determination of Site 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. 2.24); and the "Seismic Hazard Methodology for the Central and Eastern United States Program," sponsored by the Electric Power Research Institute (EPRI) (Ref. 2.25). 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. 2.26). 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 seismic analysis included loss-of-coolant accidents (LOCAs) (i.e., pipe ruptures of a spectrum of sizes including vessel rupture) and transient events. Two types of transient events were considered: those in which the power conversion system (PCS) was initially available and those in which the PCS failed as a direct consequence of the initiating event. The event trees developed in the internal-event analyses (described above) were also used to define seismically initiated accident sequences.

- *Determination of Failure Modes*: The internal-event fault trees (described above) were used in the seismic analysis, with some modification, to specify the failure modes of components, combinations of which resulted in plant system failures.

- *Determination of Fragilities*: Component seismic fragilities were obtained both from a generic fragility data base and from plant-specific fragilities estimated for components identified during a plant visit.

  The generic data base of fragility functions for seismically induced failures was originally developed as part of the Seismic Safety Margins Research Program (SSMRP) (Ref. 2.27). In that program, fragility functions for the generic categories were developed based on a combination of experimental data, design analysis reports, and an extensive survey of expert judgments, providing probability distributions of fragilities.

  Detailed fragility analyses were performed for all important structures at the studied plants. In addition, an analysis of liquefaction for the underlying soils was performed.

- *Determination of Seismic Responses*: Building and component seismic peak ground acceleration responses were computed using dynamic building models and time history analysis methods. Results from the SSMRP analysis of the Zion plant (Ref. 2.28) and methods studies (Ref. 2.23) formed the basis for assessing uncertainties in responses.

- *Computation of Core Damage Frequency*: Given the input from the five steps above, the frequencies of accident sequences, plant damage states, and core damage were

*The analysis of accident progression, containment loadings, and structural response; radioactive material transport; offsite consequences; and integrated risk for the Zion plant did not rely significantly on the previous PRA, but was essentially identical (in methods used) to the other four plant studies performed for this report.

calculated in a manner like that described above for the internal-event accident frequency analysis.

- *Estimation of Uncertainty*: The frequency distributions of individual parameters in the seismic analysis, as developed in the previous steps, were combined to yield frequency distributions of accident sequences, plant damage states, and total core damage. This process was performed using Monte Carlo techniques.

2. External-Event Methods: Fire Analysis

There were four principal steps in the fire accident frequency analysis methods used for this report. Briefly, these are:

- *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 permit the analysis staff to see the physical arrangements in each of these areas. The analysis staff had a fire zone checklist to aid in the screening analysis and in the quantification step (described below).

  Another 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 obtain answers to 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 fire locations within the power plant under study that had the greatest potential for producing accident sequences of high frequency or risk. The selection of fire locations was performed using a screening analysis, which identified potentially important fire zones and prioritized these zones based on the frequencies of fire-induced initiating events in the zone and the probabilities of subsequent failures of important equipment.

- *Accident Sequence Quantification*: After the screening analysis had eliminated all but the probabilistically significant fire zones, detailed quantification of dominant accident sequences was completed as follows:

  - Determination of the temperature response in each fire zone;

  - Computation of component fire fragilities;

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

  - Performance of operator recovery analyses (like that described above for internal-event analyses).

- *Uncertainty Analysis*: This quantification was performed using Monte Carlo techniques like those discussed above for the internal-event analysis. No expert panels were directly used to support the development of probability distributions. Distributions for needed data were developed by the analysis staff using operating experience and experimental results.

3. External-Event Methods: Other Initiating Events

In addition to the seismic and fire external-event analyses, bounding analyses were performed for other 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 initially used in these bounding analyses. If the mean initiating event frequency resulting from such an analysis was estimated to be low (e.g., less than 1E-6 per year), the external event was eliminated from further consideration. Using this logic, the bounding analyses identified those external events in need of more study.

2.2.2 Products of Accident Frequency Analysis

The accident frequency analyses performed in this study can be displayed in a variety of ways. The specific products shown in this summary report are:

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

  For Part II of this report (plant-specific results), tabular data and a histogram-type plot are used to represent the distribution of total core damage frequency. This histogram displays the fraction of Latin hypercube sampling (LHS) observations falling within each interval.* Figure 2.2 displays an example histogram (on the right side of the figure). Four measures of the probability distribution are identified in Figure 2.2 (and throughout this report):

- Mean (arithmetic average or expected value);
- Median (50th percentile value);
- 5th percentile value; and
- 95th percentile value.

In some circumstances, the calculated probability distributions extend to very small values. When this occurs, the staff has chosen to group together all observations below a specific value. This grouped set of observations is displayed apart from (but on the same figure as) the probability distribution.

A second display of accident frequency 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. 2.2) provides a summary of these four specific measures in a simple graphical form.

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 internal, seismic, and fire accident initiators.

The NRC-sponsored review of the second draft of this report includes some cautions on the interpretation of low accident frequencies (Ref. 2.29). These cautions are noted on appropriate figures throughout the remainder of this report.

- The definitions and estimated frequencies of plant damage states.

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

- Loss of all ac electric power (station blackout);
- Transient events with failure of the reactor protection system (ATWS events);
- Other transient events;
- LOCAs resulting from reactor coolant system 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 interfacing-system LOCAs).

Figure 2.3 is an example display of these results. In this figure, a pie chart is used to display the mean value of the total core damage frequency distribution for each of these plant damage states.

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 this summary report, 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. These results are provided in Section 3.2.2 of Chapter 3 and like numbered sections in Chapters 4 through 7.
- *Measures of Importance of 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

*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.

**Plant damage states were defined in these risk analyses at two levels. "Summary" plant damage states were defined for use in this report and were created by combining much more detailed damage states that consider more specific types of failures and convey much more detailed information to the accident progression analysis. These more detailed plant damage states were used in the actual risk calculations. An example of the level of detail may be found in Appendix B; the contractor reports underlying this report provide and discuss the complete set of plant damage states for all plants (Refs. 2.3 through 2.7 and 2.10 through 2.14).

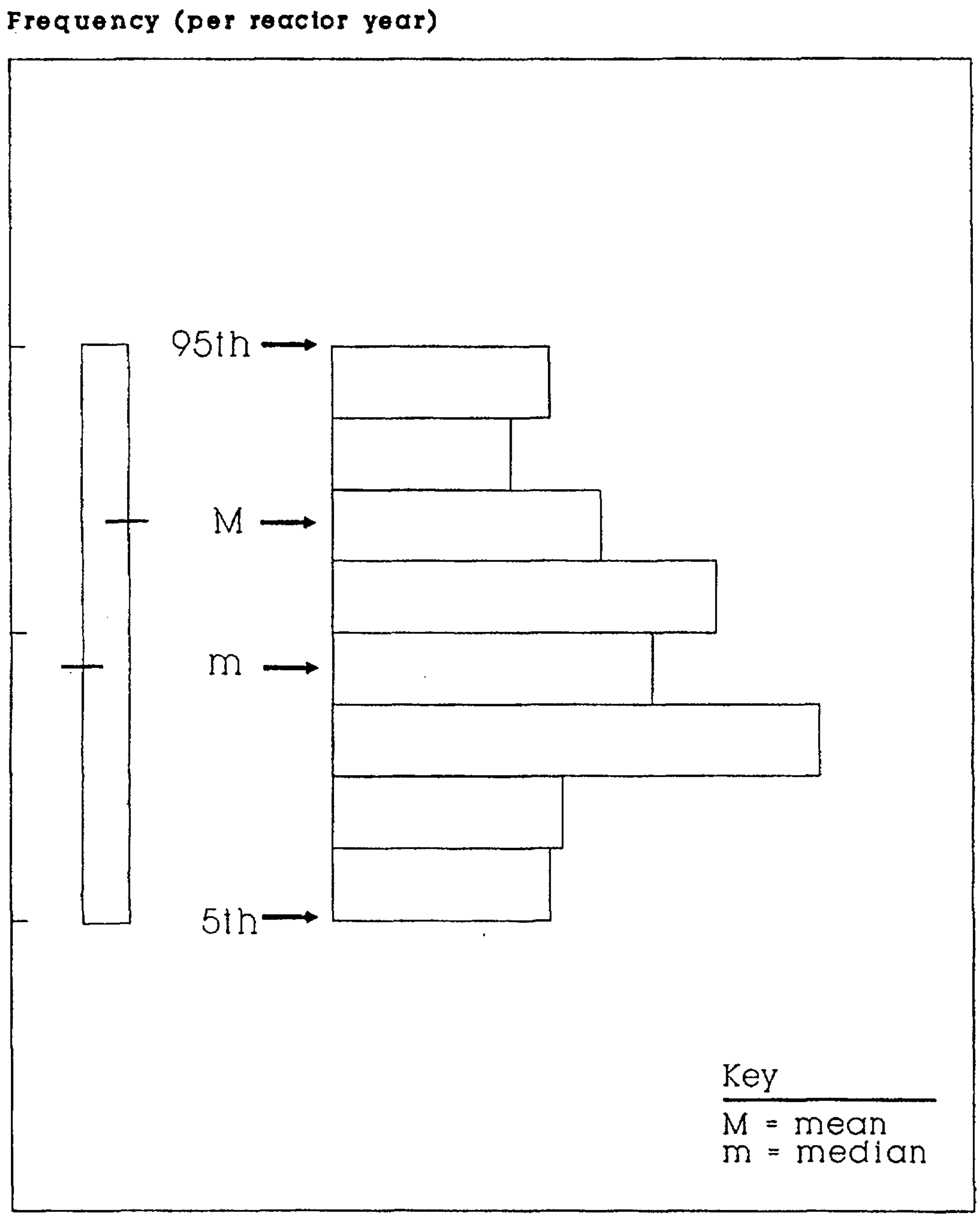
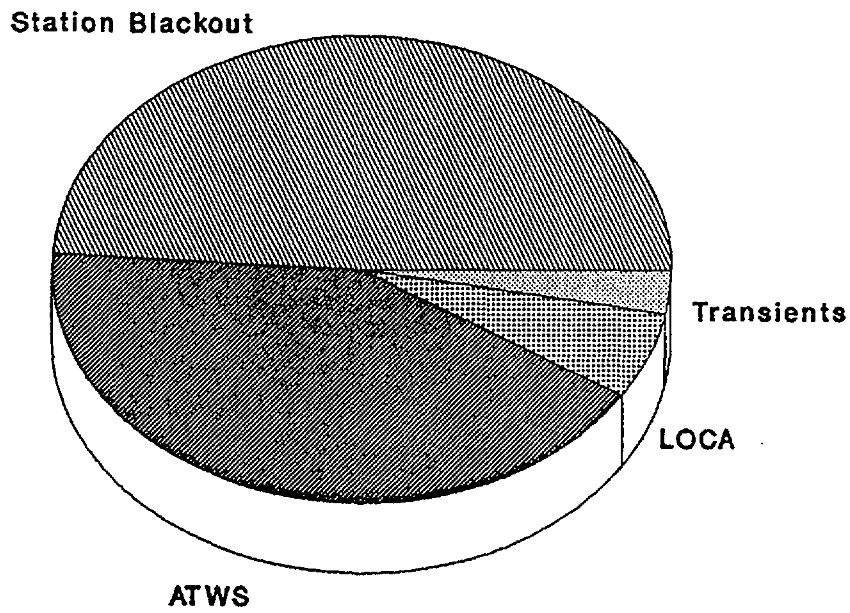


Figure 2.2 Example display of core damage frequency distribution.



Total Mean Core Damage Frequency: 4.5E-6

Figure 2.3 Example display of mean plant damage state frequencies.

individual plant components or the uncertainties in such failure rates) to the total core damage frequency. While a variety of measures exist, two are discussed (qualitatively) in this summary report. The first measure 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 reducing the frequencies of these individual events. The second importance measure discussed in this summary report indicates the relative contribution of key uncertainty 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 by reductions in the uncertainty in individual events. These results are provided in Section 3.2.4 of Chapter 3 and like numbered sections in Chapters 4 through 7.

2.3 Accident Progression, Containment Loading, and Structural Response Analysis

2.3.1 Methods

The second part of the risk analysis process shown in Figure 2.1 ("Accident Progression, Containment Loading, and Structural Response") is the analysis of the progression of the accident after the core has begun to degrade. For each general type of accident, defined by the plant damage states, the analysis considers the important characteristics of the core melting process, the challenges to the containment building, and the response of the building to those challenges. Event trees were used to organize and quantify the large amounts of information used in this analysis. The event trees combined information from many sources, e.g., detailed computer accident simulations and panels of experts providing interpretations of available data.

In summary, the principal steps of the accident progression analysis are:

- *Development of Accident Progression Event Trees*: Accident progression event trees were used in this study to identify, sequentially order, and probabilistically quantify the important events in the progression of a severe accident. The development of an accident progression event tree consisted of identifying potentially important parameters to the accident progression and associated containment building structural response, determining possible values of each parameter (including dependencies on outcomes of previous parameters in the event tree), ordering the events chronologically, and defining the information needed to determine each parameter. The information base used consisted of accident and experimental data and calculational results from accident simulation computer codes, analyses of containment building structures, etc.* While the event tree development process used for this study is conceptually similar to that of other PRAs, both the complexity of the tree (the number of parameters and possible outcomes) and the supporting data base developed were substantially greater than those of other recent PRAs, so that more explicit use could be made of severe accident experimental and calculational information (additional discussion of the supporting data base is provided below).

- *Probabilistic Quantification of Event Trees*: Using the event tree structure and information base developed in the previous step, probability distributions for the most uncertain parameters in the accident progression event tree were generated in this step. As is typical of any PRA, this assignment of values was subjective, based on the interpretation of the data base by the risk analyst. For instance, the applicable data base is sometimes conflicting. The choice of which data to emphasize and use is a matter of each analyst's judgment, based on personal experience and familiarity. However, for this study, both the degree to which experts in accident analysis were used and the degree of documentation of the rationale for the probability distributions used were significantly greater than in other recent PRAs (additional discussion of the supporting data base is provided below).

- *Grouping of Event Tree Outcomes*: Accident progression event trees such as those constructed for this study produce a large set of alternative outcomes of a severe accident. As is typically done in PRAs, these outcomes were grouped into a smaller set of "accident progression bins." For this summary report, bins were defined principally according to the timing of containment building failure. This summary set of accident progression bins is subdivided into bins of greater detail in the supporting contractor reports (Refs. 2.10 through 2.14).

As noted above, the accident progression event trees developed for this study made extensive use of the available severe accident experimental and calculational data bases. The analysis staff made use of calculational results from a number of accident simulation computer codes, including the Source Term Code Package (Ref. 2.30), CONTAIN (Ref. 2.31), MELCOR (Ref. 2.32), and MELPROG (Ref. 2.33).

To support the analysis of certain key issues in the accident progression analysis, expert panels were convened. Fourteen accident progression, containment loadings, and structural response issues were considered by four panels, as shown in Table 2.3. These panels considered a wide range of information available from experiments and computer calculations. Using expert elicitation methods summarized in Section 2.7, probability distributions were developed based on the experts' interpretations of these issues. In addition to this set of key issues, probability distributions were developed for many other issues. Section C.1 of Appendix C provides a listing of such issues, using the Surry plant as an example. Similar listings for the other plants may be found in References 2.11 through 2.14.

Additional discussion of the methods used to develop and quantify the accident progression event trees may be found in Section A.3 of Appendix A. Reference 2.8 provides an extensive discussion of the methods used, suitable for the reader expert in severe accident and risk analysis.

Section B.3 of Appendix B provides a detailed example calculation showing how the accident progression analysis methods summarized above were used in the risk analyses supporting this report.

*In the accident progression analysis of seismic-initiated accidents, some additional loads on containment structures are considered for high-intensity earthquakes (e.g., structural loads resulting from motion of piping).

Table 2.3 Accident progression and containment structural issues evaluated by expert panels.

- 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)

2.3.2 Products of Accident Progression, Containment Loading, and Structural Response Analysis

The product of the accident progression and containment loading analysis is a set of accident progression bins. Each bin consists of a group of postulated accidents (with associated probabilities for each plant damage state) that has similar outcomes with respect to 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 2.2.1, in that they are defined based on their impact on the next analysis part. Quantitatively, the product consists of a matrix of conditional probabilities (as shown in Fig.2.4*), with the rows and columns defined by the sets of plant damage states and accident progression bins, respectively. The matrix defines the probabilities that an accident will have an outcome characteristic of a given accident progression bin if the accident began as one having the characteristic of a given plant damage state.

*The mean plant damage state frequencies shown in Figures 2.4 and 2.5 (and like figures in Chapters 3 through 7) may be somewhat different from those shown in tables such as Table 3.2. The data in the latter tables resulted from uncertainty analyses using a large number of variables. The frequencies shown in the figures resulted from the uncertainty analysis of only the key accident frequency issues included in the integrated task analysis.

In this summary report, 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.

  An example display of early containment failure probability is provided in Figure 2.5.* As may be seen, the probability distribution is represented by a histogram like that discussed above for core damage frequency.

**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.

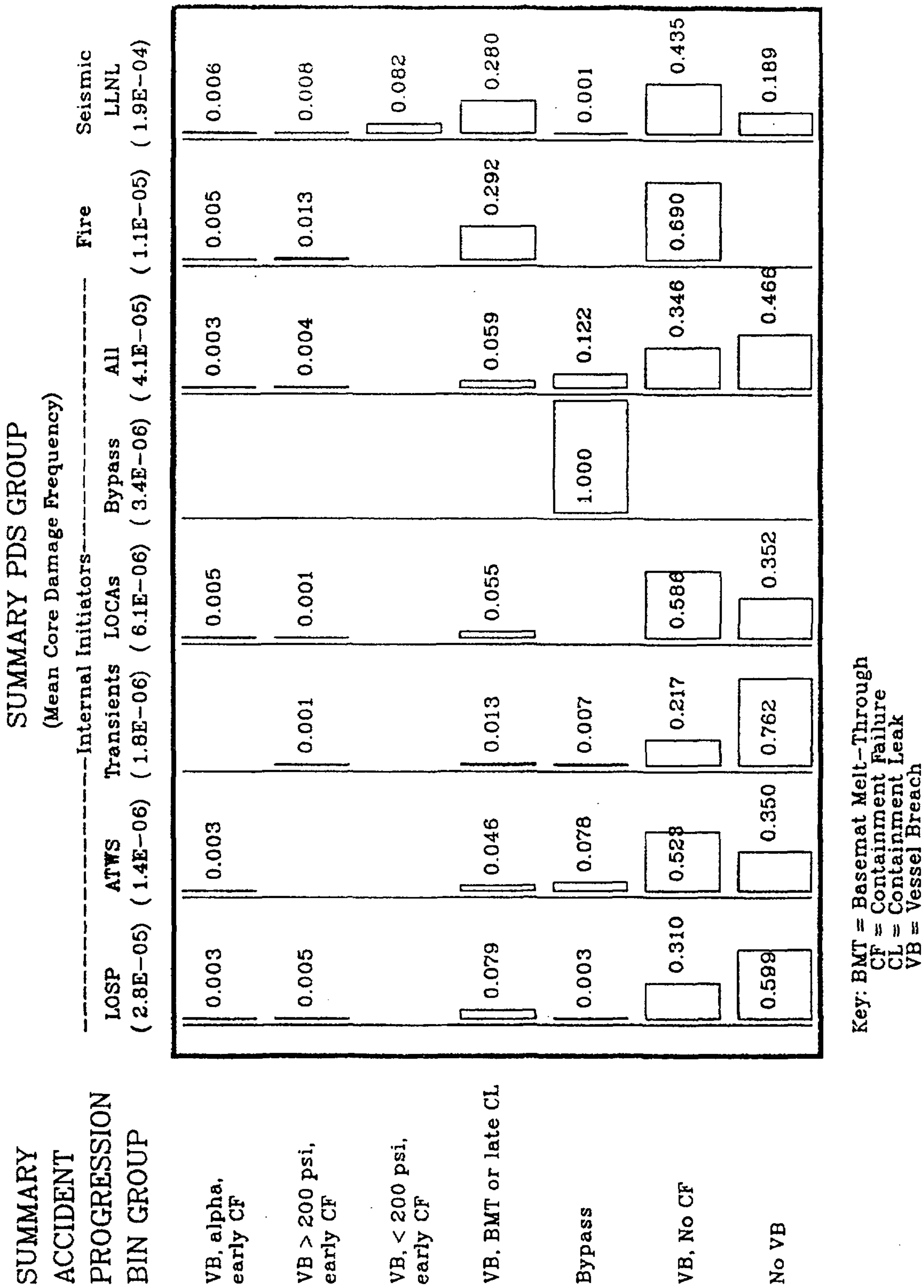


Figure 2.4 Example display of mean accident progression bin conditional probabilities.

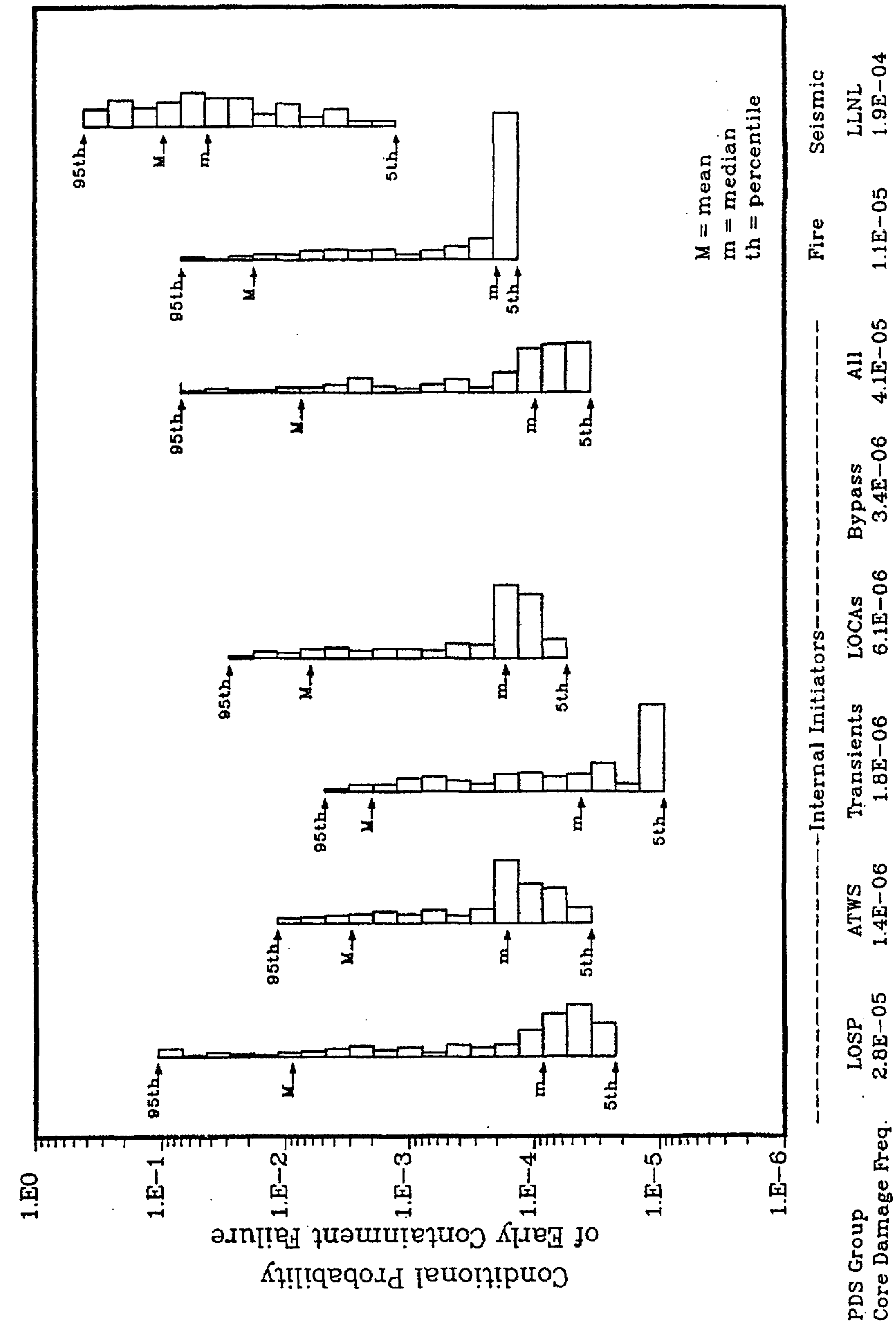


Figure 2.5 Example display of early containment failure probability distribution.

Measures of this distribution provided include:

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

- The mean conditional probability of each accident progression bin for each plant damage state.

  Figure 2.4 displays example results of the mean conditional probability of each accident progression bin for each plant damage state. Results are provided both in tabular and graphical (bar chart) forms.

2.4 Analysis of Radioactive Material Transport

2.4.1 Methods

The radioactive material transport analysis tracks the transport of the radioactive materials from the fuel to the reactor coolant system, then to the containment and other buildings, and finally into the environment. The fractions of the core inventory released to the atmosphere, and the timing and other release information needed to calculate the offsite consequences, together are termed the "source term." The removal and retention of radioactive material by natural processes, such as deposition on surfaces, and by engineered systems, such as sprays, are accounted for in each location.

Briefly, the principal steps in this analysis include:

- *Development of Parametric Models of Material Transport*: Because of the complexity and cost of radioactive material transport calculations performed with detailed codes, the number of accidents that could be investigated with these codes was rather limited. Further, no one detailed code available for the analyses contained models of all physical processes considered important to the risk analyses. Therefore, source terms for the variety of accidents of interest were calculated using simplified algorithms. The source terms were described 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 accident progressions, and nine classes of radionuclides. The release fraction at each stage of the accident and for each pathway is determined using various information such as predictions of detailed mechanistic codes, experimental data, etc. For the more important release parameters, listed in Table 2.4, probability distributions were developed by a panel of experts. The set of codes (one for each plant) used to calculate the source terms is known collectively as the "XSOR" codes (Ref. 2.34). The XSOR codes are parametric in nature; that is, they are designed to use the results of more detailed mechanistic codes or analyses as input.

Table 2.4 Source term issues evaluated by expert panel.

- 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)

Release terms are divided into two time periods, an early release and a delayed release. The timing of release is particularly important for the prediction of early health effects.

- *Detailed Analysis of Radioactive Material Transport for Selected Accident Progression Bins*: Once the basic XSOR algorithm was defined, it was necessary to insert parameters analogous to the quantification of the accident progression event tree in the previous part of the analysis. Since a quantitative uncertainty analysis was one of the objectives of this study, data on the more important parameters were constructed in the form of probability distributions. These distributions were developed based on calculations from the Source Term Code Package (STCP) (Ref. 2.30), CONTAIN (Ref. 2.31), MELCOR (Ref. 2.32), and other calculational and experimental data. The source term parameters determined by an expert panel are shown in Table 2.4. Distributions for parameters that were judged of lesser importance were evaluated by experts drawn from the analysis staff or from other groups at national laboratories. (See Section C.1 of Appendix C for a listing of such parameters for the Surry plant. Similar listings for the other plants may be found in Refs. 2.11 through 2.14.) In rare instances, single-valued estimates were used.

- *Grouping of Radioactive Releases*: For these risk analyses, radioactive releases were grouped according to their potential to cause early and latent cancer fatalities and warning time.* Through this "partitioning" process, the large number of radioactive releases calculated with the XSOR codes were collected into a small set of source term groups (30 to 60 in number). This set of groups was then used in the offsite consequence calculations discussed below.

Additional discussion of the methods used to perform the radioactive material transport analysis may be found in Section A.4 of Appendix A. Reference 2.8 provides an extensive discussion of the methods used that is suitable for the reader expert in severe accident and risk analysis.

Section B.4 of Appendix B provides a detailed example calculation showing how the radioactive material transport analysis methods summarized above were used in the risk analyses supporting this report.

*This grouping of source terms by offsite consequence effects is analogous to the grouping of accident sequences into plant damage states by their potential effect on accident progression.

2.4.2 Products of Radioactive Material Transport Analysis

The product of this part of the risk analysis is the estimate of the radioactive release magnitude, with associated energy content, time, elevation, and duration of release, for each of the specified source term groups developed in the "partitioning" process described above.

The radioactive release estimates generated in this part of the risk analysis can be displayed in a variety of ways. In this report, radioactive release magnitudes are shown in the following ways:

- Distribution of release magnitudes for each of the nine isotopic groups for selected accident progression bins.

  The results of the radioactive material transport analysis can vary in form depending on the intended use. For purposes of this report, example results that display the distribution of release magnitudes for selected accident progression bins were obtained. In Part II of this report, the results for two accident progression bins are displayed for each plant. For these selected accident progression bins, the distribution of the radioactive release magnitude (for each of the nine radionuclide groups) is characterized by the mean, median, 5th percentile, and 95th percentile. An example distribution is displayed in Figure 2.6. (Distributions of this type are constructed with the assumption that all estimated source terms are equally likely and thus do not incorporate the frequencies of the individual source terms. Recalculation of these distributions, including consideration of frequencies, does not significantly change the results.)

- Frequency distribution of radioactive releases of iodine, cesium, strontium, and lanthanum.

  Chapter 10 displays the absolute frequency* of source term release magnitudes.These results are presented in the form of complementary cumulative distribution functions (CCDFs) of the magnitude of iodine, cesium, strontium, and lanthanum releases.** This

*That is, the combined frequency of all plant damage state frequencies and conditional accident progression bin probabilities.

**These four groups are used to represent the spectrum of possible chemical groups, i.e., from chemically volatile to nonvolatile species.

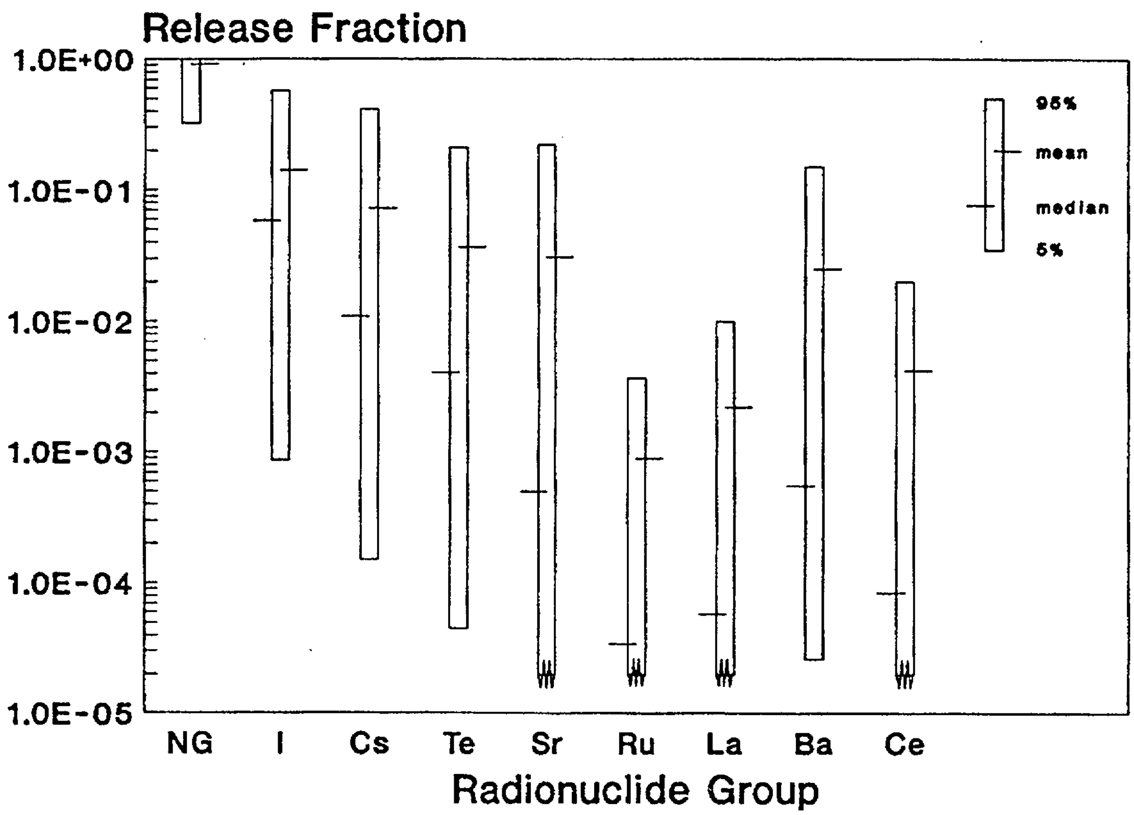


Figure 2.6 Example display of radioactive release distributions.

display provides information on the frequency of source term magnitudes exceeding a specific value for each of the plants. Figure 2.7 displays an example CCDF for one chemical group.

2.5 Offsite Consequence Analysis

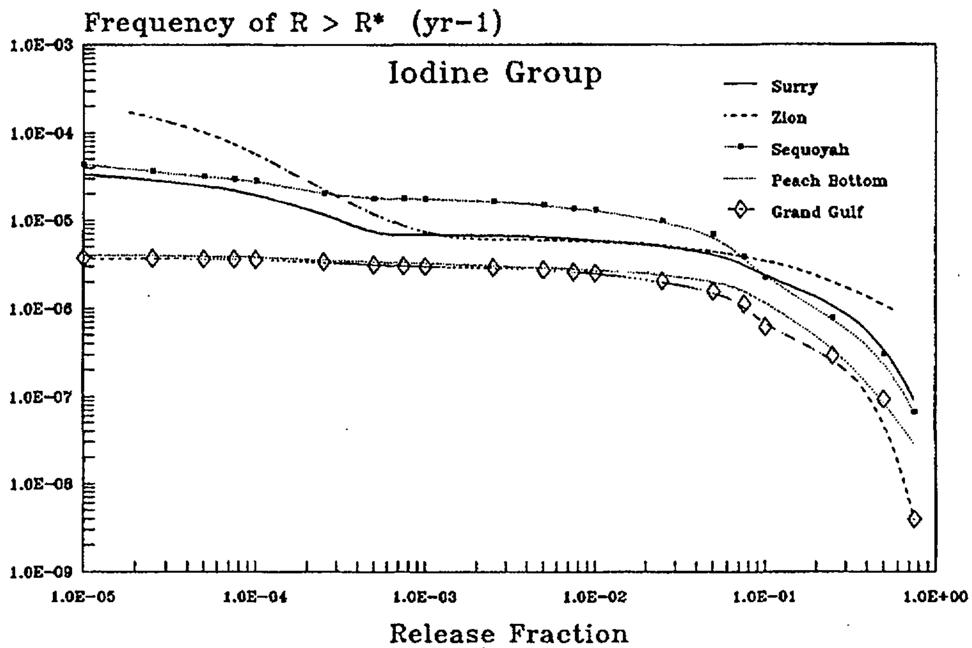
2.5.1 Methods

The severe accident radioactive releases described in the preceding section are of concern because of their potential for impacts on the surrounding environment and population. The impacts of such releases to the atmosphere can manifest themselves in a variety of early and delayed health effects, loss of habitability of areas close to the plant site, and economic losses. The fourth part of the risk analysis process shown in Figure 2.1 represents the estimation of these offsite consequences, given the radioactive releases (source term groups) generated in the previous analysis part.

There are five principal steps in the offsite consequence analysis. Briefly, these are:

- *Assessment of Pre-accident Inventories of Radioactive Material*: An assessment was made of the pre-accident inventories of each radioactive species in the reactor fuel, using information on the thermal power and refueling cycles for the plants studied. For the source term and offsite consequence analysis, the radioactive species were collected into groups of similar chemical behavior. 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.

- *Analysis of Transport and Dispersion of Radioactive Material*: The transport and dispersion of radioactive material to offsite



Note: As discussed in Reference 2.29, estimated risks at or below 1E−7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 2.7 Example display of source term complementary cumulative distribution function.

areas was modeled in two parts: the initial development of a plume in the wake of plant buildings, using models described in Reference 2.35; and the subsequent downwind transport, which used a straight-line Gaussian plume model, as described in Reference 2.36. The effect of the initial sensible energy content of the plume was included in these models so that under some conditions plume "liftoff" could occur, elevating the contained radioactive material into the atmosphere.

The dispersion models used in this report also explicitly accounted for the variability of transport and deposition with weather conditions.

Meteorological data for each specific power plant site were used. For each of a set of approximately 160 representative weather conditions, a dispersion pattern of the plume was calculated. Deposition of radioactive material from the plume onto the ground (or water bodies) beneath the plume was based on a set of experimentally derived deposition rates for dry and wet (rain) conditions.

- *Analysis of the Radiation Doses*: Using the dispersion and deposition patterns developed in the previous step and a set of dose conversion factors (which relate a concentration of a radioactive species to a dose to a given body organ) (Refs. 2.37, 2.38, and 2.39), calculations were made of the doses received by the exposed populations via direct (cloudshine, inhalation, groundshine) and indirect (ingestion, resuspension of radioactive material from the ground into the air) pathways. Site-specific population data were used in these calculations. The doses were calculated on a body organ-by-organ basis and combined into health effect estimates in a later step.

- *Analysis of Dose Mitigation by Emergency Response Actions*: Consideration was given to the mitigating effects of emergency response actions taken immediately after the accident and in the longer term. Effects included were evacuation, sheltering, and relocation of people, interdiction of milk and crops, and decontamination, temporary interdiction, and/or condemnation of land and buildings.

  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 reach the decision to evacuate and 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. 2.40). Similar relocation assumptions were made for the population outside the 10-mile planning zone. Longer-term countermeasures (e.g., crop or land interdiction) were based on EPA and Food and Drug Administration guidelines (Ref. 2.41).

  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).

- *Calculation of Health Effects*: The offsite consequence analysis calculated the following health effect measures:

  - The 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 lifetime 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).

  The health effects calculated in this analysis were based on the models of Reference 2.42. This work in turn used the work of the BEIR III report (Ref. 2.43) for its models of latent cancer effects.

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. Such an analysis is, however, planned to be performed.

Section A.5 of Appendix A provides additional discussion of the methods used for performing the offsite consequence analysis. The reader seeking extensive discussion of the methods used is directed to Reference 2.8 and to Reference 2.36, which discusses the computer code used to perform the offsite consequence analysis (i.e., the MELCOR Accident Consequence Code System (MACCS), Version 1.5).

2.5.2 Products of Offsite Consequence Analysis

The product of this part of the risk analysis process is a set of offsite consequence measures for each source term group. For this report, the specific consequence measures discussed include early fatalities, latent cancer fatalities, total population dose (within 50 miles and entire site region), and two measures for comparison with NRC's safety goals (average individual early fatality probability within 1 mile and average individual latent cancer fatality probability within 10 miles of the site boundary) (Ref. 2.44).

For display in this report, the results of the offsite consequence analyses are combined with the frequencies generated in the previous analysis steps and shown in the form of complementary cumulative distribution functions (CCDFs). This display shows the frequency of consequences occurring at a level greater than a specified amount. Figure 2.8 provides a display of such a CCDF. This information is also provided in tabular form in Chapter 11.

2.6 Uncertainty Analysis

As stated in the introduction to the chapter, 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.

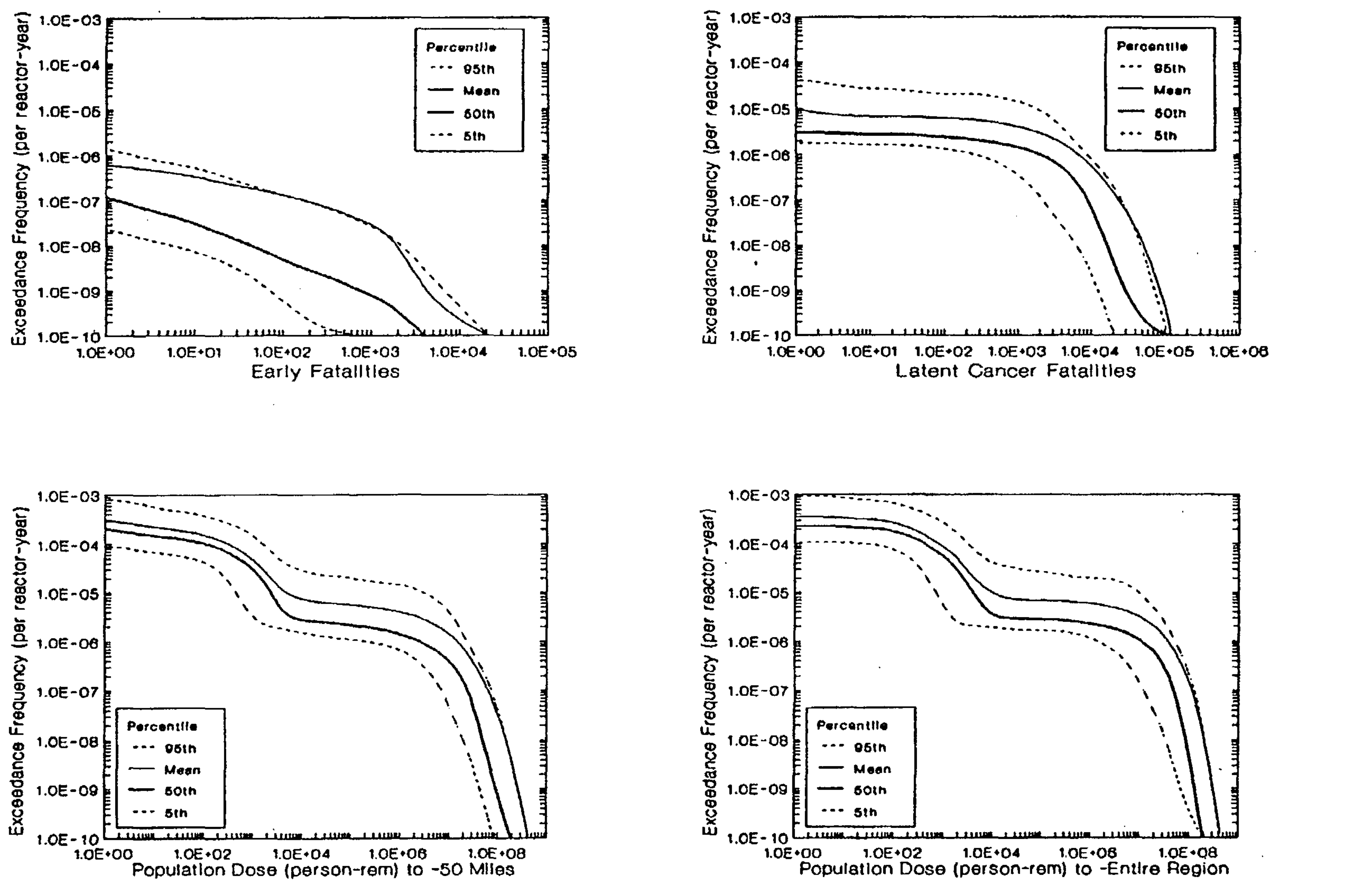
There are four steps in the performance of uncertainty analyses. Briefly, these are:

- *Scope of Uncertainty Analyses*: Important sources of uncertainty exist in all four stages of the risk analysis shown in Figure 2.1. 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 Tables 2.2 through 2.4.

- *Definition of Specific Uncertainties*: In order for uncertainties in accident phenomena to be included in the probabilistic risk analyses conducted for this study, 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.

  Parameters were chosen to be included in the uncertainty analysis if the associated uncertainties were estimated to be large and important to risk.

- *Development of Probability Distributions*: Probability distributions for input parameters were developed by a number of methods. As stated previously, distributions for many key input parameters were determined by panels of experts. The experts used a large 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 2.7. Probability distributions for some parameters believed to be of less importance to risk were generated by analysts on the project staff or by phenomenologists from several different



Note: As discussed in Reference 2.29, estimated risks at or below 1E−7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 2.8 Example display of offsite consequences complementary cumulative distribution function.

national laboratories using techniques like those employed with the expert panels. (Section C.1 of Appendix C provides a listing of parameters to which probability distributions were assigned for the Surry plant. Similar listings for the other plants may be found in Refs. 2.11 through 2.14.)

Probability distributions for many of the most important accident sequence frequency variables were generated using statistical analyses of plant data or data from other published sources.

- *Combination of Uncertainties*: A specialized Monte Carlo method, Latin hypercube sampling, 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 easily treat distributions with wide ranges and can incorporate correlations between variables. Latin hypercube sampling (Ref. 2.20) 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. 2.45). 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.

Additional discussion of uncertainty analysis methods is provided in Section A.6 of Appendix A and in detail in Reference 2.8.

2.7 Formal Procedures for Elicitation of Expert Judgment

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 2.8. The principal steps of this process are shown in Figure 2.9. Briefly, these steps are:

- *Selection of Issues*: As stated in Section 2.6, 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. 2.46). The list was further modified by the expert panels. Tables 2.2 through 2.4 list those issues studied by 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 panels to ensure a balance of "perspectives." Diversity of perspectives has been viewed by some (e.g., Refs. 2.47 and 2.48) as allowing the problem to be considered from more viewpoints and thus leading to better quality answers. The size of the panels ranged 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

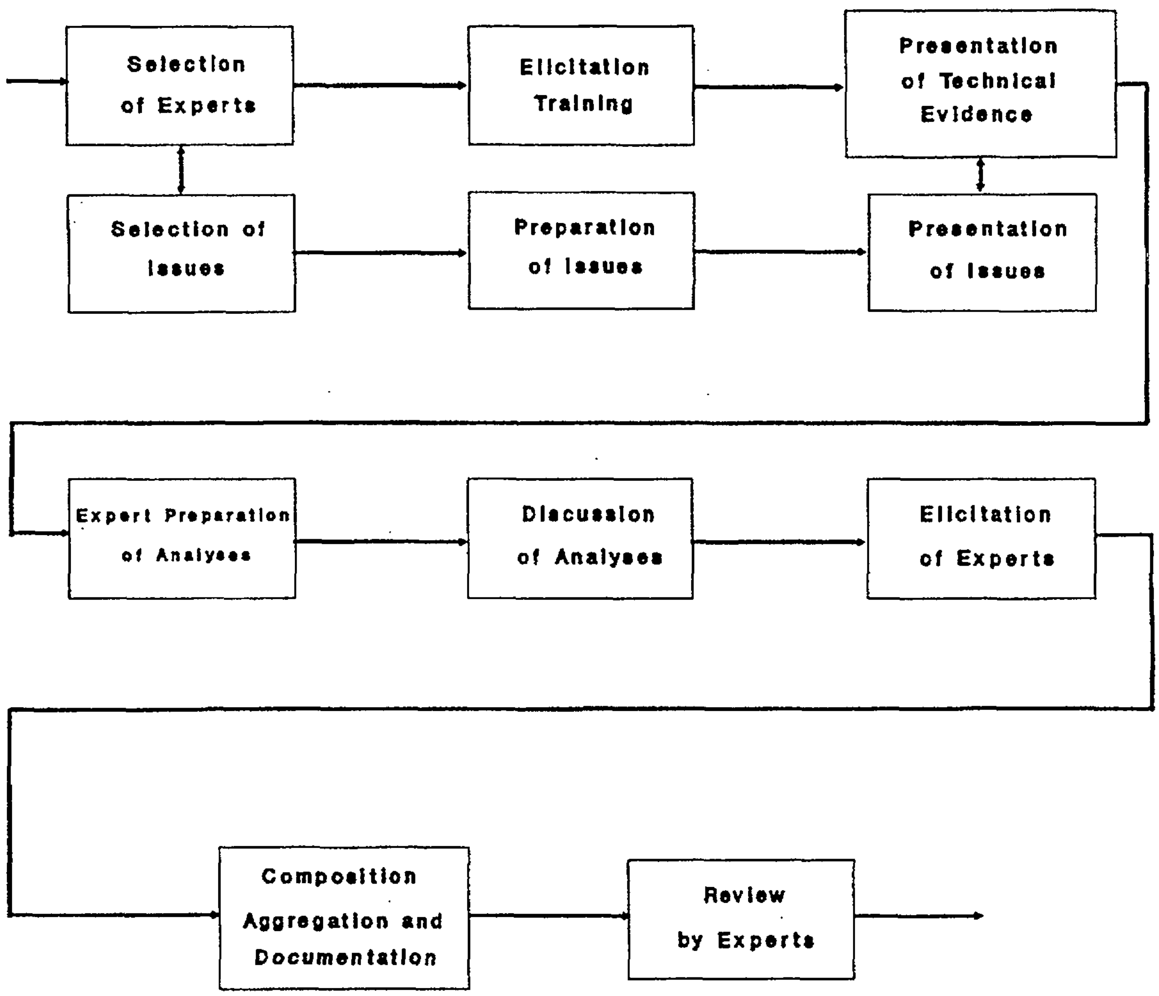


Figure 2.9 Principal steps in expert elicitation process.

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 2.49 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. 2.50).

  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 at 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. 2.51).

- *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. 2.52) 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.

2.8 Risk Integration

2.8.1 Methods

The fifth part of the risk analysis process shown in Figure 2.1 ("Risk Integration") is the integration of the other analysis products into the overall estimate of plant risk. Risk for a given consequence measure is the sum over all postulated accidents of the product of the frequency and consequence of the accident. This part of the analysis consisted of both the combination of the results of the constituent analyses and the subsequent assessment of the relative contributions of different types of accidents (as defined by the plant damage states, accident progression bins, or source term groups) to the total risk.

Appendix A provides a more detailed description of the risk integration process. In order to assist the reader seeking a detailed understanding of this process, an example calculation is provided in Appendix B. This example makes use of actual results for the Surry plant.

2.8.2 Products of Risk Integration

The risk analyses performed in this study can be displayed in a variety of ways. The specific products shown in this summary report are described below, with similar products provided for early fatality risk, latent cancer fatality risk, population dose risk within 50 miles and within the entire area surrounding the site, and for two measures related to NRC's safety goals (Ref. 2.44).

- The total risks from internal and fire events.*

  Reflecting the uncertain nature of risk results, such results can be displayed using a probability density function. For Part II of this report (plant-specific results), a histogram is used. This histogram for risk results is like that shown on the right side of Figure 2.2 for the results of the accident frequency analysis. In addition, four measures of the probability distribution are identified in Figure 2.2 (and throughout this report):

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

  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. 2.2) 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 this report can be decomposed to determine the fractional contribution of individual plant damage states and accident progression bins to the mean risk. An example display of the fractional contribution of plant damage states to mean early and latent cancer fatality risk is provided in Figure 2.10. The estimated values of these relative contributions are somewhat sensitive to the Monte Carlo sampling variation, particularly those contributions that are small. References 2.10 through 2.14 discuss this sensitivity to sampling variation in more detail. These references also include discussion of an alternative method for calculating the relative contributions to mean risk that provides somewhat different results.

- Contributions to risk uncertainty.

  Regression analyses were performed to assess the relative contributions of the uncertainty in individual parameters (or groups of parameters) to the uncertainty in risk. Results of these analyses are discussed in Part III of this report and in more detail in References 2.10 through 2.14.

*For reasons described in Chapter 1, seismic risk is not displayed or discussed in this report.

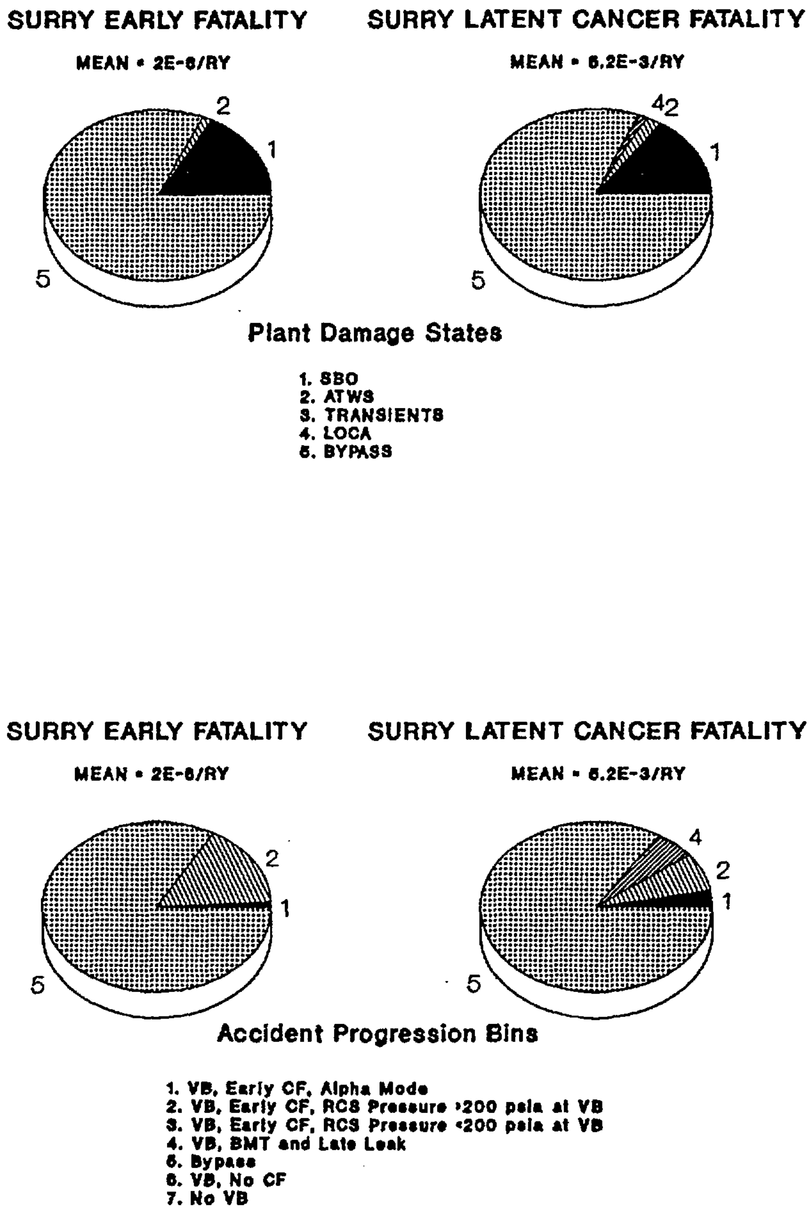


Figure 2.10 Example display of relative contributions to mean risk.

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