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

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW, Lower Level, Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the Code of Federal Regulations, and Nuclear Regulatory Commission Issuances.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. Federal Register notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Information Resources Management, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG-1150
Vol. 1

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

Final Summary Report

Manuscript Completed: October 1990
Date Published: December 1990

Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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

Page

Page

FIGURES

TABLES

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.

Overall management of the NUREG-1150 project was provided by:

Denwood Ross
Joseph Murphy
Mark Cunningham

NUREG-1150 Summary Report*

The principal authors of this summary report were:

Sarbes Acharya
Bharat Agrawal
Mark Cunningham
Richard Denning (Battelle Memorial Institute—BMI)
James Glynn
James Johnson
Pradyot Niyogi
Harold VanderMolen

Other contributors were:

Robert Bertucio (Energy Inc.—EI**)
Roger Breeding (Sandia National Laboratories—SNL)
Thomas Brown (SNL)
Allen Camp (SNL)
Wallis Cramond (SNL)
Mary Drouin (Science Applications Incorporated—SAIC)
Elaine Gorham-Bergeron (SNL)
Julie Gregory (SNL)
Frederick Harper (SNL)
Alan Kolaczkowski (SAIC)
Mark Leonard (SAIC)
Chang Park (Brookhaven National Laboratory—BNL)
Arthur Payne (SNL)
Trevor Pratt (BNL)
Martin Sattison (Idaho National Engineering Laboratory—INEL)
Timothy Wheeler (SNL)

NUREG-1150 Appendices*

This report has four appendices. The principal authors of Appendices A and B were:

Roger Breeding (SNL)
Mary Drouin (SAIC)
David Ericson, Jr. (ERC, Inc.)
Elaine Gorham-Bergeron (SNL)
Martin Sattison (INEL)

Other contributors included:

Michael Bohn (SNL)
Gary Boyd (SAROS)
Allen Camp (SNL)
Wallis Cramond (SNL)
Frederick Harper (SNL)
Jon Helton (Arizona State University—ASU)
John Lambright (SNL)
Timothy Wheeler (SNL)

*The authors and contributors noted here were responsible for the development of the second draft of NUREG-1150. Those modifications needed to produce this final version (including the development of a fifth appendix) were made by Robert Bertucio (EI**), Allen Camp (SNL), Mark Cunningham (NRC), Richard Denning (BMI), Mary Drouin (SAIC), Frederick Harper (SNL), James Johnson (NRC), Joseph Murphy (NRC), John Lambright (SNL), Trevor Pratt (BNL), Christopher Ryder (NRC), and Martin Sattison (INEL). Final technical editing was performed by Louise Gallagher; final composition was performed by M Linda McKenzie and Ina H. Schwartz.

**Now with NUS Corporation.

The principal authors of Appendices C and D were:

Nilesh Chokshi (NRC)
Richard Denning (BMI)
Mark Leonard (SAIC)
Christopher Ryder (NRC)
Stephen Unwin (BMI)
John Wreathall (SAIC)

Other contributors to these appendices were:

Sarbes Acharya (NRC)
Christopher Amos (SAIC)
Roger Breeding (SNL)
Thomas Brown (SNL)
Allen Camp (SNL)
Wallis Cramond (SNL)
Mark Cunningham (NRC)
David Ericson, Jr. (ERC, Inc.)
Elaine Gorham-Bergeron (SNL)
Julie Gregory (SNL)
Frederick Harper (SNL)
Walter Murfin (Technadyne)
Joseph Murphy (NRC)
Pradyot Niyogi (NRC)
Arthur Payne (SNL)
Timothy Wheeler (SNL)

The detailed risk analyses underlying this report were performed under contract to NRC. The NRC staff project managers for these contracts were:

Bharat Agrawal
James Johnson
Pradyot Niyogi
David Pyatt*
Richard Robinson

Principal Contractor Reports

Within the contractor organizations, the staff involved in the risk analyses were:

Sandia National Laboratories

Principal Contributors:

Christopher Amos (SAIC)
Allan Benjamin
Robert Bertucio (EI)
Michael Bohn
Gary Boyd (SAROS)
Roger Breeding
Thomas Brown
Sharon Brown (EI)
Allen Camp
Wallis Cramond
Sharon Daniel
Mary Drouin (SAIC)
Elaine Gorham-Bergeron
Julie Gregory
Frederick Harper
Eric Haskin
Jon Helton (ASU)
Sarah Higgins
Ronald Iman
Jay Johnson (SAIC)
Hong-Nian Jow
Jeffrey Julius (EI)
Alan Kolaczkowski (SAIC)
Jeffrey LaChance (SAIC)
John Lambright
Kevin Maloney
Walter Murfin (Technadyne)
Arthur Payne
Bonnie Shapiro (SAIC)
Ann Shiver
Lanny Smith
Jeremy Sprung
Teresa Sype
Timothy Wheeler

Other contributors were:

Ken Adams
Michael Allen
Kenneth Bergeron
Marshall Berman
Edward Boucheron
David Bradley
Rupert Byers
William Camp
Michael Carmel
David Chanin (Technadyne)
David Clauss
Dirk Dahlgren
Susan Dingman
Lisa Gallup (GRAM)

*Now with the U.S. Department of Energy.

Randall Gauntt
Michael Griesmeyer
Irving Hall
Phillip Hashimoto (EQE, Inc.)
Terry Heames (SAIC)
Jack Hickman
Steven Hora (U. of Hawaii)
Daniel Horschel
James Johnson (EQE)
Diane Jones (EI)
John Kelly
Stuart Lewis (SAROS)
David Kunsman
David McCloskey
Billy Marshall, Jr.
Joel Miller
David Moore (EI)
Kenneth Murata
Michael Mraz (EQE)
Nestor Ortiz
Martin Pilch
Dana Powers
Mark Quilici (EI)
Mayasandra Ravindra (EQE)
Judith Rollston (GRAM)
Martin Sherman
Michael Shortencarier
Douglas Stamps
William Tarbell
Wen Tong (EQE)
Walter Von Riesemann
Jack Walker
Jay Weingardt (SAIC)
Ginger Wilkinson
David Williams

Brookhaven National Laboratory

Principal contributors:

Erik Cazzoli
Carrie Grimshaw
Min Lee*
Chang Park
Trevor Pratt
Arthur Tingle

Other contributors:

Robert Bari
Stephen Unwin**

Idaho National Engineering Laboratory

Principal contributors:

Martin Sattison
Kevin Hall

Other contributors:

Robert Bertucio (EI)
Peter Davis (PRD Consulting)
John Young (R. Lynette & Associates***)

Additional Technical Support

Additional technical support for the five risk analyses was obtained from other organizations and individuals. These included:

University of Southern California

Ralph Keeney
Detlof von Winterfeldt
Richard John
Ward Edwards

Los Alamos National Laboratory

Mary Meyer
Jane Booker

Battelle Memorial Institute

Richard Denning

*Now with National Tsing Hwa University, Taiwan.
**Now with Battelle Memorial Institute.
***Now with SAIC.

Lee Ann Curtis
Peter Cybulskis
Hans Jordan
Rita Freeman-Kelly
Vladimir Kogan
Philip Shumacher
Stephen Unwin
Roger Wooton

Quality Assurance Teams

Quality assurance and control teams were formed to review the risk analyses. Members of these teams were:

Accident Frequency Analysis

Gary Boyd (SAROS)
David Kunsman (SNL)
Garreth Parry (NUS)
Arthur Payne (SNL)
John Wreathall (SAIC)

Risk Analysis

Kenneth Bergeron (SNL)
Gary Boyd (SAROS)
David Bradley (SNL)
Richard Denning (BMI)
Susan Dingman (SNL)
John Kelly (SNL)
David Kunsman (SNL)
Stuart Lewis (SAROS)
David Pyatt (NRC)
John Zehner (BNL)

Expert Panels

Panels of experts were used to develop probability distributions for a number of key parameters in the risk analyses. Members of the expert panels were:

Accident Frequency Issues

Barbara Bell (BMI)
Dennis Bley (Pickard, Lowe and Garrick, Inc.—PLG)
Gary Boyd (SAROS)
Robert Budnitz (Future Resource Associates, Inc.)
Larry Bustard (SNL)
Karl Fleming (PLG)
Michael Hitchler (Westinghouse)
Jerry Jackson (NRC)
Joseph Murphy (NRC)
Garreth Parry (NUS)
David Rhodes (Atomic Energy of Canada Limited)

In-Vessel Accident Phenomenological Issues

Peter Bieniarz (Risk Management Associates—RMA)
William Camp (SNL)
Vernon Denny (SAIC)
Richard Hobbins (INEL)
Steven Hodge (Oak Ridge National Laboratory—ORNL)
Robert Lutz (Westinghouse)
Michael Podowski (Rensselaer Polytechnic Institute)
Garry Thomas (Electric Power Research Institute—EPRI)
Robert Wright (NRC)

Containment Loading Issues

Louis Baker (Argonne National Laboratory)
Kenneth Bergeron (SNL)
Theodore Ginsburg (BNL)
James Metcalf (Stone and Webster Engineering Corp.—S&W)
Martin Plys (Fauske and Associates, Inc.—FAI)
Martin Sherman (SNL)
Patricia Worthington (NRC)
Alfred Torri (PLG)

Molten Core Containment Issues

David Bradley (SNL)
Michael Corradini (University of Wisconsin)

George Greene (BNL)
Michael Hazzan (S&W)
Mujid Kazimi (Massachusetts Institute of Technology)
Raj Sehgal (EPRI)

Containment Structural Response Issues

David Clauss (SNL)
Charles Miller (CCNY)
Kam Mokhtarian (Chicago Bridge and Iron, Inc.)
Joseph Rashid (ANATECH)
Subir Sen (Bechtel Power Corp.)
Richard Tolen (United Engineers and Construction)
Walter Von Riesemann (SNL)
Adolph Walser (Sargent and Lundy Engineers)
J. Randall Weatherby (SNL)
Donald Wesley (IMPELL)

Source Term Issues

Peter Bieniarz (RMA)
Andrzej Drozd (S&W)
James Gieseke (BMI)
Robert Henry (FAI)
Thomas Kress (ORNL)
Y.H. (Ben) Liu (University of Minnesota)
Dana Powers (SNL)
Richard Vogel (EPRI)
David Williams (SNL)

Other Support

The publication of this report could not have been achieved without substantial help from other NRC staff members. These included: Leslie Lancaster and Richard Robinson (technical review); Louise Gallagher (editorial review); Veronica Blackstock, Annette Spain, Jean Shipley, Mahmooda Bano, Debra Veltri, Wanda Haag (word processing support); and Joanne Johansen, M Linda McKenzie, Bonnie Epps, Jane Corley, Marianne Bender, and Jeanette Kiminas from Electronic Composition Services, Office of Administration (final report composition).

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.

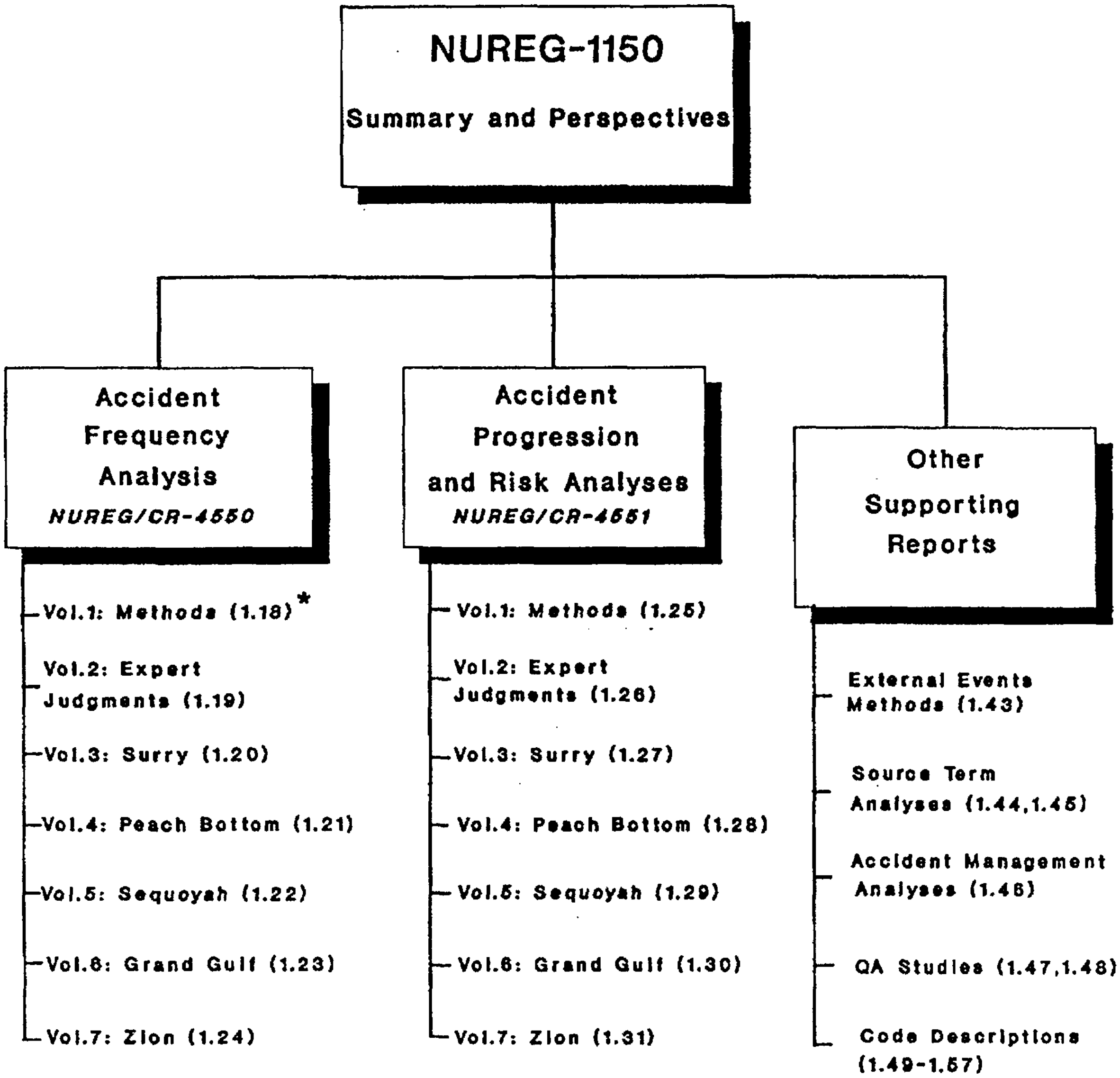
Three appendices are contained in Volume 2 of this report. Appendix A discusses in greater detail the methods used to perform the five risk analyses.* In Appendix B, an example calculation is provided to describe the flow of data through the individual elements of the NUREG-1150 risk analysis process. 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.

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

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.



*See reference list at end of Chapter 1.

Figure 1.1 Reports supporting NUREG-1150.

REFERENCES FOR CHAPTER 1

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

1.2 USNRC, "Reporting the Progress of Resolution of Unresolved Safety Issues in the NRC Annual Report," SECY-78-616, November 27, 1978.

1.3 D. D. Carlson et al., "Reactor Safety Study Methodology Applications Program," Sandia National Laboratories, NUREG/CR-1659, Vol. 1, SAND80-1897, April 1981.

1.4 H. W. Lewis et al., "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission," NUREG/CR-0400, September 1978.

1.5 USNRC, "NRC Statement on Risk Assessment and the Reactor Safety Study Report (WASH-1400) in Light of the Risk Assessment Review Group Report," January 18, 1979.

1.6 J. G. Kemeny et al., "Report of the President's Commission on the Accident at Three Mile Island," October 1979.

1.7 M. Rogovin et al., "Three Mile Island—A Report to the Commissioners and to the Public," NUREG/CR-1250, Vol. 1, January 1980.

1.8 J. T. Larkins and M. A. Cunningham, "Nuclear Power Plant Severe Accident Research Plan," USNRC Report NUREG-0900, January 1983.

1.9 G. P. Marino (Ed.), "Nuclear Power Plant Severe Accident Research Plan," USNRC Report NUREG-0900, Revision 1, April 1986.

1.10 M. Silberberg et al., "Reassessment of the Technical Bases for Estimating Source Terms," USNRC Report NUREG-0956, July 1986.

1.11 J. W. Hickman et al., "PRA Procedures Guide. A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," American Nuclear Society, NUREG/CR-2300, Vols. 1 and 2, January 1983.

1.12 USNRC, "Probabilistic Risk Assessment Reference Document," NUREG-1050, September 1984.

1.13 USNRC, "Policy Statement on Severe Reactor Accidents Regarding Future Design and Existing Plants," *Federal Register*, Vol. 50, p. 32138, August 8, 1985.

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

1.15 USNRC, "Revision of Backfitting Process for Power Reactors," *Federal Register*, Vol. 53, p. 20603, June 6, 1988.

1.16 USNRC, "Individual Plant Examinations for Severe Accident Vulnerabilities—10 CFR 50.54(f)," Generic Letter 88-20, November 23, 1988.

1.17 USNRC, "Integration Plan for Closure of Severe Accident Issues," SECY-88-147, May 25, 1988.

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

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

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

1.21 A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

1.22 R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: *Sequoyah* Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

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

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

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

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

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

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

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

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

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

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

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

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

1.35 USNRC, "Special Committee To Review the Severe Accident Risks Report," *Federal Register*, Vol. 54, p. 26124, June 21, 1989.

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

1.37 L. LeSage et al., "Report of the Special Committee on NUREG-1150, The NRC's Study of Severe Accident Risks," American Nuclear Society, June 1990.

1.38 Letter from Carlyle Michelson, Chairman, ACRS, to Kenneth M. Carr, Chairman, NRC, "Review of NUREG-1150, 'Severe Accident Risks: An Assessment of Five U.S. Nuclear Power Plants,'" November 15, 1990.

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

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

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

1.41 National Academy of Sciences, "Estimating Losses from Future Earthquakes—Panel Report," Panel on Earthquake Loss Estimation Methodology, National Academy Press, 1989.

1.42 Y. T. Lee et al., "A Comparison of Background Seismic Risks and the Incremental Seismic Risk Due to Nuclear Power Plants," *Nuclear Engineering and Design,* 53(1979), pp. 141–154.

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

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

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

1.46 A. L. Camp et al., "The Risk Management Implications of NUREG–1150 Methods and Results," Sandia National Laboratories, NUREG/CR–5263, SAND88–3100, September 1989.

1.47 P. Cybulskis, "Assessment of the XSOR Codes," Battelle Columbus Division, NUREG/CR–5346, BMI–2171, November 1989.

1.48 C. A. Dobbe et al., "Quality Assurance and Verification of the MACCS Code Version 1.5," Idaho National Engineering Laboratory, NUREG/CR–5376, EGG–2566, February 1990.

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

1.50 R. L. Iman and M. J. Shortencarier, "A Fortran 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use with Computer Models," Sandia National Laboratories, NUREG/CR–3624, SAND83–2365, June 1984.

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

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

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

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

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

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

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

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

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.

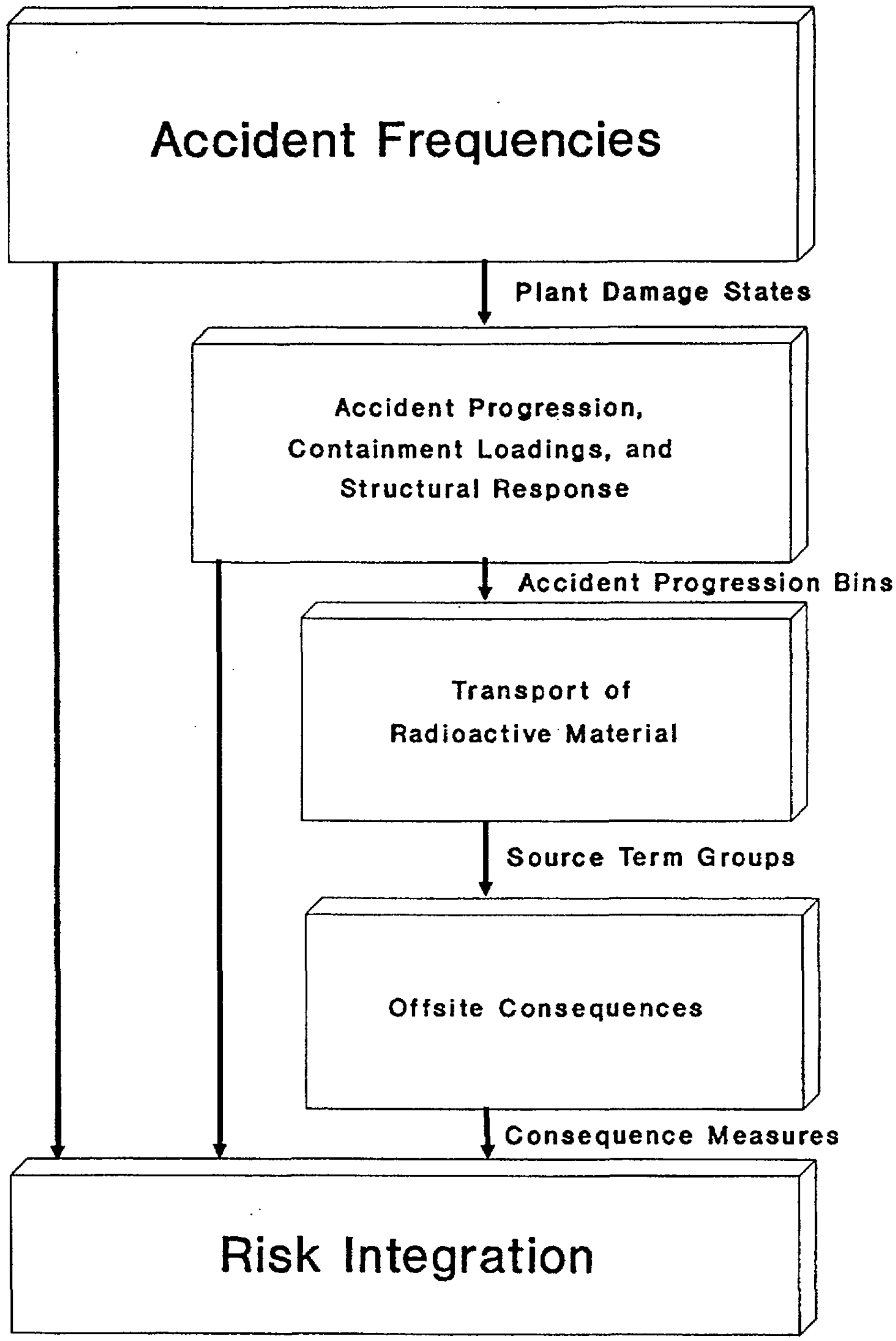


Figure 2.1 Elements of risk analysis process.

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).

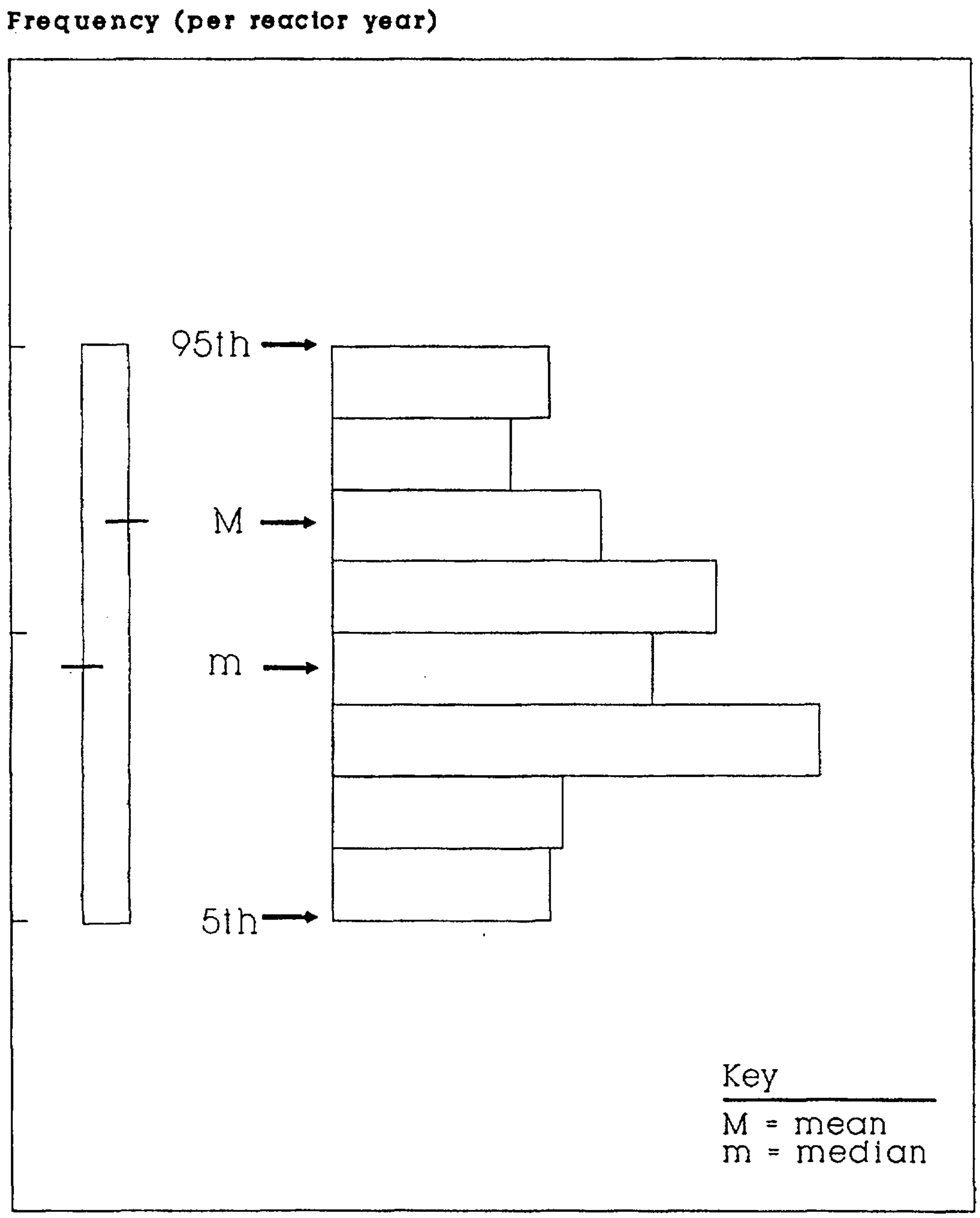


Figure 2.2 Example display of core damage frequency distribution.

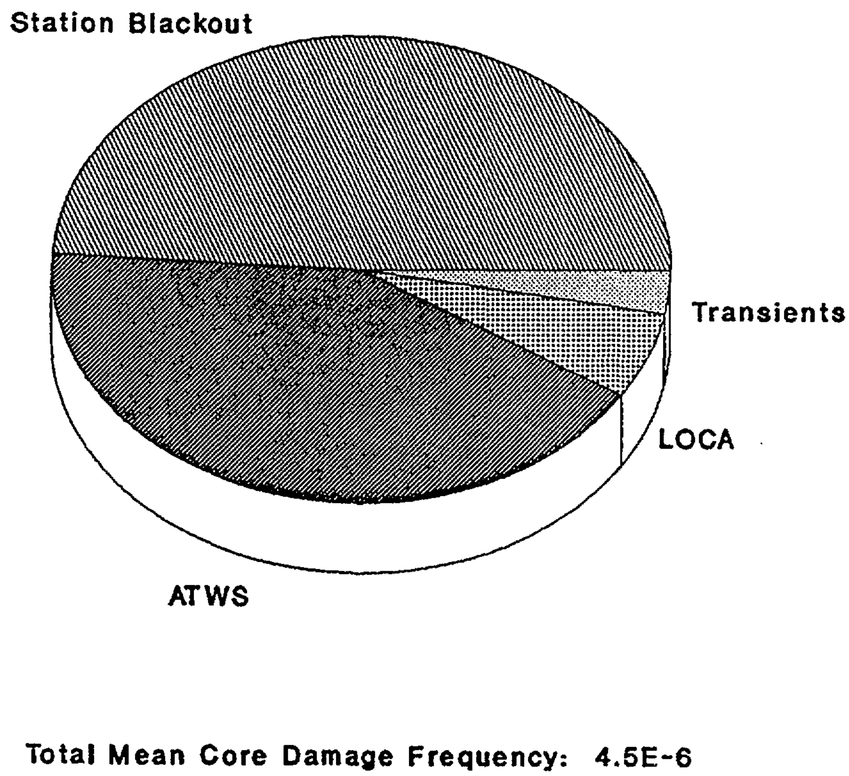


Figure 2.3 Example display of mean plant damage state frequencies.

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.

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.

*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 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.

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

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

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

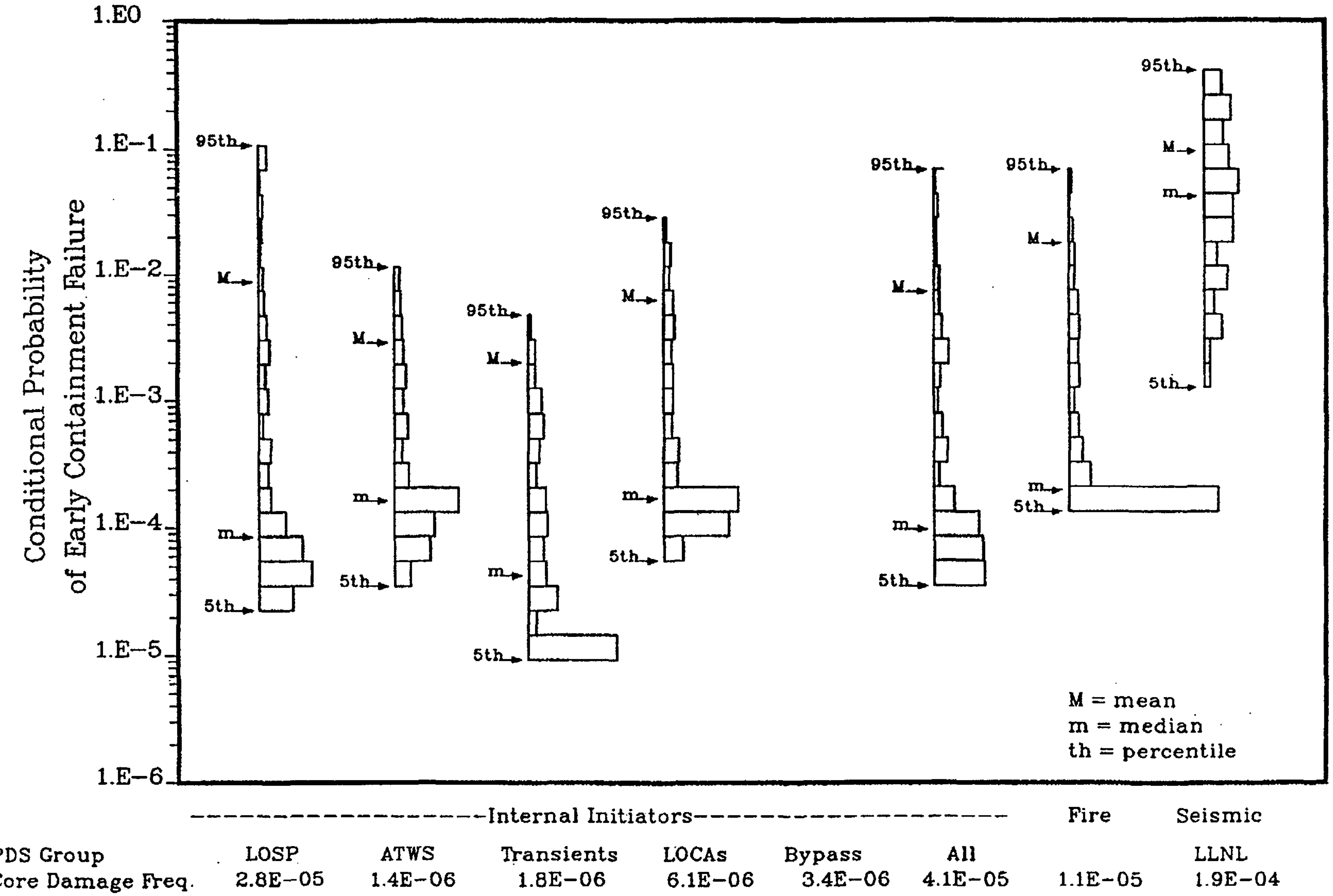


Figure 2.5 Example display of early containment failure probability distribution.

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.

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

*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.

*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.

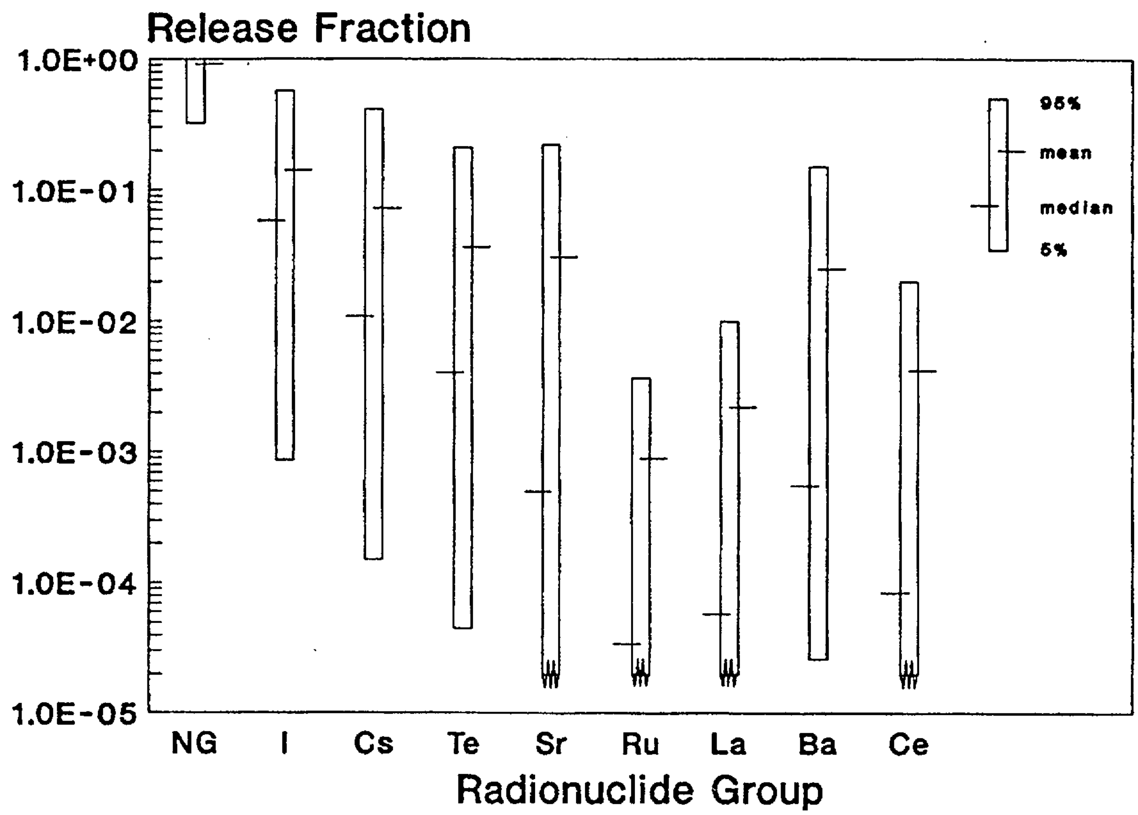


Figure 2.6 Example display of radioactive release distributions.

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

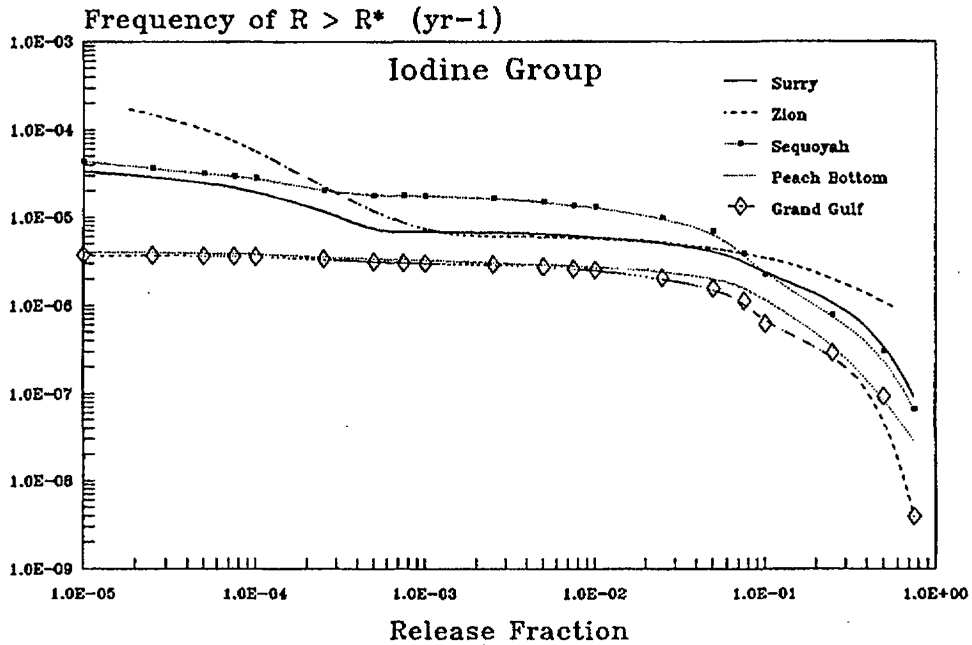
2.5.1 Methods

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.

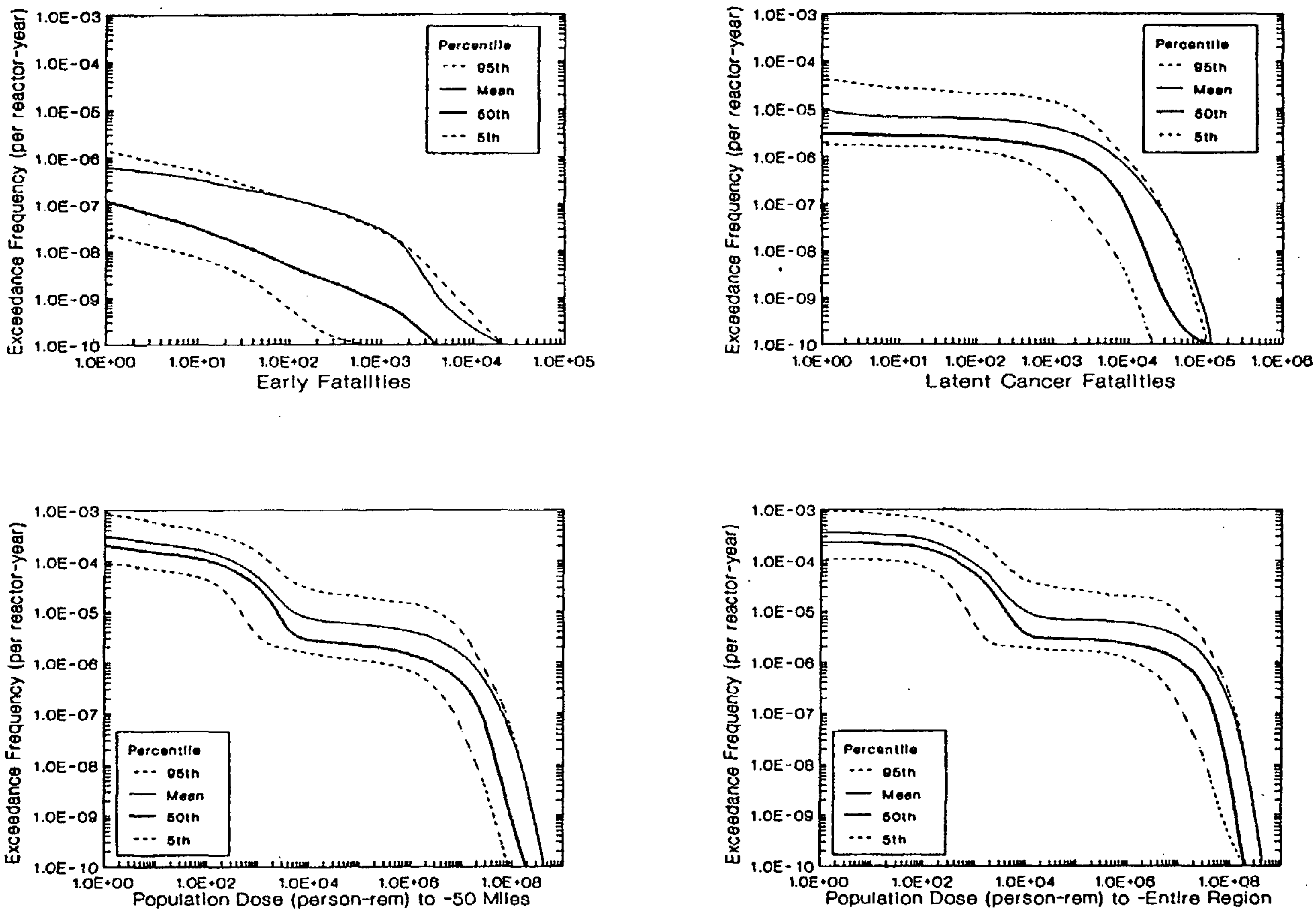
There are four steps in the performance of uncertainty analyses. Briefly, these are:

- *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

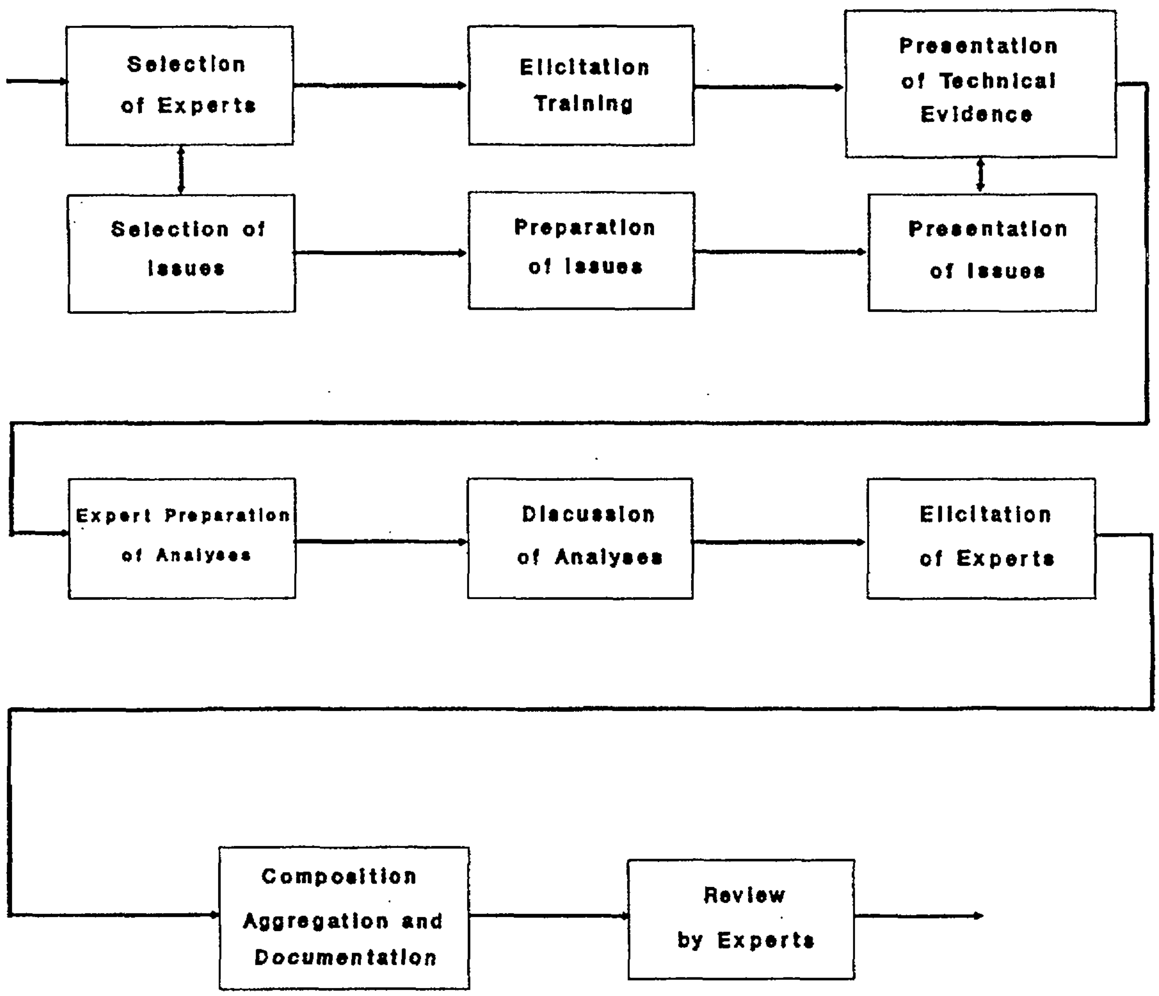


Figure 2.9 Principal steps in expert elicitation process.

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.

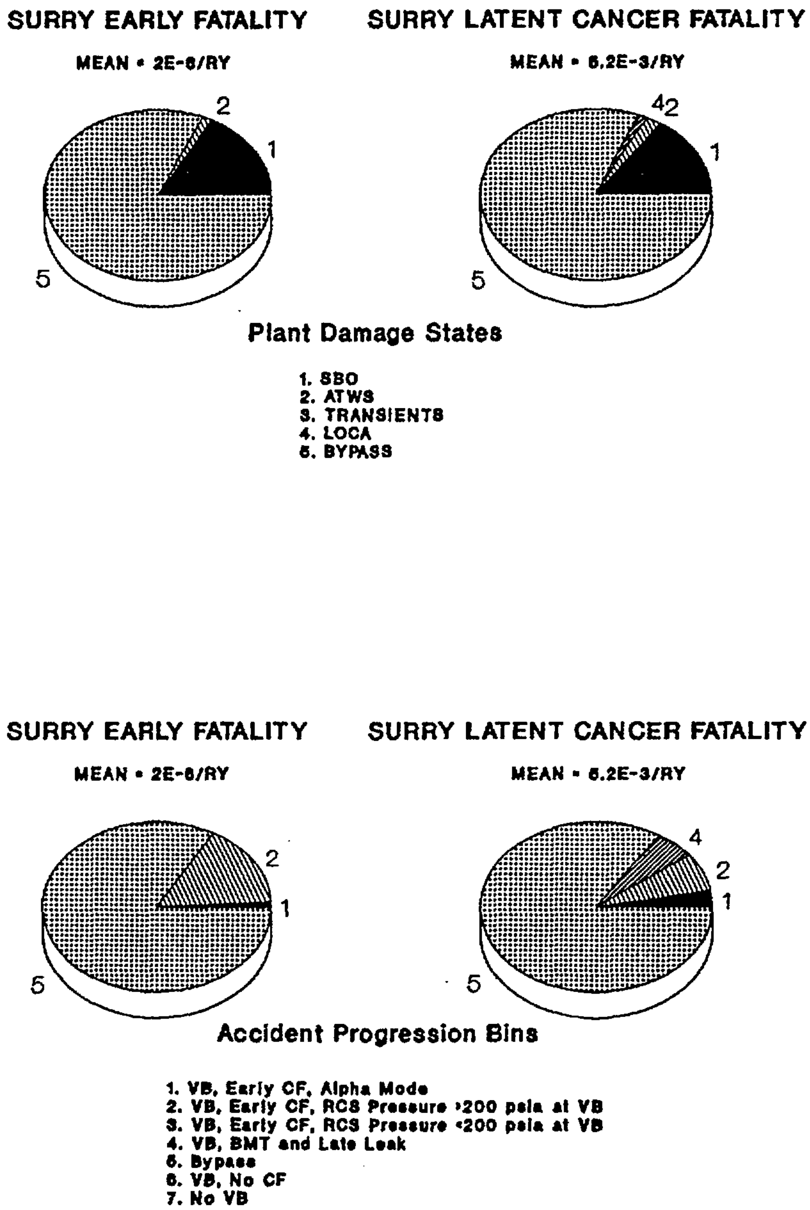


Figure 2.10 Example display of relative contributions to mean risk.

REFERENCES FOR CHAPTER 2

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

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

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

2.4 A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

2.5 R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

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

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

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

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

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

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

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

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

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

2.15 J. W. Hickman, "PRA Procedures Guide. A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," American Nuclear Society and Institute of Electrical and Electronic Engineers, NUREG/CR-2300 (2 of 2), January 1983.

2.16 USNRC, "Probabilistic Risk Assessment Reference Document," NUREG-1050, September 1984.

2.17 A. Mosleh et al., "Procedures for Treating Common Cause Failures in Safety and Reliability Studies. Procedural Framework and Examples," NUREG/CR-4780, Vol. 1, EPRI NP-5613, January 1988.

2.18 A. D. Swain III, "Accident Sequence Evaluation Program—Human Reliability Analysis Procedure," Sandia National

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

Laboratories, NUREG/CR-4772, SAND 86-1996, February 1987.

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

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

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

2.22 D. L. Berry et al., "Review and Evaluation of the Zion Probabilistic Safety Study: Plant Analysis," Sandia National Laboratories, NUREG/CR-3300, Vol. 1, SAND83-1118, May 1984.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

2.37 D. C. Kocher, "Dose Rate Conversion Factors for External Exposure to Photons

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

and Electrons," Oak Ridge National Laboratory, NUREG/CR-1918, ORNL/NUREG-79, August 1981.

2.38 International Commission on Radiological Protection, "Recommendations of ICRP," Publication 26, *Annals of ICRP,* Vol. 1, No. 3, 1977.

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

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

2.41 U.S. Department of Health and Human Services/Food and Drug Administration, "Accidental Radioactive Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies," *Federal Register,* Vol. 47, No. 205, pp. 47073–47083, October 22, 1982.

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

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

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

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

2.46 USNRC, "Reactor Risk Reference Document," NUREG-1150, Vols. 1-3, Draft for Comment, February 1987.

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

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

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

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

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

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

PART II

Summary of Plant Results

3. SURRY PLANT RESULTS

3.1 Summary Design Information

The Surry Power Station is a two-unit site. Each unit, designed by the Westinghouse Corporation, is a three-loop pressurized water reactor (PWR) rated at 2441 MWt (788 MWe) and is housed in a subatmospheric containment designed by Stone and Webster Engineering Corporation. The balance of plant systems were engineered and built by Stone and Webster Engineering Corporation. Located on the James River near Williamsburg, Virginia, Surry 1 started commercial operation in 1972. Some important system design features of the Surry plant are described in Table 3.1. A general plant schematic is provided in Figure 3.1.

This chapter provides a summary of the results obtained in the detailed risk analyses underlying this report (Refs. 3.1 and 3.2). A discussion of perspectives with respect to these results is provided in Chapters 8 through 12.

3.2 Core Damage Frequency Estimates

3.2.1 Summary of Core Damage Frequency Estimates

The core damage frequency and risk analyses performed for this study considered accidents initiated by both internal and external events (Ref. 3.1). The core damage frequency results obtained from internal events are provided in graphical form, displayed as a histogram, in Figure 3.2 (Section 2.2.2 discusses histogram development). The core damage frequency results obtained from both internal and external events are provided in tabular form in Table 3.2.

The Surry plant was previously analyzed in the Reactor Safety Study (RSS) (Ref. 3.3). The RSS calculated a point estimate core damage frequency from internal events of 4.6E–5 per year. The present study calculated a total median core damage frequency from internal events of 2.3E–5 per year. For a detailed discussion of, and insights into, the comparison between this study and the RSS, see Chapter 8.

3.2.1.1 Internally Initiated Accident Sequences

A detailed description of accident sequences important at the Surry plant is provided in Reference 3.1. For this summary report, the accident sequences described in that report have been grouped into five summary plant damage states. These are:

- Station blackout,
- Large and small loss-of-coolant accidents (LOCAs),
- Anticipated transients without scram (ATWS),
- All other transients except station blackout and ATWS, and
- Interfacing-system LOCA and steam generator tube rupture.

The relative contributions of these groups to the mean internal-event core damage frequency at Surry are shown in Figure 3.3. From Figure 3.3, it is seen that station blackout sequences are the largest contributors to mean core damage frequency. It should be noted that the plant configuration was modeled as of March 1988 and thus does not reflect implementation of the station blackout rule.

Within the general class of station blackout accidents, the more probable combinations of failures leading to core damage are:

- Loss of onsite and offsite ac power and failure of the auxiliary feedwater (AFW) system. All core heat removal is unavailable after failure of AFW. Station blackout results in the unavailability of the high-pressure injection system, the containment spray system, and the inside and outside containment spray recirculation systems. For station blackout at Unit 1 alone, it was assessed that one high-pressure injection (HPI) pump at Unit 2 would not be sufficient to provide feed and bleed cooling through the crossconnect while at the same time provide charging flow to Unit 2. Core damage was estimated to begin in approximately 1 hour if AFW and HPI flow had not been restored by that time.

- Loss of onsite and offsite ac power results in the unavailability of the high-pressure injection system, the containment spray system, the inside and outside containment spray recirculation systems, and the motor-driven auxiliary feedwater pumps. While the loss of all ac power does not affect instrumentation at the start of the station blackout, a long

Table 3.1 Summary of design features: Surry Unit 1.

1. Coolant Injection Systems	a. High-pressure safety injection and recirculation system with 2 trains and 3 pumps. b. Low-pressure injection and recirculation system with 2 trains and 2 pumps. c. Charging system provides normal makeup flow with safety injection crosstie to Unit 2.
2. Steam Generator Heat Removal Systems	a. Power conversion system. b. Auxiliary feedwater system (AFWS) with 3 trains and 3 pumps (2 MDPs, 1 TDP)* and crosstie to Unit 2 AFWS.
3. Reactivity Control Systems	a. Control rods. b. Chemical and volume control systems.
4. Key Support Systems	a. dc power provided by 2-hour design basis station batteries. b. Emergency ac power provided by 1 dedicated and 1 swing diesel generator (both self-cooled). c. Component cooling water provides cooling to RCP thermal barriers. d. Service water is gravity-fed system that provides heat removal from containment following an accident.
5. Containment Structure	a. Subatmospheric (10 psia). b. 1.8 million cubic feet. c. 45 psig design pressure. d. Reinforced concrete.
6. Containment Systems	a. Spray injection initiated at 25 psia with 2 trains and 2 pumps. b. Inside spray recirculation initiated (with 2-minute time delay) at 25 psia with 2 trains and 2 pumps (both pumps inside containment). c. Outside spray recirculation initiated (with 5-minute time delay) at 25 psia with 2 trains and 2 pumps (both pumps outside containment). d. Inside and outside spray recirculation systems are the only sources of containment heat removal after a LOCA.

*MDP — Motor-Driven Pump.
TDP — Turbine-Driven Pump.

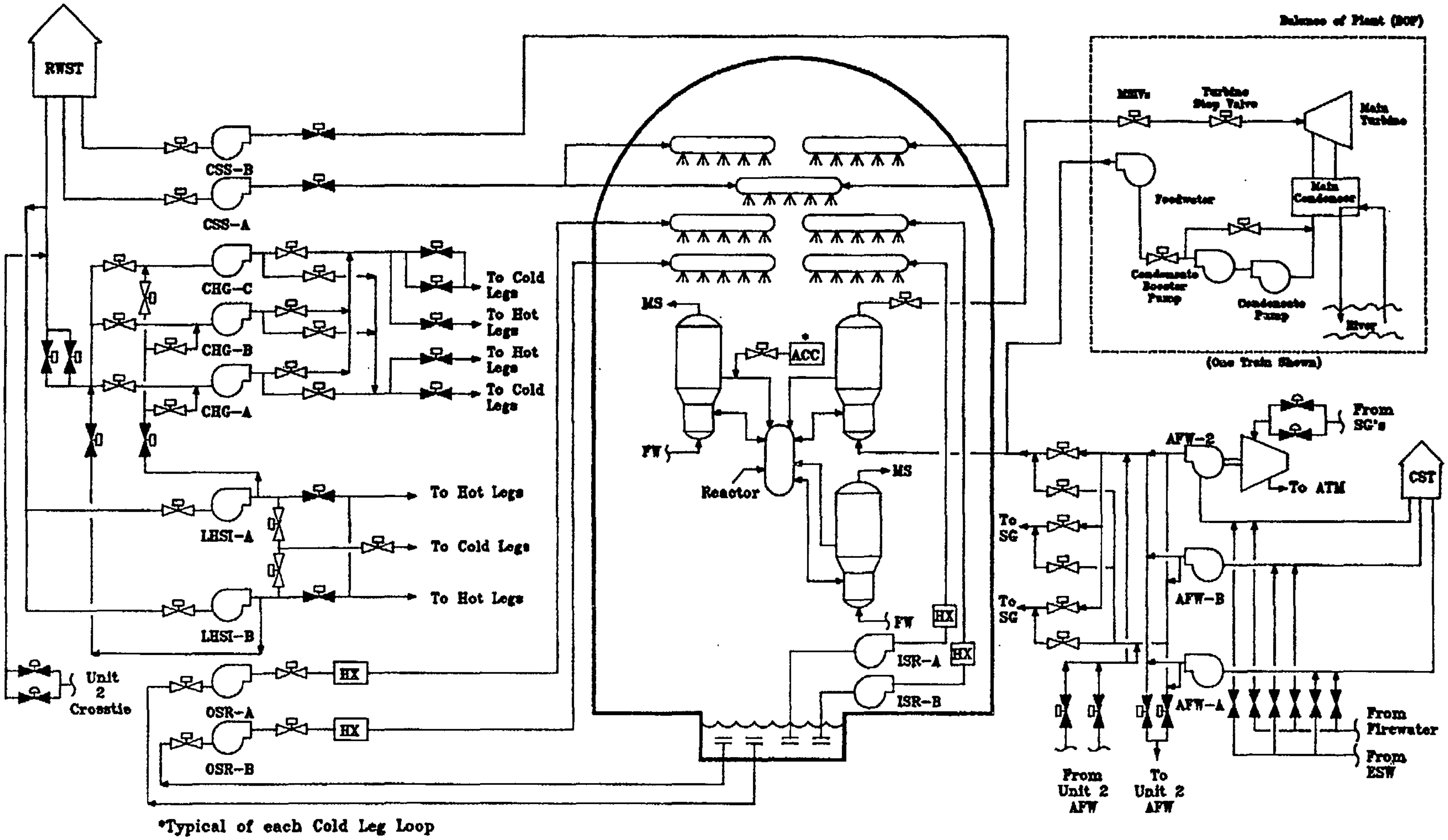


Figure 3.1 Surry plant schematic.

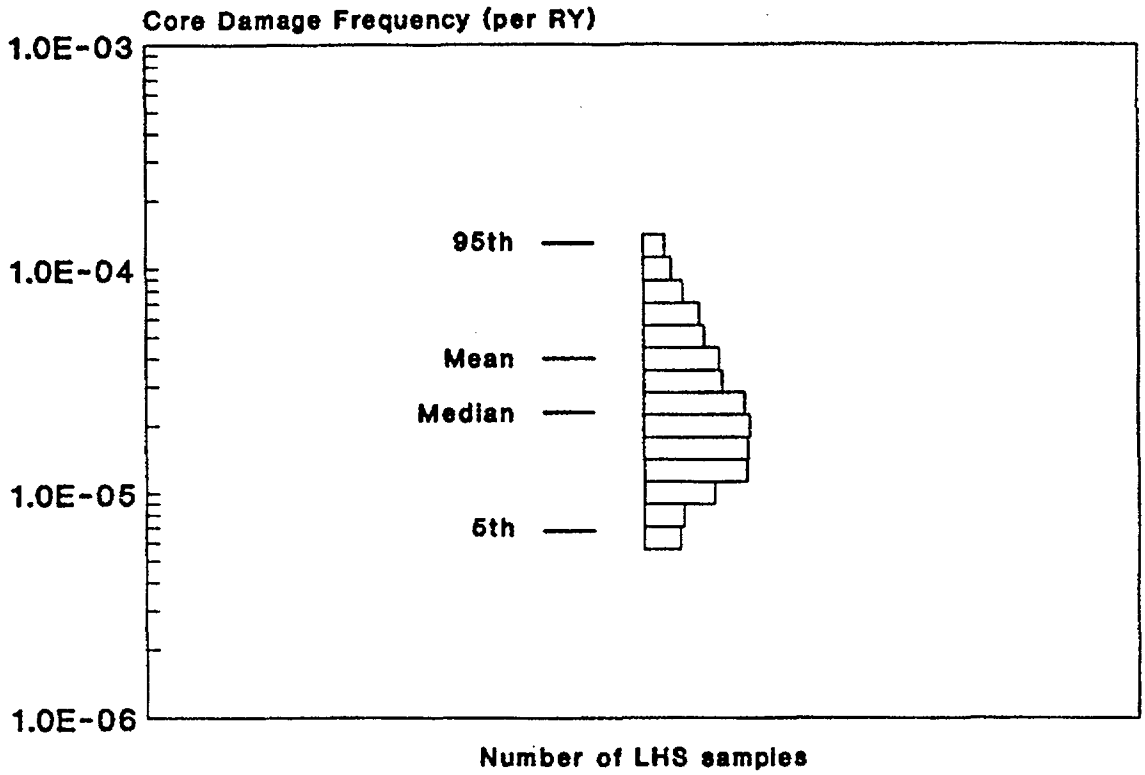


Figure 3.2 Internal core damage frequency results at Surry.*

Table 3.2 Summary of core damage frequency results: Surry.*

	5%	Median	Mean	95%
Internal Events	6.8E-6	2.3E-5	4.0E-5	1.3E-4
Station Blackout				
Short Term	1.1E-7	1.7E-6	5.4E-6	2.3E-5
Long Term	6.1E-7	8.2E-6	2.2E-5	9.5E-5
ATWS	3.2E-8	4.2E-7	1.6E-6	5.9E-6
Transient	7.2E-8	6.9E-7	2.0E-6	6.0E-6
LOCA	1.2E-6	3.8E-6	6.0E-6	1.6E-5
Interfacing LOCA	3.8E-11	4.9E-8	1.6E-6	5.3E-6
SGTR	1.2E-7	7.4E-7	1.8E-6	6.0E-6
External Events**				
Seismic (LLNL)	3.9E-7	1.5E-5	1.2E-4	4.4E-4
Seismic (EPRI)	3.0E-7	6.1E-6	2.5E-5	1.0E-4
Fire	5.4E-7	8.3E-6	1.1E-5	3.8E-5

*As discussed in Reference 3.4, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

**See "Externally Initiated Accident Sequences" in Section 3.2.1.2 for discussion.

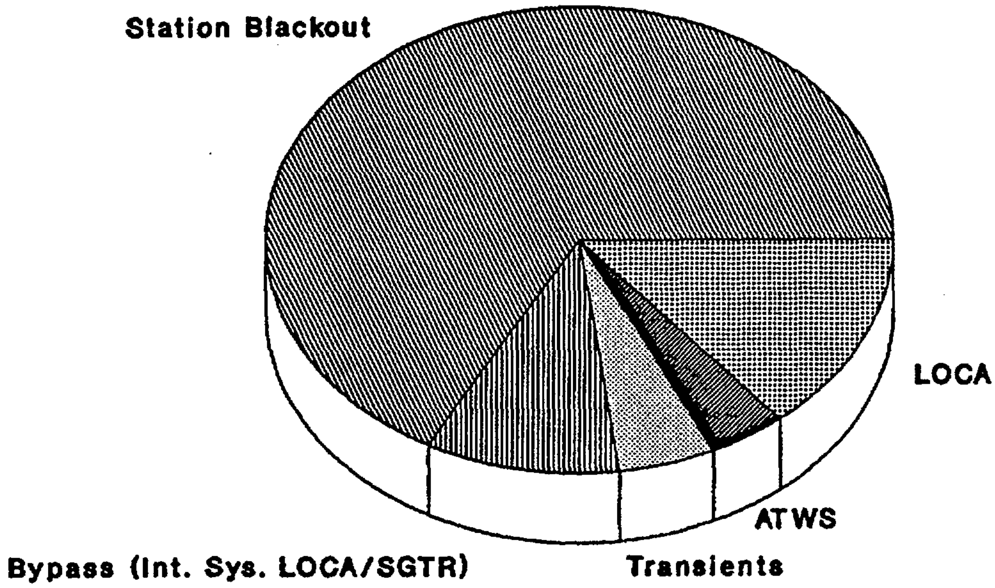


Figure 3.3 Contributors to mean core damage frequency from internal events at Surry.

duration station blackout leads to battery depletion and subsequent loss of vital instrumentation. Battery depletion was concluded to occur after approximately 4 hours. The ability to subsequently provide decay heat removal with the turbine-driven AFW pump is lost because of the loss of all instrumentation and control power. Using information from Reference 3.5, approximately 3 hours beyond the time of battery depletion was allowed for restoration of ac power before core uncovery would occur.

- Loss of onsite and offsite ac power, followed by a reactor coolant pump seal LOCA due to loss of all seal cooling. Station blackout also results in the unavailability of the HPI system, as well as the auxiliary feedwater motor-driven pumps, the containment spray system, and the inside and outside spray recirculation systems. Continued coolant loss through the failed seals, with unavailability of the HPI system, leads to core uncovery.

Within the general class of LOCAs, the more probable combinations of failures are:

- LOCA with an equivalent diameter of greater than 6 inches in the reactor coolant system (RCS) piping with failure of the low-pressure injection or recirculation system. Recovery of equipment is unlikely for the system failures assessed to be most likely and, because the break size is sufficiently large, the time to core uncovery is approximately 5 to 10 minutes, leaving virtually no time for recovery actions. All containment heat removal systems are available. The dominant contributors to failure of the low-pressure recirculation function are the common-cause failure of the refueling water storage tank (RWST) isolation valves to close, common-cause failure of the pump suction valves to open, common-cause failure of the discharge isolation valves to the hot legs to open, or miscalibration of the RWST level sensors.

- Intermediate-size LOCAs with an equivalent diameter of between 2 and 6 inches in the

RCS piping with failure of the low-pressure injection or recirculation core cooling system. All containment heat removal systems are available, but the continued heatup and boiloff of primary coolant leads to core uncovery in 20 to 50 minutes. The dominant contributors to low-pressure injection failure are common-cause failure of the low-pressure injection (LPI) pumps to start or plugging of the normally open LPI injection valves.

- Small-size LOCAs with an equivalent diameter of between 1/2 and 2 inches in the RCS piping with failure of the HPI system. All containment heat removal systems are available, but the continued heatup and boiloff of primary coolant leads to core uncovery in 1 to 8 hours. The dominant contributors to HPI system failures are hardware failures of the check valves in the common suction and discharge line of all three charging pumps or common-cause failure of the motor-operated valves in the HPI discharge line.

Within the general class of containment bypass accidents, the more probable combinations of failures are:

- An interfacing-system LOCA resulting from a failure of any one of the three pairs of check valves in series that are used to isolate the high-pressure RCS from the LPI system. The failure modes of interest for Event V are rupture of valve internals on both valves or failure of one valve to close upon repressurization (e.g., during a return to power from cold shutdown) combined with rupture of the other valve. The resultant flow into the low-pressure system is assumed to result in failure (rupture) of the low-pressure piping or components outside the containment boundary. Although core inventory makeup by the high-pressure systems is initially available, inability to switch to recirculation would eventually lead to core damage approximately 1 hour after the initial failure. Because of the location of the postulated system failure (outside containment), all containment mitigating systems are bypassed.

- A steam generator tube rupture (SGTR) accident initiated by the double-ended guillotine rupture of one steam generator (SG) tube. (Multiple tube ruptures may be possible but were not considered in this analysis.) If the operators fail to depressurize the reactor coolant system in a timely manner (in about 45 minutes), there is a high probability that water will be forced through the safety relief valves (SRVs) on the steam line from the affected SG. The probability that the SRVs will fail to reclose under these conditions is also estimated to be very high (near 1.0). Failure to close (gag the SRVs) by a local, manual action results in a non-isolable path from the RCS to the environment. After the entire contents of the refueling water storage tank are pumped through the broken SG tube, the core uncovers. The onset of core degradation is thus not expected until about 10 hours after the start of the accident.

3.2.1.2 Externally Initiated Accident Sequences

A detailed description of accident sequences initiated by external events important at the Surry plant is provided in Part 3 of Reference 3.1. The accident sequences described in that reference have been divided into two main types for this study. These are:

- Seismic, and
- Fire.

A scoping study has also been performed to assess the potential effects of other externally initiated accidents (Ref. 3.1, Part 3). This analysis indicated that the following external-event sources could be excluded based on the low frequency of the initiating event:

- Air crashes,
- Hurricanes,
- Tornados,
- Internal flooding, and
- External flooding.

1. Seismic Accident Frequency Analysis

The relative contribution of classes of seismically and fire-initiated accidents to the total mean frequency of externally initiated core damage accidents is provided in Figure 3.4. As may be seen, seismically initiated loss of offsite power plant transients and transients that (through cooling system failures) lead to reactor coolant pump seal LOCAs are the most likely causes of externally caused core damage accidents. For these two accident initiators, the more probable combinations of system failures are:

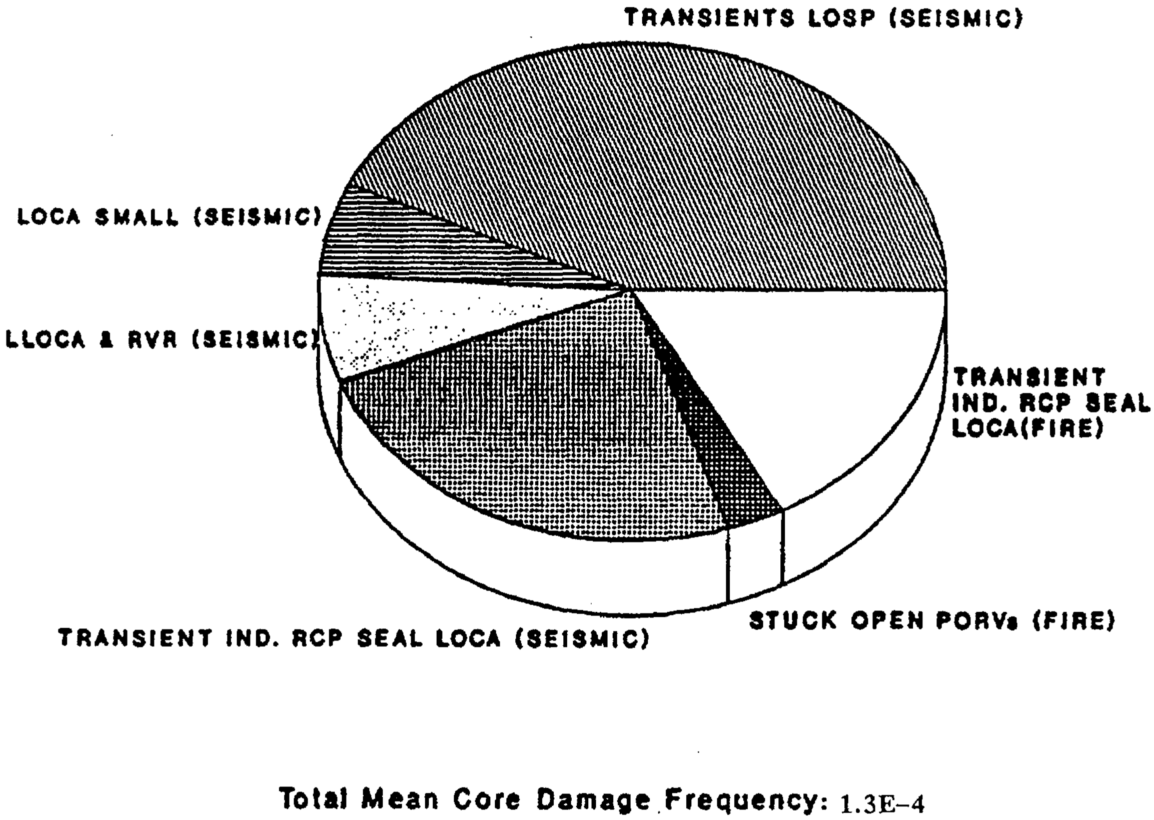


Figure 3.4 Contributors to mean core damage frequency from external events (LLNL hazard curve) at Surry.

- Transient-initiated accident sequences resulting from loss of offsite power in conjunction with failures of the auxiliary feedwater system and failure of the feed and bleed mode of core cooling. These result from either seismically induced diesel generator failures (causing station blackout and eventual battery depletion) or from seismically induced failure of the condensate storage tank in conjunction with power-operated relief valve (PORV) failures.

- Loss of offsite power (LOSP) due to seismically induced failure of ceramic insulators in the switchyard, with simultaneous (seismic) failure of both high-pressure injection (HPI) and component cooling water (CCW) systems (the redundant sources of seal cooling). Failures of HPI result from seismic failures of the refueling water storage tank or emergency diesel generator load panels, while seismic failures of the diesels or the CCW heat exchanger supports result in loss of the CCW system.

As discussed in Chapter 2, the seismic analysis in this report made use of two sets of hazard curves from Lawrence Livermore National Laboratory (LLNL) (Ref. 3.6) and the Electric Power Research Institute (EPRI) (Ref. 3.7). The above accident sequences are dominant for both sets of hazard curves. In addition, the differences between the seismic risk estimates shown in Table 3.2 for the LLNL and the EPRI cases are due entirely to the differences between the two sets of hazard curves. That is, the system models, failure rates, and success logic were identical for both estimates.

The seismic hazard associated with the curves developed by EPRI was significantly less than that of the LLNL curves. Differences between these curves result primarily from differences between the methodology and assumptions used to develop the hazard curves. In the LLNL program, considerable emphasis was placed on a wide range of uncertainty in the ground-motion attenuation models, while a relatively coarse set of seismic tectonic provinces was used in characterizing each site. By contrast, in the EPRI program considerable emphasis was placed on a fine zonation for the tectonic provinces, and very little uncertainty in the ground-motion attenuation was considered. In any case, it is the difference between the two sets of hazard curves that causes the differences between the numeric estimates in Table 3.2.

2. Fire Accident Frequency Analysis

The fire-initiated accident frequency analyses performed for this report considered the impact of fires beginning in a variety of separate locations within the plant. Those locations found to be most important were:

- Emergency switchgear room,
- Control room,
- Auxiliary building, and
- Cable vault and tunnel.

In the emergency switchgear room, a fire is assumed to fail either control or power cables for both HPI and CCW, leading directly to a reactor coolant pump seal LOCA. No additional random failures were required for this sequence to lead to core damage. (Credit was given for operator recovery by crossconnecting the Unit 2 HPI system.) The identical scenario arises as the result of fires postulated in the auxiliary building and the cable vault and tunnel. Thus, fires in these three areas both cause the initiating event (a seal LOCA) and fail the system required to mitigate the scenario (i.e., HPI).

In the control room, a fire in a bench board was determined to lead to spurious actuation of a PORV with smoke-induced abandonment of the control room. A low probability of successful operator recovery actions from the remote shutdown panel (RSP) was assessed since the PORV closure status is not displayed at the RSP. In addition, the PORV block valve controls in the RSP are not routed independently of the control room bench board and thus may not function.

The frequency of fire-initiated accident scenarios in other locations contributed less than 10 percent to the total fire-initiated core damage frequency.

3.2.2 Important Plant Characteristics (Core Damage Frequency)

Characteristics of the Surry plant design and operation that have been found to be important in the analysis of core damage frequency include:

1. Crossties Between Units

The Surry plant has numerous crossties between similar systems at Units 1 and 2. Some of these were installed in order to comply with requirements of 10 CFR Part 50, Appendix R (fire protection) (Ref. 3.8) or high-energy line-break threats, and some were installed for operational reasons. Crossties exist for the auxiliary feedwater system, the charging pump system, the charging pump cooling system, and the refueling water storage tanks. These crossties are subject to technical specifications, their potential use is included in the plant operating procedures, and they are reviewed in operator training. The availability of such crossties was estimated to reduce the internal-event core damage frequency by approximately a factor of 3.

2. Diesel Generators

Surry is a two-unit site with three emergency diesel generators (DGs), one of which is a swing diesel (which can be aligned to one unit or the other), while many other PWR plants have dedicated diesels for each safety-grade power train (i.e., four DGs for a two-unit site). Each DG is self-cooled and supplied with a dedicated battery (independent of the batteries providing power to the vital dc buses) for starting. The latter two factors eliminate potential common-cause failure modes found important at other plants in this study (e.g., Peach Bottom and Grand Gulf). The Surry site also has a gas turbine generator. However, administrative procedures and design characteristics of support equipment (e.g., dc batteries and compressed air) preclude its use during a station blackout accident.

3. Reactor Coolant Pump Seals

At Surry, there are two diverse and independent methods for providing reactor coolant pump seal cooling: the component cooling water system and the charging system (which has its own dedicated cooling system). The only common support systems for seal cooling are ac and dc power. As such, reactor coolant pump seal LOCAs have been found important only in station blackout sequences. This is in contrast to some other PWR plants that have a dependency between charging pumps and the component cooling water system and thus greater potential for loss of seal cooling. Without cooling, the seals were expected to degrade or fail. The probability of seal failure upon loss of seal cooling was studied in detail by the expert panel elicitation (Ref. 3.9). Reflecting this, the Surry analyses have found that station blackout accident sequences with significant seal leakage are important contributors to the total frequency of core damage.

4. **Battery Capacity**

For the Surry plant, the station Class 1E battery depletion time following station blackout has been estimated to be 4 hours (Ref. 3.5). The inability to ensure availability for longer times contributes significantly to the frequency of core damage resulting from station blackout accident sequences. The batteries are designed and tested for 2 hours. A 4-hour battery depletion time is considered realistic because of the margin in the design and possible load shedding.

5. **Capability for Feed and Bleed Core Cooling**

In the Surry plant, the high-pressure injection system and the power-operated relief valves have the capability to provide feed and bleed core cooling in the event of loss of the cooling function of the steam generators. This capability to provide core cooling through feed and bleed is estimated to result in approximately a factor of 1.4 reduction in core damage frequency. Without the crossties of auxiliary feedwater to Unit 2, which enhances overall reliability of the auxiliary feedwater system, the benefit of feed and bleed cooling would be much greater.

3.2.3 Important Operator Actions

The estimation of accident sequence and total core damage frequencies depends substantially on the credit given to operating crews in performing actions before and during an accident. Failure to perform these actions correctly and reliably will have a substantial impact on estimated core damage frequency. For the Surry plant, actions found to be important are discussed below.

During loss of offsite power and station blackout, important actions required to be taken by the operating crew to prevent core damage include:

- Align alternative source of condensate to condensate storage tank

  The primary source of condensate for the AFW system is a 100,000-gallon tank. This is nominally sufficient for the duration of most station blackout events. But in the event that a steam generator becomes faulted, the increased AFW flow would require the provision of additional condensate water. This would involve manual local actions.

- Isolate condenser water box

  Surry has a somewhat unique gravity-fed service water system that relies on the head difference between the intake canal and the discharge canal to provide flow through service water heat exchangers. The intake canal is normally supplied with water by the circulating water pumps. These pumps are not provided with emergency power and are thus unavailable after a loss of offsite power. The condenser at each unit is provided with four inlet and four outlet isolation valves. These isolation valves are provided with emergency power. Each inlet isolation valve is provided with a hand wheel, located in the turbine building, in order to allow manual condenser isolation during station blackout to avoid draining the canal.

- Cool down and depressurize the RCS

  The Emergency Contingency Actions (ECAs) call for depressurization of the secondary side of the steam generators during a station blackout to provide cooldown and depressurization of the reactor coolant system. This action is done through manual, local valve lineups.

During steam generator tube rupture, the most important operator action is to cool down and depressurize the RCS within approximately 45 minutes after the event in order to prevent lifting the relief valves on the damaged steam generator. Other possible recovery actions considered in this accident sequence include: provision of an alternative source of steam generator feed flow in response to a loss of feed flow; crossconnect of HPI from Unit 2 or opening of alternative injection paths in response to failure of safety injection flow; and isolation of a damaged, faulted steam generator.

During small-break and medium-break LOCA accident sequences, two human actions are principally important in response to loss of core coolant injection or recirculation. These are:

- Cool down and depressurize the RCS

  RCS cooldown and depressurization is the procedure directed for all small-break LOCAs. This event is important to reduce the pressure in the RCS and thus reduce the leak rate. Successful cooldown and depressurization of the RCS will delay the need to go to recirculation cooling.

- Crossconnect high-pressure injection (HPI)

  In the event that HPI pumps or water sources are unavailable at Unit 1, HPI flow can be provided via a crosstie with the Unit 2 charging system. This crosstie requires an operator to locally open and/or close valves in the charging pump area. It was estimated that the crossconnect of HPI would require 15 to 20 minutes. This and other timing considerations were such that the HPI crossconnect was considered viable only for small and very small LOCAs.

3.2.4 Important Individual Events and Uncertainties (Core Damage Frequency)

As discussed in Chapter 2, the process of developing a probabilistic model of a nuclear power plant involves the combination of many individual events (initiators, hardware failures, operator errors, etc.) into accident sequences and eventually into an estimate of the total frequency of core damage. After development, such a model can also be used to assess the relative importance and contribution of the individual events. The detailed studies underlying this report have been analyzed using several event importance measures. The results of the analyses using two measures, "risk reduction" and "uncertainty" importance, are summarized below.

- Risk (core damage frequency) reduction importance measure (internal events)

  The risk-reduction importance measure is used to assess the change in core damage frequency as a result of setting the probability of an individual event to zero. Using this measure, the following individual events were found to cause the greatest reduction in the estimated core damage frequency if their probabilities were set to zero:

  - Loss of offsite power initiating event. The core damage frequency would be reduced by approximately 61 percent.

  - Failure of diesel generator number one to start. The core damage frequency would be reduced by approximately 25 percent.

  - Probability of not recovering ac electric power between 3 and 7 hours after loss of offsite power. The core damage frequency would be reduced by approximately 24 percent.

  - Failure to recover diesel generators. The core damage frequency would be reduced by approximately 18 to 21 percent.

- Uncertainty importance measure (internal events)

  A second importance measure used to evaluate the core damage frequency results is the uncertainty importance measure. For this measure, the relative contribution of the uncertainty of groups of component failures and basic events to the uncertainty in total core damage frequency is calculated. Using this measure, the following event groups were found to be most important:

  - Probabilities of diesel generators failing to start when required;

  - Probabilities of diesel generators failing to run for 6 hours;

  - Frequency of loss of offsite power; and

  - Frequency of interfacing-system LOCA.

It should be noted that many events each contribute a small amount to the uncertainty in core damage frequency; no single event dominates the uncertainty.

3.3 Containment Performance Analysis

3.3.1 Results of Containment Performance Analysis

The Surry containment system uses a subatmospheric concept in which the containment building housing the reactor vessel, reactor coolant system, and secondary system's steam generator is maintained at 10 psia. The containment building is a reinforced concrete structure with a volume of 1.8 million cubic feet. Its design basis pressure is 45 psig, whereas its mean failure pressure is estimated to be 126 psig. As previously discussed in Chapter 2, the method used to estimate accident loads and containment structural response for Surry made extensive use of expert judgment to interpret and supplement the limited data available.

The potential for early Surry containment failure is of major interest in this risk analysis. The principal threats identified in the Surry risk analyses (Ref. 3.2) as potentially leading to early containment failure are: (1) pressure loads, i.e., hydrogen combustion and direct containment heating due to ejection of molten core material via the rapid expulsion of hot steam and gases from the reactor coolant system; and (2) in-vessel steam explosions leading to vessel failure with the vessel upper head being ejected and impacting the containment building dome area (the so-called alpha-mode failure). Containment bypass (such as failures of reactor coolant system isolation check valves in the emergency core cooling system or steam generator tubes) is another serious threat to the integrity of the containment system.

The results of the Surry containment analysis are summarized in Figures 3.5 and 3.6. Figure 3.5 displays information in which the conditional probabilities of seven containment-related accident progression bins; e.g., VB, alpha, early CF, are presented for each of seven plant damage states; e.g., loss of offsite power. This information indicates that, on a plant damage state frequency-weighted average,* the conditional mean probability from internally initiated accidents of: (1) early containment failure is about 0.01, (2) late containment failure (basemat melt-through or leakage) is about 0.06, (3) direct bypass of the containment is about 0.12, and (4) no containment failure is 0.81. Figure 3.6 further displays the conditional probability distribution of early containment failure for each plant damage state to show the estimated range of uncertainties in these containment failure predictions. The important conclusions to be drawn from the information in Figures 3.5 and 3.6 are: (1) the mean conditional probability of early containment failure from internal events is low; i.e., less than 0.01; (2) the principal containment release mechanism is bypass due to interfacing-system LOCA; and (3) external initiating events such as fire and earthquakes produce higher early and late containment failure probabilities.

*Each value in the column in Figure 3.5 labeled "All" is obtained by calculating the products of individual accident progression bin conditional probabilities for each plant damage state and the ratio of the frequency of that plant damage state to the total core damage frequency.

The accident progression analyses performed for this report are particularly noteworthy in that, for core melt accidents at Surry, there is a high probability that the reactor coolant system (RCS) will be at relatively low pressures (less than 200 psi) at the time of molten core penetration of the lower reactor vessel head, thereby reducing the potential for direct containment heating (DCH). There are several reasons for concluding that the RCS will be at low system pressure such as: stuck-open PORVs, operator depressurization, failed reactor coolant pump seals, induced failures of RCS piping due to high temperatures, and the relative "mix" of plant damage states (i.e., for the frequency of plant damage states initially at high versus low RCS pressures). Accordingly, it has been concluded that the potential for early containment failure due to the phenomenon of DCH is less in the risk analyses underlying this report relative to previous studies (Ref. 3.10) on the basis of a combination of higher probabilities of low RCS pressures (discussed above), lower calculated pressures given direct containment heating, and greater estimated strength of the Surry containment building (Ref. 3.2). (See Section C.5 of Appendix C for additional discussion of DCH and why its importance is now less.)

Additional discussions on containment performance (for all studied plants) are provided in Chapter 9.

3.3.2 Important Plant Characteristics (Containment Performance)

Characteristics of the Surry plant design and operation that are unique to the containment building during core damage accidents include:

1. **Subatmospheric Containment Operation**

   The Surry containment is maintained at a subatmospheric pressure (10 psia) during operation with a continual monitoring of the containment leakage. As a result, the likelihood of pre-existing leaks of significant size is negligible.

2. **Post-Accident Heat Removal System**

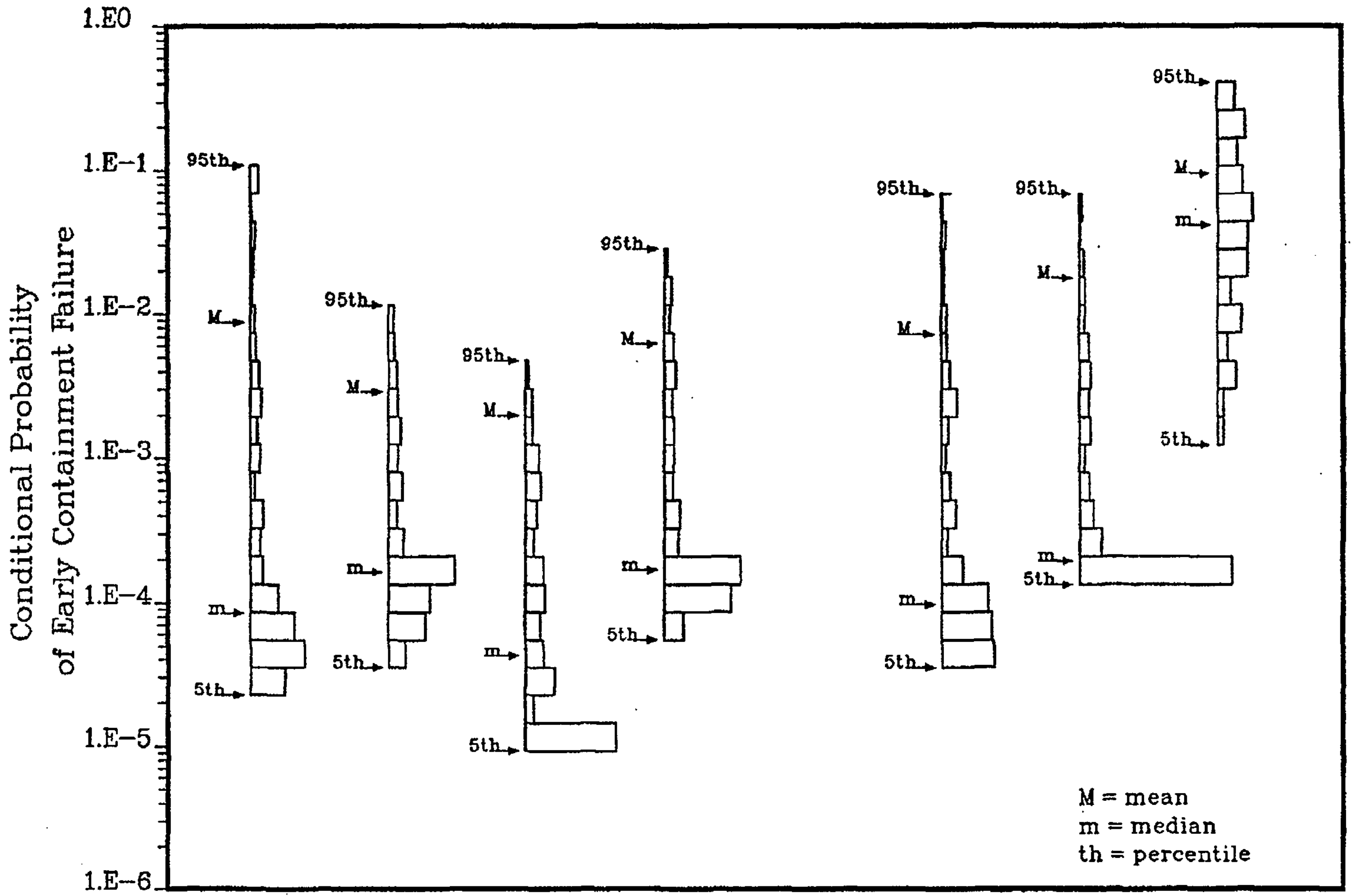
   The Surry containment does not have fan cooler units that are qualified for post-accident heat removal as do some other PWR plants. Containment (and core) heat removal

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

SUMMARY PDS GROUP (Mean Core Damage Frequency)

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

Figure 3.5 Conditional probability of accident progression bins at Surry.



	------------------Internal Initiators------------------						Fire	Seismic
PDS Group	LOSP	ATWS	Transients	LOCAs	Bypass	All		LLNL
Core Damage Freq.	2.8E-05	1.4E-06	1.8E-06	6.1E-06	3.4E-06	4.1E-05	1.1E-05	1.9E-04

Figure 3.6 Conditional probability distributions for early containment failure at Surry.

following an accident is provided by the containment spray recirculation system, whereas, in some PWR plants, post-accident heat removal can also be provided by the residual heat removal system heat exchangers in the emergency core cooling system.

3. **Reactor Cavity Design**

   The reactor cavity area is not connected directly with the containment sump area. As a result, if the containment spray systems fail to operate during an accident, the reactor cavity will be relatively dry. The amount of water in the cavity can have a significant influence on phenomena that can occur after reactor vessel lower head failure, such as magnitude of containment pressurization from direct containment heating and post-vessel failure steam generation, the formation of coolable debris beds, and the retention of radioactive material released during core-concrete interactions.

4. **Containment Building Design**

   The containment volume and high failure pressure provide considerable capacity for accommodation of severe accident pressure loads.

3.4 Source Term Analysis

3.4.1 Results of Source Term Analysis

In the Surry plant, the absolute frequency of an early failure of the containment* due to the loads produced in a severe accident is small. Although the absolute frequency of containment bypass is also small, for internal accident initiators it is greater than the absolute early failure frequency. Thus, bypass sequences are the more likely means of obtaining a large release of radioactive material. Figure 3.7 illustrates the distribution of source terms associated with the accident progression bin representing containment bypass. The range of release fractions is quite large, primarily as the result of the range of parameters provided by the experts. The magnitude of the release for many of the elemental groups is also large, indicative of a potentially serious accident. Typically, consequence analysis codes only predict the occurrence of early fatalities in the surrounding population when the release fractions of the volatile groups (iodine, cesium, and tellurium) exceed approximately 10 percent (Ref. 3.11). For the bypass accident progression bin, the median value for the volatile radionuclides is approximately at the 10 percent level whereas for the early containment failure bin not shown, the releases are lower. The median values are somewhat smaller than 10 percent, but the ranges extend to approximately 30 percent.

In contrast to the large source term for the bypass bin, Figure 3.8 provides the range of source terms predicted for an accident progression bin involving late failure of the containment. The fractional release of radionuclides for this bin is several orders of magnitude smaller than for the bypass bin, except for iodine, which can be reevolved late in the accident. It should be noted that, for many of the elemental groups, the mean of the distribution falls above the 95th percentile value. For distributions that occur over a range of many orders of magnitude, sampling from the extreme tail of the distribution (at the high end) can dominate and cause this result.

Additional discussion on source term perspectives is provided in Chapter 10.

3.4.2 Important Plant Characteristics (Source Term)

Plant design features that affect the mode and likelihood of containment failure also influence the magnitude of the source term. These features were described in the previous section. Plant features that have a more direct influence on the source term are described in the following paragraphs.

1. **Containment Spray System**

   The Surry plant has an injection spray system that uses the refueling water storage tank as a water source and a recirculation spray system that recirculates water from the containment sump. Sprays are an effective means for removing airborne radioactive aerosols. For sequences in which sprays operate throughout the accident, it is most likely that the containment will not fail and the leakage to the environment will be minor. If the containment does fail late in the accident following extended spray operation, analyses indicate that the release of aerosols will be extremely small. Even in a station blackout case with delayed recovery of sprays, condensation of steam from the air, and a subsequent hydrogen explosion that fails containment, Source Term Code Package (STCP) analyses indicate that spray operation results in substantially reduced source terms (Ref. 3.12).

*In this section, the absolute frequencies of early containment failure are discussed (i.e., including the frequencies of the plant damage states). This is in contrast to the previous section, which discusses conditional failure probabilities (i.e., given that a plant damage state occurs).

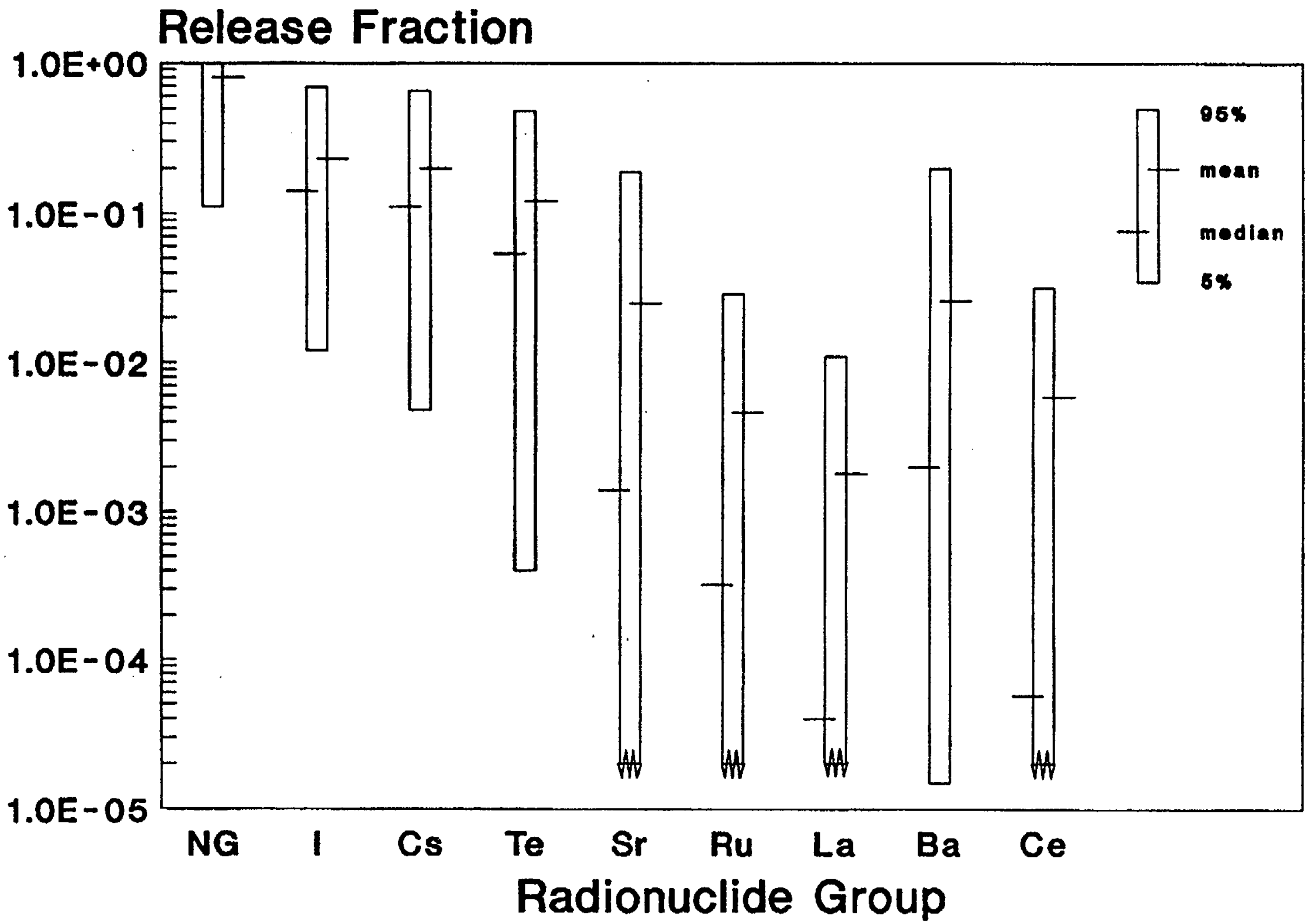


Figure 3.7 Source term distributions for containment bypass at Surry.

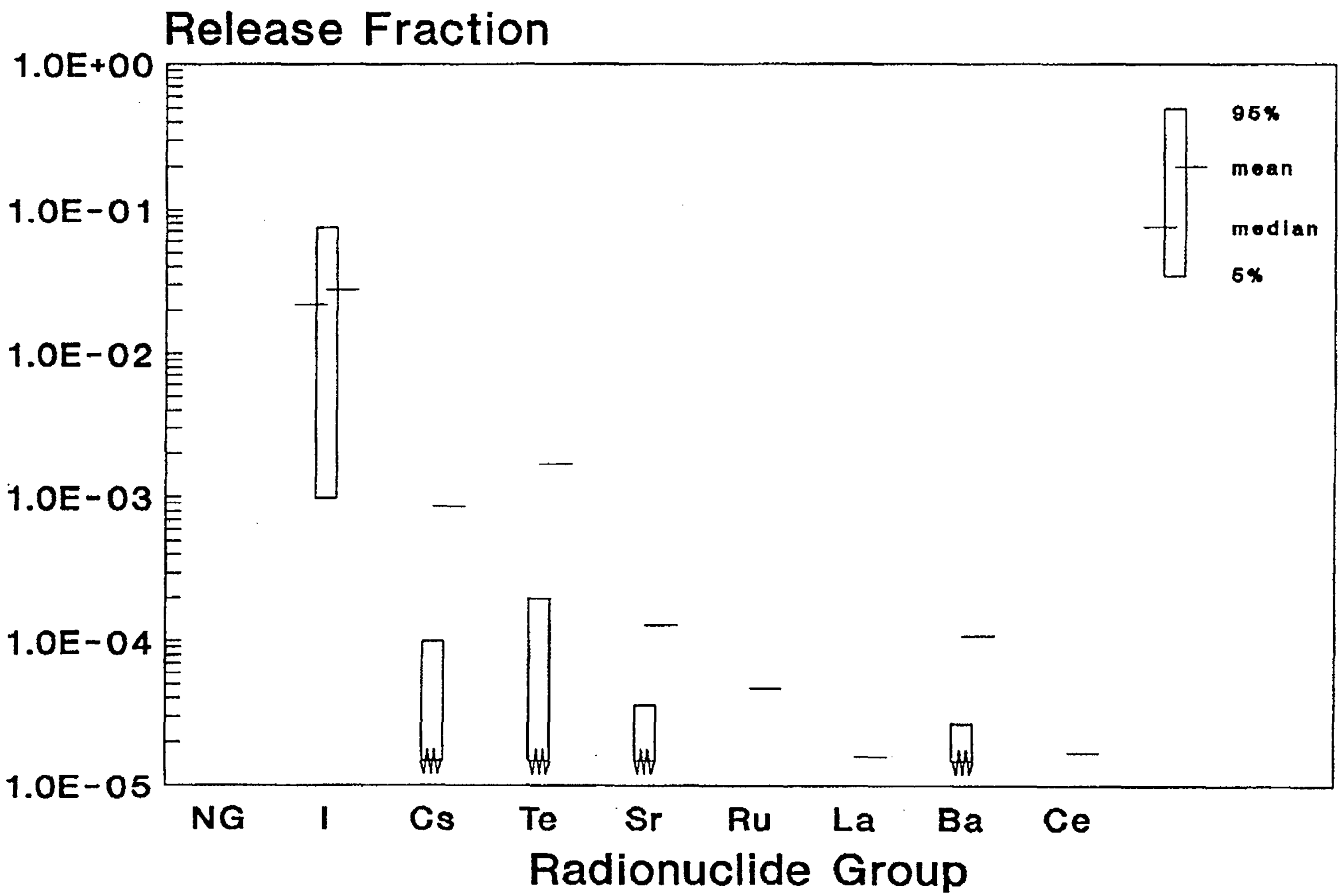


Figure 3.8 Source term distributions for late containment failure at Surry.

Sprays are not always effective in reducing the source term, however. The risk-dominant containment bypass sequences are largely unaffected by operation of the spray systems. Early containment failure scenarios involving high-pressure melt ejection have a component of the release that occurs almost simultaneously with containment failure, for which the sprays would not be effective.

In addition to removing aerosols from the atmosphere, containment sprays are an important source of water to the reactor cavity at Surry, which is otherwise dry. A coolable debris bed can be established in the cavity, preventing interactions between the hot core and concrete. If a coolable debris bed is not formed, a pool of water overlaying the hot core as it attacks concrete can effectively mitigate the release of radioactive material to the containment from this interaction.

2. **Cavity Configuration**

   Water collecting on the floor of the Surry containment cannot flow into the reactor cavity. As a result, the cavity will be dry at the time of vessel meltthrough unless the containment spray system has operated. As discussed earlier, water in the cavity can have a substantial effect on mitigating or eliminating the release of radioactive material from the molten core-concrete interaction.

3.5 Offsite Consequence Results

Figures 3.9 and 3.10 display the frequency distributions in the form of graphical plots of complementary cumulative distribution functions (CCDFs) of four offsite consequence measures—early fatalities, latent cancer fatalities, and the 50-mile and entire site region population exposures (in person-rems). The CCDFs in Figures 3.9 and 3.10 include contributions from all source terms associated with reactor accidents caused by the internal initiating events and fire, respectively. Four CCDFs, namely, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs, are shown for each consequence measure.

Surry plant-specific and site-specific parameters were used in the consequence analysis for these CCDFs. The plant-specific parameters included source terms and their frequencies, the licensed thermal power (2441 MWt) of the reactor, and the approximate physical dimensions of the power plant building complex. The site-specific parameters included exclusion area radius (520 meters), meteorological data for 1 full year collected at the site meteorological tower, the site region population distribution based on the 1980 census data, topography (fraction of the area that is land—the remaining fraction is assumed to be water), land use, agricultural practice and productivity, and other economic data for up to 1,000 miles from the Surry plant.

The consequence estimates displayed in these figures have incorporated the benefits of the following protective measures: (1) evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), (2) early relocation of the remaining population only from the heavily contaminated areas both within and outside the 10-mile EPZ, and (3) decontamination, temporary interdiction, or condemnation of land, property, and foods contaminated above acceptable levels.

The population density within the Surry 10-mile EPZ is about 230 persons per square mile. The average delay time before evacuation (after a warning prior to radionuclide release) from the 10-mile EPZ and average effective evacuation speed used in the analyses were derived from information contained in a utility-sponsored Surry evacuation time estimate study (Ref. 3.13) and the NRC requirements for emergency planning.

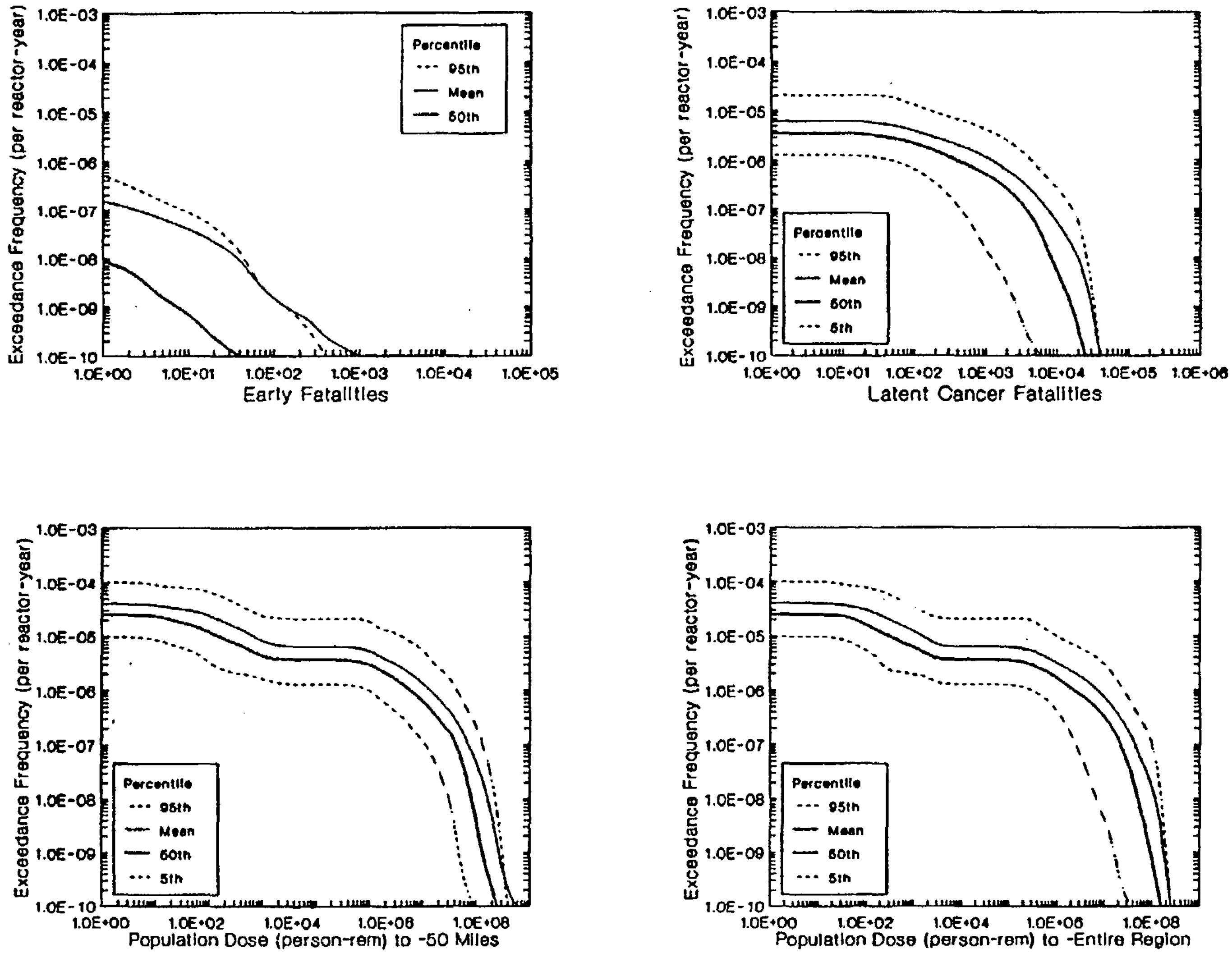
The results displayed in Figures 3.9 and 3.10 are discussed in Chapter 11.

3.6 Public Risk Estimates

3.6.1 Results of Public Risk Estimates

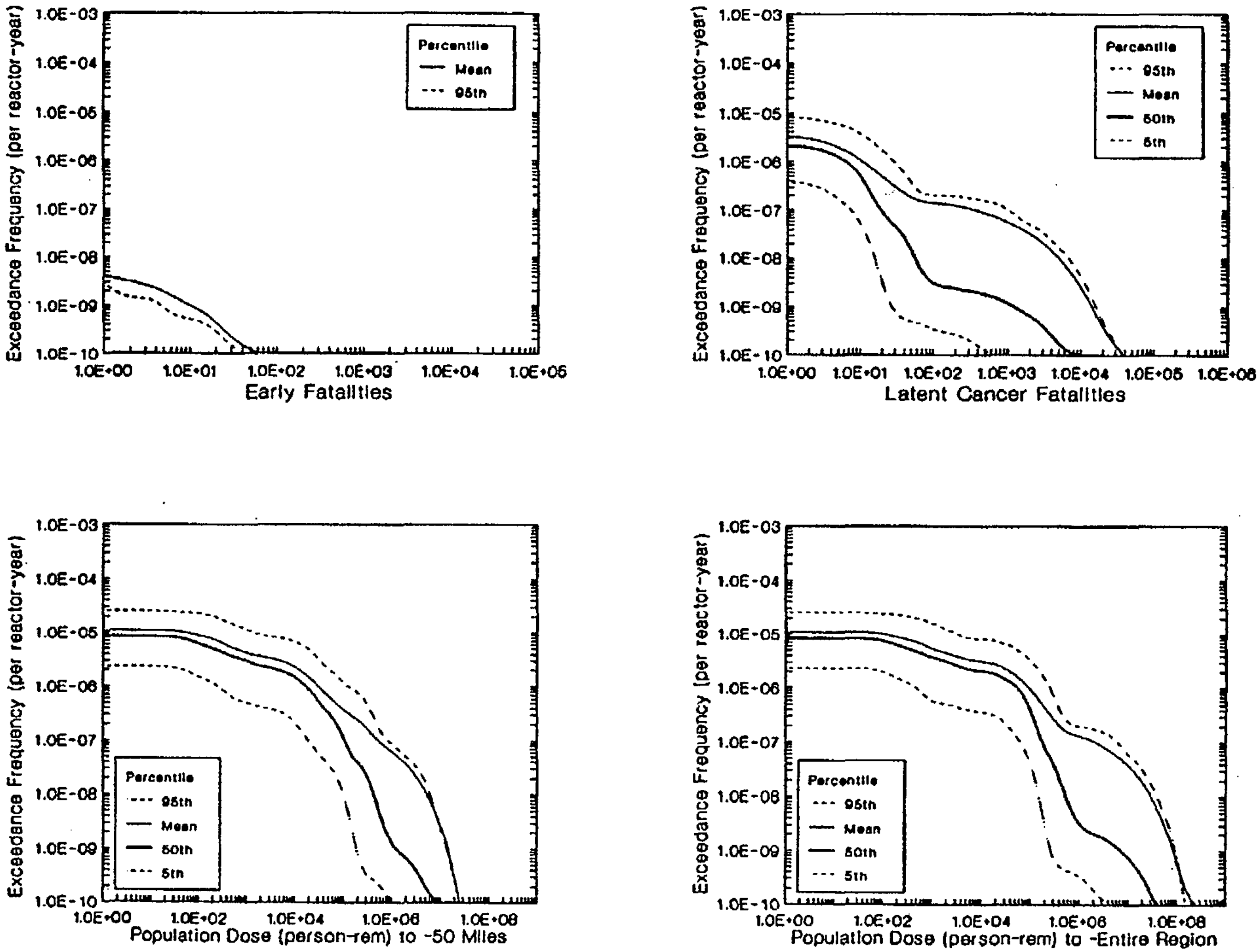
A detailed description of the results of the Surry risk analysis is provided in Reference 3.2. For this summary report, results are provided for the following measures of public risk:

- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Surry exclusion area boundary, and
- Individual latent cancer fatality risk in the population within 10 miles of the Surry site.



Note: As discussed in Reference 3.4, consequences at frequencies estimated 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 3.9 Frequency distributions of offsite consequence measures at Surry (internal initiators).



Note: As discussed in Reference 3.4, consequences at frequencies estimated 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 3.10 Frequency distributions of offsite consequence measures at Surry (fire initiators).

The first four of the above measures are commonly used measures in nuclear power plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 3.14).

3.6.1.1 Internally Initiated Accident Sequences

The results of the risk studies using the above measures are provided in Figures 3.11 through 3.13 for internally initiated accidents. The figures display the variabilities in mean risks estimated from the meteorology-averaged conditional mean values of the consequence measures. For the first two measures, the results of the first risk study of Surry, the Reactor Safety Study (Ref. 3.3), are also provided. As may be seen, both the early fatality risks and latent cancer fatality risks are lower than those of the Reactor Safety Study. The early fatality risk distribution, however, has a longer tail at the low end indicating a belief by the experts that there is a finite probability that risks may be orders of magnitude lower than those of the Reactor Safety Study. The risks of population dose within 50 miles of the plant site as well as within the entire site region are very low. Individual early fatality and latent cancer fatality risks are well below the NRC safety goals.

For the early and latent cancer fatality risk measures, the Reactor Safety Study values lie in the upper portions of the present risk range. This is because of the current estimates of better containment performance and source terms. The estimated probability of early containment failure in this study is significantly lower than the Reactor Safety Study values. The source term ranges of the Reactor Safety Study are comparable with the upper portions of the present study. The median core damage frequencies of the two studies, however, are about the same (2.3E-5 per reactor year for this study compared to 4.6E-5 per reactor year for the Reactor Safety Study). A more detailed comparison between results is provided in Chapters 12.

The risk results shown in Figure 3.11 have been analyzed to determine the relative contributions of plant damage states and containment-related accident progression bins to mean risk. The results of this analysis are provided in Figures 3.14 and 3.15. As may be seen, the mean early and latent cancer fatality risks of the Surry plant are principally due to accidents that bypass the containment building (interfacing-system LOCA (Event V) and steam generator tube ruptures). Details of these accident sequences are provided in Section 3.2.1.1. It should be noted from these discussions that for the steam generator tube rupture accident, if corrective or protective actions are taken (e.g., alternative sources of water are made available, emergency response is initiated*) before the refueling water storage tank water is totally depleted, i.e., within about a 10-hour period after start of the accident, risks from this accident may be substantially reduced.

3.6.1.2 Externally Initiated Accident Sequences

The Surry plant has been analyzed for two externally initiated accidents: earthquakes and fire (see Section 3.2.1.2). The fire risk analysis has been performed, including estimates of consequences and risk, while the seismic analysis has been conducted up to the containment performance (as discussed in Chapter 2). Sensitivity analyses of seismic risk at Surry are provided in Reference 3.2.

Results of fire risk analysis (variabilities in mean risks estimated from meteorology-averaged conditional mean values of the consequence measures) of Surry are shown in Figures 3.16 through 3.18 for the early fatality, latent cancer fatality, population dose (within 50 miles of the site and within the entire site region), and individual early and latent cancer fatality risks. As can be seen, the risks from fire are substantially lower than those from internally initiated events.

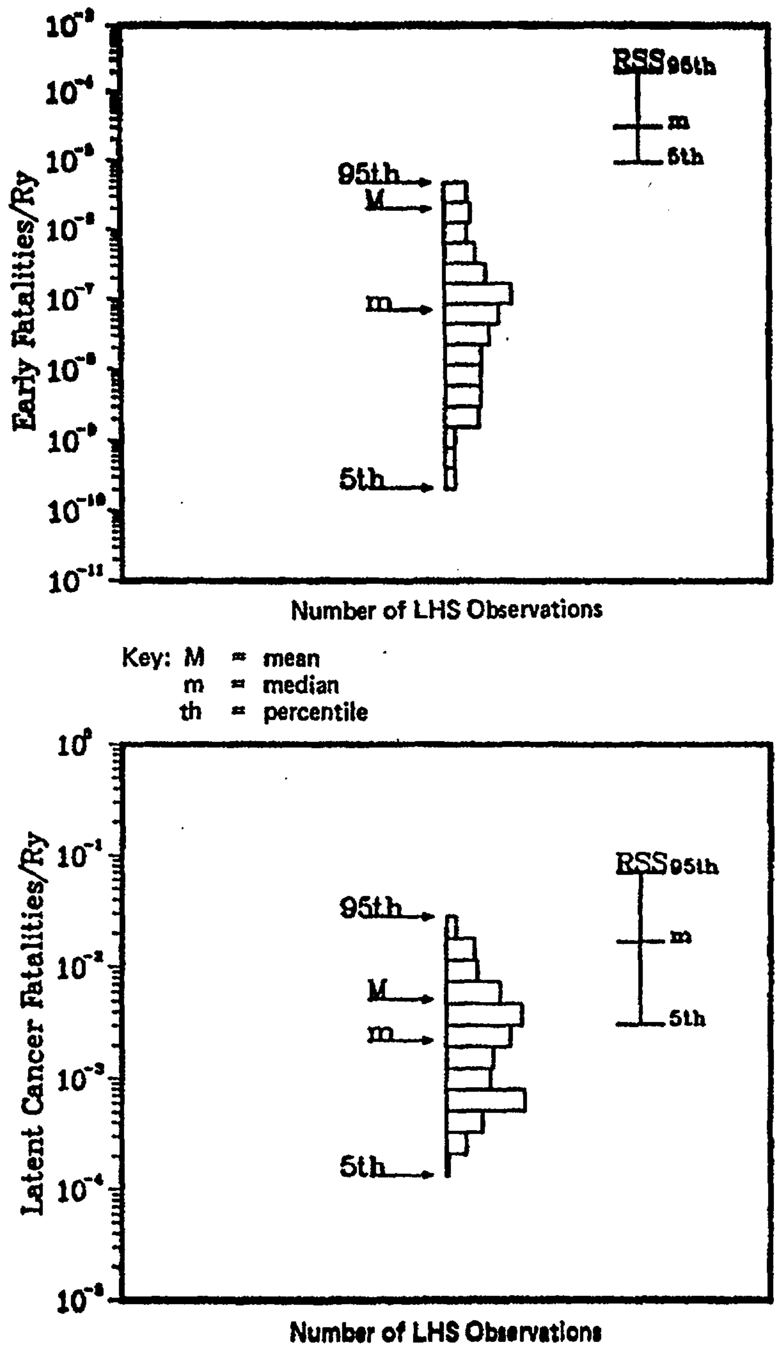
Major contributors to early and latent cancer fatality risks are shown in Figure 3.19. (Note that there are no bypass initiating events in the fire plant damage state.) The most risk-important sequence is a fire in the emergency switchgear room that leads to loss of ac power throughout the station. The principal risk-important accident progression bin is early containment failure with the reactor coolant system at high pressure (>200 psia) at vessel breach leading to direct containment heating.

Additional discussion of risk perspectives (for all five plants studied) is provided in Chapter 12.

3.6.2 Important Plant Characteristics (Risk)

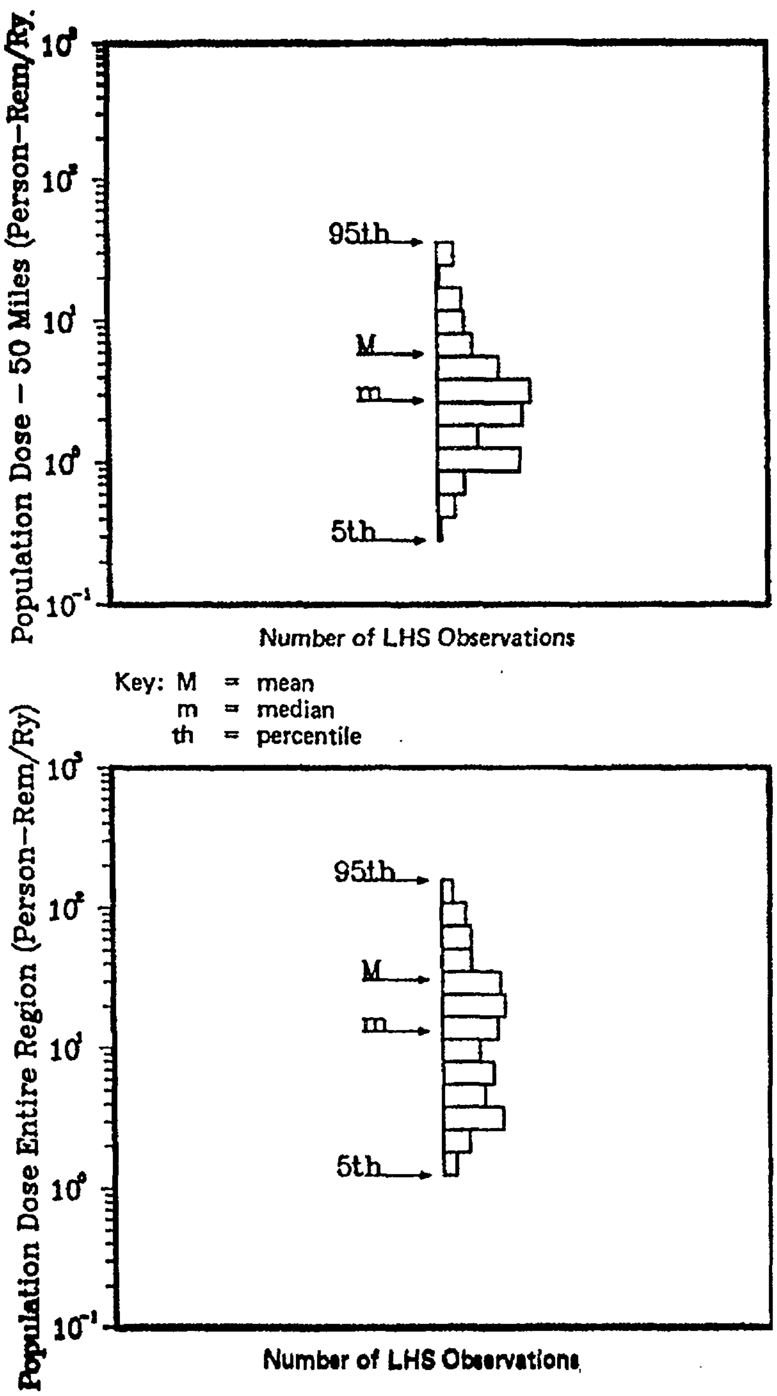
The plant characteristics discussed in Section 3.2.2 that were important in the analysis of core damage frequency were primarily related to the station blackout accident sequences and have not been found to be important in the risk analysis.

*See Chapter 11 for sensitivity of offsite consequences to alternative modes of emergency response.



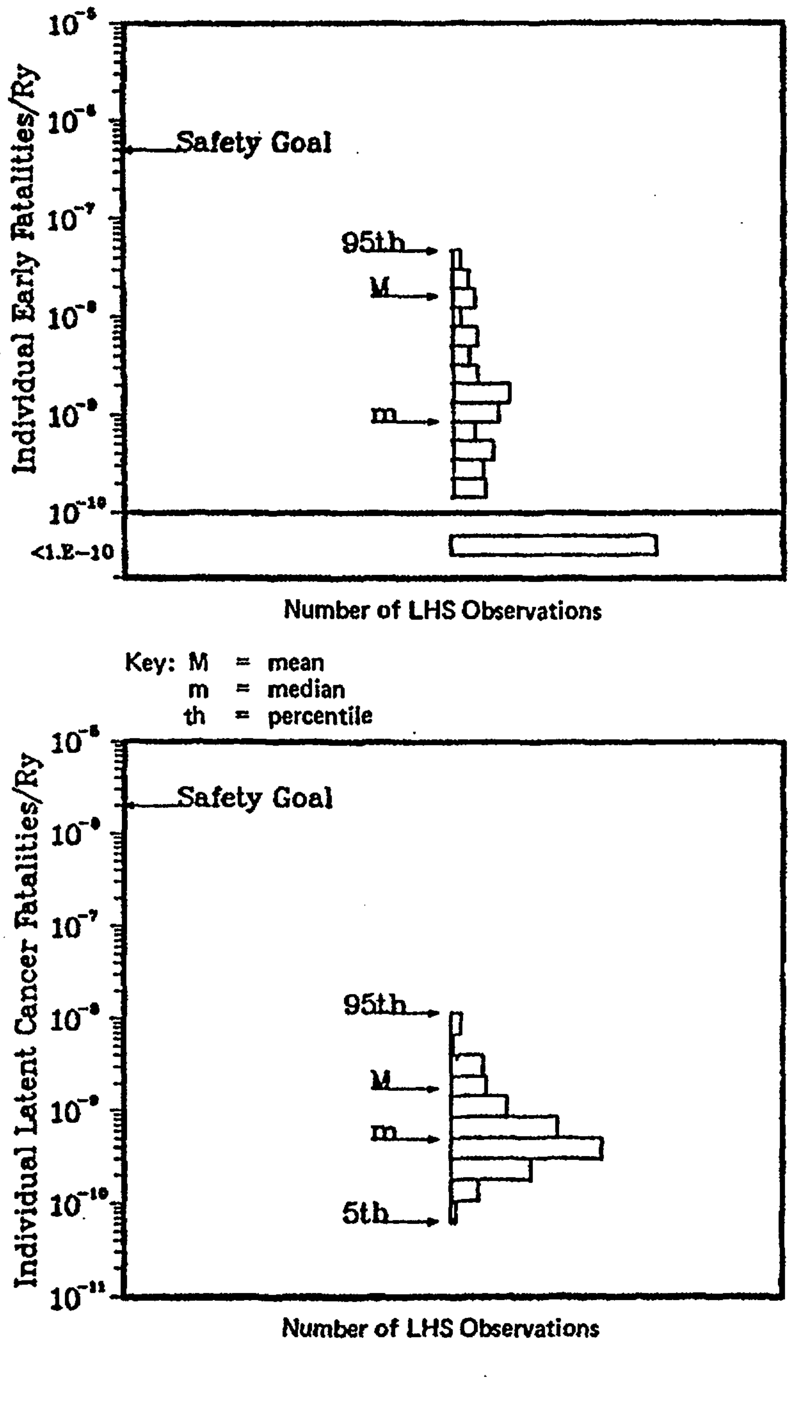
Note: As discussed in Reference 3.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 3.11 Early and latent cancer fatality risks at Surry (internal initiators).



Note: As discussed in Reference 3.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 3.12 Population dose risks at Surry (internal initiators).



Note: As discussed in Reference 3.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 3.13 Individual early and latent cancer fatality risks at Surry (internal initiators).

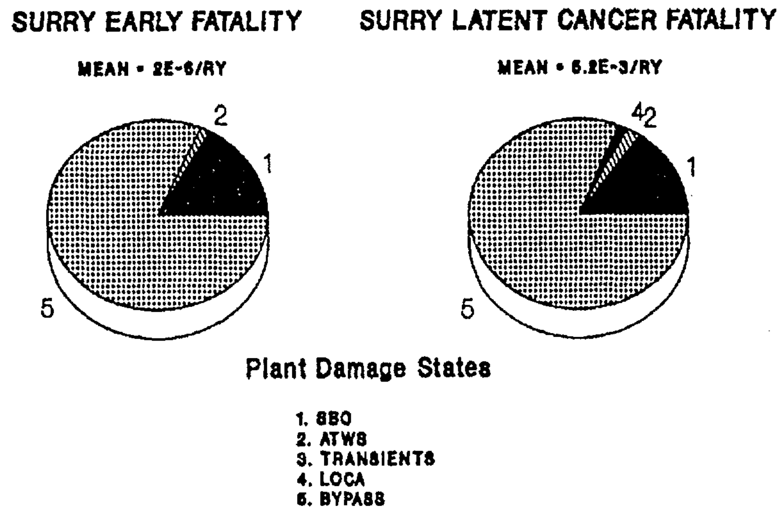


Figure 3.14 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Surry (internal initiators).

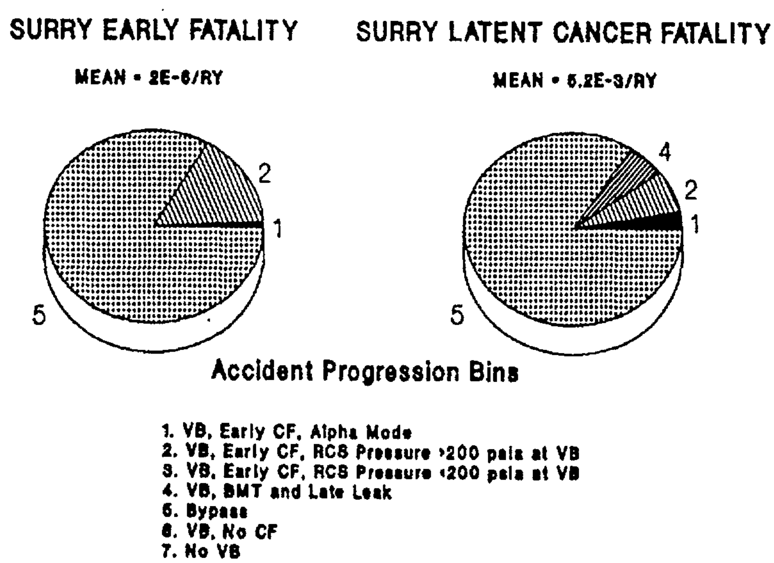
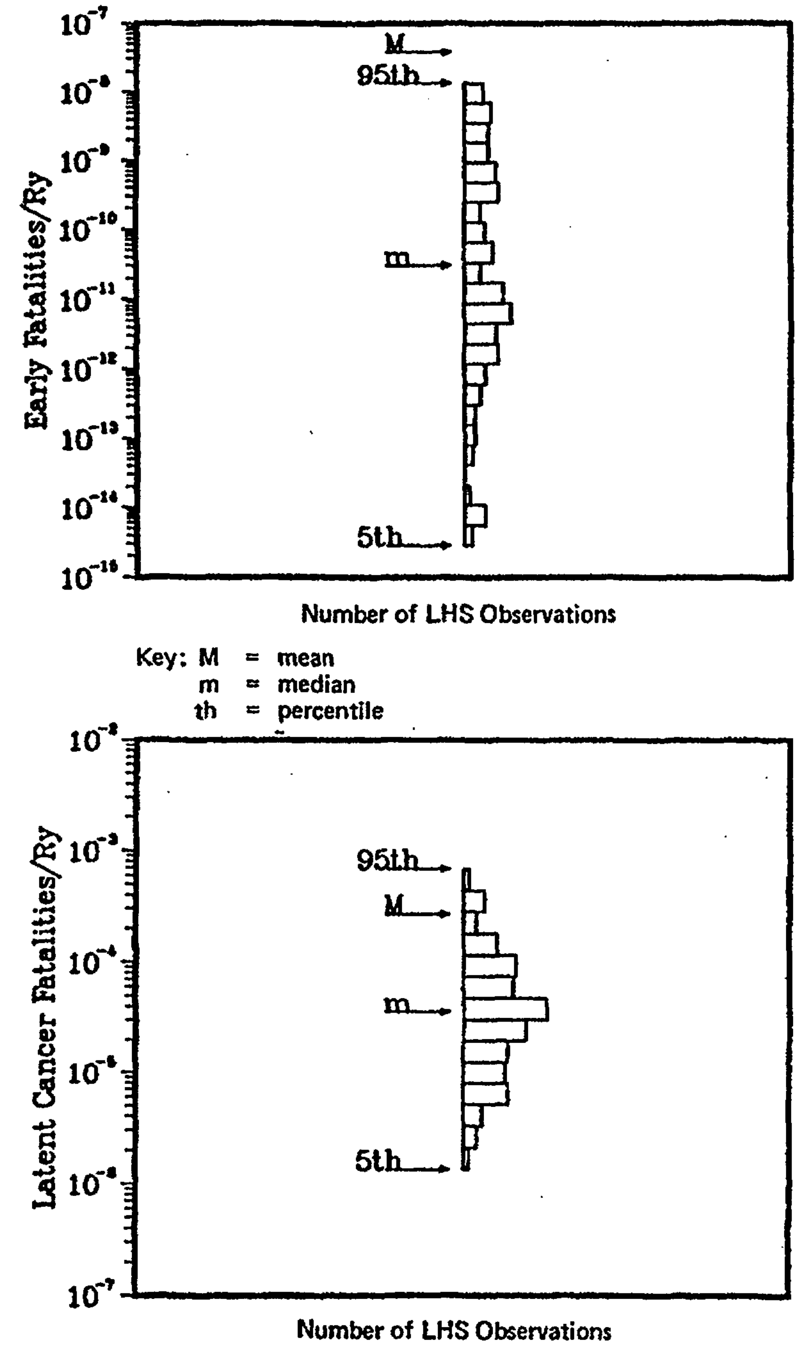
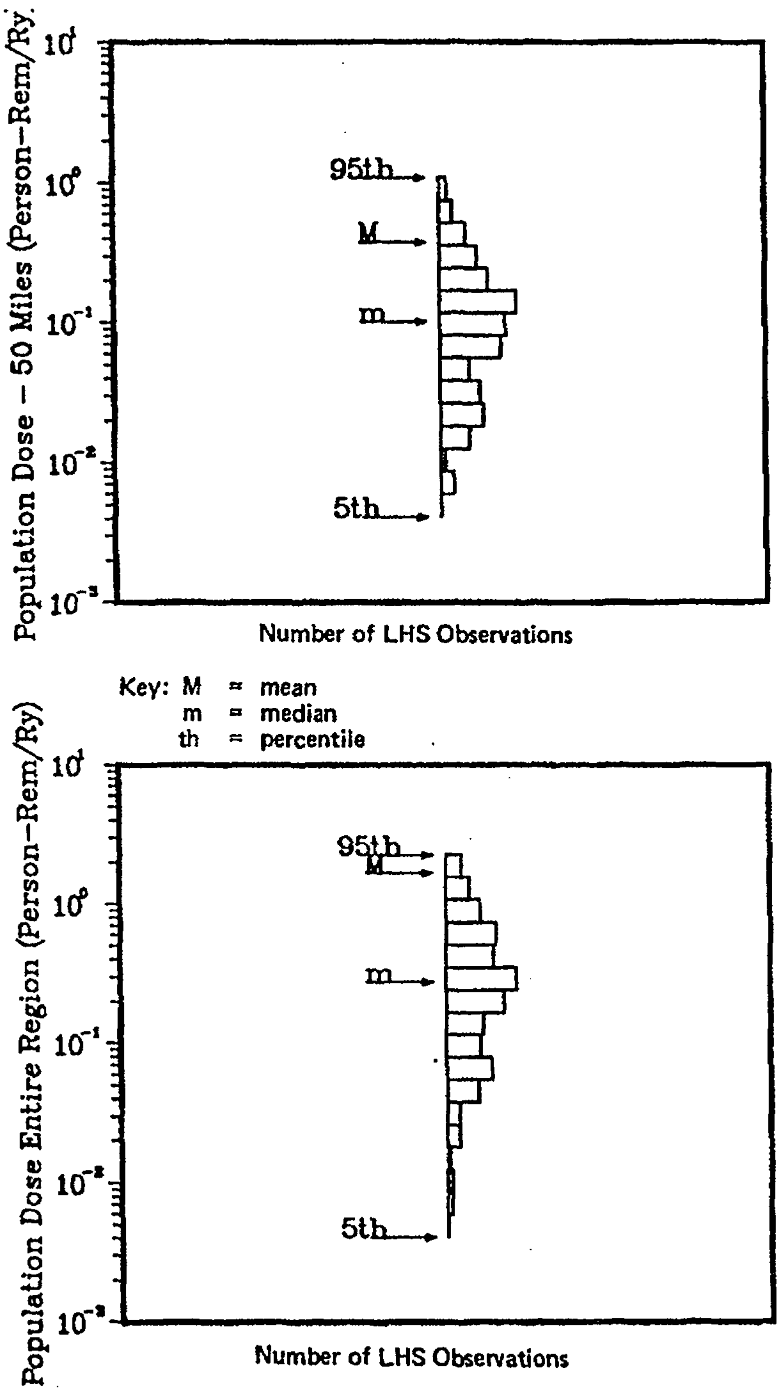


Figure 3.15 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Surry (internal initiators).



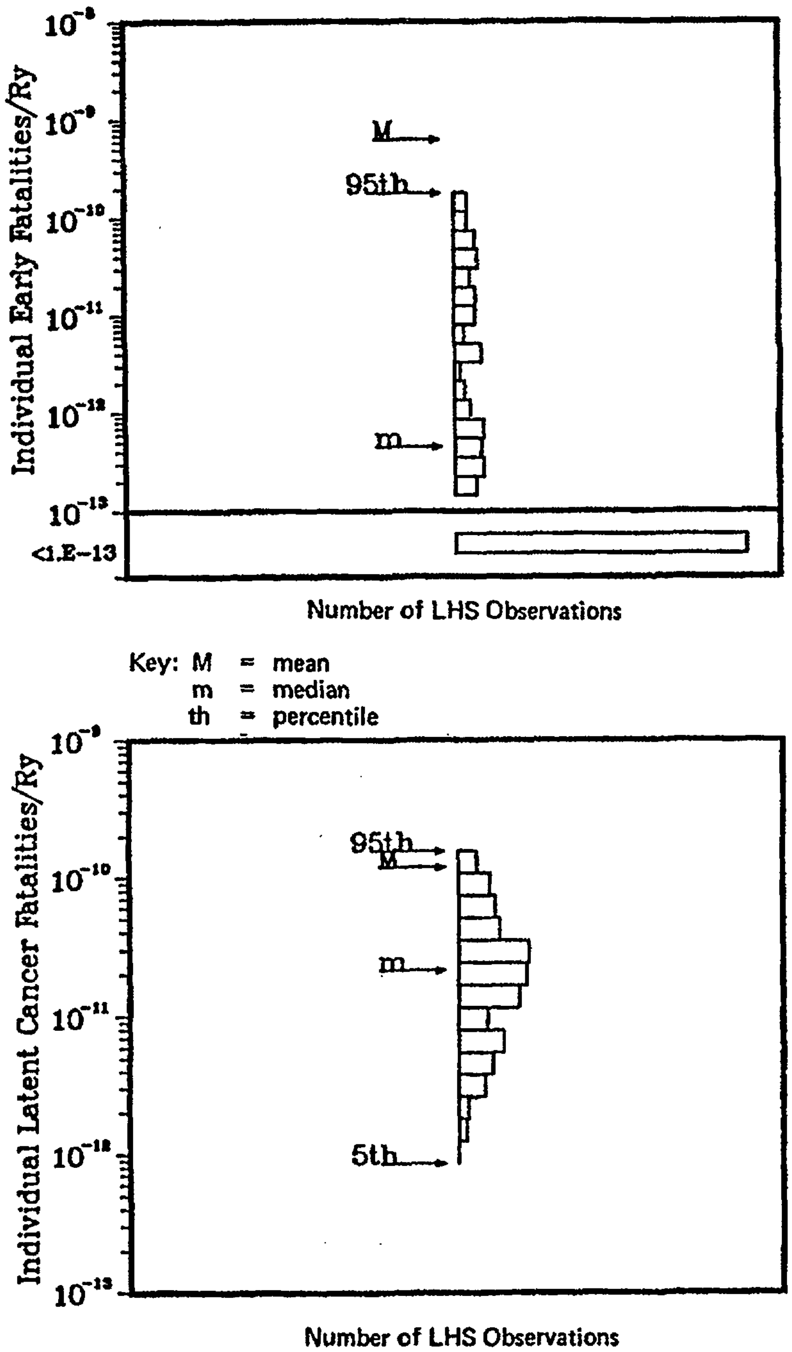
Note: As discussed in Reference 3.4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 3.16 Early and latent cancer fatality risks at Surry (fire initiators).



Note: As discussed in Reference 3.4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 3.17 Population dose risks at Surry (fire initiators).



Note: As discussed in Reference 3.4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 3.18 Individual early and latent cancer fatality risks at Surry (fire initiators).

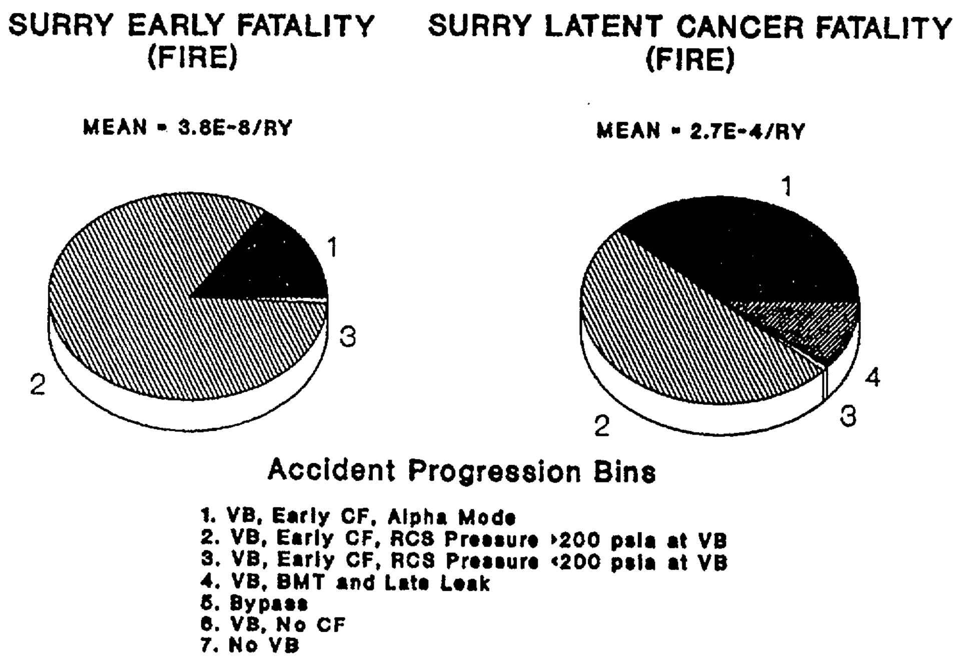


Figure 3.19 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Surry (fire initiators).

That is, because of the high consequences of the containment bypass sequences and low frequency of early containment failures, Event V and SGTR were more important risk contributors in the Surry analysis. The following general observations can be made from the risk results:

- The Surry containment appears robust, with a low conditional probability of failure (early or late). This is responsible, to a large extent, for the low risk estimates for the Surry plant. (In comparison with other plants studied in this report, risks for Surry are relatively high; but, in the absolute sense, these risks are very low and are well below NRC safety goals, as can be seen in Chapter 12.)

- Early fatality risk is dominated by bypass accidents, primarily from an interfacing-system LOCA. This accident leads to rapid core damage; the radioactive release is assessed to take place before evacuation is complete. Steam generator tube rupture accident sequences with stuck-open SRVs result in very late core melt; evacuation is assessed to be complete before the release is estimated to occur.

- The configuration of low-pressure piping outside the containment leads to a high probability that the release from an interfacing-system LOCA would be partially scrubbed by overlaying water. If the release were to take place without such scrubbing, the contribution to early fatality risk would be higher.

- Depressurization of the reactor coolant system by deliberate or inadvertent means plays an important role in the progression of severe accidents at Surry in that it decreases the probability of containment failure by high-pressure melt ejection and direct containment heating.

- Risks from accidents initiated by fires are dominated by early containment failures and are estimated to be much lower than those from internally initiated accidents.

REFERENCES FOR CHAPTER 3

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

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

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

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

3.5 A. Kolaczkowski and A. Payne, "Station Blackout Accident Analyses," Sandia National Laboratories, NUREG/CR-3226, SAND82-2450, May 1983.

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

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

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

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

3.10 USNRC, "Reactor Risk Reference Document," NUREG-1150, Vols. 1-3, Draft for Comment, February 1987.

3.11 G. D. Kaiser, "The Implications of Reduced Source Terms for Ex-Plant Consequence Modeling," Executive Conference on the Ramifications of the Source Term (Charleston, SC), March 12, 1985.

3.12 R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios—PWR, Subatmospheric Containment Design," Battelle Columbus Division, NUREG/CR-4624, Vol. 3, BMI-2139, July 1986.

3.13 P. R. C. Voorhees, "Surry Nuclear Power Station Estimation of Evacuation Times," prepared for Virginia Power Company, March 1981.

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

4. PEACH BOTTOM PLANT RESULTS

4.1 Summary Design Information

The Peach Bottom Atomic Power Station is a General Electric boiling water reactor (BWR-4) unit of 1065 MWe capacity housed in a Mark I containment constructed by Bechtel Corporation. Peach Bottom Unit 2, analyzed in this study, began commercial operation in July 1974 under the operation of Philadelphia Electric Company (PECo). Some important system design features of the Peach Bottom plant are described in Table 4.1. A general plant schematic is provided in Figure 4.1.

This chapter provides a summary of the results obtained in the detailed risk analyses underlying this report (Refs. 4.1 and 4.2). A discussion of perspectives with respect to these results is provided in Chapters 8 through 12.

4.2 Core Damage Frequency Estimates

4.2.1 Summary of Core Damage Frequency Estimates

The core damage frequency and risk analyses performed for this study considered accidents initiated by both internal and external events (Refs. 4.1 and 4.2). The core damage frequency results obtained from internal events are displayed in graphical form as a histogram in Figure 4.2 (Section 2.2.2 discusses histogram development). The core damage frequency results obtained from internal and external events are provided in tabular form in Table 4.2.

The Peach Bottom plant was previously analyzed in the Reactor Safety Study (RSS) (Ref. 4.3). The RSS calculated a total point estimate core damage frequency from internal events of 2.6E-5 per year. This study calculated a total median core damage frequency from internal events of 1.9E-6 per year with a corresponding mean value of 4.5E-6. For a detailed discussion of, and insights into, the comparison between this study and the RSS, see Chapter 8.

4.2.1.1 Internally Initiated Accident Sequences

A detailed description of accident sequences important at the Peach Bottom plant is provided in Reference 4.1. For this summary report, the accident sequences described in that report have been grouped into four summary plant damage states. These are:

- Station blackout,
- Anticipated transient without scram (ATWS),
- Loss-of-coolant accidents (LOCAs), and
- Transients other than station blackout and ATWS.

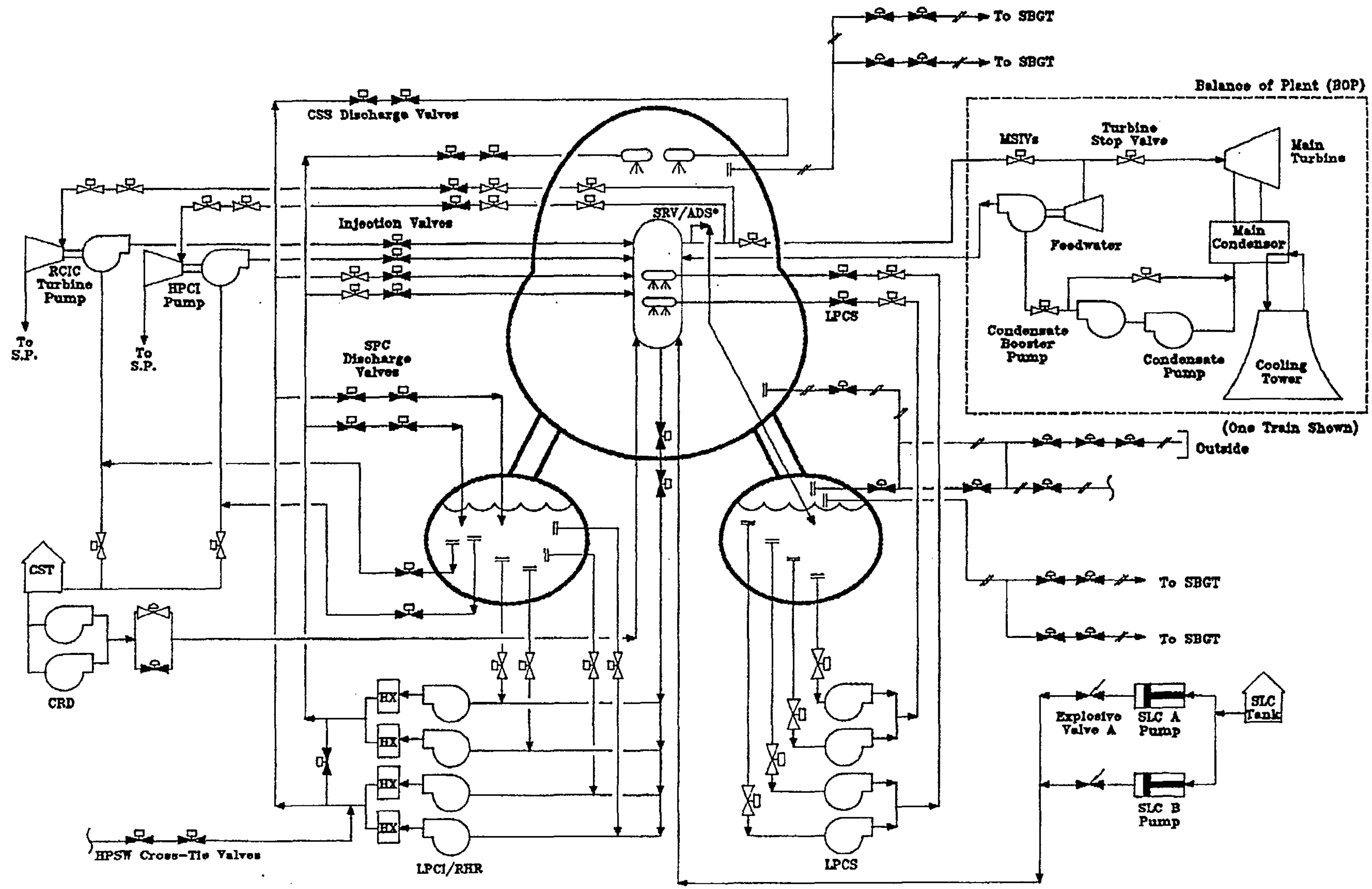
The relative contributions of these groups to mean internal-event core damage frequency at Peach Bottom are shown in Figure 4.3. From Figure 4.3, it may be seen that station blackout sequences as a class are the largest contributor to mean core damage frequency. It should be noted that the plant configuration (as analyzed for this study) does not reflect modifications that may be required in response to the station blackout rule.

Within the general class of station blackout accidents, the more probable combinations of failures leading to core damage are:

- Loss of onsite and offsite ac power results in the loss of all core cooling systems (except high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC), both of which are ac independent in the short term) and all containment heat removal systems. HPCI or RCIC (or both) systems function but ultimately fail at approximately 10 hours because of battery depletion or other late failure modes (e.g., loss of room cooling effects). Core damage results in approximately 13 hours as a result of coolant boiloff.
- Loss of offsite power occurs followed by a subsequent failure of all onsite ac power. The diesel generators fail to start because of failure of all the vital batteries. Without ac and dc power, all core cooling systems (including HPCI and RCIC) and all containment heat removal systems fail. Core damage begins in approximately 1 hour as a result of coolant boiloff.
- Loss of offsite power occurs followed by a subsequent failure of a safety relief valve to reclose. All onsite ac power fails because the diesel generators fail to start and run from a variety of faults. The loss of all ac power fails most of the core cooling systems and all the containment heat removal systems. HPCI and RCIC (which are ac independent) are available and either or both initially function

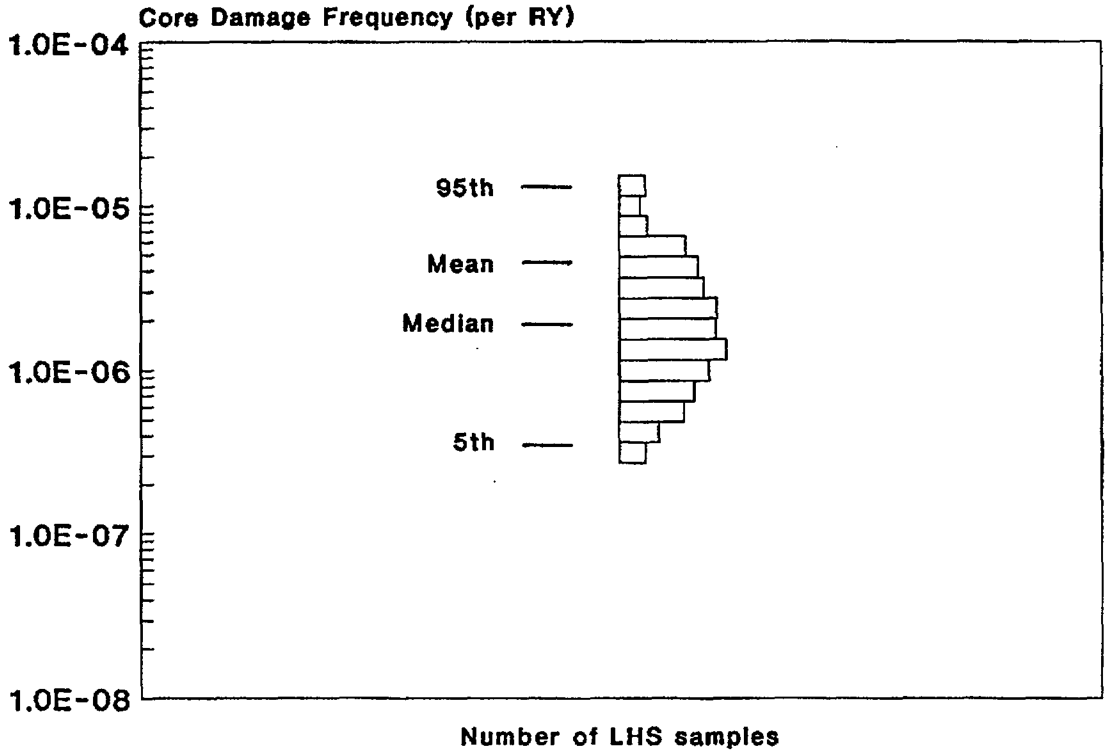
Table 4.1 Summary of design features: Peach Bottom Unit 2.

1. Coolant Injection Systems	a.	High-pressure coolant injection system provides coolant to the reactor vessel during accidents in which system pressure remains high, with 1 train and 1 turbine-driven pump.
	b.	Reactor core isolation cooling system provides coolant to the reactor vessel during accidents in which system pressure remains high, with 1 train and 1 turbine-driven pump.
	c.	Low-pressure core spray system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with 2 trains and 4 motor-driven pumps.
	d.	Low-pressure coolant injection system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with 2 trains and 4 pumps.
	e.	High-pressure service water crosstie system provides coolant makeup source to the reactor vessel during accidents in which normal sources of emergency injection have failed (low RPV pressure), with 1 train and 4 pumps for crosstie.
	f.	Control rod drive system provides backup source of high-pressure injection, with 2 pumps/210 gpm (total)/1,100 psia.
	g.	Automatic depressurization system for depressurizing the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant to the reactor vessel: 5 ADS relief valves/capacity 820,000 lb/hr. In addition, there are 6 non-ADS relief valves.
2. Key Support Systems	a.	dc power with up to approximately 10–12-hour station batteries.
	b.	Emergency ac power from 4 diesel generators shared between 2 units.
	c.	Emergency service water provides cooling water to safety systems and components shared by 2 units.
3. Heat Removal Systems	a.	Residual heat removal/suppression pool cooling system to remove heat from the suppression pool during accidents, with 2 trains and 4 pumps.
	b.	Residual heat removal/shutdown cooling system to remove decay heat during accidents in which reactor vessel integrity is maintained and reactor at low pressure, with 2 trains and 4 pumps.
	c.	Residual heat removal/containment spray system to suppress pressure and remove decay heat in the containment during accidents, with 2 trains and 4 pumps.
4. Reactivity Control Systems	a.	Control rods.
	b.	Standby liquid control system, with 2 parallel positive displacement pumps rated at 43 gpm per pump, but each with 86 gpm equivalent because of the use of enriched boron.
5. Containment Structure	a.	BWR Mark I.
	b.	0.32 million cubic feet.
	c.	56 psig design pressure.
6. Containment Systems	a.	Containment venting—drywell and wetwell vents used when suppression pool cooling and containment sprays have failed to reduce primary containment pressure.



*Typical arrangement (5 ADS SRVs and 6 non-ADS SRVs)

Figure 4.1 Peach Bottom plant schematic.



Note: As discussed in Reference 4.4, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

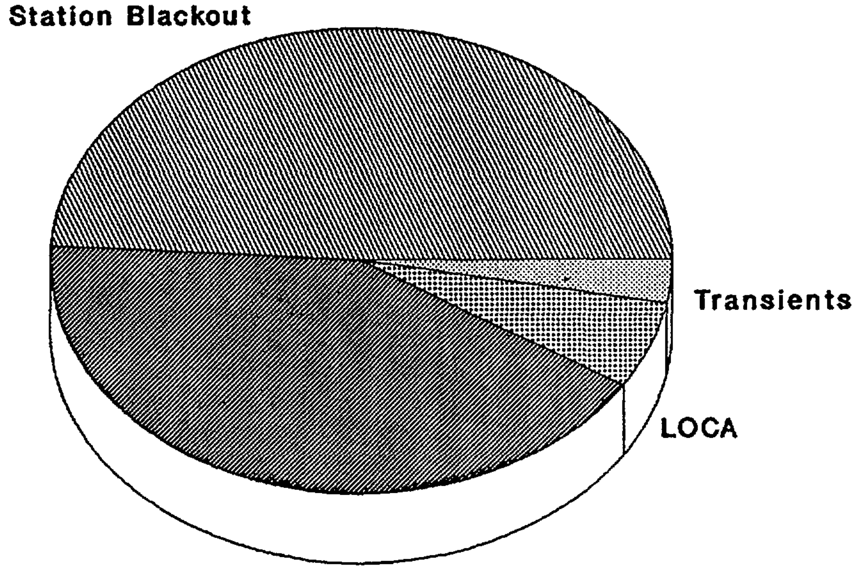
Figure 4.2 Internal core damage frequency results at Peach Bottom.

Table 4.2 Summary of core damage frequency results: Peach Bottom.*

	5%	Median	Mean	95%
Internal Events	3.5E-7	1.9E-6	4.5E-6	1.3E-5
Station Blackout	8.3E-8	6.2E-7	2.2E-6	6.0E-6
ATWS	3.1E-8	4.4E-7	1.9E-6	6.6E-6
LOCA	2.5E-9	4.4E-8	2.6E-7	7.8E-7
Transient	6.1E-10	1.9E-8	1.4E-7	4.7E-7
External Events**				
Seismic (LLNL)	5.3E-8	4.4E-6	7.7E-5	2.7E-4
Seismic (EPRI)	2.3E-8	7.1E-7	3.1E-6	1.3E-5
Fire	1.1E-6	1.2E-5	2.0E-5	6.4E-5

*Note: As discussed in Reference 4.4, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

**See "Externally Initiated Accident Sequences" in Section 4.2.1.2 for discussion.



Total Mean Core Damage Frequency: 4.5E-6

Figure 4.3 Contributors to mean core damage frequency from internal events at Peach Bottom.

but ultimately fail at approximately 10 hours because of battery depletion or other late failure modes (e.g., loss of room cooling effects). Core damage results in 10 to 13 hours as a result of coolant boiloff.

Within the general class of anticipated transient without scram accidents, the more probable combinations of failures leading to core damage are:

- Transient (e.g., loss of feedwater) occurs followed by a failure to trip the reactor because of mechanical faults in the reactor protection system (RPS) and closure of the main steam isolation valves (MSIVs). The standby liquid control system (SLCS) does not function (primarily because of operator failure to actuate), but the HPCI does start. However, increased suppression pool temperatures fail the HPCI. Low-pressure coolant injection (LPCI) is unavailable and all core cooling is lost. Core damage occurs in approximately 20 minutes to several hours, depending on the time at which the LPCI fails because of different LPCI failure modes.

- Transient occurs followed by a failure to scram (mechanical faults in the RPS) and closure of the MSIVs. SLCS is initiated but HPCI fails to function because of random faults. The operator fails to depressurize after HPCI failure and therefore the low-pressure core cooling systems cannot inject. Core damage occurs in approximately 15 minutes.

Within the general class of LOCAs, the more probable combination of failures leading to core damage is:

- A medium-size LOCA (i.e., break size of approximately 0.004 to 0.1 ft^2) occurs. HPCI works initially but fails because of low steam pressure. The low-pressure core cooling systems fail to actuate primarily because of miscalibration faults of the pressure sensors, which do not "permit" the injection valves to open. All core cooling is lost and core damage occurs in approximately 1 to 2 hours following the initiating event.

4.2.1.2 Externally Initiated Accident Sequences

A detailed description of accident sequences initiated by external events important at the Peach Bottom plant is provided in Part 3 of Reference 4.1. The accident sequences described in that reference have been grouped into two main types for this study. These are:

- Seismic, and
- Fire.

A scoping study has also been performed to assess the potential effects of other externally initiated accidents (Ref. 4.1, Part 3). This analysis indicated that the following external-event sources could be excluded based on the low frequency of the initiating event:

- Aircraft crashes,
- Hurricanes,
- Tornados,
- Internal flooding, and
- External flooding.

1. Seismic Accident Frequency Analysis

The relative contribution of classes of seismically and fire-initiated accidents to the total mean frequency of externally initiated core damage accidents is provided in Figure 4.4. As may be seen, the dominant seismic scenarios are transient (38%) and LOCA sequences (27%) with the other contributors being substantially less. For these two seismic accident initiators, the more probable combinations of system failures are:

- The transient sequence results from seismically induced failure of ceramic insulators in the switchyard causing loss of offsite power (LOSP) in conjunction with loss of onsite ac power. This latter results primarily from loss of the emergency service water (ESW) system (which provides the jacket cooling for the emergency diesel generators) and/or direct failures of 4 kV buses or the diesel generators themselves. The vast majority of failures are seismically induced.
- The large LOCA sequence is initiated by postulated seismically induced failures of the supports on the recirculation pumps. Core damage results from this initiator in conjunction with seismically induced failures of the low-pressure injection systems. The latter requires ac power, and the dominant sources of failure of onsite ac power are the ESW or emergency diesel generator seismic failures as discussed above.

As discussed in Chapter 2, the seismic analysis in this report made use of two sets of hazard curves from Lawrence Livermore National Laboratory (LLNL) (Ref. 4.5) and the Electric Power Research Institute (EPRI) (Ref. 4.6). The differences between the seismic core damage frequencies shown in Table 4.2 for the LLNL and the EPRI cases are due entirely to the differences between the two sets of hazard curves. That is, the system models, failure rates, and success logic were identical for both estimates.

The seismic hazard associated with the curves developed by EPRI was significantly less than that of the LLNL curves. Differences between these curves result primarily from differences between the methodology and assumptions used to develop the hazard curves. In the LLNL program, considerable emphasis was placed on a wide range of uncertainty in the ground-motion attenuation models, while a relatively coarse set of seismic tectonic provinces was used in characterizing each site. By contrast, in the EPRI program considerable emphasis was placed on a fine zonation for the tectonic provinces, and very little uncertainty in the ground-motion attenuation was considered. In any case, it is the difference between the two sets of hazard curves that causes the differences between the numeric estimates in Table 4.2.

2. Fire Accident Frequency Analysis

The fire-initiated accident frequency analyses performed for this report considered the impact of fires beginning in a variety of separate locations within the plant. Those locations found to be most important were:

- Emergency switchgear rooms,
- Control room, and
- Cable-spreading room.

No other plant locations contributed more than 1.0E–8 per year to the core damage frequency.

Fires in the cable-spreading room are assumed to require manual plant trip and to fail the high-pressure injection and depressurization systems, namely: high pressure core injection (HPCI), reactor core isolation cooling (RCIC), control rod drive (CRD), and automatic depressurization systems (ADS). In each case, the failure occurs because of fire damage to the control cables.

Fires in the emergency switchgear rooms failed offsite power and in some instances portions of the emergency service water system, and core damage occurs because of a station blackout sequence involving additional random failures of the emergency service water system (which provides jacket cooling to the diesel generators).

Finally, two fire scenarios were identified for the control room, both of which involve manual plant trip and abandonment of the control room. One scenario involved random failure of the RCIC system and a reasonable probability that the operators fail to recover the plant using HPCI or ADS in conjunction with LPCI from the remote shutdown panel. The other scenario failed the RCIC system because of a fire in its control cabinet but allowed for recovery from the remote shutdown panel.

4.2.2 Important Plant Characteristics (Core Damage Frequency)

Characteristics of the Peach Bottom plant design and operation that have been found to be important in the analysis of core damage frequency include:

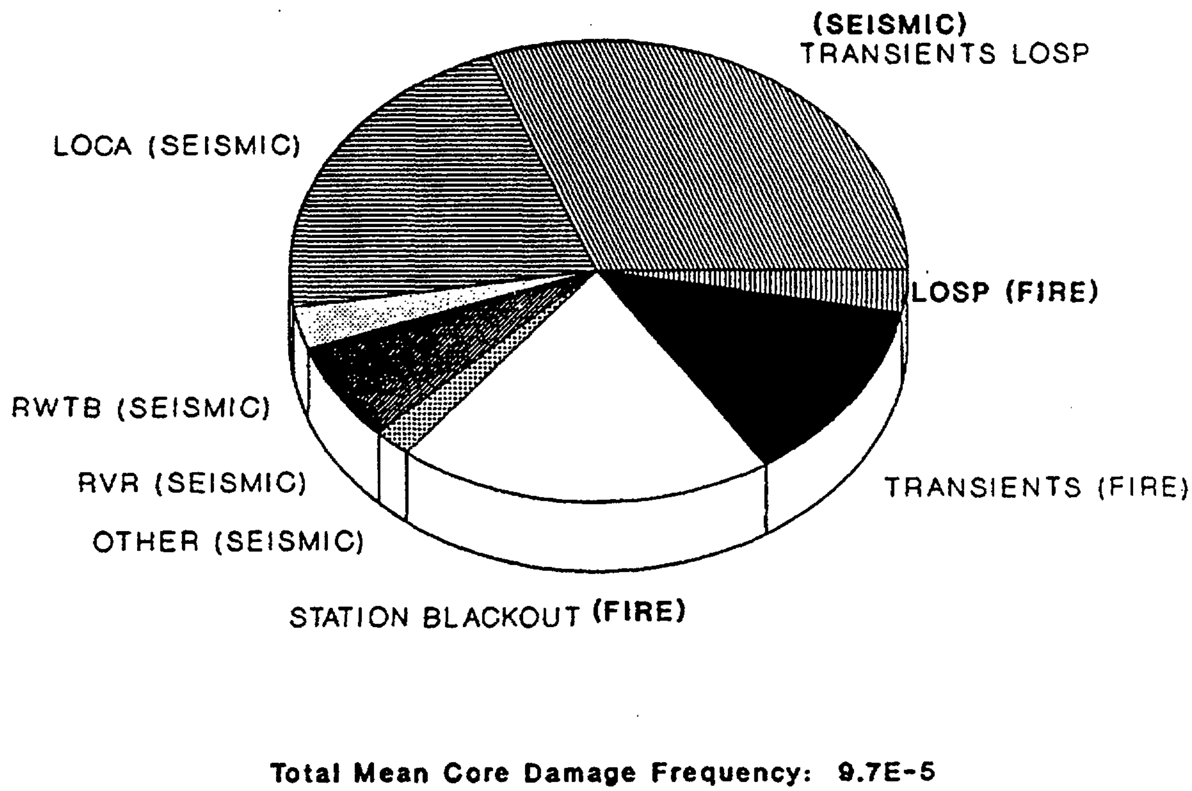


Figure 4.4 Contributors to mean core damage frequency from external events (LLNL hazard curve) at Peach Bottom.

1. **High-Pressure Service Water System Crosstie**

   The high-pressure service water (HPSW) system, if the reactor vessel has been depressurized, can inject raw water to the reactor vessel via the residual heat removal injection lines. Most components of HPSW are located outside the reactor building and thus are not affected by any potential severe reactor building environment that could cause other injection systems to fail in some accidents. Therefore, this system offers diversity, as well as redundancy, and affects many different types of sequences. The Peach Bottom operators are trained to use this system and can do so from the control room. An extensive cleanup program would, however, be required after the system is initiated.

2. **Redundancy and Diversity of Water Supply Systems**

   At Peach Bottom, there are many redundant and diverse systems to provide water to the reactor vessel. They include:

High-pressure core injection (HPCI) with 1 pump;

Reactor core isolation cooling (RCIC) with 1 pump;

Control rod drive (CRD) with 2 pumps (both pumps required);

Low-pressure core spray (LPCS) with 4 pumps;

Low-pressure core injection (LPCI) with 4 pumps;

Condensate with 3 pumps; and

High-pressure service water (HPSW) with 4 pumps.

Because of this redundancy of systems, LOCAs and transients other than station blackout and ATWS are small contributors to the core damage frequency.

CRD, condensate, and HPSW pumps are located outside the reactor building (generally away from potentially severe environments) and represent excellent secondary high- and low-pressure coolant systems if normal injection systems fail. These systems are not available during station blackout.

3. **Redundancy and Diversity of Heat Removal Systems**

At Peach Bottom, there are several diverse means for heat removal. These systems are:

Main steam/feedwater system;

Suppression pool cooling mode of residual heat removal (RHR);

Shutdown cooling mode of RHR;

Containment spray system mode of RHR; and

Containment venting.

This diversity has greatly reduced the importance of transients with long-term loss of heat removal.

4. **Diesel Generators**

Peach Bottom is a two-unit site with four emergency diesels shared between the two units. One diesel can supply the necessary power for both units. DC power to start the diesels is supplied from vital dc station batteries. The four emergency diesels share a common service water system that provides oil cooling, jacket, and air cooling. The Peach Bottom emergency diesels historically have had a failure-to-start probability that is much better than the industry average, e.g., a factor of ~10 lower failure probability.

5. **Battery Capacity**

Philadelphia Electric Company (PECo) has performed analyses of the battery life based on the current station blackout procedures. PECo estimates that the station batteries at Peach Bottom are capable of lasting at least 12 hours in a station blackout. They have revised their station blackout procedure to include load shedding in order to ensure a longer period of injection and accident monitoring. The ability to ensure availability for 12 hours reduces the frequency of core damage resulting from station blackout accident sequences.

6. **Emergency Service Water (ESW) System**

The ESW system provides cooling water to selected equipment during a loss of offsite power. The system has two full capacity self-cooled pumps whose suction is from the Conowingo pond and a backup third pump with a separate water source. Failure of the ESW system would quickly fail operating diesel generators and potentially fail the low-pressure core spray (LPCS) pumps and the RHR pumps. The HPCI pumps and RCIC pumps would fail (in the long term) from a loss of their room cooling after a loss of the ESW system.

It should be noted that there is an outstanding issue regarding the need for ESW that involves whether or not the LPCS/RHR pumps actually require ESW cooling. PECo has stated that these pumps are designed to operate with working fluid temperatures approaching 160°F without pump cooling. This implies that in scenarios where the ESW system has been lost, these pumps could still operate; some RHR pumps would be placed in the suppression pool cooling mode and therefore keep the working fluid at less than 160°F. It is felt that there is significant validity to these arguments. However, because it is uncertain whether the suppression pool water can be maintained below 160°F in some sequences and whether PECo has properly accounted for pump heat addition to the system, the analysis summarized here assumes these LPCS/RHR pumps will fail upon loss of ESW cooling.

7. **Automatic and Manual Depressurization System**

The automatic depressurization system (ADS) is designed to depressurize the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant. The ADS consists of five safety relief valves capable of being manually opened. The operator may manually initiate the ADS or may depressurize the reactor vessel, using the six additional relief valves that are not connected to the ADS logic. The ADS valves are located inside the containment; however, the instrument nitrogen and the dc power required to operate the valves are supplied from outside the containment.

8. **Standby Liquid Control (SLC) System**

The SLC system provides a backup method that is redundant but independent of the control rods to establish and maintain the reactor subcritical. The suction for the SLC system comes from a control tank that has sodium pentaborate in solution with demineralized water. Most of the SLC system is located in the reactor building outside the drywell. Local access to the SLC system could be affected by containment failure or containment venting.

9. **Venting Capability**

The primary containment venting system at Peach Bottom is used to prevent containment pressure limits from being exceeded. There are several vent paths:

- 2-inch torus vent to standby gas treatment (SBGT),
- 6-inch integrated leak rate test (ILRT) pipe from the torus,
- 18-inch torus vent path,
- 18-inch torus supply path,
- 2-inch drywell vent to SBGT,
- Two 3-inch drywell sump drain lines,
- 6-inch ILRT line from drywell,
- 18-inch drywell vent path, and
- 18-inch drywell supply path.

The types of sequences on which venting has the most effect are transients with long-term loss of decay heat removal. The chance of survival of the containment is increased with venting; therefore, the core damage frequency from such sequences is reduced.

If the reactor is at decay heat loads, venting using the 6-inch ILRT line or equivalent as a minimum is sufficient to lessen the containment pressure. However, in an ATWS sequence, three to four of the large 18-inch vent pathways need to be used in order to achieve the same effect. It is preferable to use a vent pathway from the torus rather than from the drywell because of the scrubbing of radioactive material coming through the suppression pool.

It is significant to note that the 6-inch ILRT line is a solid pipe rather than ductwork, so that venting by means of this pipe does not create a severe environment within the reactor building; use of the 18-inch lines will result in failure of the ductwork and severe environments within the reactor building.

10. **Location of Control Rod Drive (CRD) Pumps**

The CRD pumps at Peach Bottom are not located in the reactor building (like most plants) but are in the turbine building. Therefore, in a severe accident where severe environments are sometimes created, the CRD pumps are not subjected to these environments and can continue to operate.

4.2.3 Important Operator Actions

The emergency operating procedures (EOPs) at Peach Bottom direct the operator to perform certain actions depending on the plant conditions or symptoms (e.g., reactor vessel level below top of active fuel). Different accident sequences can have similar symptoms and therefore the same "recovery" actions. The operator actions that either are important in reducing accident frequencies or are contributing to accident frequencies are discussed and can apply to many different accident sequences.

The quantification of these human failure events was based on an abbreviated version of the THERP method (Ref. 4.7). These failure events include the following:

- Actuate core cooling

  In an accident where feedwater is lost (which includes condensate), the reactor vessel water level starts to decrease. When Level 2 is reached, HPCI and RCIC should be automatically actuated. If Level 1 is reached, the automatic depressurization system (ADS) should be actuated with automatic actuation of the low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI). If these systems fail to actuate, the operator can attempt to manually actuate them from the control room. In addition, the operator can attempt to recover the power conversion system (PCS) (i.e., feedwater) or manually initiate control rod drive (CRD) (i.e., put CRD in its enhanced flow mode). If automatic depressurization failure was one of the faults, the operator can manually depressurize so that LPCS and LPCI can inject. Lastly, the operator also has the option to align the HPSW to LPCI for another core cooling system.

- Establish containment heat removal

  Besides core cooling, the operator must also establish containment heat removal (CHR). Without CHR, the potential exists for operating core cooling systems to fail. If an accident occurs, the EOPs direct the operator to initiate the suppression pool cooling mode of residual heat removal (RHR) after the suppression pool temperature reaches 95°F. The operator closes the LPCI injection valves and the heat exchanger bypass valves and opens the suppression pool discharge valves. He also ensures that the proper service water system train is operating. With suppression pool cooling (SPC) functioning, CHR is being performed. If system faults preclude the use of SPC, the operator has other means to provide CHR. He can actuate other modes of RHR such as shutdown cooling or containment spray; or the operator can vent the containment to remove the heat.

- Restore service water

  Many of the components/systems require cooling water from the emergency service water (ESW) system in order to function. If the ESW pumps fail, the operator can manually start the emergency cooling water pump, which is a backup to the ESW pumps.

Specifically for station blackout, there are certain actions that can be performed by the operating crew:

- Recovering ac power

  Station blackout is caused by the loss of all ac power, i.e., both offsite and onsite power. Restoring offsite power or repairing the diesel generators was included in the analysis. The quantification of these human failure events was derived from historical data (i.e., actual time required to perform these repairs) and not by performing a human reliability analysis on these events.

Transients where reactor trip does not occur (i.e., ATWS) involve accident sequences where the phenomena are more complex. The operator actions were evaluated in more detail (using the SLIM-MAUD* method performed by Brookhaven National Laboratory (Ref. 4.8)) than for the regular transients. These actions include the following:

- Manual scram

  A transient that demands the reactor to be tripped occurs, but the reactor protection system (RPS) fails from electrical faults. The operator can then manually trip the reactor by first rotating the collar on the proper scram buttons and then depressing the buttons, or he can put the reactor mode switch in the "shutdown" position.

- Insert rods manually

  If the electrical faults fail both the RPS and the manual trip, the operator can manually insert the control rods one at a time.

- Actuate standby liquid control (SLC)

  With the reactor not tripped, reactor power remains high; the reactor core is not at decay heat levels. This can present problems since the CHR systems are only designed to decay heat removal capacity. However, the SLC system (manually activated) injects sodium pentaborate that reduces reactor power to decay heat levels. The EOPs direct the operator to actuate SLC if the reactor power is above 3 percent and before the suppression pool temperature reaches 110°F. The operator obtains the SLC keys (one per pump) and inserts the keys into the switches and turns only one to the "on" position.

- Inhibit automatic depressurization system (ADS)

  In an ATWS condition, the operator is directed to inhibit the ADS if he has actuated SLC. The operator must put both ADS switches in the inhibit mode.

*SLIM-MAUD is a computer algorithm for transforming man-man and man-machine information into probability statements.

- Manually depressurize reactor

  If the high-pressure coolant injection (HPCI) fails, inadequate high-pressure core cooling occurs. Because the ADS was inhibited, when Level 1 is reached, ADS will not occur and the operator must manually depressurize so that low-pressure core cooling can inject.

4.2.4 Important Individual Events and Uncertainties (Core Damage Frequency)

As discussed in Chapter 2, the process of developing a probabilistic model of a nuclear power plant involves the combination of many individual events (initiators, hardware failures, operator errors, etc.) into accident sequences and eventually into an estimate of the total frequency of core damage. After development, such a model can also be used to assess the relative importance and contribution of the individual events. The detailed studies underlying this report have been analyzed using several event importance measures. The results of the analyses using two measures, "risk reduction" and "uncertainty" importance, are summarized below.

- Risk (core damage frequency) reduction importance measure (internal events)

  The risk-reduction importance measure is used to assess the change in core damage frequency as a result of setting the probability of an individual event to zero. Using this measure, the following individual events were found to cause the greatest reduction in core damage frequency if their probabilities were set to zero:

  - Mechanical failure of the reactor protection system. The core damage frequency would be reduced by approximately 52 percent.
  - Transient initiators with the power conversion system available. The core damage frequency would be reduced by approximately 47 percent.
  - Loss of offsite power initiating event. The core damage frequency would be reduced by approximately 39 percent.
  - Operator failure to restore the standby liquid control system after testing. The core damage frequency would be reduced by approximately 25 percent.
  - Operator failure to initiate emergency heat sink. The core damage frequency would be reduced by approximately 17 percent.
  - Operator failure to actuate standby liquid control system. The core damage frequency would be reduced by approximately 16 percent.
  - Operator miscalibrates reactor pressure sensors. The core damage frequency would be reduced by approximately 12 percent.

  Note that the top risk-reduction events do not necessarily appear in the most frequent sequences since the latter sequences may result from the cumulative influence of many lesser contributors.

- Uncertainty importance measure (internal events)

  A second importance measure used to evaluate the core damage frequency analysis results is the uncertainty importance measure. For this measure, the relative contribution of the uncertainty of individual events to the uncertainty in total core damage frequency is calculated. Using this measure, the following events were found to be most important:

  - Mechanical failure of the reactor protection system.
  - Failure of the diesel generators to continue to run once started.
  - Loss of offsite power or transients with the power conversion system available.
  - Miscalibration of the reactor pressure sensors by the operator.
  - Operator failure to restore the standby liquid control system after testing.

4.3 Containment Performance Analysis

4.3.1 Results of Containment Performance Analysis

The Peach Bottom Mark I containment design concept consists of a pressure-suppression containment system that houses the reactor vessel, the reactor coolant recirculating loops, and other branch connections to the reactor coolant system. The containment design consists of a light-bulb-shaped drywell and a water-filled toroidal-shaped suppression pool. Both the drywell and the suppression pool are freestanding steel shells with the drywell region backed by a reinforced concrete structure. The containment system has a volume of 320,000 cubic feet and is designed to withstand a peak pressure of 56 psig resulting from a primary system loss-of-coolant accident. The estimated mean failure pressure for Peach Bottom's containment system is 148 psig, which is very similar to that for large PWR containment designs. However, its small free volume relative to other containment types significantly limits its capacity to accommodate noncondensible gases generated in severe accident scenarios in addition to increasing its potential to come into contact with molten core material. The complexity of the events occurring in severe accidents has made predictions of when and where Peach Bottom's containment would fail heavily reliant on the use of expert judgment to interpret and supplement the limited data available.

The potential for early containment failure (before or within roughly 2 hours after reactor vessel breach) is of principal concern in Peach Bottom's risk analysis. For the Peach Bottom Mark I type of containment, the principal mechanisms that can cause its early failure are (1) drywell shell meltthrough due to its interaction with the molten core material released from the breached reactor pressure vessel, (2) overpressure failure of the drywell due to rapid direct containment heating following reactor vessel breach, and (3) stretching of the drywell head bolts (due to internal pressurization) causing a direct leakage path from the system. Possible overpressure failures due to hydrogen combustion effects are of negligible probability for Peach Bottom since the containment is inerted. In addition to the early modes of containment failure, core damage sequences can also result in late containment failure or no containment failure at all.

The results of the Peach Bottom containment analysis are summarized in Figures 4.5 and 4.6. Figure 4.5 contains a display of information in which the conditional probabilities of 10 containment-related accident progression bins; e.g., V.B-early WWF - >200, are presented for each of six plant damage states, such as station blackout. This information indicates that, on a plant damage state frequency-weighted average,* the mean conditional probability from internally initiated accidents of: (1) early wetwell failure is about 0.03, (2) early drywell failure is about 0.52, (3) late failure of either the wetwell or drywell is about 0.04, and (4) no containment failure is about 0.27. Figure 4.6 further displays the conditional probability distribution of early containment failure for each plant damage state, thereby providing the estimated range of uncertainties in these containment failure predictions. The important conclusions that can be drawn from the information in these two figures are: (1) there is a high mean probability (i.e., 50%) that the Peach Bottom containment will fail early for the dominant plant damage states; (2) early containment failures will primarily occur in the drywell structure resulting in a bypass of the suppression pool's scrubbing effects for radioactive material released after vessel breach; and (3) the principal cause of early drywell failure is drywell shell meltthrough. The data further indicate that the early containment failure probability distributions for most plant damage states are quite broad. Also presented in these displays of containment failure information is evidence that there is a high probability of early containment failure during external events such as fire and earthquakes. Specifically, the seismic analysis indicates that the conditional probability of early containment failure from all causes, i.e., direct containment structural failure or related failure from the effects of a core damage event, could be as high as 0.9.

*Each value in the column in Figure 4.5 labeled "All" is obtained by summing the products of individual accident progression bin conditional probabilities for each plant damage state and the ratio of the frequency of that plant damage state to the total core damage frequency.

Additional discussion on containment performance (for all studied plants) is provided in Chapter 9.

4.3.2 Important Plant Characteristics (Containment Performance)

Characteristics of the Peach Bottom containment design and operation that are important during core damage accidents include:

1. **Containment Inerting**

   The Peach Bottom containment is maintained in an inerted state, i.e., nitrogen filled. This inerted containment condition significantly reduces the chance of hydrogen combustion in the containment, thereby removing a major threat to its failure. However, hydrogen combustion in the reactor building is a possibility for some severe accident sequences.

2. **Drywell Sprays**

   The Peach Bottom drywell contains a spray header that can be used to mitigate the effects of the actions of molten core material on the floor of the drywell. In particular, the spray system may provide sufficient water to prevent the molten core material from coming into contact with the drywell shell and potentially causing its failure.

PLANT DAMAGE STATE
(Mean Core Damage Frequency)

ACCIDENT PROGRESSION BIN	Internal Initiators: SBO (2.08E−06)	Internal Initiators: LOCAs (1.50E−07)	Internal Initiators: ATWS (1.93E−06)	Internal Initiators: Transients (1.81E−07)	Internal Initiators: All (4.34E−06)	Fire (1.98E−05)	Seismic LLNL (7.52E−05)
VB > 200psi, early WWF	0.045		0.006		0.022	0.045	0.028
VB < 200 psi, early WWF	0.012	0.028	0.006	0.026	0.011	0.004	0.027
VB > 200 psi, early DWF	0.436		0.330		0.341	0.529	0.369
VB < 200 psi, early DWF	0.133	0.360	0.194	0.356	0.183	0.070	0.489
VB, late WWF	0.007			0.002	0.003	0.009	0.006
VB, late DWF	0.061	0.074	0.015	0.074	0.047	0.086	0.074
VB, CV	0.074	0.003	0.207	0.016	0.110	0.020	0.007
No CF	0.121	0.536	0.127	0.512	0.184	0.159	
No VB	0.112		0.091	0.014	0.089	0.078	
No Core Damage			0.024		0.010		

VB = Vessel Breach
WWF = Wetwell Failure
DWF = Drywell Failure
CV = Containment Venting
CF = Containment Failure

Figure 4.5 Conditional probability of accident progression bins at Peach Bottom.

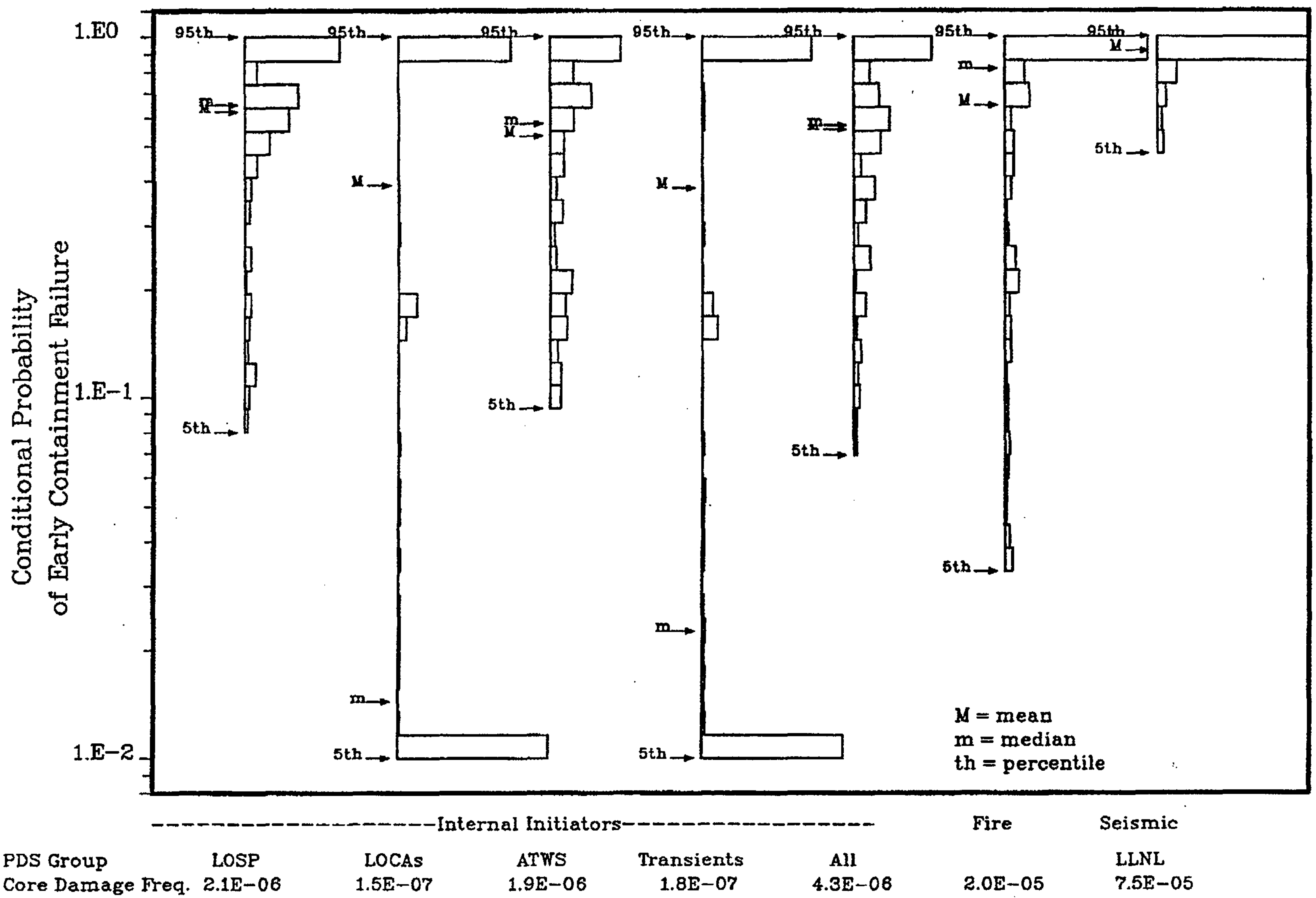


Figure 4.6 Conditional probability distributions for early containment failure at Peach Bottom.

4.4 Source Term Analysis

4.4.1 Results of Source Term Analysis

Failure of the drywell shell following vessel meltthrough is a characteristic of the risk-dominant accident progression bins for the Peach Bottom plant. Figure 4.7 illustrates the source terms for the early failure accident progression bin in which the reactor coolant system is pressurized (> 200 psi) at the time of vessel failure. In comparison with the bypass release that was illustrated for Surry in Figure 3.7, the core fractions of the volatile groups (iodine, cesium, and tellurium) released to the environment are slightly reduced. For the majority of accident sequences in Peach Bottom, the radionuclides released from fuel in-vessel must pass through the suppression pool where substantial decontamination is possible. In sequences where the drywell spray system is operable, the ex-vessel release will also be mitigated by the spray or an overlaying pool of water. Both the in-vessel and ex-vessel releases will receive further attenuation in the reactor building before release to the environment. Even if the decontamination factor of some of these stages is small, the overall effect is to make the likelihood of a very large release quite small.

The Peach Bottom plant has instituted emergency operating procedures to vent the containment in the wetwell region to avoid failure by overpressurization. Figure 4.8 shows the source terms for the accident progression bin in which the containment is vented and no subsequent failure of the containment occurs. The source terms for the volatile radionuclide groups are less than those for the early drywell failure bin discussed previously. In both cases, scrubbing of the in-vessel release by the suppression pool has the principal mitigating influence on the environmental release. The release fractions for the less volatile groups are smaller for the vented accident progression bin but only by approximately a factor of one-half. There are two reasons why the differences between the environmental release of the ex-vessel species for the vented and drywell failure cases are not greater. The decontamination capability of the suppression pool for ex-vessel release, in which the flow is through the downcomers, is somewhat less than for the in-vessel release, which passes through spargers on the safety relief lines. Thus, even though the ex-vessel release must pass through the pool for the vented case, the decontamination factor may be small. The ex-vessel release for the drywell failure accident progression bin will at least be subjected to decontamination in the reactor building and possibly to sprays and scrubbing by an overlaying water layer.

The range of uncertainty in the release for the barium and strontium radionuclide groups is particularly evident. The spread between the mean and median is two orders of magnitude. Although the release is likely to be quite small, the mean value of the release is as high as the mean value for the tellurium release.

Additional discussion on source term perspectives is provided in Chapter 10.

4.4.2 Important Plant Characteristics (Source Term)

1. **Reactor Building**

   The Peach Bottom containment is located within a reactor building. A release of radioactive material to the reactor building will undergo some degree of decontamination before release to the environment. An important consideration in determining the magnitude of building decontamination is whether hydrogen combustion occurs in the building and whether combustion is sufficiently energetic to fail the building. The range of decontamination factors for the reactor building used in the study is from 1.1 to 10 with a median value of 3 for typical accident conditions.

2. **Pressure-Suppression Pool**

   The pressure-suppression pool is particularly effective in the reduction of the in-vessel release component of the source terms for Peach Bottom. The range of decontamination factors used is from 1.2 to 4000 with a median of 80 for flow through the safety relief valve lines.

   The submergence is less and bubble size is larger for flow through the downcomers than for the spargers through which the in-vessel release is most likely to enter the pool. As a result, the decontamination factor for the ex-vessel release or any in-vessel release that passes through the drywell is smaller, ranging from approximately 1 to 90 with a median of 10. Furthermore, the likelihood of failure of the drywell at the time of vessel meltthrough is predicted to be high. For scenarios involving early drywell failure, the suppression pool would be bypassed during the period of core-concrete interaction and radionuclide release.

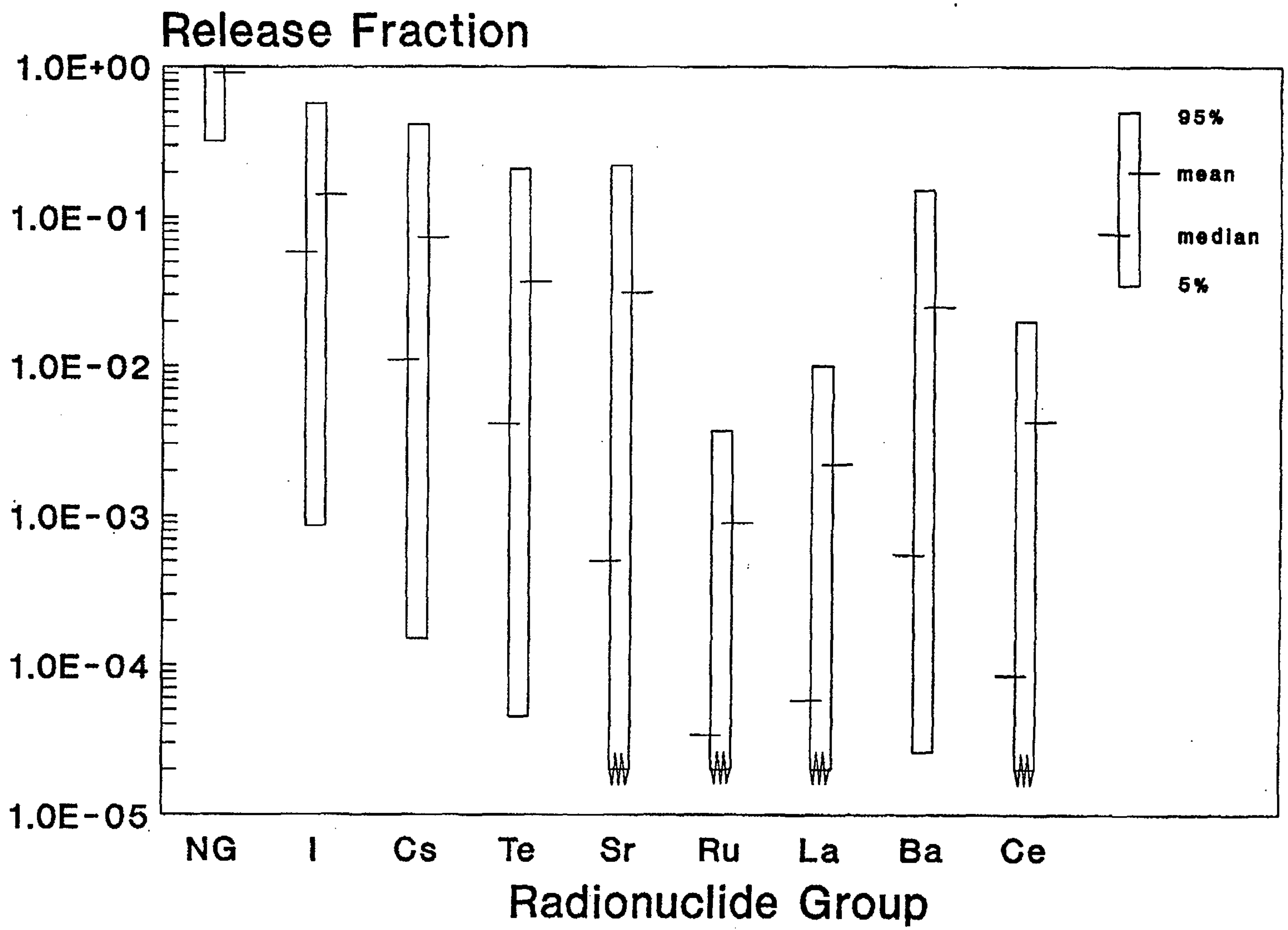


Figure 4.7 Source term distributions for early failure in drywell at Peach Bottom.

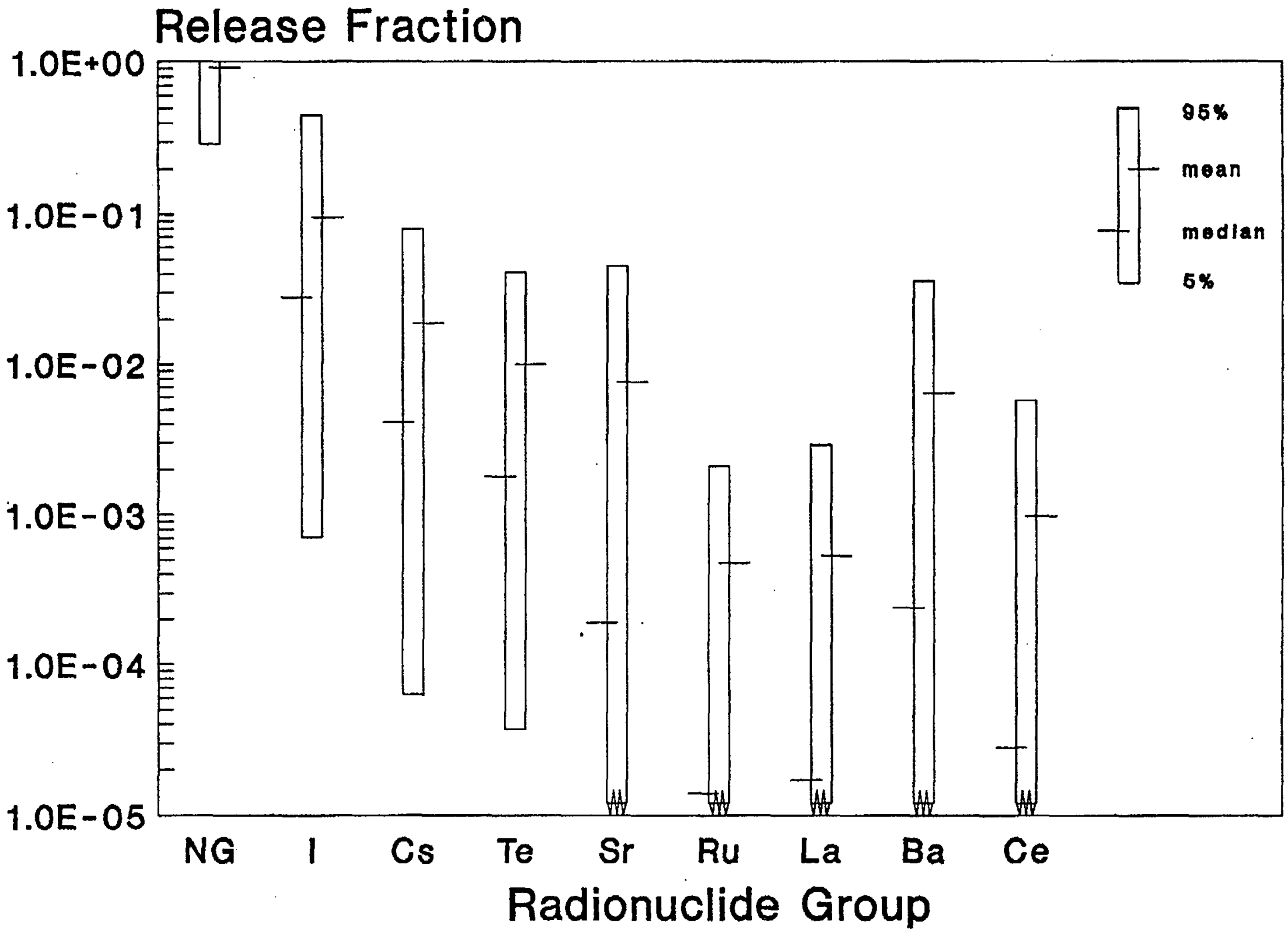


Figure 4.8 Source term distributions for vented containment at Peach Bottom.

3. **Venting**

   The Peach Bottom containment can be vented from the wetwell air space. By preventing containment failure, venting can potentially prevent some scenarios from becoming core damage accidents. In scenarios that proceed to fuel melting, venting can lead to the mitigation of the release of radioactive material to the environment by ensuring that the release passes through the suppression pool. The effect of venting on core damage frequency is described in Chapter 8. Figure 4.8 illustrates the source term characteristics for the venting accident progression bins. Although the source terms are somewhat less than for the early drywell failure accident progression bin, the uncertainties in the release fractions are quite broad. At the high end of the uncertainty range, it is possible that 40 percent of the core inventory of iodine could be released to the environment.

   The effectiveness of venting to mitigate severe accident release of radioactive material is limited in the Peach Bottom analyses because of the high likelihood of early drywell failure, particularly as the result of direct attack of the shell by molten core debris. If direct attack of the containment shell is determined not to lead to failure or if effective means are found to preclude failure, the effectiveness of venting could be greater. However, considering the range of uncertainties in the source term analyses, the predicted consequences of vented accident progression bins are not necessarily minor.

4.5 Offsite Consequence Results

Figures 4.9 and 4.10 display the frequency distributions in the form of graphical plots of the complementary cumulative distribution functions (CCDFs) of four offsite consequence measures—early fatalities, latent cancer fatalities, and the 50-mile and entire site region population exposures (in person-rems). The CCDFs in Figures 4.9 and 4.10 include contributions from all source terms associated with reactor accidents caused by the internal initiating events and fire, respectively. Four CCDFs, namely, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs, are shown for each consequence measure.

Peach Bottom plant-specific and site-specific parameters were used in the consequence analysis for these CCDFs. The plant-specific parameters included source terms and their frequencies, the licensed thermal power (3293 MWt) of the reactor, and the approximate physical dimensions of the power plant building complex. The site-specific parameters included exclusion area radius (820 meters), meteorological data for 1 full year collected at the site meteorological tower, the site region population distribution based on the 1980 census data, topography (fraction of the area that is land—the remaining fraction is assumed to be water), land use, agricultural practice and productivity, and other economic data for up to 1,000 miles from the Peach Bottom plant.

The consequence estimates displayed in these figures have incorporated the benefits of the following protective measures: (1) evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), (2) early relocation of the remaining population only from the heavily contaminated areas both within and outside the 10-mile EPZ, and (3) decontamination, temporary interdiction, or condemnation of land, property, and foods contaminated above acceptable levels.

The population density within the Peach Bottom 10-mile EPZ is about 90 persons per square mile. The average delay time before evacuation (after a warning prior to radionuclide release) from the 10-mile EPZ and average effective evacuation speed used in the analyses were derived from information contained in a utility-sponsored Peach Bottom evacuation time estimate study (Ref. 4.9) and the NRC requirements for emergency planning.

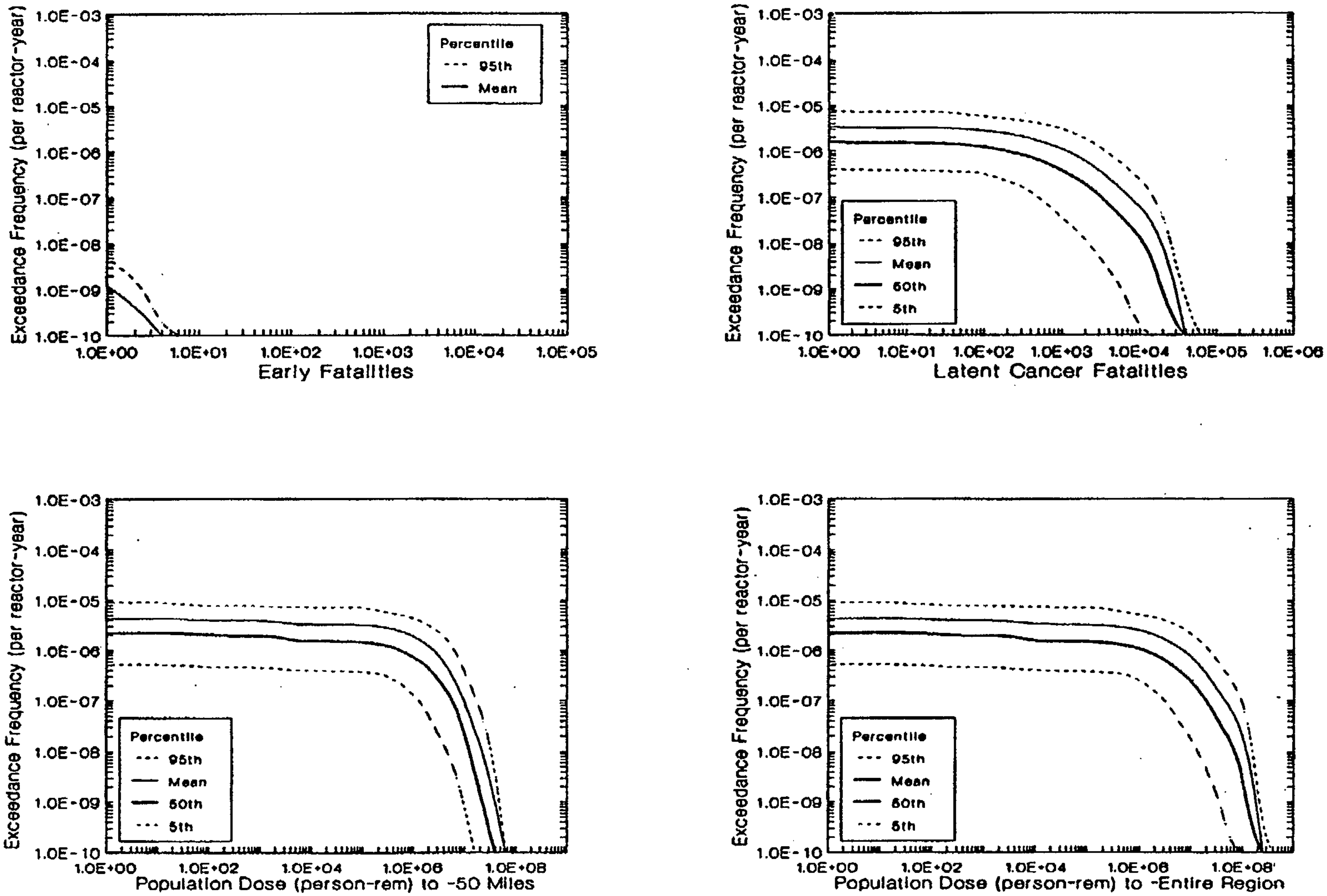
The results displayed in Figures 4.9 and 4.10 are discussed in Chapter 11.

4.6 Public Risk Estimates

4.6.1 Results of Public Risk Estimates

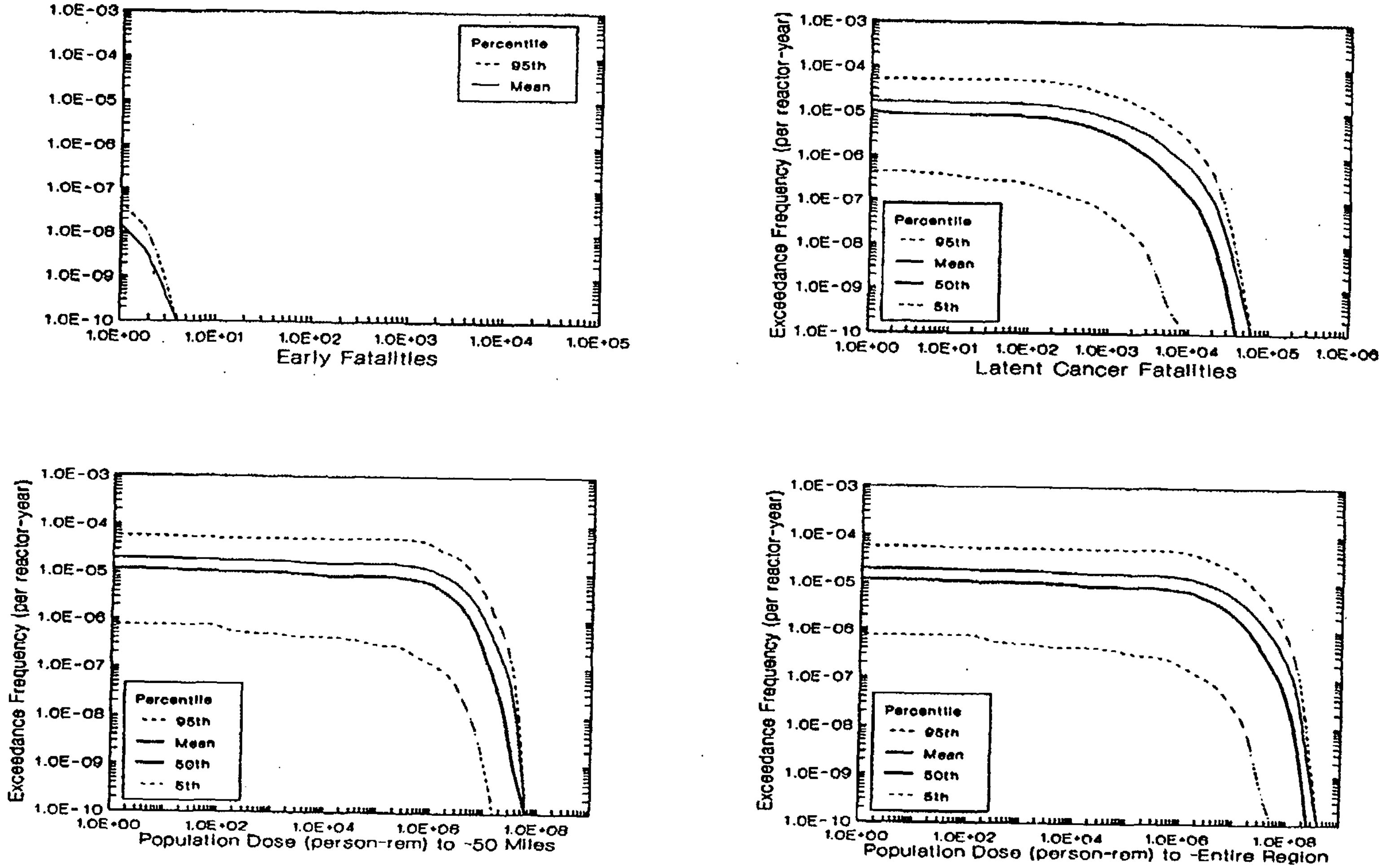
A detailed description of the results of the Peach Bottom risk is provided in Reference 4.2. For this summary report, results are provided for the following measures of public risk:

- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Peach Bottom exclusion area boundary, and
- Individual latent cancer fatality risk in the population within 10 miles of the site.



Note: As discussed in Reference 4.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 4.9 Frequency distributions of offsite consequence measures at Peach Bottom (internal initiators).



Note: As discussed in Reference 4.4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 4.10 Frequency distributions of offsite consequence measures at Peach Bottom (fire initiators).

The first four of the above measures are commonly used measures in nuclear power plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 4.10).

4.6.1.1 Internally Initiated Accident Sequences

The results of the risk studies using the above measures are shown in Figures 4.11 through 4.13. The figures display the variabilities in mean risks estimated from the meteorology-averaged conditional mean values of the consequence measures. For the first two measures, the results of the first risk study of Peach Bottom, the Reactor Safety Study (Ref. 4.3), are also provided. As may be seen, the early fatality risk from Peach Bottom is estimated to be very low. Latent cancer fatality risks are lower than those of the Reactor Safety Study. The risks of population dose and individual early fatality risk are also very low, and the individual latent cancer fatality risk is orders of magnitude lower than the NRC safety goals. These comparisons are discussed in more detail in Chapter 12.

The risk results shown in Figure 4.11 have been analyzed to determine the relative contributions of plant damage states and accident progression bins to mean risk. The results of this analysis are provided in Figures 4.14 and 4.15. As can be seen from these figures, and from the supporting document (Ref. 4.2), the major contributors to both early and latent cancer fatality risks are from station blackout (SBO) and anticipated transients without scram (ATWS). The dominant accident progression bins are early containment failure and drywell failure caused by drywell meltthrough and loads at vessel breach (due to direct containment heating, steam blowdown, or quasistatic pressure from steam explosion).

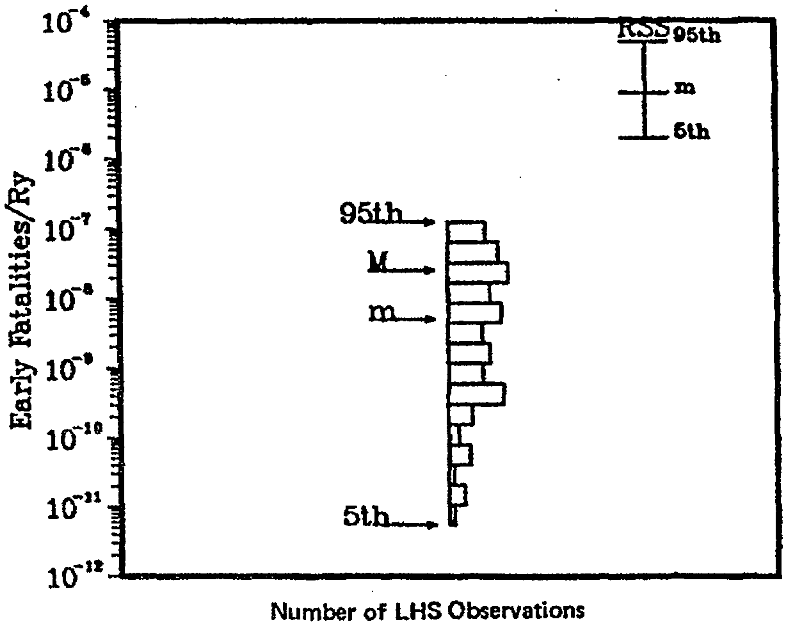
4.6.1.2 Externally Initiated Accident Sequences

As discussed in Section 4.2.1.2, the Peach Bottom plant has been analyzed for two externally initiated accidents: earthquakes and fire. The fire risk analysis has been performed through the estimates for consequences and risk measures, whereas, as explained in Chapter 2, the seismic analysis has been conducted up to containment performance. Sensitivity analyses of seismic risk at Peach Bottom are provided in Reference 4.2.

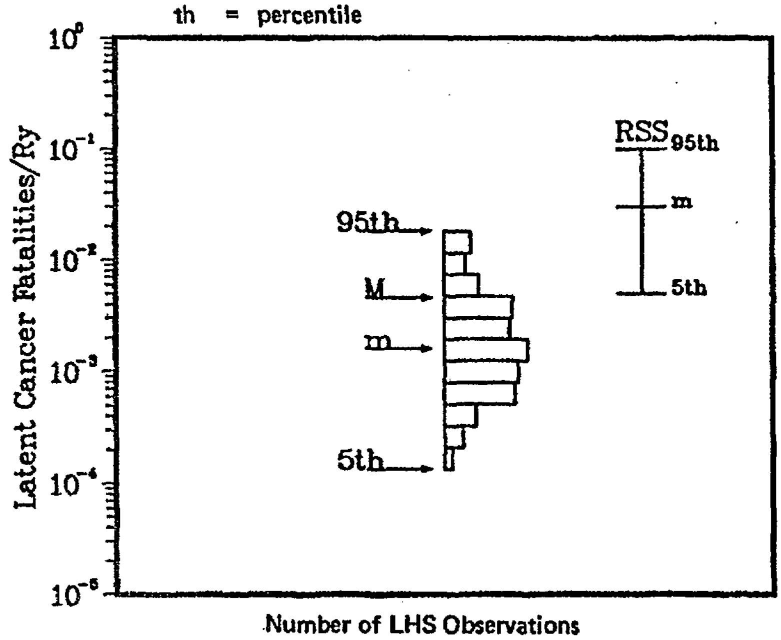
Results of fire risk analysis (variabilities in mean risks estimated from the meteorology-averaged conditional mean values of the consequence measures) of Peach Bottom are shown in Figures 4.16 through 4.18 for early fatality, latent cancer fatality, population dose (within 50 miles of the site and within the entire site region), and individual early and latent cancer fatality risks. Major contributions to early and latent cancer fatality risks are shown in Figure 4.19. As can be seen, early and latent cancer fatality risks for fire at Peach Bottom are dominated by early containment failure and drywell failure caused by drywell meltthrough and loads at vessel breach. Other risk measures are slightly higher than those for internally initiated events but well below NRC safety goals.

4.6.2 Important Plant Characteristics (Risk)

The risk from the internal events are driven by long-term station blackout (SBO) and anticipated transients without scram (ATWS). The dominance of these two plant damage states can be attributed to both general BWR characteristics and plant-specific design. BWRs in general have more redundant systems that can inject into the reactor vessel than PWRs and can readily go to low pressure and use their low-pressure injection systems. This means that the dominant plant damage states will be driven by events that fail a multitude of systems (i.e., reduce the redundancy through some common-mode or support system failure) or events that only require a small number of systems to fail in order to reach core damage. The station blackout plant damage state satisfies the first of these requirements in that all systems ultimately depend upon ac power, and a loss of offsite power is a relatively high probability event. The total probability of losing ac power long enough to induce core damage is relatively high, although still low for a plant with Peach Bottom's design. The ATWS scenario is driven by the small number of systems that are needed to fail and the high stress upon the operators in these sequences.

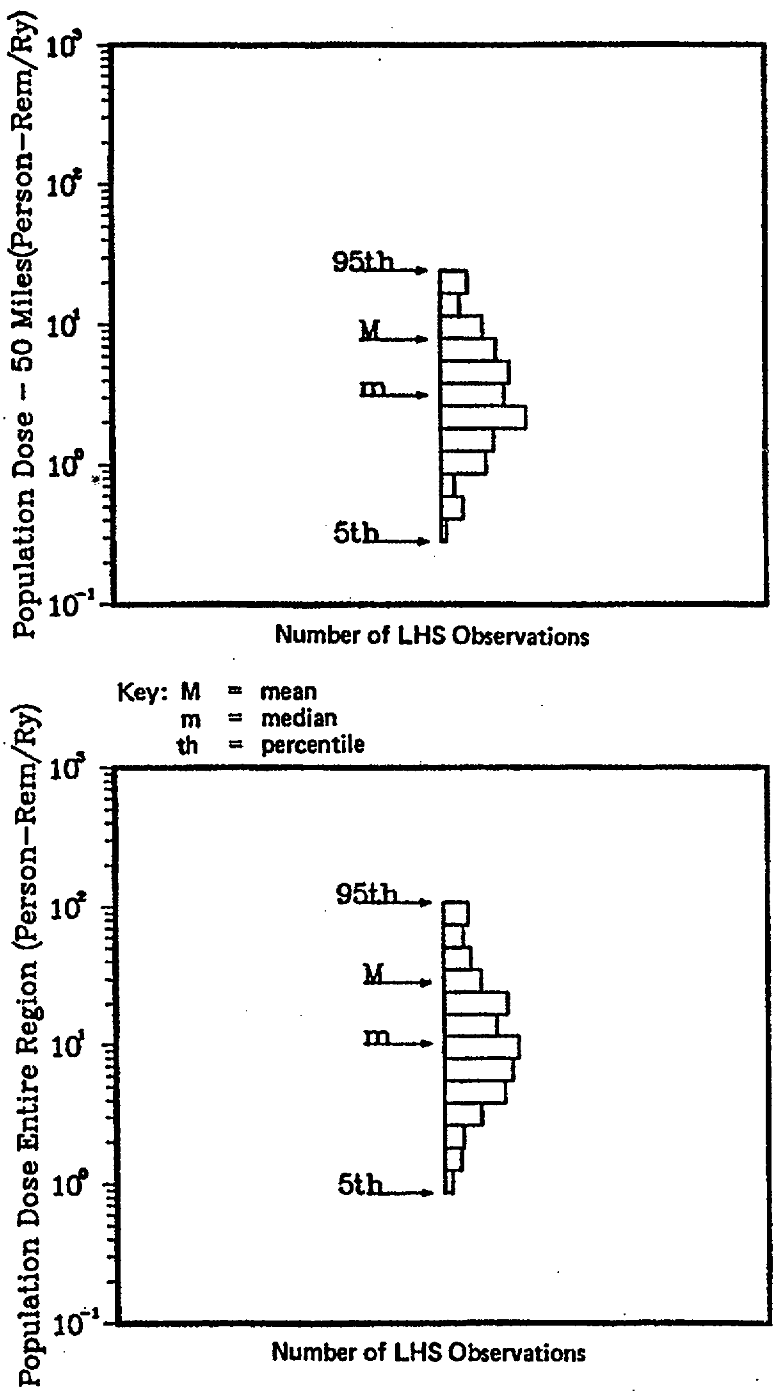


Key: M = mean
m = median
th = percentile



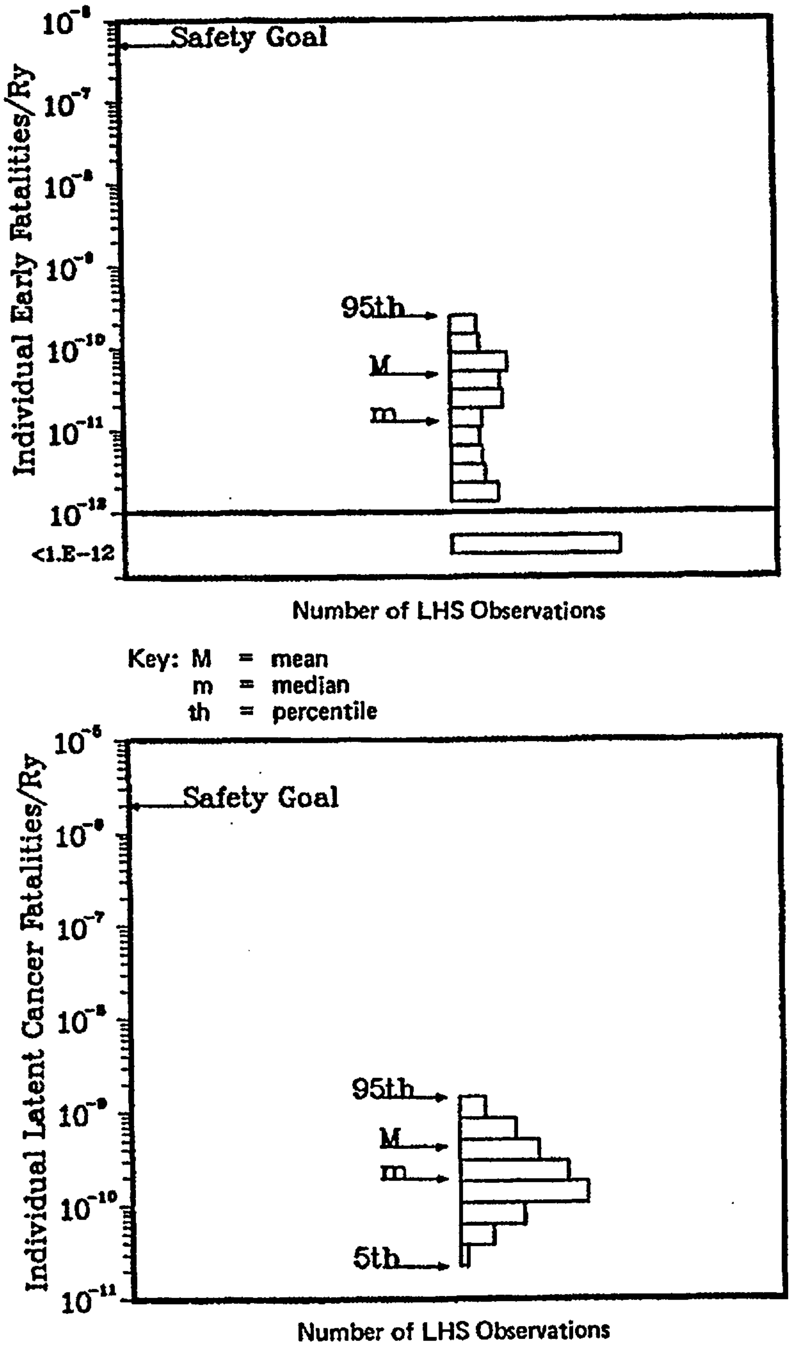
Note: As discussed in Reference 4.4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 4.11 Early and latent cancer fatality risks at Peach Bottom (internal initiators).



Note: As discussed in Reference 4.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 4.12 Population dose risks at Peach Bottom (internal initiators).



Note: As discussed in Reference 4.4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 4.13 Individual early and latent cancer fatality risks at Peach Bottom (internal initiators).

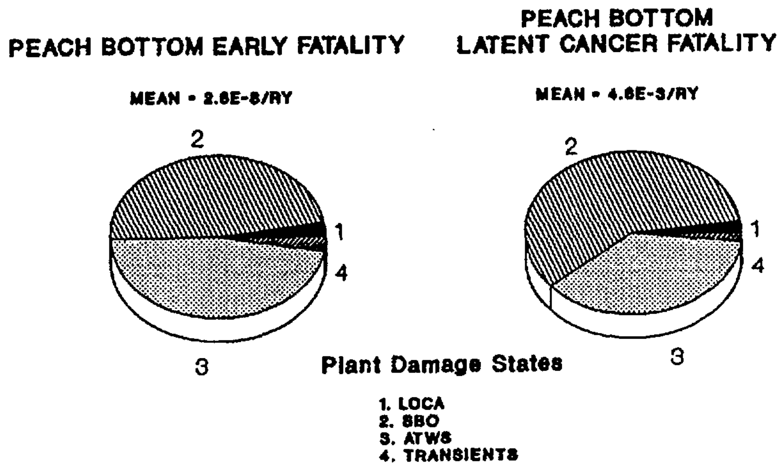


Figure 4.14 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Peach Bottom (internal initiators).

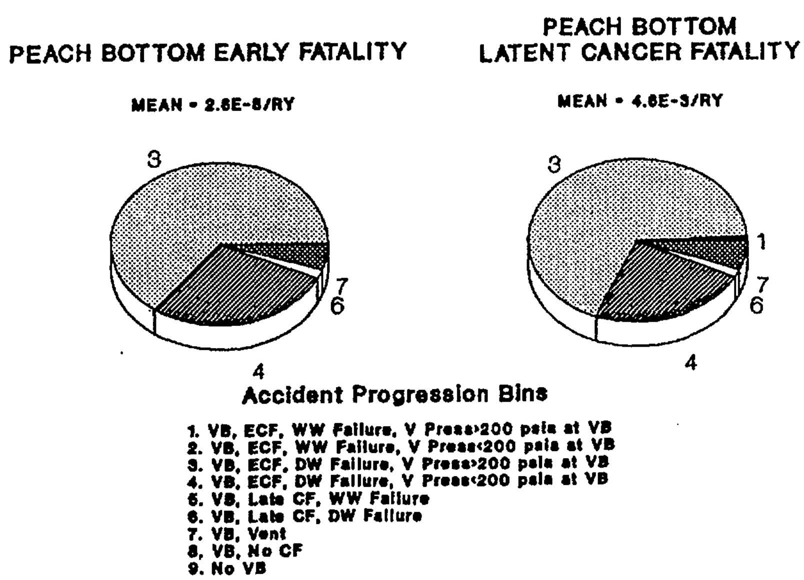
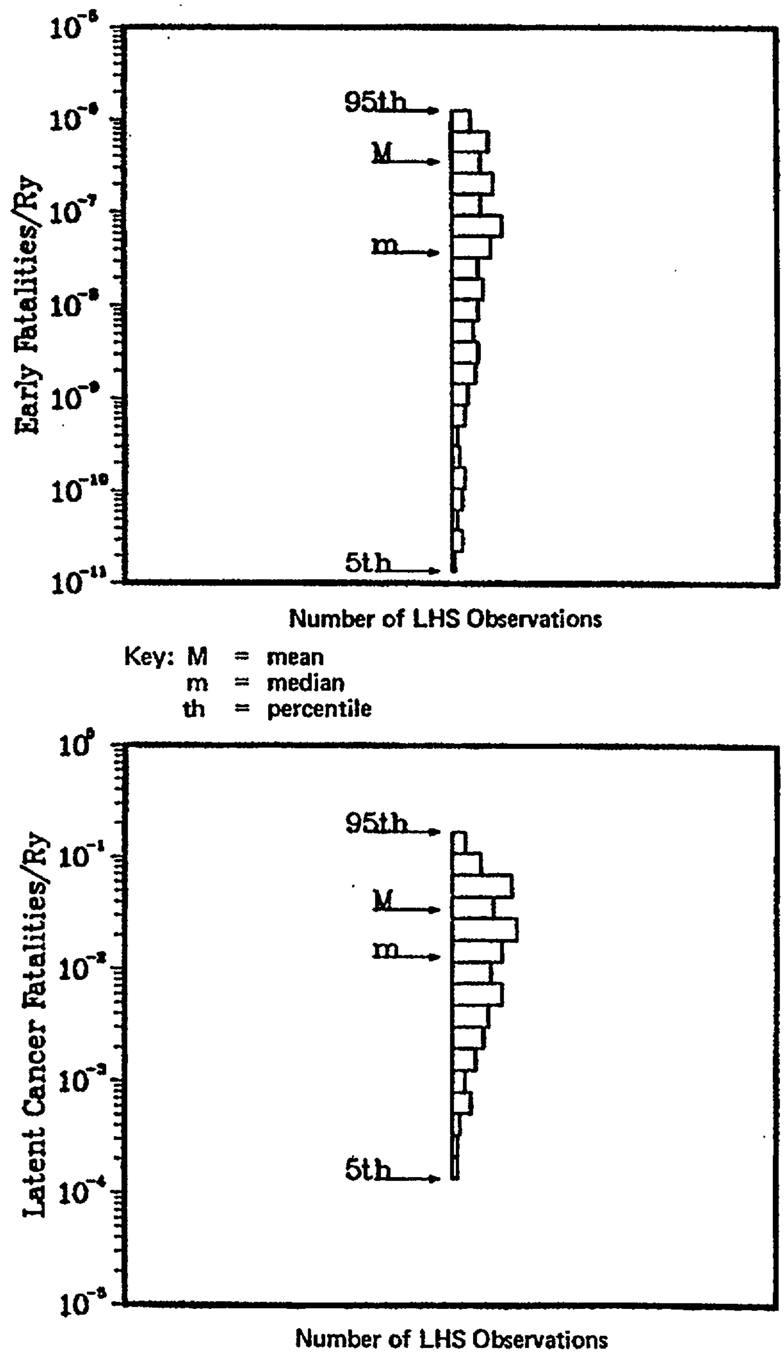
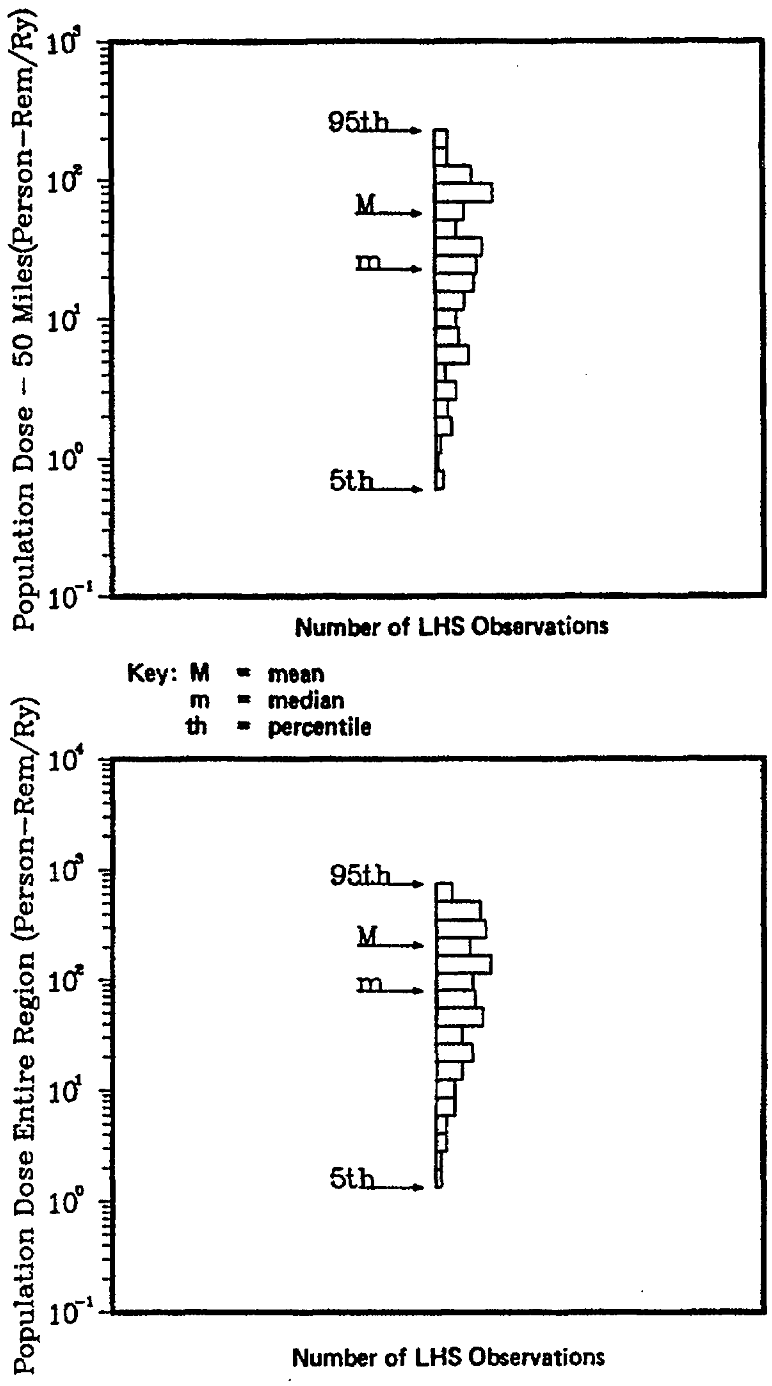


Figure 4.15 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Peach Bottom (internal initiators).



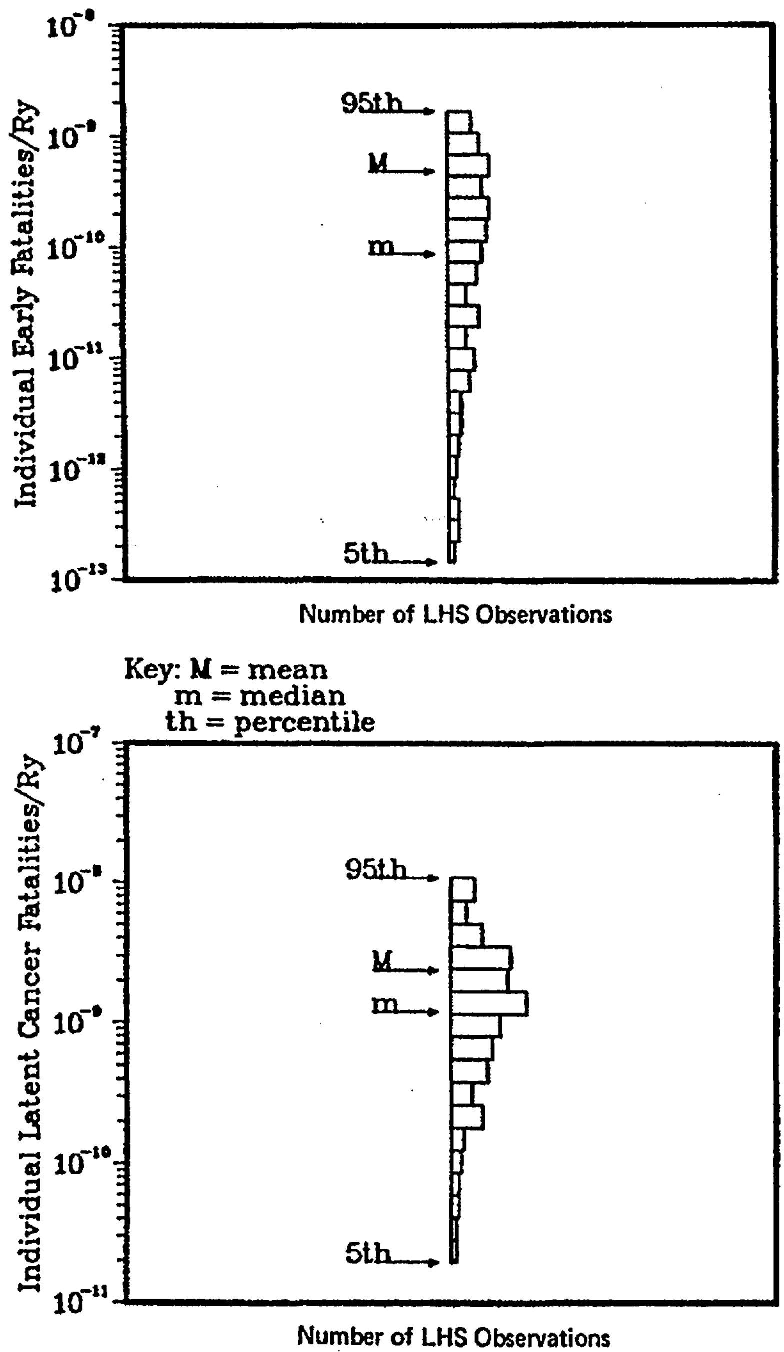
Note: As discussed in Reference 4.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 4.16 Early and latent cancer fatality risks at Peach Bottom (fire initiators).



Note: As discussed in Reference 4.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 4.17 Population dose risks at Peach Bottom (fire initiators).



Note: As discussed in Reference 4.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 4.18 Individual early and latent cancer fatality risks at Peach Bottom (fire initiators).

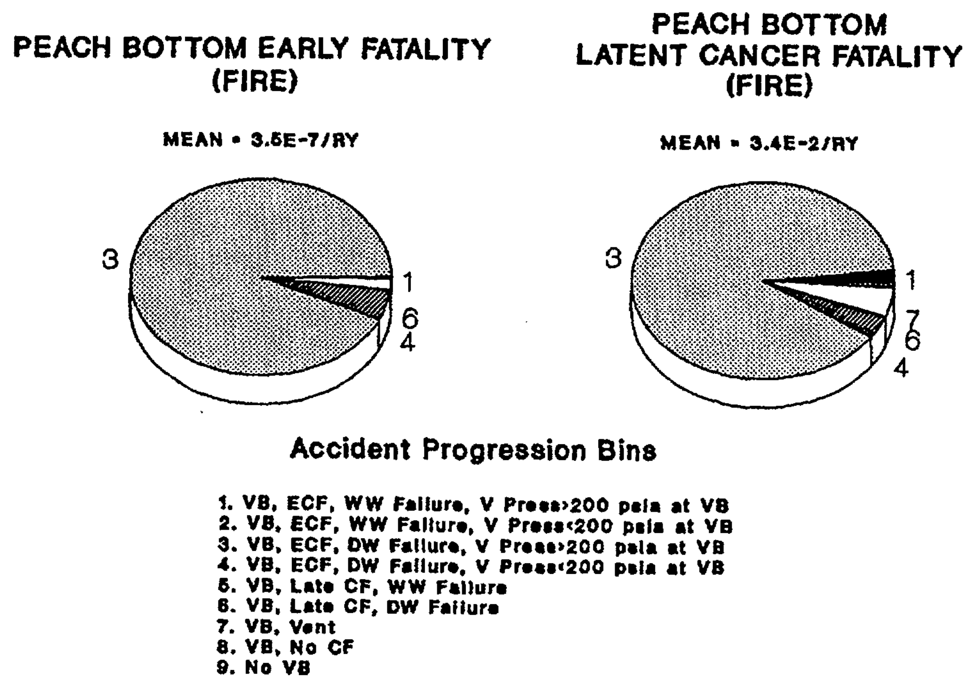


Figure 4.19 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Peach Bottom (fire initiators).

REFERENCES FOR CHAPTER 4

4.1 A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

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

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

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

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

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

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

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

4.9 Philadelphia Electric Company, "Evacuation Time Estimates with the Plume Exposure Pathway Emergency Planning Zone for the Peach Bottom Atomic Power Station," Rev. 0, July 1982.

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

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

5. SEQUOYAH PLANT RESULTS

5.1 Summary Design Information

The Sequoyah Nuclear Power Plant is a two-unit site. Each unit, designed by Westinghouse Corporation, is a four-loop pressurized water reactor (PWR) rated at 1148 MWe and is housed in an ice condenser containment. The balance of plant systems were engineered and built by the utility, the Tennessee Valley Authority. Sequoyah 1 started commercial operation in 1981. Some important design features of the Sequoyah plant are described in Table 5.1. A general plant schematic is provided in Figure 5.1.

This chapter provides a summary of the results obtained in the detailed risk analyses underlying this report (Refs. 5.1 and 5.2). A discussion of perspectives with respect to these results is provided in Chapters 8 through 12.

5.2 Core Damage Frequency Estimates

5.2.1 Summary of Core Damage Frequency Estimates

The core damage frequency and risk analyses performed for this study considered accidents initiated only by internal events (Ref. 5.1); no external-event analyses were performed. The core damage frequency results obtained are provided in tabular form in Table 5.2 and in graphical form, displayed as a histogram, in Figure 5.2 (Section 2.2.2 discusses histogram development). This study calculated a total median core damage frequency from internal events of 3.7E–5 per year.

5.2.1.1 Internally Initiated Accident Sequences

Twenty-three individual accident sequences were identified as important to the core damage frequency estimates for Sequoyah. A detailed description of these accident sequences is provided in Reference 5.1. For the purpose of discussion here, the accident sequences have been grouped into five summary plant damage states. These are:

- Station blackout,
- Loss-of-coolant accidents (LOCAs),
- Anticipated transients without scram (ATWS),
- Transients other than station blackout and ATWS, and
- Interfacing-system LOCA and steam generator tube rupture (bypass accidents).

The relative contributions of these groups to the total mean core damage frequency at Sequoyah is shown in Figure 5.3. It is seen that loss-of-coolant accidents as a group are the largest contributors to core damage frequency. Within the general class of loss-of-coolant accidents, the most probable combinations of failures are:

- Intermediate (2" < D < 6"), small (1/2 < D < 2"), and very small (D < 1/2") size LOCAs in the reactor coolant system piping followed by failure of high-pressure or low-pressure emergency coolant recirculation from the containment sump. Coolant recirculation from the containment sump can fail because of valve failures, pump failures, plugging of drains or strainers, or operator failure to correctly reconfigure the emergency core cooling system (ECCS) equipment for the recirculation mode of operation.

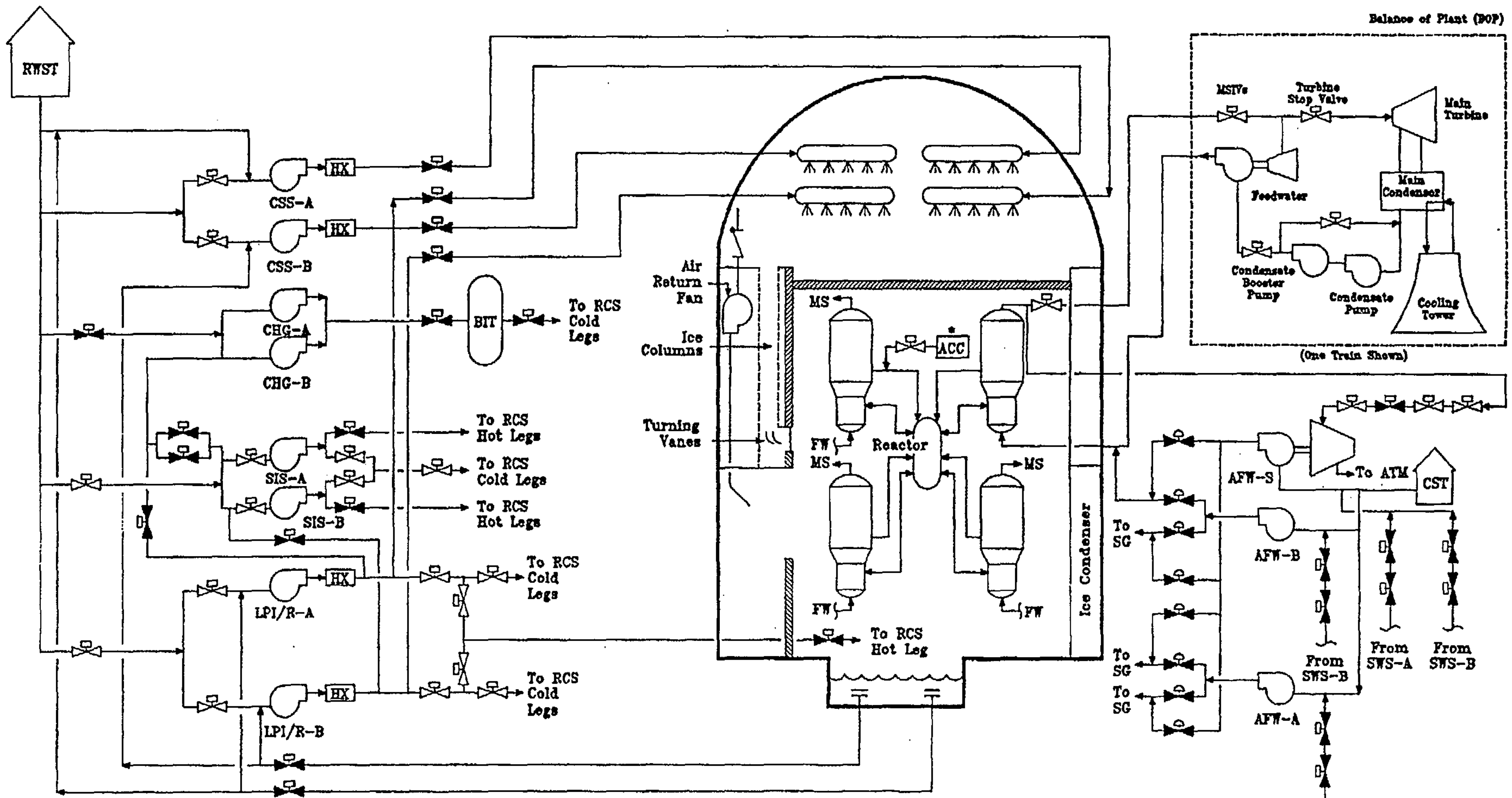
Station blackout sequences as a group are the second largest contributor to core damage frequency. Within this group, the most probable combinations of failures are:

- Station blackout with failure of the auxiliary feedwater (AFW) system. Core uncovery is caused by failure of the AFW system to provide steam generator feed flow, thus causing gradual heatup and boiloff of reactor coolant. Station blackout also results in the unavailability of the high-pressure injection systems for feed and bleed. The dominant contributors to this sequence are the station blackout followed by initial turbine-driven AFW pump unavailability due to mechanical failure or maintenance outage, or failure of the operator to open air-operated valves after depletion of the instrument air supply.
- Station blackout with initial AFW operation that fails at a later time because of battery depletion or station blackout, with reactor coolant pump (RCP) seal LOCA because of loss of all RCP seal cooling. Station blackout results in a loss of seal injection flow to the RCPs and a loss of component cooling water to the RCP thermal barriers. This condition results in vulnerability of the RCP seals to

Table 5.1 Summary of design features: Sequoyah Unit 1.

1. Coolant Injection System	a. Charging system provides safety injection flow, emergency boration, feed and bleed cooling, and normal seal injection flow to the RCPs,* with 2 centrifugal pumps. b. RHR system provides low-pressure emergency coolant injection and recirculation following LOCA, with 2 trains and 2 pumps. c. Safety injection system provides high head safety injection and feed and bleed cooling, with 2 trains and 2 pumps.
2. Steam Generator Heat Removal Systems	a. Power conversion system. b. Auxiliary feedwater system, with 3 trains and 3 pumps (2 MDPs, 1 TDP).*
3. Reactivity Control Systems	a. Control rods. b. Chemical and volume control systems.
4. Key Support Systems	a. dc power, with 2-hour station batteries. b. Emergency ac power, with 2 diesel generators for each unit, each diesel generator dedicated to a 6.9 kV emergency bus (these buses can be crosstied to each other via a shutdown utility bus). c. Component cooling water provides cooling water to RCP* thermal barriers and selected ECCS equipment, with 5 pumps and 3 heat exchangers for both Units 1 and 2. d. Service water system, with 8 self-cooled pumps for both Units 1 and 2.
5. Containment Structure	a. Ice condenser. b. 1.2 million cubic feet. c. 10.8 psig design pressure.
6. Containment Systems	a. Spray system provides containment pressure-suppression during the injection phase following a LOCA and also provides containment heat removal during the recirculation phase following a LOCA. b. System of igniters installed to burn hydrogen. c. Air-return fans to circulate atmosphere through the ice condenser and keep containment atmosphere well mixed.

*MDP: Motor-Driven Pump
TDP: Turbine-Driven Pump
RCP: Reactor Coolant Pump



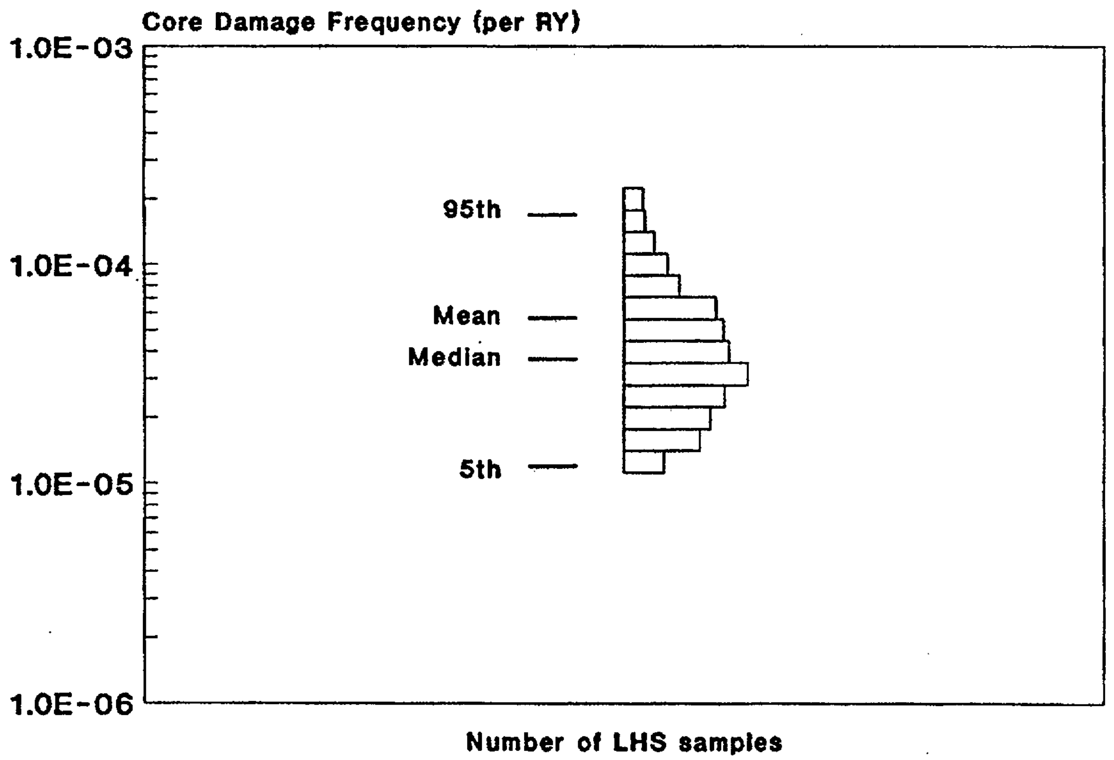
*Typical of each Cold Leg Loop

Figure 5.1 Sequoyah plant schematic.

Table 5.2 Summary of core damage frequency results: Sequoyah.*

	5%	Median	Mean	95%
Internal Events	1.2E-5	3.7E-5	5.7E-5	1.8E-4
Station Blackout				
Short Term	4.2E-7	3.8E-6	9.6E-6	3.6E-5
Long Term	1.0E-7	1.4E-6	5.0E-6	1.7E-5
ATWS	4.3E-8	5.3E-7	1.9E-6	7.5E-6
Transient	2.5E-7	1.1E-6	2.6E-6	7.2E-6
LOCA	4.4E-6	1.8E-5	3.6E-5	1.2E-4
Interfacing LOCA	1.5E-11	2.0E-8	6.5E-7	2.1E-6
SGTR	2.4E-8	4.1E-7	1.7E-6	7.1E-6

*As discussed in Reference 5.3, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).



Note: As discussed in Reference 5.3, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 5.2 Internal core damage frequency results at Sequoyah.

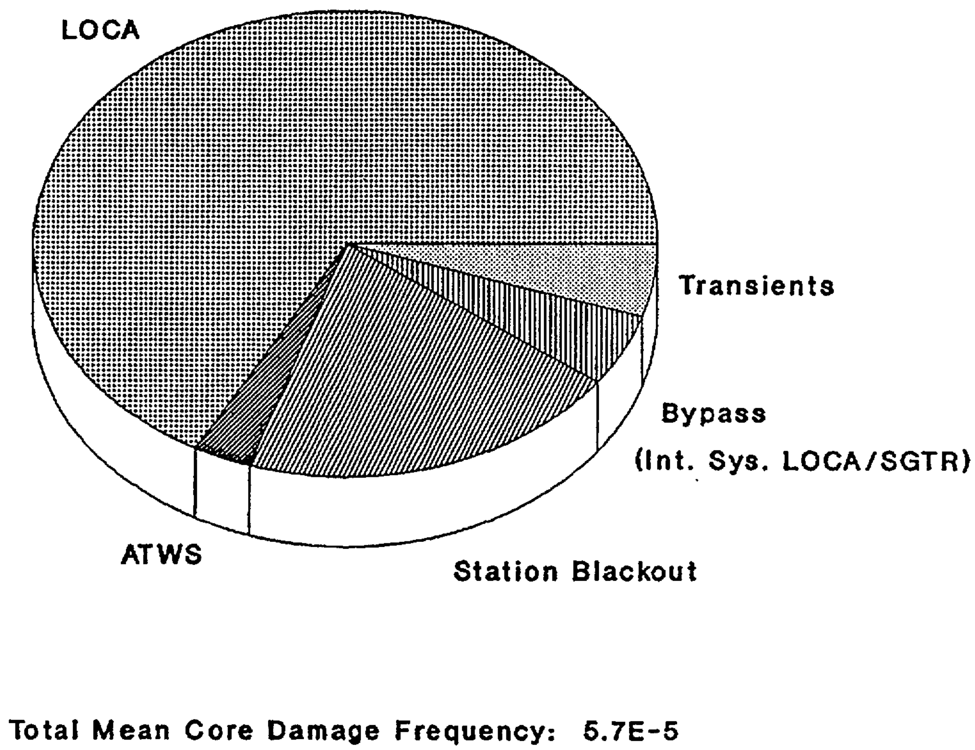


Figure 5.3 Contributors to mean core damage frequency from internal events at Sequoyah.

failure. The failure to restore ac power and safety injection flow following any seal LOCA leads to core uncovery. The time to core uncovery following onset of a seal LOCA is a function of the leak rate and whether or not the operator takes action to depressurize the reactor coolant system.

Within the general group of containment bypass accidents, the more probable combinations of failure are:

- Steam generator tube rupture, followed by failure to depressurize the reactor coolant system (RCS). Subsequent failure to depressurize the RCS in the long term and thus limit RCS leakage leads to continued blowdown through the steam generator and eventual core uncovery. An important event in this sequence is the initial failure of the operator to depressurize within 45 minutes after the tube rupture. This leads to a relief valve demand in the secondary cooling system. The steam generator safety valve will be demanded if the power-operated relief valve is blocked. Subsequent failure of the PORV or safety valve to reclose leads to direct loss of RCS inventory to the atmosphere. Failure of subsequent efforts to recover the sequence by RCS depressurization or closure of the PORV or safety valve leads to refueling water storage tank inventory depletion and eventual core uncovery.

- Failure of RCS pressure isolation leading to LOCAs in systems interfacing with the reactor coolant system (by overpressurization of low-pressure piping in the interfacing system). These sequences comprise 2 percent of the total core damage frequency but are important contributors to risk because they create a direct release path to the environment. These accidents are of special interest because they prevent ECCS operation in the recirculation mode and lead to containment bypass.

5.2.2 Important Plant Characteristics (Core Damage Frequency)

Characteristics of the Sequoyah plant design and operation that have been found to be important in the analysis of core damage frequency include:

1. **Electric Power Crossconnects Between Units 1 and 2**

   The Sequoyah electric power system design includes the capability to crosstie the 6.9 kV emergency buses at Unit 1 and Unit 2 and includes the capability to energize dc battery boards at Unit 1 from the batteries at Unit 2. These crossties help reduce the frequency of station blackout at Unit 1 and significantly reduce the possibility of battery depletion as an important contributor for those station blackouts that are postulated to occur. The crossties reduce the station blackout core damage frequency by less than a factor of 2. As station blackout sequences only account for 20 percent of the total core damage frequency, the crossties reduce total core damage frequency by approximately 10 percent.

2. **Transfer to Emergency Core Cooling and Containment Spray System Recirculation Mode**

   The process for switching the emergency core cooling system and the containment spray system from the injection mode to the recirculation mode at Sequoyah involves a series of operator actions that must be accomplished in a relatively short time (~20 minutes) and are only partially automated. Therefore, operator action is required to maintain core cooling when switching over to the recirculation mode. Single operator errors during switchover from injection to recirculation following a small LOCA can lead directly to core uncovery. Recirculation failure can also result from common-cause failures affecting the entire emergency core cooling system and containment spray system. These failures include level sensor miscalibration for the refueling water storage tank and failure to remove the upper containment compartment drain plugs after refueling.

3. **Loss of Coolant from Interfacing-System LOCA**

   Interfacing-system LOCA results from failures of any one of the four pairs of series check valves used to isolate the high-pressure RCS from the low-pressure injection system. The resultant flow into the low-pressure system is assumed to result in rupture of the low-pressure piping or components outside the containment boundary. Although core inventory makeup by the high-pressure injection system is initially available, the inability to switch to the recirculation mode would eventually lead to core damage. Because of the location of the postulated LOCA, all containment safeguards are bypassed.

   The failure scenarios of interest are those that produce a sudden large backleakage from the RCS that cannot be accommodated by relief valves in the low-pressure systems. Interfacing-system LOCA could therefore occur in two ways:

   a. Random or dependent rupture of valve internals on both valves. Rupture of the upstream valve would go undetected until rupture of the second valve occurred, and

   b. Rupture of the downstream valve combined with the failure of the upstream valve to be closed on demand. This scenario has an extremely low probability at Sequoyah because the check valve testing procedures require leak rate testing after each valve use.

   If an interfacing-system LOCA should occur, a potential recovery action was identified and considered in the analysis in which the operator may be able to isolate the interfacing-system LOCA by closing the appropriate low-pressure injection cold leg isolation valve.

4. **Diesel Generators**

   Sequoyah is a two-unit site with four diesel generator units. Each diesel is dedicated to a particular (6.9 kV) emergency bus at one of the units. Each diesel generator can only be connected to its dedicated emergency bus. However, the 6.9 kV buses can be crosstied to each other through the use of the shutdown utility bus, thus providing an indirect way to crosstie diesels and emergency buses. The diesel generators have dedicated batteries for starting and can be loaded on the emergency buses manually or with alternative power supplies. Emergency ac power is therefore not as susceptible to failures of the station batteries as at those plants where station batteries are used for diesel startup.

5. **Containment Design**

   The ice condenser containment design is important to estimates of core damage frequency because of the spray actuation setpoints. The relatively low-pressure setpoints result in spray actuation for a significant percentage of small LOCAs. The operation of the sprays will deplete the refueling water storage tank (RWST) in approximately 20 minutes, thus requiring fast operator intervention to switch over to recirculation mode. The reduced time available for operator action results in an increased human error rate for recirculation alignment associated with this time interval.

5.2.3 Important Operator Actions

Several operator actions are very important in preventing core uncovery. These actions are discussed in this section with respect to the accident sequence in which they occur.

- Switchover to ECCS recirculation in a small LOCA

  There are four major operator actions during recirculation switchover:

  - Switchover of high-pressure emergency core cooling system (ECCS) from injection to recirculation.
  - Isolation of ECCS suction from RWST.
  - Switchover of containment spray system (CSS) from injection to recirculation, including isolation of suction from the RWST.
  - Valving in component cooling water (CCW) to the residual heat removal (RHR) heat exchangers.

- Control of containment sprays during small LOCAs

  Virtually all small LOCAs will result in automatic containment spray actuation. If the operator does not control sprays early during a small LOCA, the RWST level will decrease and switchover to recirculation will be required.

  All actions are performed in the main control room at one location. The time for diagnosis is relatively short (~20 minutes) for determining if the event is actually a LOCA and anticipating whether high-pressure recirculation will be needed when the low RWST level alarm is actuated.

- Feed and bleed cooling

  For accident sequences in which main and auxiliary feedwater are unavailable, feed and bleed cooling can be used to remove decay heat from the core. The operator is instructed to initiate feed and bleed cooling if steam generator levels drop below 25 percent. This point is reached approximately 30 minutes after auxiliary feedwater (AFW) and main feedwater become unavailable.

- Anticipated transients without scram (ATWS)

  Five operator actions could potentially be required during an ATWS sequence, depending on the particular course of the sequence. These events are:

  - Manual reactor trip.
  - Trip turbine if not done automatically.
  - Start AFW if not started automatically.
  - Open block valve on power-operated relief valve (PORV) within 2 minutes if PORV is isolated previous to initiating event.
  - Emergency boration, if manual trip failed.

  Due to the fast-acting nature of an ATWS, all ATWS actions must be performed from memory.

- Steam generator tube rupture

  Steam generator tube rupture (SGTR) accident sequences are considered to begin with a double-ended rupture of a single steam generator tube. Very shortly thereafter, a safety injection signal will occur on low RCS pressure. The immediate concern for the operator, after identifying the event as an SGTR, is to identify and isolate the ruptured steam generator. There are three possible operator actions during an SGTR. These are:

  - Cool down and depressurize the RCS very shortly (~45 minutes) after the event in order to prevent lifting the relief valves on the affected steam generator;

- Restore the main feedwater flow in the event of a loss of auxiliary feed flow; and

- Isolate the steam generator that contains the ruptured tube.

- Interfacing-system LOCA recovery action

  The two RHR trains are physically isolated from each other and are provided with system isolation capability. To recover from an interfacing-system LOCA in the RHR system and to continue core cooling, the break must first be isolated and the reactor coolant system refilled. Since the RHR valves are not designed to close against the pressure differentials present during the blowdown, isolation of the affected loop and operation of the unaffected loop must be accomplished following blowdown. The RHR valves can be closed from the control room. No credit for local action is given because of the steam environment following the blowdown.

5.2.4 Important Individual Events and Uncertainties (Core Damage Frequency)

As discussed in Chapter 2, the process of developing a probabilistic model of a nuclear power plant involves the combination of many individual events (initiators, hardware failures, operator errors, etc.) into accident sequences and eventually into an estimate of the total frequency of core damage. After development, such a model can also be used to assess the importance of the individual events. The detailed studies underlying this report have been analyzed using several event importance measures. The results of the analyses using two measures, "risk reduction" and "uncertainty" importance, are summarized below.

- Risk (core damage frequency) reduction importance measure (internal events)

  The risk-reduction importance measure is used to assess the change in core damage frequency as a result of setting the probability of an individual event to zero. Using this measure, the following individual events were found to cause the greatest reduction in core damage frequency if their probabilities were set to zero:

  - Very small LOCA initiating event. The core damage frequency will be reduced by approximately 38 percent.

  - Operator fails to control sprays during a small LOCA. The core damage frequency will be reduced by approximately 37 percent.

  - Loss of offsite power initiating event. The core damage frequency will be reduced by approximately 21 percent.

  - Operator failure to properly align high-pressure recirculation. The core damage frequency will be reduced by approximately 15 to 20 percent.

  - Failure to recover diesel generators within 1 hour. The core damage frequency will be reduced by approximately 14 percent.

  - Failure to recover ac power within 1 hour. The core damage frequency will be reduced by approximately 13 percent.

  - Intermediate LOCA initiating events. The core damage frequency will be reduced by approximately 12 percent.

  - Small LOCA initiating events. The core damage frequency will be reduced by approximately 13 percent.

- Uncertainty importance measure (internal events)

  A second importance measure used to evaluate the core damage frequency analysis results is the uncertainty importance measure. For this measure, the relative contribution of the uncertainty of individual events to the uncertainty in total core damage frequency is calculated. Using this measure, the largest contributors to uncertainty in the results are the human error probabilities for failure to reconfigure the ECCS for high-pressure recirculation. All other events contribute relatively little to the uncertainty in overall core damage frequency.

5.3 Containment Performance Analysis

5.3.1 Results of Containment Performance Analysis

The Sequoyah primary containment consists of a pressure-suppression containment system, i.e., ice condenser, which houses the reactor pressure vessel, reactor coolant system, and the steam generators for the secondary side steam supply system. The containment system is comprised of a steel vessel surrounded by a concrete shield building enclosing an annular space. The internal containment volume, which has a total capacity of 1.2 million cubic feet, is divided into two major compartments connected by the ice condenser system, with the reactor coolant system occupying the lower compartment. The ice condenser is essentially a cold storage ice-filled room 50 feet in height, bounded on one side by the steel containment wall. The design basis pressure for Sequoyah's ice condenser containment is 10.8 psig, whereas its estimated mean failure pressure is 65 psig. This low-pressure design combined with the relatively small free volume made hydrogen control a design basis consideration, i.e., recombiners, and also a major consideration with respect to containment integrity for severe accidents, i.e., igniters and air-return fans. Similar to other containment design analyses for this study, the estimate of where and when Sequoyah's containment will fail relied heavily on the use of expert judgment to interpret and supplement the limited data available (Ref. 5.4).

The potential for early containment failure has been of considerable concern for Sequoyah since the steel containment has such a low design pressure. The principal mechanisms threatening the containment are hydrogen combustion effects, overpressurization due to direct containment heating, failure of the wall by direct contact with molten core material, and isolation failures.

The results of the Sequoyah containment analysis are summarized in Figures 5.4 and 5.5. Figure 5.4 displays information in which the conditional probabilities of ten containment-related accident progression bins; e.g., VB-early CF (during CD), are presented for each of five plant damage states. This information indicates that, on a frequency-weighted average,* the mean conditional probability from internal events of (1) early containment failure due to effects such as hydrogen combustion, direct containment heating, and wall contact failure is 0.07, (2) late containment failure due primarily to basemat meltthrough is 0.21, (3) containment bypass is 0.06, and (4) probability of no containment failure or no vessel breach is 0.66. It should be noted, however, that the conditional probabilities of early containment failure for the loss of offsite power (LOSP) plant damage state are considerably higher than the averaged values, i.e., about 0.13 for LOSP sequences involving vessel breach and 0.17 when those LOSP sequences having no vessel breach are included. Figure 5.5 further develops the conditional probability distribution of early containment failure for each of the plant damage states, providing the estimated range of uncertainties in the containment failure predictions. Overall conclusions that can be drawn from this information are discussed in Chapter 9. However, it should be noted that Sequoyah's early containment failure probability depends heavily on the accuracy of our predictions of core arrest probability, direct containment heating, hydrogen combustion, and wall attack effects.

Additional discussions on containment performance (for all studied plants) are provided in Chapter 9.

5.3.2 Important Plant Characteristics (Containment Performance)

Characteristics of the Sequoyah design and operation that are important to containment performance include:

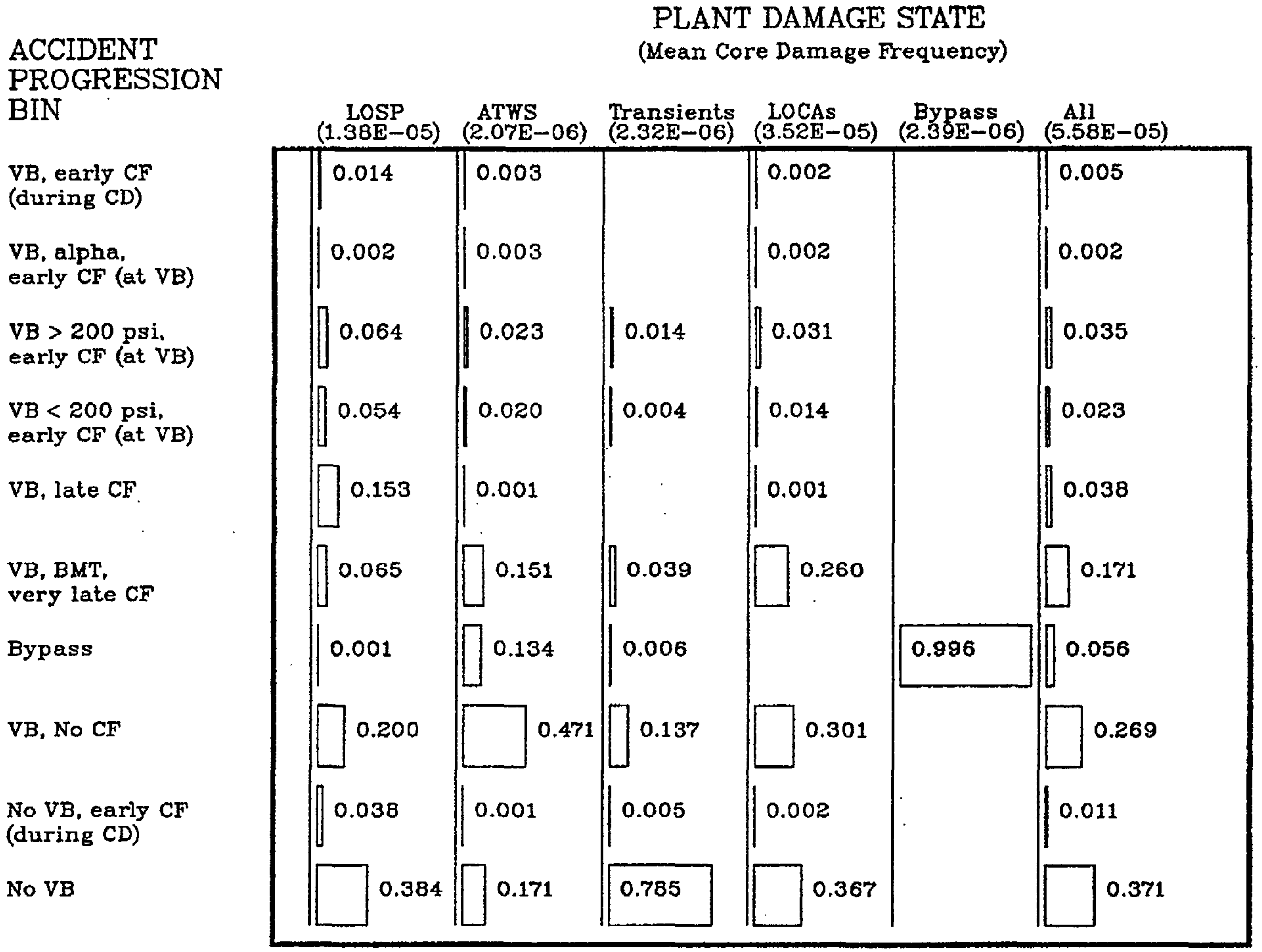
1. **Pressure-Suppression Design**

   The Sequoyah ice condenser suppression design can have a significant effect on certain accident sequence risk results. For example, the availability of ice in the ice condenser can reduce the risk significantly from events involving steam or direct containment heating threats to the containment. In contrast, its availability during some station blackout sequences can result in a potentially combustible hydrogen concentration at the exit of the ice bed. Further discussion of the ice condenser pressure-suppression system relative to other PWR dry containments is contained in Chapter 9.

2. **Hydrogen Ignition System**

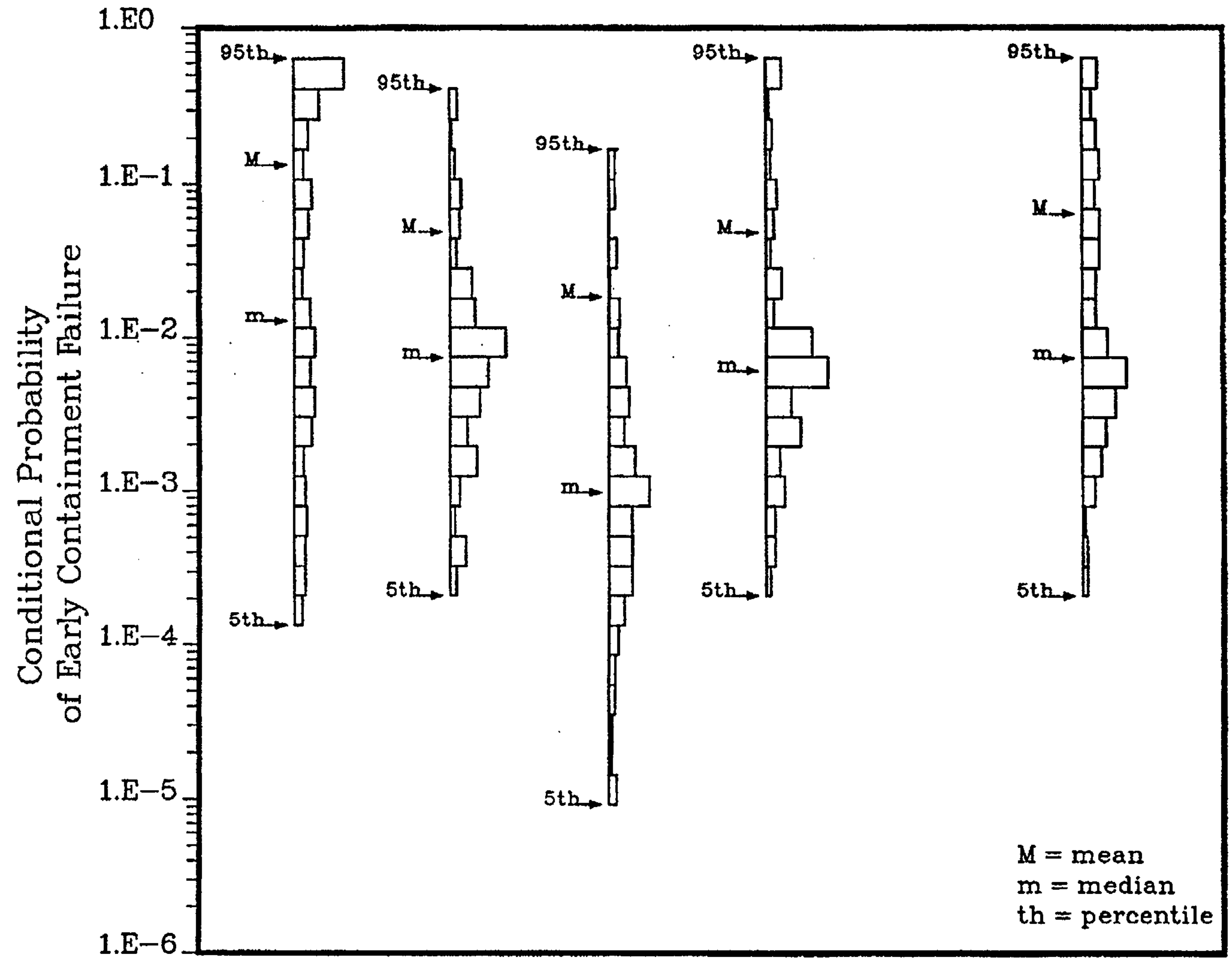
   The Sequoyah hydrogen ignition system will significantly reduce the threat to containment from uncontrolled hydrogen combustion

*Each value in the column in Figure 5.4 labeled "All" is obtained by calculating the products of individual accident progression bin conditional probabilities for each plant damage state and the ratio of the frequency of that plant damage state to the total core damage frequency.



BMT = Basemat Meltthrough
CF = Containment Failure
VB = Vessel Breach
CD = Core Degradation

Figure 5.4 Conditional probability of accident progression bins at Sequoyah.



PDS Group	LOSP	ATWS	Transients	LOCAs	Bypass	All
Core Damage Freq.	1.4E-05	2.1E-06	2.3E-06	3.5E-05	2.4E-06	5.6E-05

Figure 5.5 Conditional probability distributions for early containment failure at Sequoyah.

effects except for station blackout sequences. However, when power is recovered following a station blackout, if the igniters are turned on before the air-return fans have diluted the hydrogen concentration at or above the ice beds, the ignition could trigger a detonation or deflagration that could fail containment. These blackout sequences, however, represent a small fraction of the overall frequency of core damage.

3. **Lower Compartment Design**

   The design and construction of the seal table is such that if the reactor coolant system is at an elevated pressure upon vessel breach, the core debris is likely to get into the seal table room, which is directly in contact with the containment, and melt through the wall causing a break of containment. The design of the reactor cavity, however, does have the potential to cool the molten core debris and also mitigate the effects of potential direct containment heating events for those sequences where water is in the reactor cavity.

5.4 Source Term Analysis

5.4.1 Results of Source Term Analysis

The absolute frequencies of early containment failure from severe accident loads and of containment bypass are predicted to be similar for the Sequoyah plant (Ref. 5.2). Figure 5.6 illustrates the release fractions for an early containment failure accident progression bin. The mean values for the release of the volatile radionuclide groups are approximately 10 percent, indicative of an accident with the potential for causing early fatalities. The in-vessel releases in these accidents can be subject to decontamination by the ice bed or by containment sprays following release to the containment. The sprays require ac power and are, therefore, not available prior to power recovery in station blackout plant damage states. The decontamination factor of the ice bed is also affected by the unavailability of the recirculation fans during station blackout.

The location and mode of containment failure are particularly important for early containment failure accident progression bins. A substantial fraction of the early failures result in subsequent bypass of the ice bed. In particular, if the containment ruptures as the result of a sudden, high-pressure load, such as from hydrogen deflagration, the damage to the containment wall could be extensive and is likely to result in bypass.

In most accident sequences for Sequoyah, there is substantial water in the cavity that can either prevent core-concrete attack, if a coolable debris bed is formed, or mitigate the release of radionuclides during core-concrete attack by scrubbing in the overlaying water pool. As a result, a large release to the environment of the less volatile radionuclides that are released from fuel during core-concrete attack is unlikely for the Sequoyah plant.

In the station blackout plant damage state, containment failure can occur late in the accident as the result of hydrogen combustion following power recovery. Figure 5.7 illustrates the source terms for a late containment failure accident progression bin in which it is unlikely that water would be available to scrub the core-concrete releases. In this case, decontamination by the ice bed is important in mitigating the environmental release. As discussed previously, for very wide ranges of uncertainty covering many orders of magnitude, one or more high results can dominate the mean such that it falls above the 95th percentile.

5.4.2 Important Plant Characteristics (Source Term)

1. **Ice Condenser**

   In addition to condensing steam, the ice beds can trap radioactive aerosols and vapors in a severe accident. The extent of decontamination is very sensitive to the volume fraction of steam in the flowing gas, which in turn depends on whether the air-return fans are operational. For a single pass through the ice condenser with high steam fraction, the range of decontamination factor used in this study was from 1.3 to 35 with a median of 7 for the in-vessel release and less than half as effective for the core-concrete release. For the low steam fraction scenarios with a single pass through the ice beds, the lower bound was approximately 1.1, the upper bound 8, and the median 2. The values used for multiple passes through the ice bed when the containment is intact and the air-return fans are running are only slightly larger, with a median value of 3. Thus, the credit for ice bed retention is substantially less than the values used for the decontamination effectiveness of suppression pools in the BWRs.

2. **Cavity Configuration**

   The Sequoyah reactor cavity will be flooded if there is sufficient water on the containment floor to overflow into the cavity. If the contents of the refueling water storage tank are

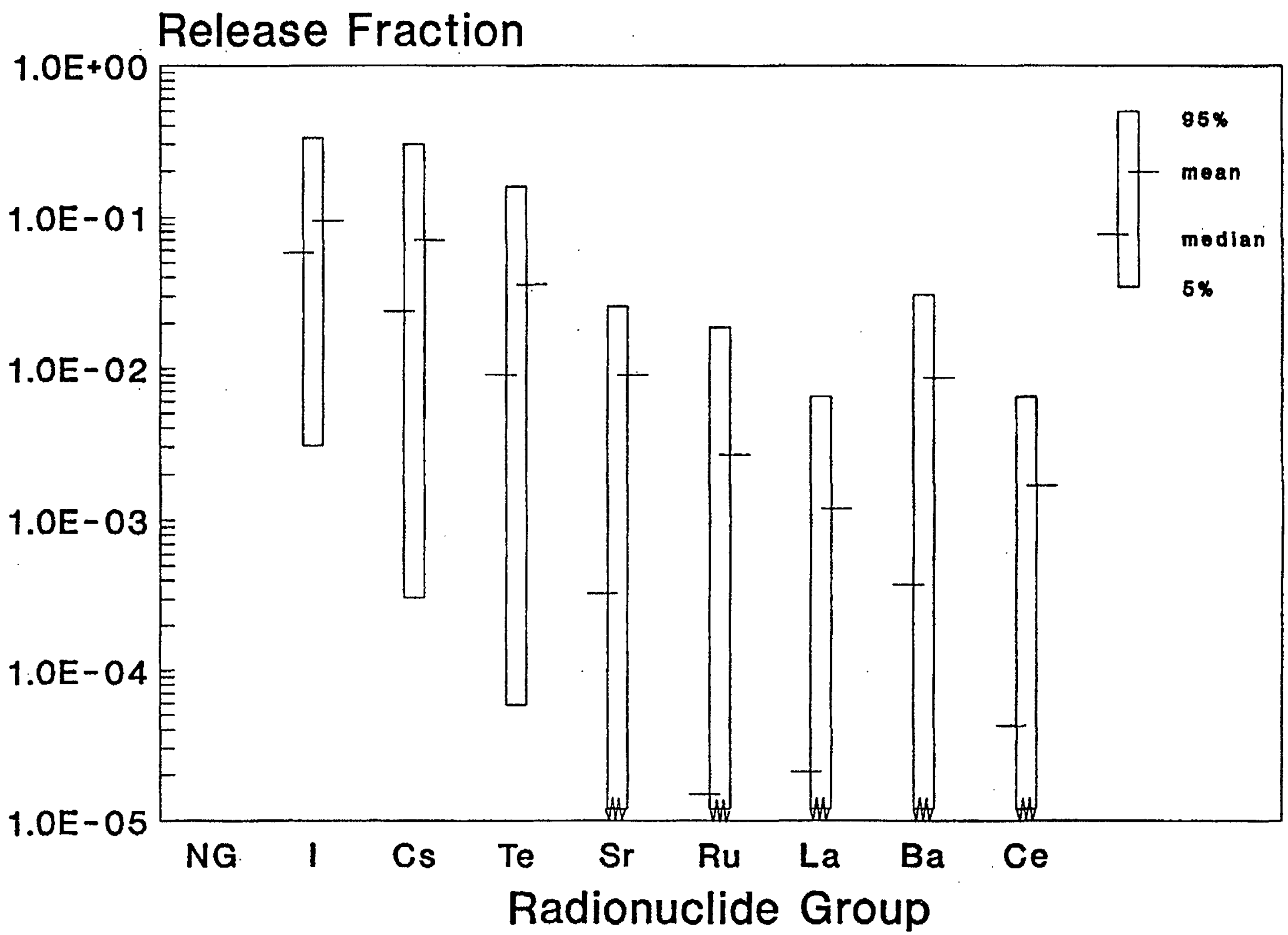


Figure 5.6 Source term distributions for early containment failure at Sequoyah.

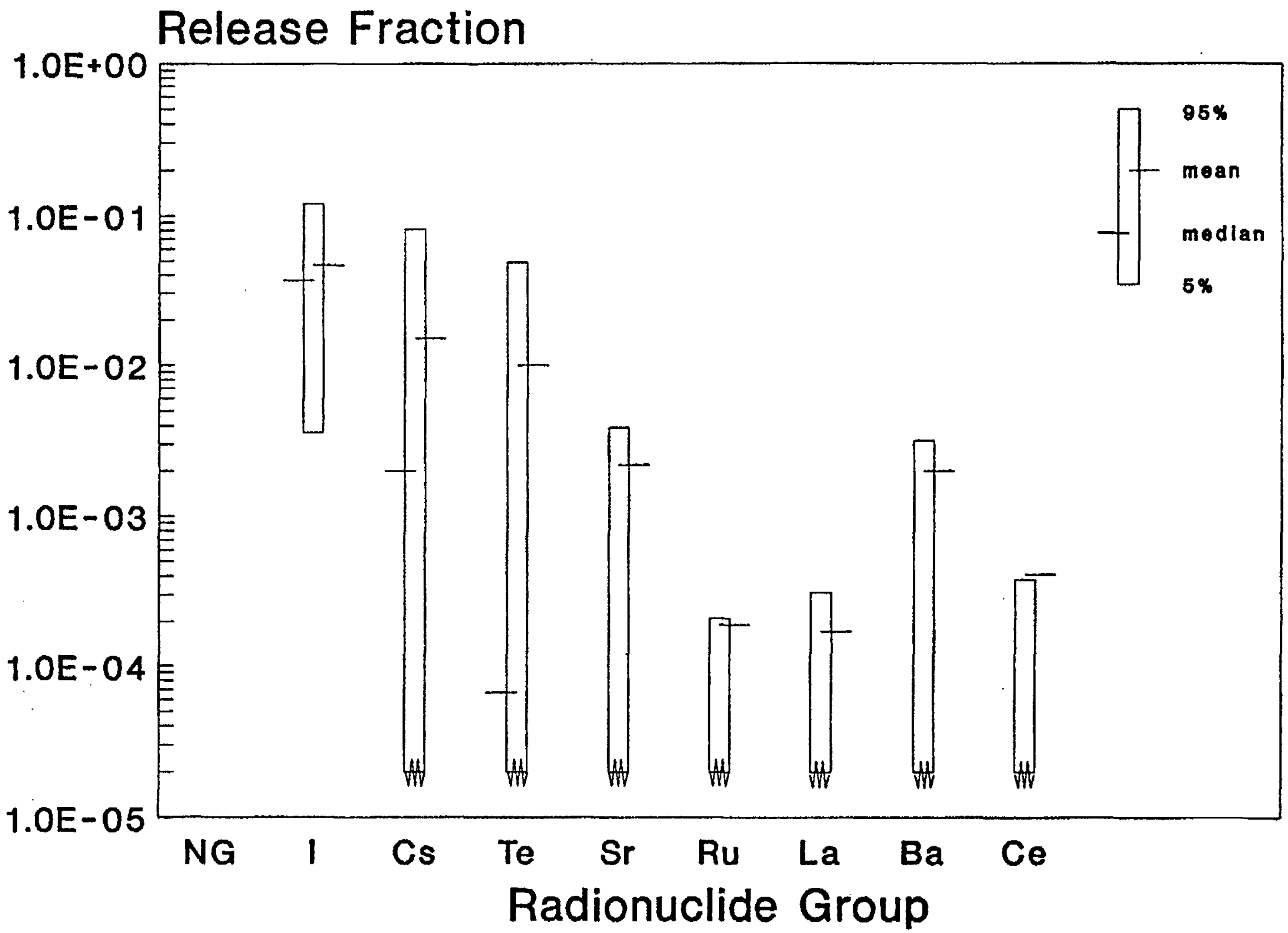


Figure 5.7 Source term distributions for late containment failure at Sequoyah.

discharged into the containment (e.g., by the spray system) and there is substantial ice melting, the water level in the cavity can be as high as 40 feet, extending to the level of the reactor coolant system hot legs. A decontamination factor for the deep water pool was used in the analyses, which ranged from approximately 4 to 9,000 with a median value of approximately 10 for the less volatile radionuclides released ex-vessel. If neither source of water to the containment is available, however, there will be no water in the cavity.

3. **Spray System**

   The Sequoyah containment has a spray system in the upper compartment to condense steam that bypasses the ice beds and for use after the ice has melted. As in the Surry plant, the spray system has the potential to dramatically reduce the airborne concentration of radioactive material if the containment remains intact for an extended period of time.

5.5 Offsite Consequence Results

Figure 5.8 displays the frequency distributions in the form of graphical plots of the complementary cumulative distribution functions (CCDFs) of four offsite consequence measures—early fatalities, latent cancer fatalities, and the 50-mile and entire site region population exposures (in person-rems). These CCDFs include contributions from all source terms associated with reactor accidents caused by internal initiating events. Four CCDFs, namely, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs, are shown for each consequence measure.

Sequoyah plant-specific and site-specific parameters were used in the consequence analysis for these CCDFs. The plant-specific parameters included source terms and their frequencies, the licensed thermal power (3423 MWt) of the reactor, and the appropriate physical dimensions of the power plant building complex. The site-specific parameters included exclusion area radius (585 meters), meteorological data for 1 full year collected at the site meteorological tower, the site region population distribution based on the 1980 census data, topography (fraction of the area that is land—the remaining fraction is assumed to be water), land use, agricultural practice and productivity, and other economic data for up to 1,000 miles from the Sequoyah plant.

The consequence estimates displayed in these figures have incorporated the benefits of the following protective measures: (1) evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), (2) early relocation of the remaining population only from the heavily contaminated areas both within and outside the 10-mile EPZ, and (3) decontamination, temporary interdiction, or condemnation of land, property, and foods contaminated above acceptable levels.

The population density within the Sequoyah 10-mile EPZ is about 120 persons per square mile. The average delay time before evacuation (after a warning prior to radionuclide release) from the 10-mile EPZ and average effective evacuation speed used in the analyses were derived from information contained in a utility-sponsored Sequoyah evacuation time estimate study (Ref. 5.5) and the NRC requirements for emergency planning.

The results displayed in Figure 5.8 are discussed in Chapter 11.

5.6 Public Risk Estimates

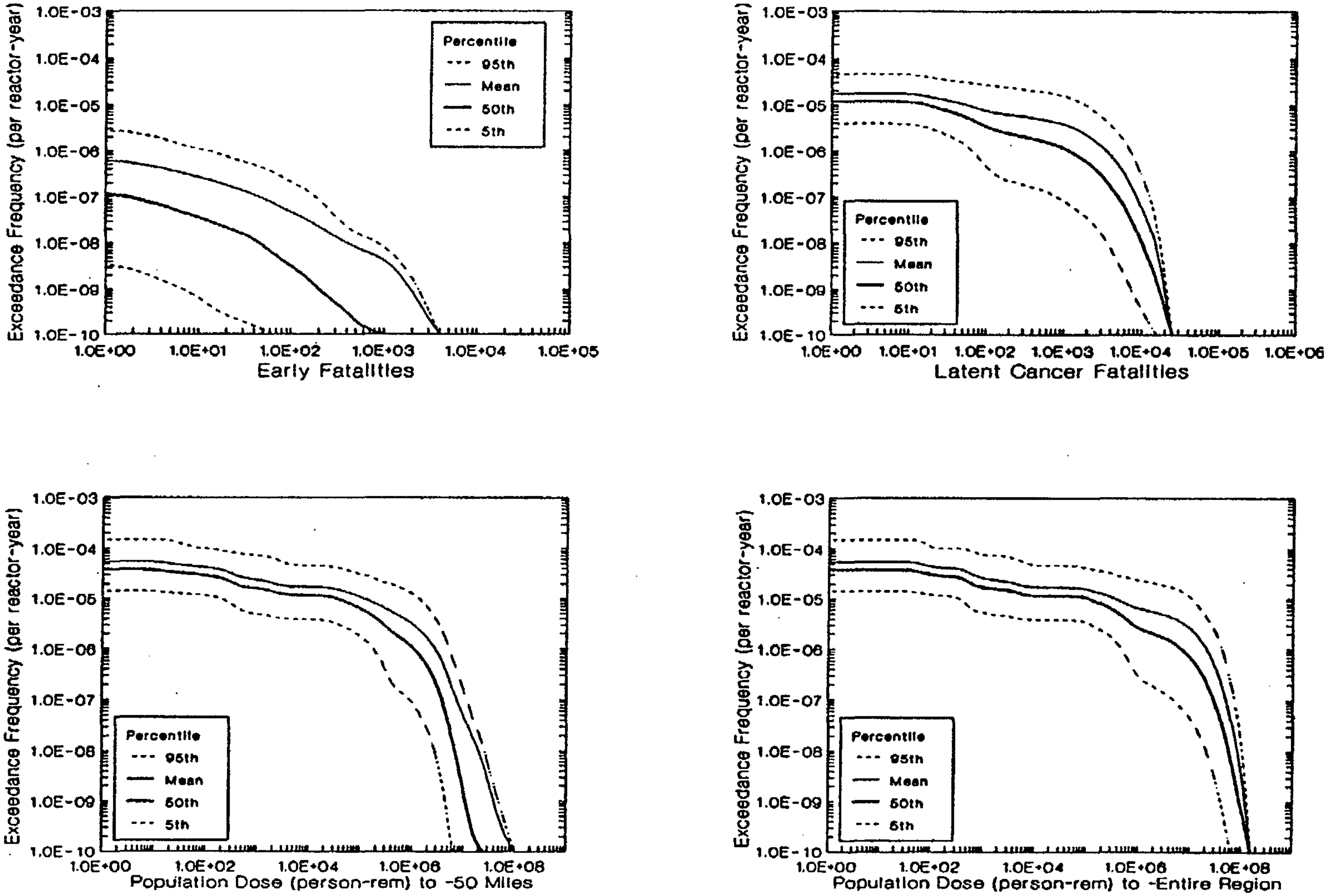
5.6.1 Results of Public Risk Estimates

A detailed description of the results of the Sequoyah risk is provided in Reference 5.2. For this summary report, results are provided for the following measures of public risk:

- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Sequoyah boundary, and
- Individual latent cancer fatality risk in the population within 10 miles of the Sequoyah site.

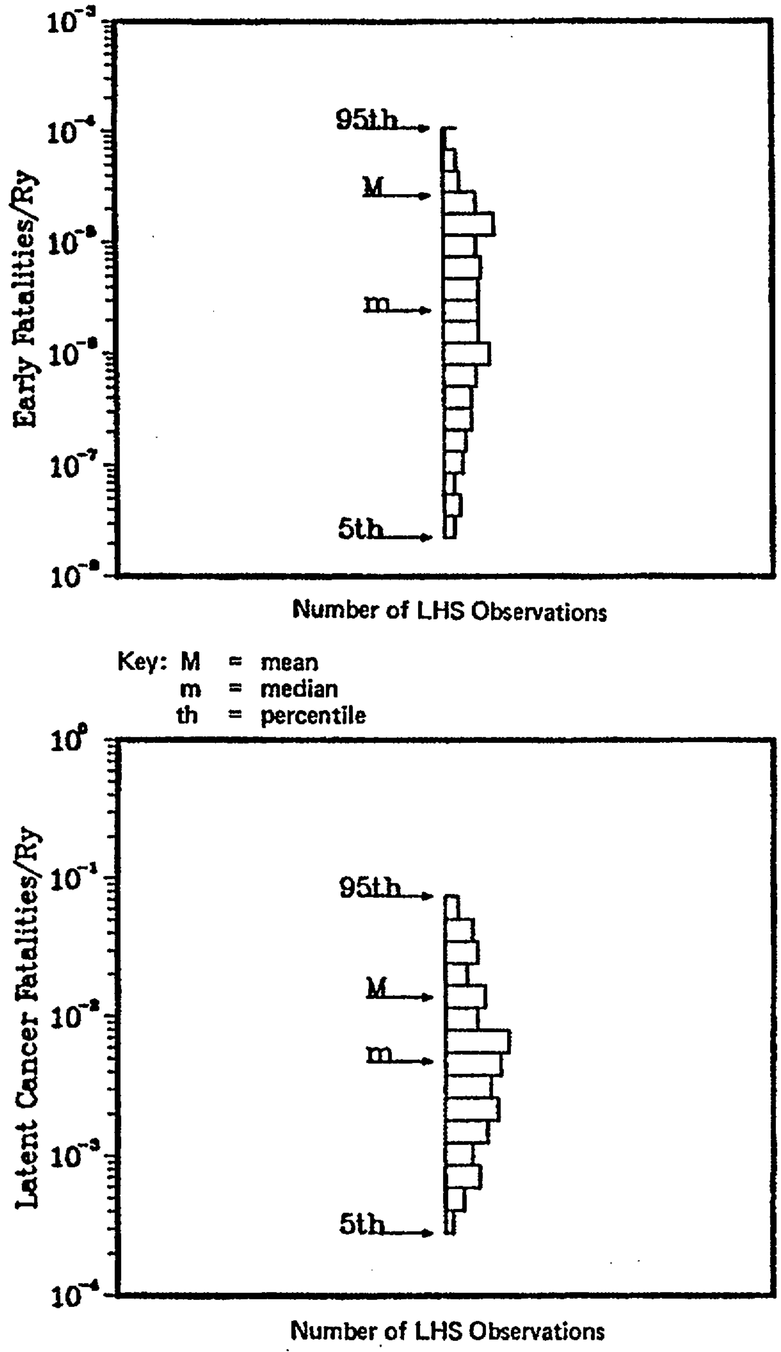
The first four of the above measures are commonly used measures in nuclear power plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 5.6).

The results of Sequoyah risk analysis using the above measures are shown in Figures 5.9 through 5.11. The figures display the variabilities in mean risks estimated from the meteorology-averaged mean values of the consequence measures. The



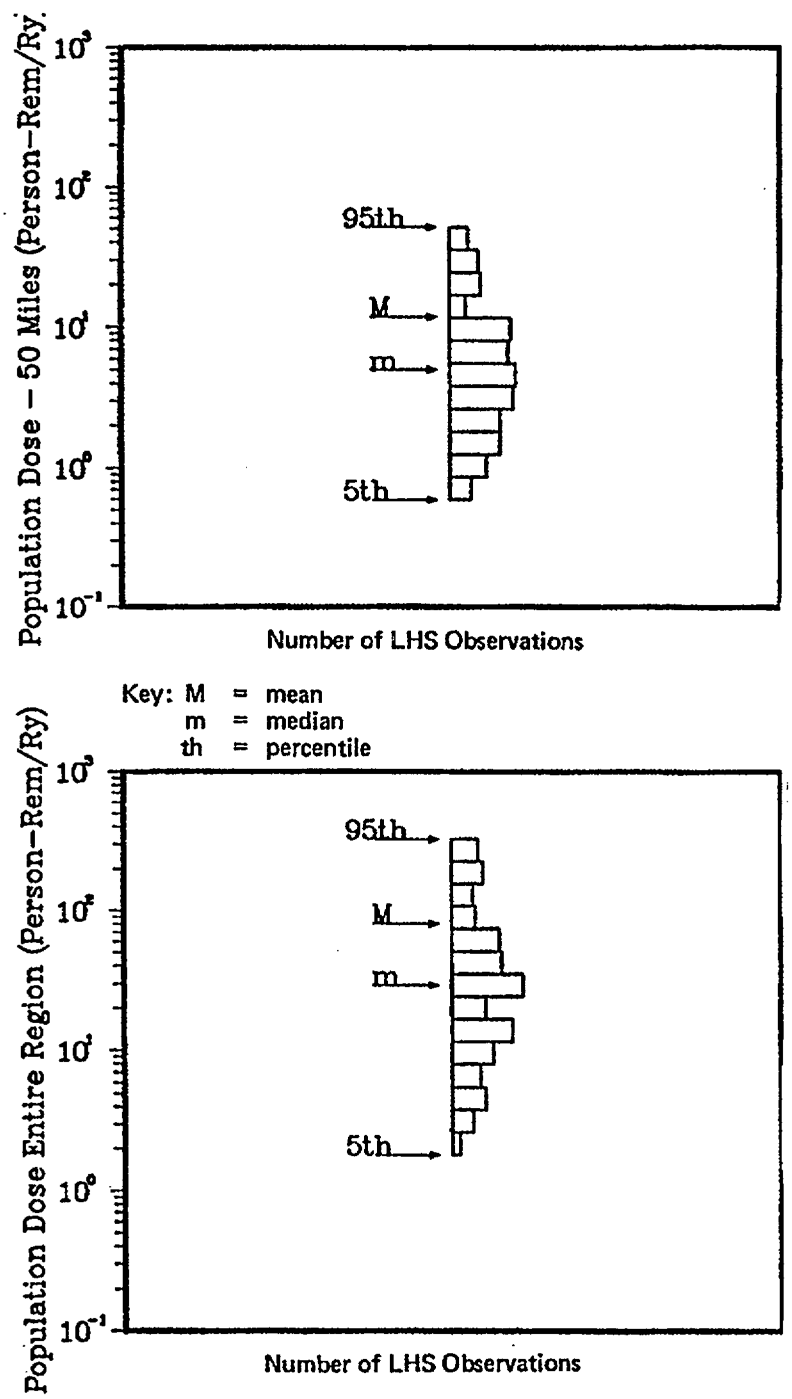
Note: As discussed in Reference 5.3, estimated consequences at frequencies at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 5.8 Frequency distributions of offsite consequence measures at Sequoyah (internal initiators).



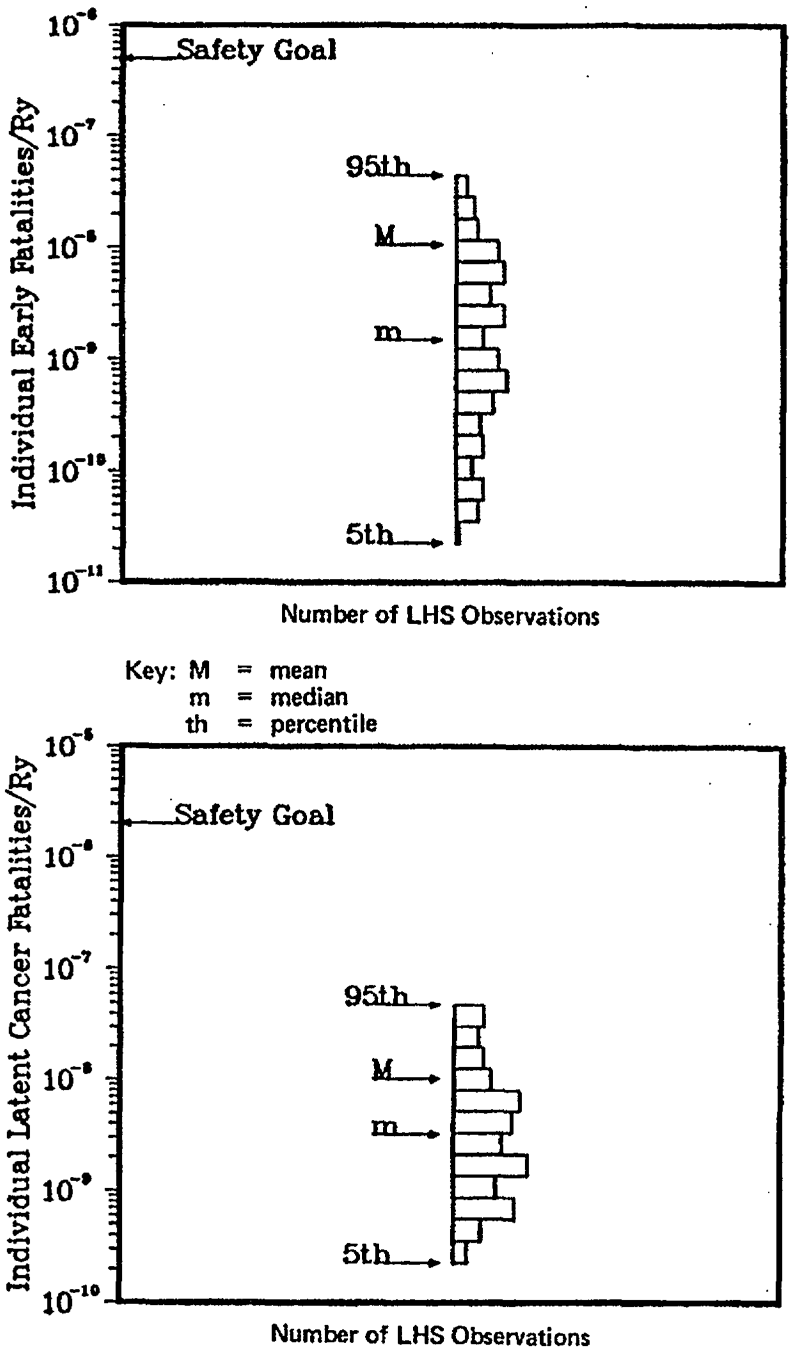
Note: As discussed in Reference 5.3, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 5.9 Early and latent cancer fatality risks at Sequoyah (internal initiators).



Note: As discussed in Reference 5.3, 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 5.10 Population dose risks at Sequoyah (internal initiators).



Note: As discussed in Reference 5.3, 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 5.11 Individual early and latent cancer fatality risks at Sequoyah (internal initiators).

early and latent cancer fatality risks, while quite low in absolute value, are higher than those from the Surry plant analysis (see Chapter 3). Other risk measure estimates are slightly higher than the Surry estimates. The individual early fatality and latent cancer fatality risks are well below the NRC safety goals. Detailed comparisons of results are provided in Chapter 12.

The risk results shown in Figure 5.9 have been analyzed to identify the relative contributions to mean risk of plant damage states and accident progression bins. These results are presented in Figures 5.12 and 5.13. As may be seen, the dominant contributor of early fatality risk is the bypass accident group, and particularly the interfacing-system LOCA (the V sequence); whereas the largest contributions to the latent cancer fatality risk came from the station blackout and bypass accident groups. For early fatality risk, the dominant contributor to risk is from accident sequences where the containment is bypassed, whereas, for latent cancer fatality risk, major accident progression bin contributors are bypass accidents and early containment failures. The accident progression bin involving accidents with no vessel breach appears as a contributor to early and latent cancer fatality risks. This bin possesses risk potential because of early containment failure due to hydrogen events from loss of offsite power in which ac power is recovered and breach is arrested and also from accidents involving steam generator tube rupture in which vessel breach is arrested.

5.6.2 Important Plant Characteristics (Risk)

Sequoyah risk analysis indicates that bypass sequences dominate early fatality risk. Timing is a key factor in this sequence in relation to evacuation. The release characteristics also contribute to the large effect of early fatalities because of the large magnitude of unmitigated source terms and the low energy of the first release. The low energy plume is not lofted over the evacuees but is held low to the ground after release. Another class of accidents that is important to early fatality risk is station blackout. It is the early containment failure (that is, failure of containment at and before vessel breach) associated with this accident class that contributes to early fatality risk.

An interfacing-system LOCA at Sequoyah will discharge into the auxiliary building where decontamination by automatically activated fire sprays is likely. Neither the probability of actuation nor the decontamination factor has been well established. The effects of an interfacing-system LOCA could either be higher or lower than those that have been calculated in this study.

Approximately equal contributions to latent cancer fatality risk come from station blackout and bypass. The bypass sequences contribute because of the large source terms and the bypass of any mitigating systems. The only other major contribution to latent cancer fatality comes from the LOCA sequences, mainly due to containment failures at vessel breach with high (> 200 psia) reactor coolant system pressure.

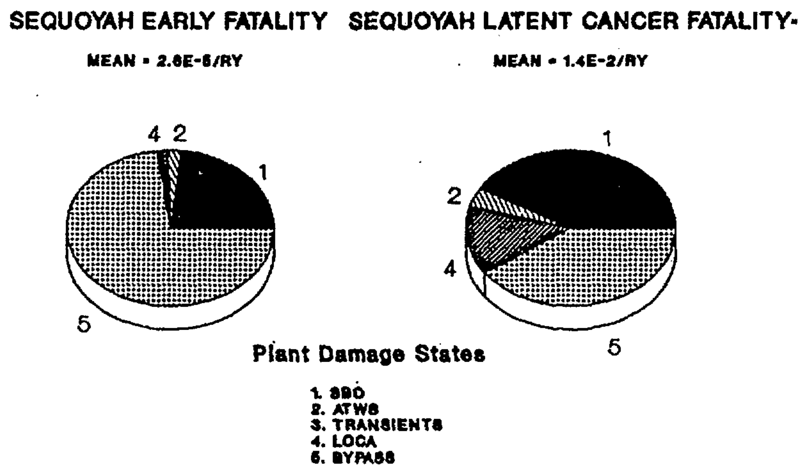


Figure 5.12 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Sequoyah (internal initiators).

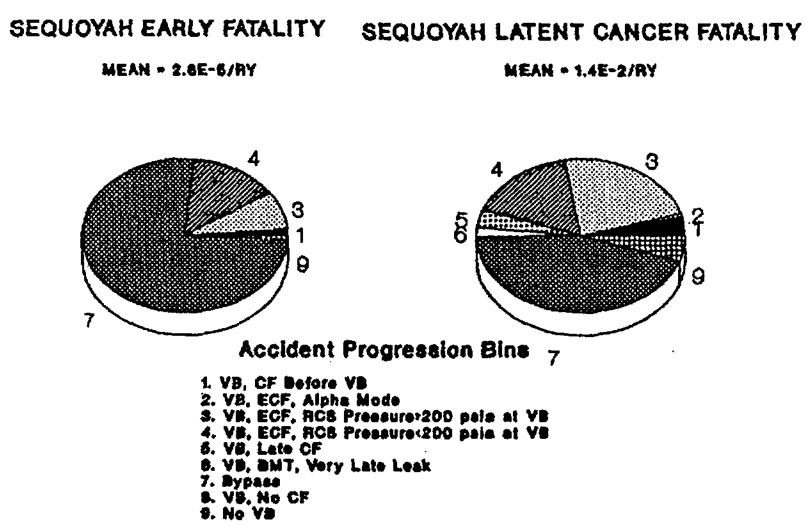


Figure 5.13 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Sequoyah (internal initiators).

REFERENCES FOR CHAPTER 5

5.1 R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

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

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

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

5.5 Tennessee Department of Transportation, "Evacuation Time Estimates with the Plume Exposure Pathway Emergency Planning Zone," prepared for Sequoyah Nuclear Plant, June 1987.

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

6. GRAND GULF PLANT RESULTS

6.1 Summary Design Information

The Grand Gulf Nuclear Station is a General Electric boiling water reactor (BWR-6) unit of 1250 MWe capacity housed in a Mark III containment. Grand Gulf Unit 1, constructed by Bechtel Corporation, began commercial operation in July 1985 and is operated by Entergy Operations. Some important design features of the Grand Gulf plant are described in Table 6.1. A general plant schematic is provided in Figure 6.1.

This chapter provides a summary of the results obtained in the detailed risk analyses underlying this report (Refs. 6.1 and 6.2). A discussion of perspectives with respect to these results is provided in Chapters 8 through 12.

6.2 Core Damage Frequency Estimates

6.2.1 Summary of Core Damage Frequency Estimates

The core damage frequency and risk analyses performed for this study considered accidents initiated only by internal events (Ref. 6.1). The core damage frequency results obtained are provided in tabular form in Table 6.2 and in graphical form, displayed as a histogram, in Figure 6.2. (Section 2.2.2 discusses histogram development.) This study calculated a total median core damage frequency from internal events of 1.2E-6 per year.

The Grand Gulf plant was previously analyzed in the Reactor Safety Study Methodology Applications Program (RSSMAP) (Ref. 6.3). A point estimate core damage frequency of 3.6E-5 from internal events was calculated in that study. A point estimate core damage frequency of 2.1E-6 was calculated in this analysis for purposes of comparison. A point estimate is calculated from the sum of all the cut-set frequencies, where each of the cut-set frequencies is the product of the point estimates (usually means) of the events in the cut sets.

6.2.1.1 Internally Initiated Accident Sequences

A detailed description of accident sequences important at the Grand Gulf plant is provided in Reference 6.1. For this report, the accident sequences described in that reference have been divided into two summary plant damage states. These are:

- Station blackout, and
- Anticipated transients without scram (ATWS).

The relative contributions of these groups to mean internal-event core damage frequency at Grand Gulf are shown in Figure 6.3. It may be seen that station blackout accident sequences as a class are the largest contributors to core damage frequency. It should be noted that the plant configuration as analyzed does not reflect the implementation of the station blackout rule.

Within the general class of station blackout accidents, the more probable combinations of failures leading to core damage are:

- Loss of offsite power occurs followed by the successful cycling of the safety relief valves (SRVs). Onsite ac power fails because all three diesel generators fail to start and run as a result of either hardware or common-cause faults. The loss of all ac power (i.e., station blackout) results in the loss of all core cooling systems (except for the reactor core isolation cooling (RCIC) system) and all containment heat removal systems. The RCIC system, which is ac independent, independently fails to start and run. All core cooling is lost, and core damage occurs in approximately 1 hour after offsite power is lost.
- Station blackout accident that is similar to the one described above except that one SRV fails to reclose and sticks open. Core damage occurs in approximately 1 hour after offsite power is lost.

In addition to these two short-term accident scenarios, this study also considered long-term station blackout accidents. In these accidents, loss of offsite power occurs and all three diesel generators fail to start or run. The safety relief valves cycle successfully and RCIC starts and maintains proper coolant level within the reactor vessel. However, ac power is not restored in these long-term scenarios, and RCIC eventually fails because of high turbine exhaust pressure, battery depletion, or other long-term effects. Core damage occurs approximately 12 hours after offsite power is lost.

Table 6.1 Summary of design features: Grand Gulf Unit 1.

1. Coolant Injection Systems	a. High-pressure core spray (HPCS) system provides coolant to reactor vessel during accidents in which system pressure remains high or low, with 1 train and 1 MDP.*
	b. Reactor core isolation cooling system provides coolant to the reactor vessel during accidents in which system pressure remains high, with 1 train and 1 TDP.*
	c. Low-pressure core spray system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with 1 train and 1 MDP.*
	d. Low-pressure coolant injection system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with 3 trains and 3 pumps.
	e. Standby service water crosstie system provides coolant makeup source to the reactor vessel during accidents in which normal sources of emergency injection have failed, with 1 train and 1 pump (for crosstie).
	f. Firewater system is used as a last resort source of low-pressure coolant injection to the reactor vessel, with 3 trains, 1 MDP,* 2 diesel-driven pumps.
	g. Control rod drive system provides backup source of high-pressure injection, with 2 pumps/238 gpm (total)/1103 psia.
	h. Automatic depressurization system (ADS) depressurizes the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant to the reactor vessel, with 8 relief valves/capacity of 900,000 lb/hr. In addition, there are 12 non-ADS relief valves.
	i. Condensate system used as a backup injection source.
2. Heat Removal Systems	a. Residual heat removal/suppression pool cooling system removes decay heat from the suppression pool during accidents, with 2 trains and 2 pumps.
	b. Residual heat removal/shutdown cooling system removes decay heat during accidents in which reactor vessel integrity is maintained and reactor is at low pressure, with 2 trains and 2 pumps.
	c. Residual heat removal/containment spray system suppresses pressure in the containment during accidents, with 2 trains and 2 pumps.
3. Reactivity Control Systems	a. Control rods.
	b. Standby liquid control system, with 2 parallel positive displacement pumps rated at 43 gpm per pump.

*TDP -Turbine-Driven Pump
MDP - Motor-Driven Pump

Table 6.1 (Continued)

4. Key Support Systems	a. dc power with 12-hour station batteries. b. Emergency ac power, with 2 diesel generators and third diesel generator dedicated to HPCS but with crossties. c. Suppression pool makeup system provides water from the upper containment pool to the suppression pool following a LOCA. d. Standby service water provides cooling water to safety systems and components.
5. Containment Structure	a. BWR Mark III. b. 1.67 million cubic feet. c. 15 psig design pressure.
6. Containment Systems	a. Containment venting is used when suppression pool cooling and containment sprays have failed to reduce primary containment pressure. b. Hydrogen igniter system prevents the buildup of large quantities of hydrogen inside the containment during accident conditions.

Within the general class of ATWS accidents, the most probable combination of failures leading to core damage is:

- Transient initiating event occurs followed by a failure to trip the reactor because of mechanical faults in the reactor protection system (RPS). The standby liquid control system (SLCS) is not actuated and the high-pressure core spray (HPCS) system fails to start and run because of random hardware faults. The reactor is not depressurized and therefore the low-pressure core cooling system cannot inject. All core cooling is lost; core damage occurs in approximately 20 to 30 minutes after the transient initiating event occurs.

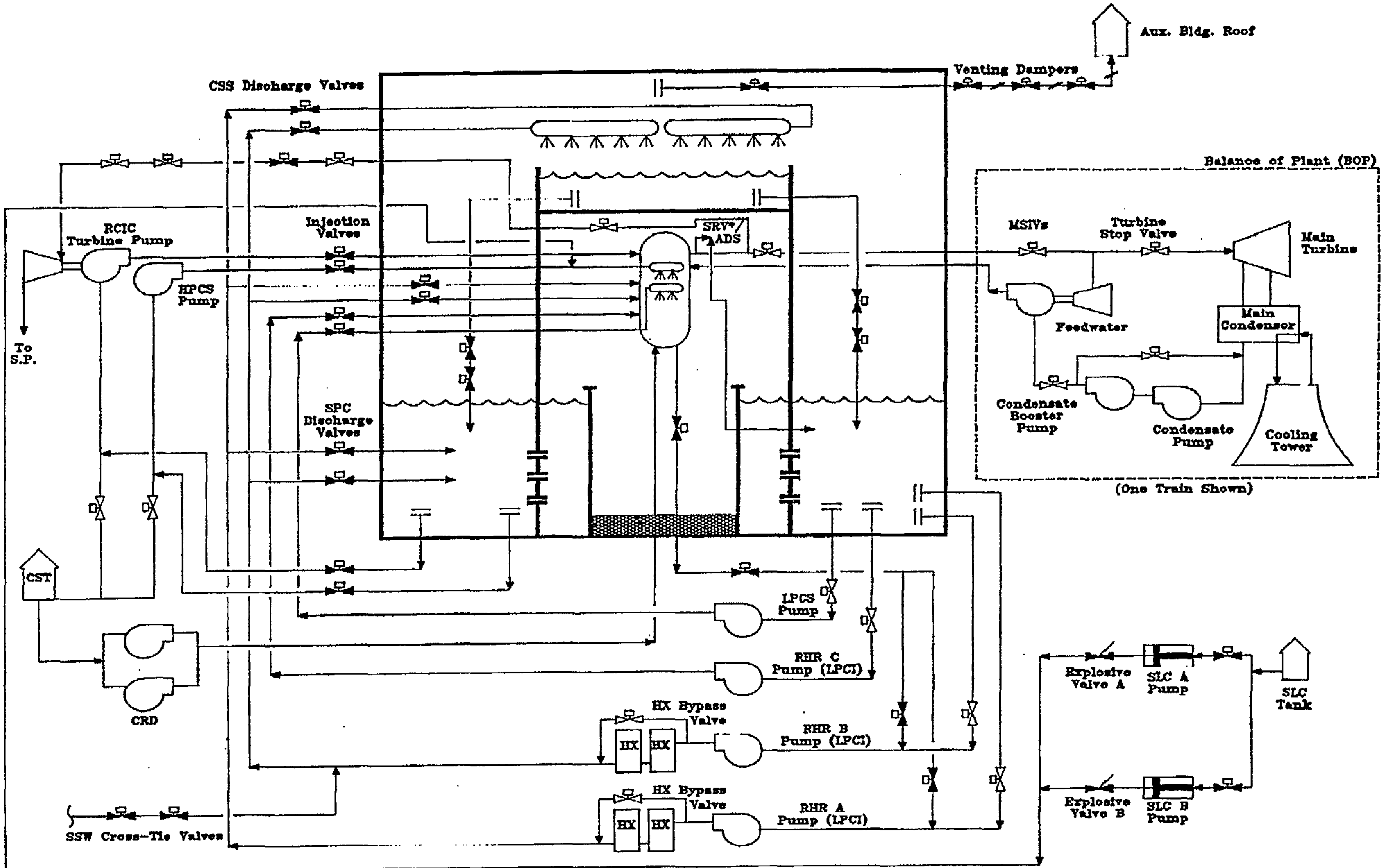
6.2.2 Important Plant Characteristics (Core Damage Frequency)

Characteristics of the Grand Gulf plant design and operation that have been found to be important in the analysis of core damage frequency include:

1. **Firewater System as Source of Coolant Makeup**

   The firewater system as a core coolant injection system can be used as a backup (last resort) source of low-pressure coolant injection to the reactor vessel. The system has two diesel-driven pumps, making it operational under station blackout conditions as long as dc power is available. The potential use of this system is estimated to reduce the total core damage frequency by approximately a factor of 1.5.

   The reason for the relatively small impact on the total core damage frequency is twofold. The firewater system is a low-pressure system; the reactor pressure must be maintained below approximately 125 psia for firewater to be able to inject. If an accident occurs in which core cooling is immediately lost, the core becomes uncovered in less time than that required to align and activate the firewater system. If core cooling is provided and then lost in the long term (e.g., at approximately greater than 4 hours after the start of the accident), firewater can provide sufficient makeup to prevent core damage. However, the dominant sequences at Grand Gulf are accidents where core cooling is lost immediately.



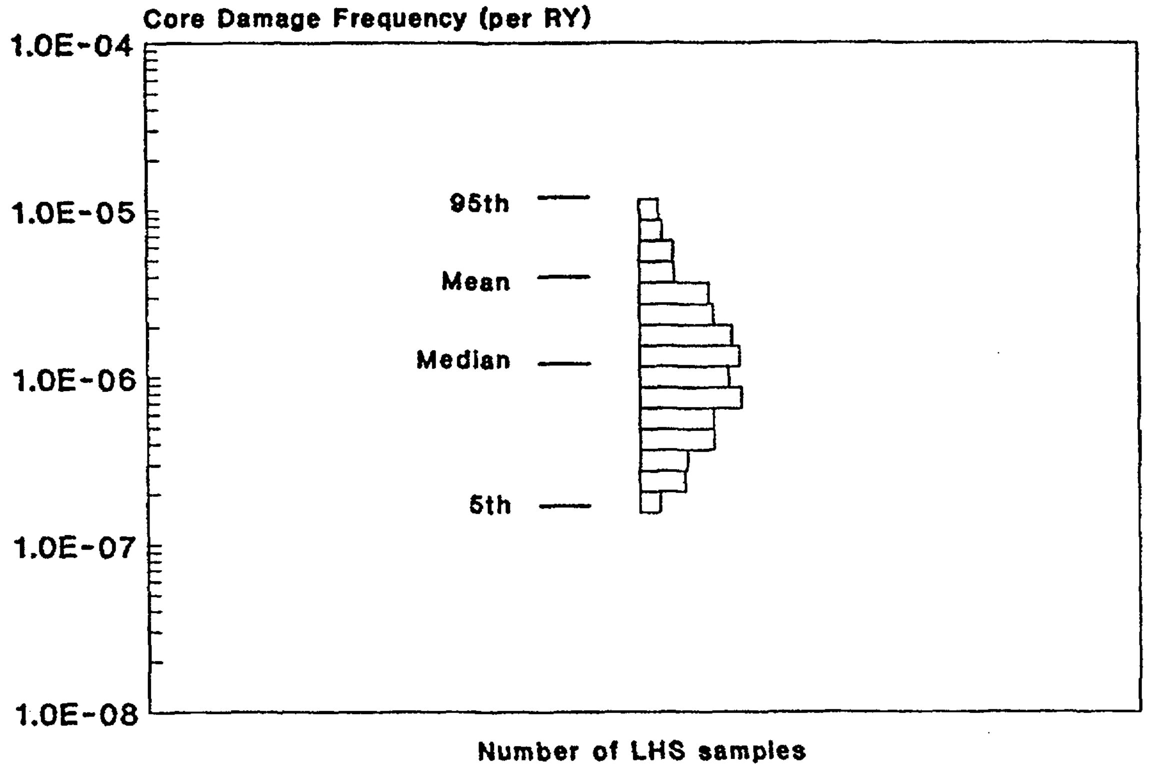
*Typical arrangement (8 ADS SRVs and 12 non-ADS SRVs)

Figure 6.1 Grand Gulf plant schematic.

Table 6.2 Summary of core damage frequency results: Grand Gulf.*

	5%	Median	Mean	95%
Internal Events	1.7E-7	1.2E-6	4.0E-6	1.2E-5
ATWS	8.5E-10	1.9E-8	1.1E-7	5.1E-7
Station Blackout	1.3E-7	1.1E-6	3.9E-6	1.1E-5

*As discussed in Reference 6.4, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).



Note: As discussed in Reference 6.4, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 6.2 Internal core damage frequency results at Grand Gulf.

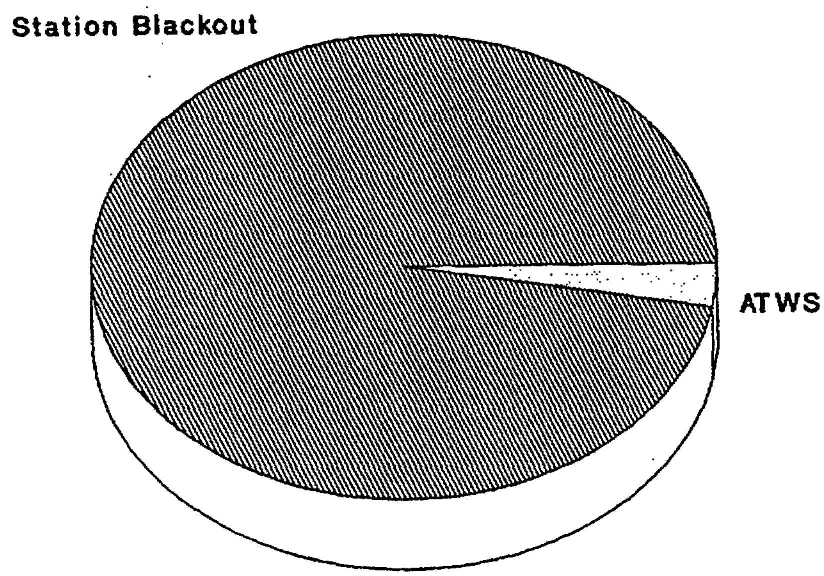


Figure 6.3 Contributors to mean core damage frequency from internal events at Grand Gulf.

2. **High-Pressure Core Spray (HPCS) System**

   The HPCS system consists of a single train with motor-operated valves and a motor-driven pump and provides coolant to the reactor vessel during accidents in which pressure is either high or low. The bearings and seals of the HPCS pump are cooled by the pumped fluid. If the temperature of this water exceeds design limits, the potential exists for the HPCS pump to fail. The bearings are designed to operate for no more than 24 hours at a temperature of 350°F. The peak temperature achieved in any of the accidents analyzed is approximately 325°F. Even if the seals were to experience some leakage, the resultant HPCS room environment would not adversely affect the operability of the pump. The availability of an HPCS system with such design characteristics is estimated to reduce the core damage frequency by approximately a factor of 7. The HPCS is powered by a dedicated diesel generator when required so that this system is truly an independent system.

3. **Capability of Pumps to Operate with Saturated Water**

   The emergency core cooling pumps that depend on the pressure-suppression pool as their water source during accident conditions have been designed to pump saturated water. Thus, if the pool becomes saturated because of containment venting or containment failure, the core cooling systems are not lost but can continue to cool the reactor core.

4. **Redundancy and Diversity of Water Supply Systems**

   At Grand Gulf, there are many redundant and diverse systems to provide water to the reactor vessel. They include:

   HPCS with 1 pump;

   Reactor core isolation cooling (RCIC) with 1 pump;

   Control rod drive (CRD) with 2 pumps (both are required for core cooling);

Condensate with 3 pumps;

Low-pressure core spray (LPCS) with 1 pump;

Low-pressure coolant injection (LPCI) with 3 pumps;

Standby service water (SSW) crosstie with 1 pump; and

Firewater system with 3 pumps.

Because of the redundancy of systems for LOCAs and transients, core cooling loss as a result of independent random failures is of low probability. However, in a station blackout, except for RCIC and firewater, the core cooling systems are lost with a probability of unity because they require ac power.

5. **Redundancy and Diversity of Heat Removal Systems**

At Grand Gulf there are several diverse means for heat removal. These systems are:

Main steam/feedwater system with 3 trains;

Suppression pool cooling mode of residual heat removal (RHR) with 2 trains;

Shutdown cooling mode of RHR with 2 trains;

Containment spray system mode of RHR with 2 trains; and

Containment venting with 1 train.

Although the various modes of RHR have common equipment (e.g., pumps), there is still enough redundancy and diversity that, for non-station-blackout accidents, independent random failures again are small contributors to the core damage frequency.

6. **Automatic and Manual Depressurization System**

The automatic depressurization system (ADS) is designed to depressurize the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant to the reactor vessel. The ADS consists of eight safety relief valves capable of being manually opened. The operator may manually initiate the ADS or may depressurize the reactor vessel, using the 12 relief valves that are not connected to the ADS logic. The ADS valves are located inside the containment.

6.2.3 Important Operator Actions

The emergency operating procedures (EOPs) at Grand Gulf direct the operator to perform certain actions depending on the plant conditions or symptoms (e.g., reactor vessel level below the top of active fuel). Different accident sequences can have similar symptoms and therefore the same "recovery" actions. Operator actions that are important include the following:

- Actuate core cooling

  In an accident where feedwater is lost (which includes condensate), the reactor water level starts to decrease. When Level 2 (-41.6 inches) is reached, high-pressure core spray (HPCS) and reactor core isolation cooling (RCIC) should be automatically actuated. If Level 1 (-150.3 inches) is reached, the ADS should occur with automatic actuation of the low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI). If the reactor level sensors are miscalibrated, these systems will not automatically actuate. The operator has many other indications to determine both the reactor water level and the fact that core coolant makeup is not occurring. Manual actuation of these systems is required if such failures occur in order to prevent core damage.

- Establish containment heat removal

  Besides core cooling, the operator must also establish containment heat removal (CHR). If an accident occurs, the EOPs direct the operator to initiate the suppression pool cooling mode of RHR when the suppression temperature reaches 95°F. The operator closes the LPCI valves and the heat exchanger bypass valves and opens the suppression pool discharge valves. He also ensures that the proper service water system train is operating. With suppression pool cooling (SPC) functioning, CHR is being performed. If system faults preclude the use of SPC, the operator has other means to provide CHR. He can actuate other modes of RHR such as shutdown cooling or containment spray, or the operator can vent the containment to remove the energy.

- Establish room cooling through natural circulation

  The heating, ventilating, and air conditioning (HVAC) system provides room cooling support to a variety of systems. If HVAC is lost, design limits can be exceeded and equipment (i.e., pumps) can fail. If these conditions occur, the operator can open doors to certain rooms and establish a natural circulation/ventilation that prevents the room temperature from exceeding the design limits of the equipment.

For station blackout accidents, there are certain actions that can be performed by the operating crew as follows:

- Crosstie division 1 or 2 loads to HPCS diesel generator

  In a station blackout where the HPCS diesel generator is available, the operator can choose to crosstie this diesel to one of the other divisions. The operator might choose this option when (1) the HPCS system fails and core cooling is required, or (2) in the long term (e.g., longer than 8 hours) containment heat removal is required to prevent containment failure. If the operator chooses to crosstie, the operator must shed all the loads from the HPCS diesel and then open and close certain breakers. He can then load certain systems from either division 1 or from division 2.

- Align firewater

  In an accident, particularly station blackout, where core cooling was initially available (for approximately 4 hours) and then lost, the firewater system can provide adequate core cooling. The operator must align the firewater hoses to the proper injection lines (described in the procedure) and then open the injection valves.

- Depressurize reactor via RCIC steam line

  In a station blackout, the diesel generators have failed and only dc power is available (in certain sequences). If core cooling is being provided with firewater, then the reactor must remain at low pressure, which requires that at least one safety relief valve (SRV) must remain open. For the SRV to remain open, dc power is required. However, without the diesel generator recharging the battery, the battery will eventually deplete, the SRV will close, and the reactor will repressurize, which causes the loss of the firewater. The operator can maintain the reactor pressure low by opening the valves on the RCIC steam line. This provides a vent path from the reactor to the suppression pool.

- Recovering ac power

  Station blackout is caused by the loss of all ac power, both offsite and onsite power. Restoring offsite power or repairing the diesel generators was included in the analysis. The quantification of these human failure events was derived from historical data (i.e., actual time required to perform these repairs) and not by performing human reliability analysis on these events.

Transients where reactor trip does not occur (i.e., ATWS) involve accident sequences where the phenomena are more complex. The operator actions were evaluated in more detail (Ref. 6.5) than for the regular transient-initiated accident. These actions include the following:

- Manual scram

  A transient occurs that demands the reactor to be tripped, but the reactor protection system (RPS) fails because of electrical faults. The operator can then manually trip the reactor by first rotating the collar on proper scram buttons and then depressing the buttons, or he can put the reactor mode switch in the "shutdown" position.

- Insert rods manually

  If the electrical faults fail both the RPS and the manual trip, the operator can manually insert the control rods one a time.

- Actuate standby liquid control (SLC) system

  With the reactor not tripped, reactor power remains high; the reactor core is not at decay heat levels. This can present problems since the containment heat removal systems are only designed to decay heat removal capacity. However, the SLC system (manually actuated) injects sodium pentaborate that reduces reactor power to decay heat levels. The EOPs direct the operator to actuate SLC if the reactor power is above 4 percent and before the suppression pool temperature reaches 110°F. The operator obtains the SLC keys (one per pump) from the shift supervisor's desk, inserts the keys into the switches, and turns both to the "on" position.

- Inhibit automatic depressurization system (ADS)

  In an ATWS condition, the operator is directed to inhibit the ADS if he has actuated

SLC. The operator must put both ADS switches (key locked) in the inhibit mode.

- Manually depressurize reactor

  If HPCS fails, inadequate high-pressure core cooling occurs. When Level 1 is reached, ADS will not occur because the ADS was inhibited, and the operator must manually depressurize so that low-pressure core cooling can inject. The operator can either press the ADS button (which overrides the inhibit) or manually open one SRV at a time.

6.2.4 Important Individual Events and Uncertainties (Core Damage Frequency)

As discussed in Chapter 2, the process of developing a probabilistic model of a nuclear power plant involves the combination of many individual events (initiators, hardware failures, operator errors, etc.) into accident sequences and eventually into an estimate of the total frequency of core damage. After development, such a model can also be used to assess the importance of the individual events. The detailed studies underlying this report have been analyzed using several event importance measures. The results of the analyses using two measures, "risk reduction" and "uncertainty" importance, are summarized below.

- Risk (core damage frequency) reduction importance measure (internal events)

  The risk-reduction importance measure is used to assess the change in core damage frequency as a result of setting the probability of an individual event to zero. Using this measure, the following individual events were found to cause the greatest reduction in core damage frequency if their probabilities were set to zero.

  - Loss of offsite power initiating event. The core damage frequency would be reduced by approximately 92 percent.

  - Failure to restore offsite power in 1 hour. The core damage frequency would be reduced by approximately 70 percent.

  - Failure of the RCIC turbine-driven pump to run. The core damage frequency would be reduced by approximately 48 percent.

  - Failure to repair hardware faults of diesel generator in 1 hour. The core damage frequency would be reduced by approximately 46 percent.

  - Failure of a diesel generator to start. The core damage frequency would be reduced by approximately 23 to 32 percent, depending on the diesel generator.

  - Common-cause failure of the vital batteries. The core damage frequency would be reduced by approximately 20 percent.

- Uncertainty importance measure (internal events)

  A second importance measure used to evaluate the core damage frequency analysis results is the uncertainty importance measure. For this measure, the relative contribution of the uncertainty of individual events to the uncertainty in total core damage frequency is calculated. Using this measure, the following events were found to be most important:

  - Loss of offsite power;

  - Failure of the diesel generators to run, given start;

  - Individual and common-cause failure of the diesel generators to start;

  - Standby service water motor-operated valves (MOVs) fail to open; and

  - High-pressure core spray and RCIC MOVs fail to function.

6.3 Containment Performance Analysis

6.3.1 Results of Containment Performance Analysis

The Grand Gulf pressure-suppression containment design is of the Mark III type in which the reactor vessel, reactor coolant circulating loops, and other branch connections to the reactor coolant system are housed within the drywell structure. The drywell structure in turn is completely contained within an outer containment structure with the two volumes communicating through the water-filled vapor suppression pool. The outer containment building is a steel-lined reinforced concrete structure with a volume of 1.67 million cubic feet that is designed for a peak pressure of 15 psig resulting from a reactor coolant system loss-of-coolant accident. For this same design basis accident, the inner concrete drywell structure is designed for a peak pressure of 30 psig. The mean failure pressure for Grand Gulf's containment structure has been estimated to be 55 psig. This estimated containment failure pressure for Grand Gulf is much lower than the Peach Bottom Mark I estimated failure pressure of 148 psig; however, Grand Gulf's free volume is several times larger. The availability of Grand Gulf's large volume removed the design basis need to inert the containment against failure from hydrogen combustion following design basis accidents; however, subsequent severe accident considerations after the TMI accident resulted in the installation of hydrogen igniters. For the severe accident sequences developed in this analysis, hydrogen combustion remains the major threat to Grand Gulf's containment integrity (in the station blackout accidents dominating the frequency of core damage, igniters are not operable). Similar to other containment design analyses, the estimate of where and when Grand Gulf's containment system will fail relied heavily on the use of expert judgment to interpret the limited data available.

The potential for early containment and/or drywell failure for Grand Gulf as compared to Peach Bottom's Mark I suppression-type containment involves significantly different considerations. Of particular significance with regard to the potential for large radioactive releases from Grand Gulf is the prediction of the combined probabilities of simultaneous early containment and drywell failures, which in turn produce a direct radioactive release path to the environment. The results of these analyses for Grand Gulf are shown in Figures 6.4 and 6.5. Figure 6.4 displays information in which the eight conditional probabilities of containment-related accident progression bins; e.g., VB-early CF-no SPB, are presented for each of four plant damage states, e.g., ATWS. This information indicates that, on a plant damage state frequency-weighted average* for internally initiated events, there are mean conditional probabilities of (1) 0.23 that the integrity of the drywell and the outer containment will be sufficiently affected that substantial bypass of the suppression pool will occur; (2) 0.24 for early containment failure with no bypass of the suppression pool pathway from the drywell; (3) 0.12 for late containment failure with pool bypass; (4) 0.23 for late containment failure but no pool bypass; and (5) 0.09 for no containment failure.

*Each value in the column in Figure 6.4 labeled "All" is a frequency-weighted average obtained by summing the products of individual accident progression bin conditional probabilities for each plant damage state and the ratio of the frequency of that plant damage state to the total core damage frequency.

Further examination of these data, broken down on the basis of the timing of reactor vessel breach and the nature of the containment threat, indicate: (1) prior to reactor vessel breach, hydrogen combustion and slow steam overpressurization effects lead to frequency-weighted mean conditional probabilities of containment failure of 0.20 and 0.05, respectively; (2) at reactor vessel breach, hydrogen combustion effects lead to a 0.24 conditional mean probability of containment failure; (3) prior to reactor vessel breach, hydrogen combustion effects lead to 0.12 conditional mean probability of drywell failure; (4) at reactor vessel breach, steam explosion and direct containment heating effects can lead to pedestal failures and a 0.16 conditional mean probability of drywell failure from both pedestal and overpressure effects; and (5) dynamic loads from hydrogen detonations have a small effect on the structural integrity of either the containment or the drywell.

Figure 6.5 further displays plots of Grand Gulf's conditional probability distribution for each plant damage state, thereby providing the estimated range of uncertainties in the outer containment failure predictions. The important conclusions that can be drawn from the information are (1) there is a relatively high mean conditional probability of early containment failure with a large bypass of the suppression pool's scrubbing effects, i.e., 0.23; (2) there is a high mean probability of early containment failure, i.e., 0.48; and (3) the principal threat to the combined efficacy of the Mark III containment and drywell is hydrogen combustion effects.

Additional discussions on containment performance (for all studied plants) are provided in Chapter 9.

6.3.2 Important Plant Characteristics (Containment Performance)

Characteristics of the Grand Gulf design and operation that are important during core damage accidents include:

1. **Drywell-Wetwell Configuration**

   With the reactor vessel located inside the drywell, which in turn is completely surrounded by the outer containment building, there needs to be a combination of failures in both structures to provide a direct release path to the environment that bypasses the suppression pool, e.g., hydrogen combustion

SUMMARY ACCIDENT PROGRESSION BIN GROUP	SUMMARY PDS GROUP (Mean Core Damage Frequency)				
	STSB (3.85E-06)	LTSB (1.04E-07)	ATWS (1.12E-07)	Transients (1.87E-08)	All (4.09E-06)
VB, early CF, early SPB, no CS	0.168	0.292	0.006	0.011	0.158
VB, early CF, early SPB, CS	0.031	0.017	0.237	0.202	0.049
VB, early CF, late SPB	0.006	0.005	0.003	0.003	0.007
VB, early CF, no SPB	0.182	0.531	0.505	0.331	0.218
VB, late CF	0.308	0.129	0.074	0.232	0.284
VB, venting	0.032	0.003	0.109	0.075	0.038
VB, No CF	0.053	0.003	0.036	0.092	0.050
No VB	0.201	0.015	0.025	0.050	0.180

CF = Containment Failure
CS = Containment Sprays
CV = Containment Venting
SPB = Suppression Pool Bypass
VB = Vessel Breach

Figure 6.4 Conditional probability of accident progression bins at Grand Gulf.

impairing the function of both the drywell and containment.

2. **Containment Volume**

The Grand Gulf containment volume is much larger than that of a Mark I containment and as such can accommodate significant quantities of noncombustible gases before failure even though its estimated failure pressure is less than half that of a Mark I containment. Its low design pressure, however, makes it susceptible to failure from hydrogen combustion effects in those cases where the igniters are not working.

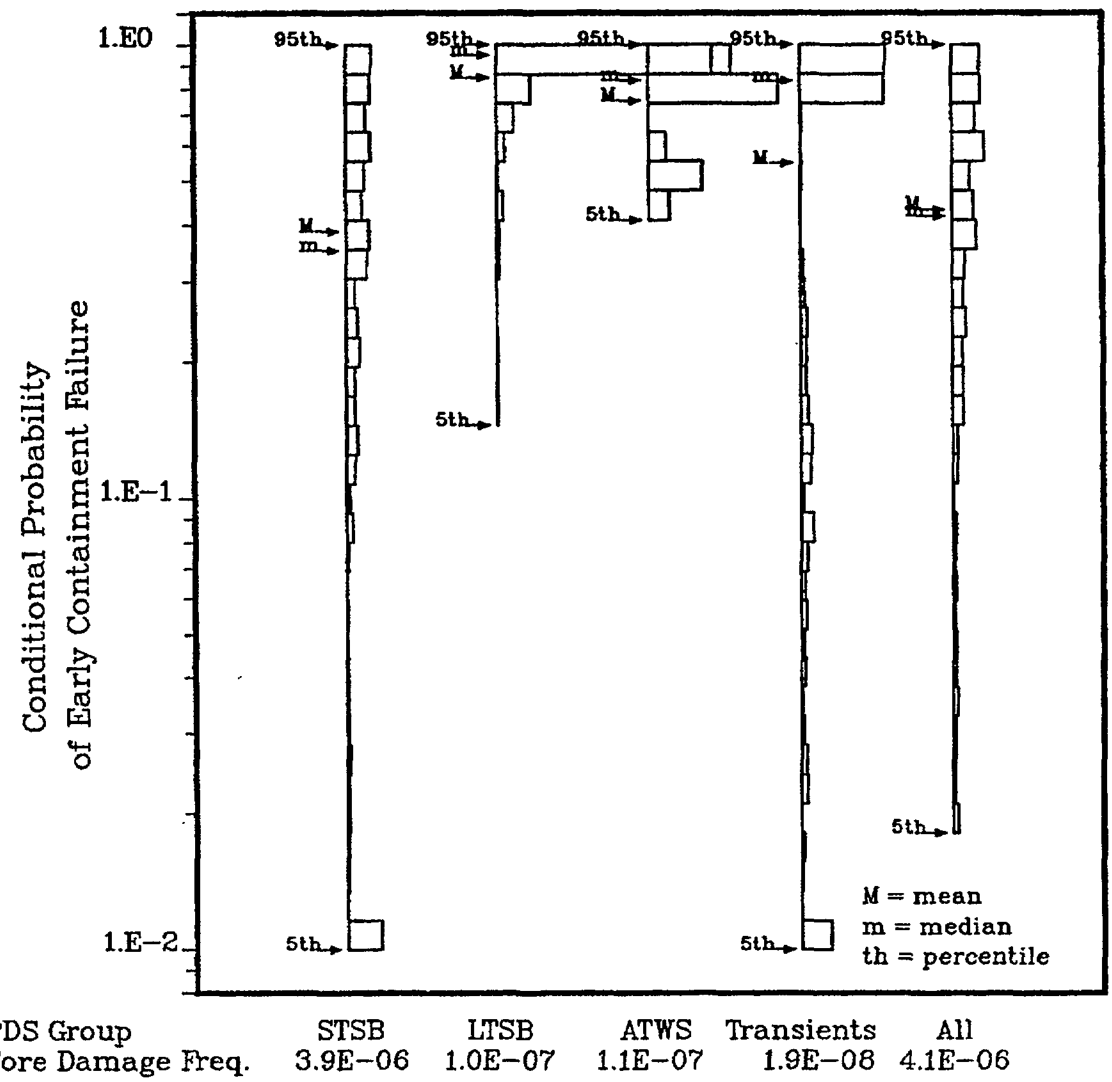


Figure 6.5 Conditional probability distributions for early containment failure at Grand Gulf.

3. **Hydrogen Ignition System**

The Grand Gulf containment hydrogen ignition system is capable of maintaining the concentration of hydrogen from severe accidents in manageable proportions for many severe accidents. However, for station blackout accident sequences, the igniter system is not operable. When power is restored, the ignition system will be initiated; potentially the containment has high hydrogen concentrations. Some potential then exists for a deflagration causing simultaneous failures of both the containment building and the drywell structure.

4. **Containment Spray System**

The Grand Gulf containment spray system has the capability to condense steam and reduce the amount of radioactive material released to the environment for specific accident sequences. However, for some sequences, i.e., loss of ac power, its eventual initiation upon power recovery and that of the hydrogen ignition system could result in subsequent hydrogen combustion that has some potential to fail the containment and drywell.

6.4 Source Term Analysis

6.4.1 Results of Source Term Analysis

A key difference between the Peach Bottom (Mark I) design and Grand Gulf (Mark III) design is the wetwell/drywell configuration. If the drywell remains intact in the accident and the mode of containment failure does not result in loss of the suppression pool, leakage to the environment must pass through the pool and be subject to decontamination.

Figures 6.6 and 6.7 illustrate the effect of drywell integrity in mitigating the environmental release of radionuclides for early containment failure. In Figure 6.6, both the drywell and the containment fail early and sprays are not available. The median release for the volatile radionuclides is approximately 10 percent, indicative of a large release with the potential for causing early fatalities. For the early containment failure accident progression bin with the drywell intact, as illustrated in Figure 6.7, the environmental source terms are reduced, since the flow of gases escaping the containment after vessel breach must also pass through the suppression pool before being released to the environment.

Additional discussion on source term perspectives (for all studied plants) is provided in Chapter 10.

6.4.2 Important Plant Characteristics (Source Term)

1. **Suppression Pool**

   The pressure-suppression pool at Grand Gulf provides the potential for substantial mitigation of the source terms in severe accidents. Since transient-initiated accidents represent a large contribution to core damage frequency, the in-vessel release of radionuclides is almost always subject to pool decontamination. Only a fraction of such accident sequences (in which a vacuum breaker sticks open in a safety relief valve discharge line) releases radionuclides directly to the drywell in this phase of the accident. The pool decontamination factors used for the Grand Gulf design for the in-vessel release range from 1.1 to 4000, with a median of 60. For the ex-vessel release component, the pool is less effective. The decontamination factors range from 1 to 90 with a median of 7.

2. **Wetwell-Drywell Configuration**

   If the drywell remains intact in a severe accident at Grand Gulf, the radionuclide release would be forced to pass through the suppression pool and the source term would be substantially mitigated. However, the likelihood of drywell failure is estimated to be quite significant, such that early failure with suppression pool bypass occurs approximately one-quarter of the time if core melting and vessel breach occur.

3. **Pedestal Flooding**

   The pedestal region communicates with the drywell region through drains in the drywell floor. The amount of water in the pedestal region depends on whether the upper water pool has been dumped into the suppression pool, on the quantity of condensate storage that has been injected into the containment, and on the transient pressurization of the containment building resulting from hydrogen burns. The effect of water in the pedestal is either to result in debris coolability or to mitigate the source term to containment of the radionuclides released during core-concrete interaction. Water in the pedestal does, however, also introduce some potential for a steam explosion that can damage the drywell.

4. **Containment Sprays**

   Containment sprays can have a mitigating effect on the release of radionuclides under conditions in which both the containment and drywell have failed. In other accident scenarios in which the in-vessel and ex-vessel releases must pass through the suppression pool before reaching the outer containment region, sprays are not nearly as important. This is, in part, because the source term has already been reduced and, in part, because the decontamination factors for suppression pools and containment sprays are not multiplicative since they selectively remove similar-sized aerosols.

6.5 Offsite Consequence Results

Figure 6.8 displays the frequency distributions in the form of graphical plots of the complementary cumulative distribution functions (CCDFs) of four offsite consequence measures—early fatalities, latent cancer fatalities, and the 50-mile and the entire site region population exposures (in person-rems). These CCDFs include contributions from all source terms associated with reactor accidents caused by internal initiating events. Four CCDFs,

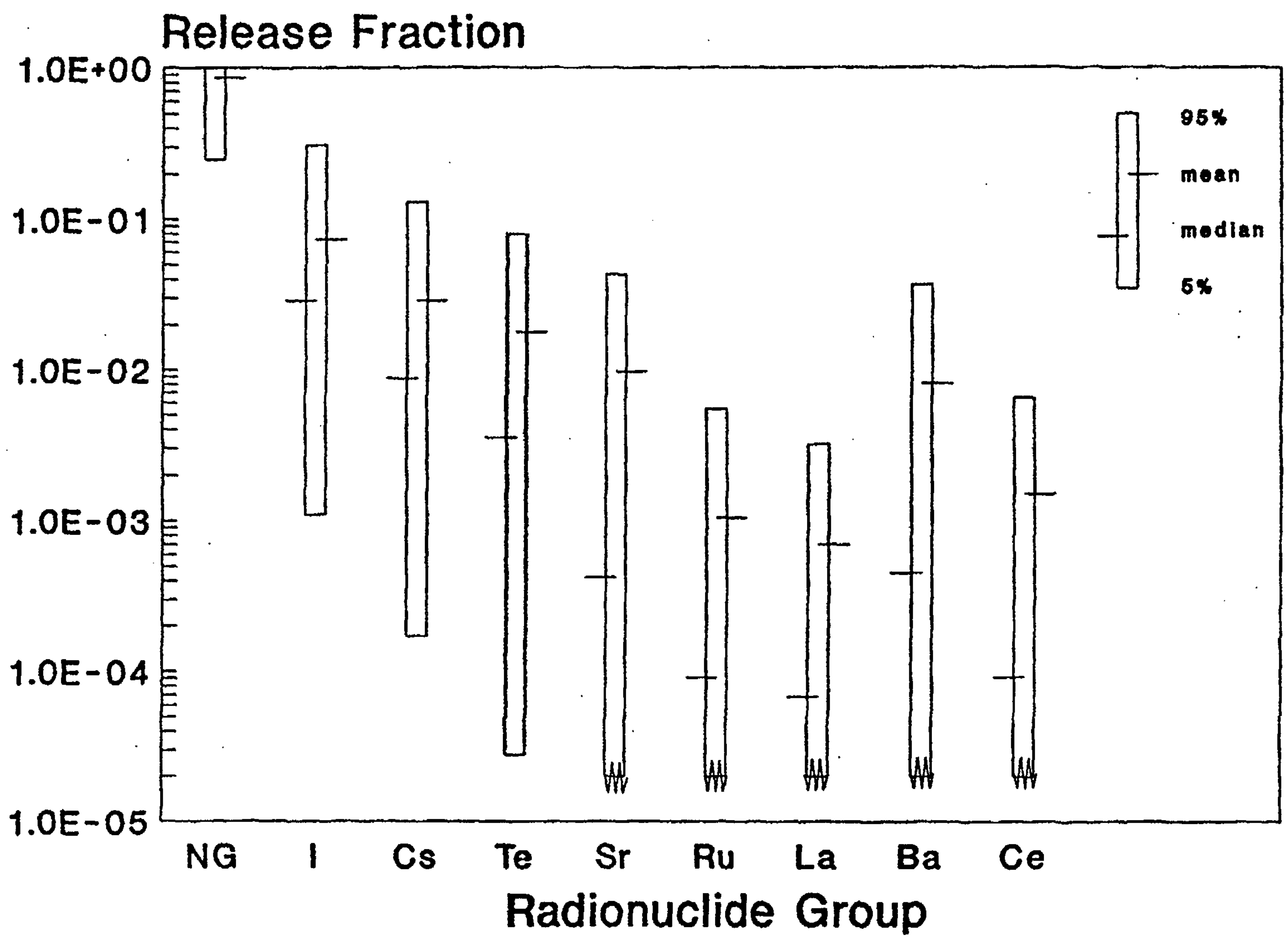


Figure 6.6 Source term distributions for early containment failure with drywell failed and sprays unavailable at Grand Gulf.

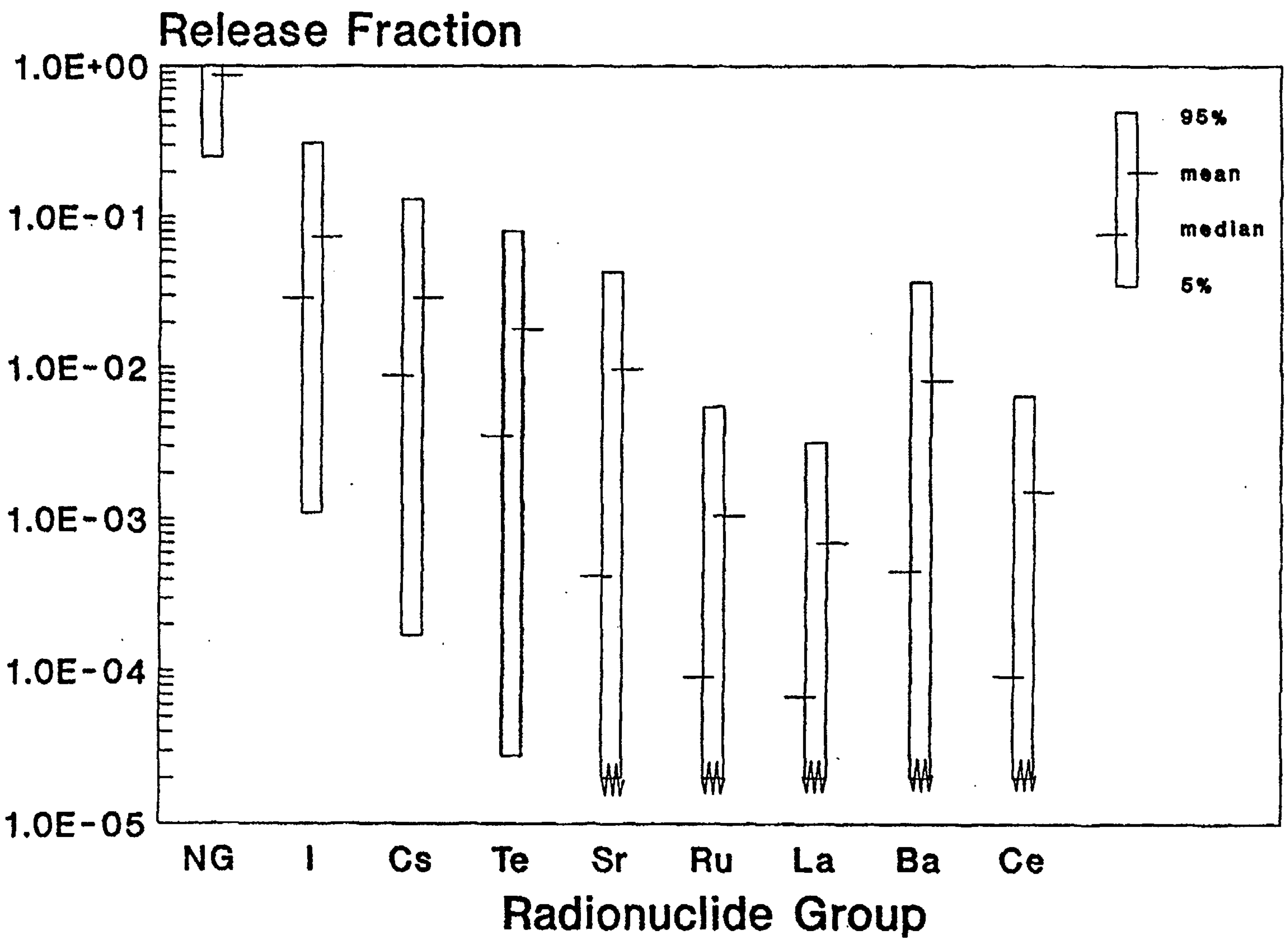
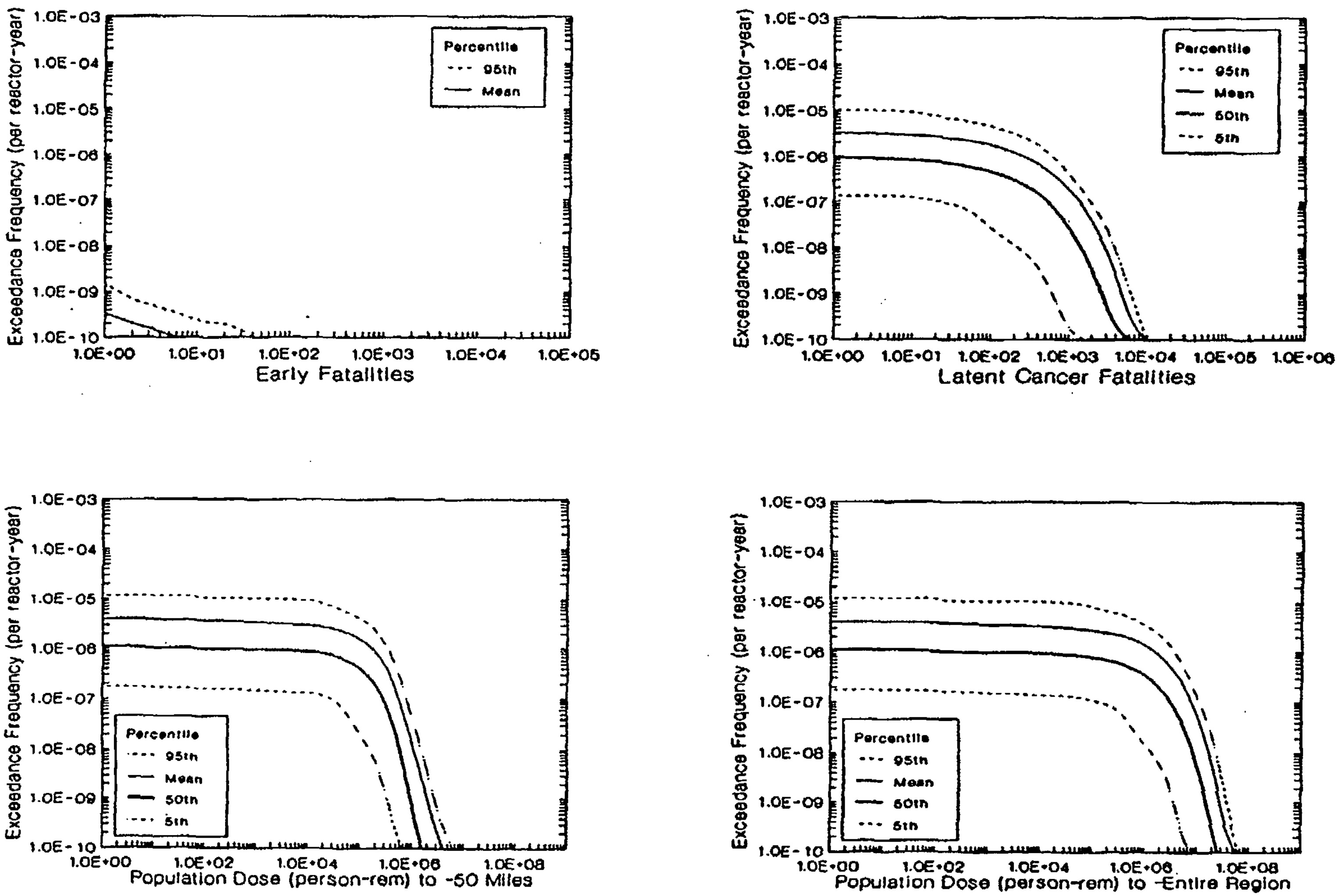


Figure 6.7 Source term distributions for early containment failure with drywell intact at Grand Gulf.



Note: As discussed in Reference 6.4, estimated consequences at frequencies 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 6.8 Frequency distributions of offsite consequence measures at Grand Gulf (internal initiators).

namely, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs, are shown for each consequence measure.

Grand Gulf plant-specific and site-specific parameters were used in the consequence analyis for these CCDFs. The plant-specific parameters included source terms and their frequencies, the licensed thermal power (3833 MWt) of the reactor, and the approximate physical dimensions of the power plant building complex. The site-specific parameters included exclusion area radius (696 meters), meteorological data for 1 full year collected at the meteorological tower, the site region population distribution based on the 1980 census data, topography (fraction of the area that is land—the remaining fraction is assumed to be water), land use, agricultural practice and productivity, and other economic data for up to 1,000 miles from the Grand Gulf plant.

The consequence estimates displayed in these figures have incorporated the benefits of the following protective measures: (1) evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), (2) early relocation of the remaining population only from the heavily contaminated areas both within and outside the 10-mile EPZ, and (3) decontamination, temporary interdiction, or condemnation of land, property, and foods contaminated above acceptable levels.

The population density within the Grand Gulf 10-mile EPZ is about 30 persons per square mile. The average delay time before evacuation (after a warning prior to radionuclide release) from the 10-mile EPZ and average effective evacuation speed used in the analyses were derived from information contained in a utility-sponsored Grand Gulf evacuation time estimate study (Ref. 6.6) and the NRC requirements for emergency planning.

The results displayed in Figure 6.8 are discussed in Chapter 11.

6.6 Public Risk Estimates

6.6.1 Results of Public Risk Estimates

A detailed description of the results of the Grand Gulf risk analysis is provided in Reference 6.2. For this summary report, results are provided for the following measures of public risk:

- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Grand Gulf exclusion area boundary, and
- Individual latent cancer fatality risk in the population within 10 miles of the Grand Gulf site.

The first four of the above measures are commonly used measures in nuclear power plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 6.7).

The results of the Grand Gulf risk studies using the above measures are shown in Figures 6.9 through 6.11. The figures display the variabilities in mean risks estimated from meteorology-averaged conditional mean values of the consequence measures. In comparison to the risks from the other plants in this study, Grand Gulf has the lowest risk estimates. The results are much below those of the Reactor Safety Study (Ref. 6.8). The individual early and latent cancer fatality risks are far below the NRC safety goals. Details of the comparison of results are provided in Chapter 12.

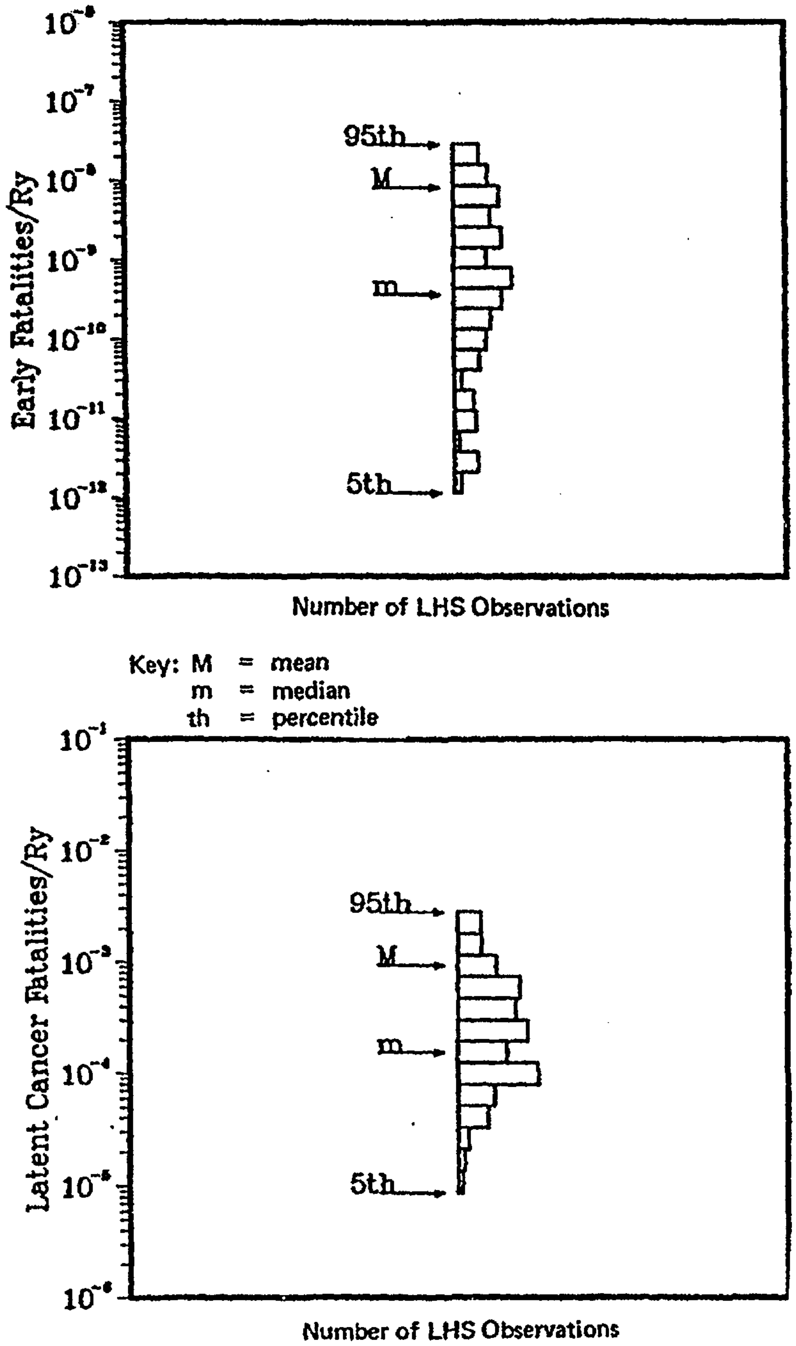
The results in Figure 6.9 have been analyzed to identify the relative contributions of accident sequences and containment failure modes to mean risk. These results are presented in Figures 6.12 and 6.13. As may be seen, the mean early fatality risk at Grand Gulf is dominated by short-term station blackout sequences. The majority of early fatality risk is associated with the coincidence of early containment failure and early suppression pool bypass.

The mean latent cancer fatality risk is also dominated by the short-term station blackout group. The major contributors to risk are from (1) early containment and early suppression pool bypass, and (2) late containment failure.

6.6.2 Important Plant Characteristics (Risk)

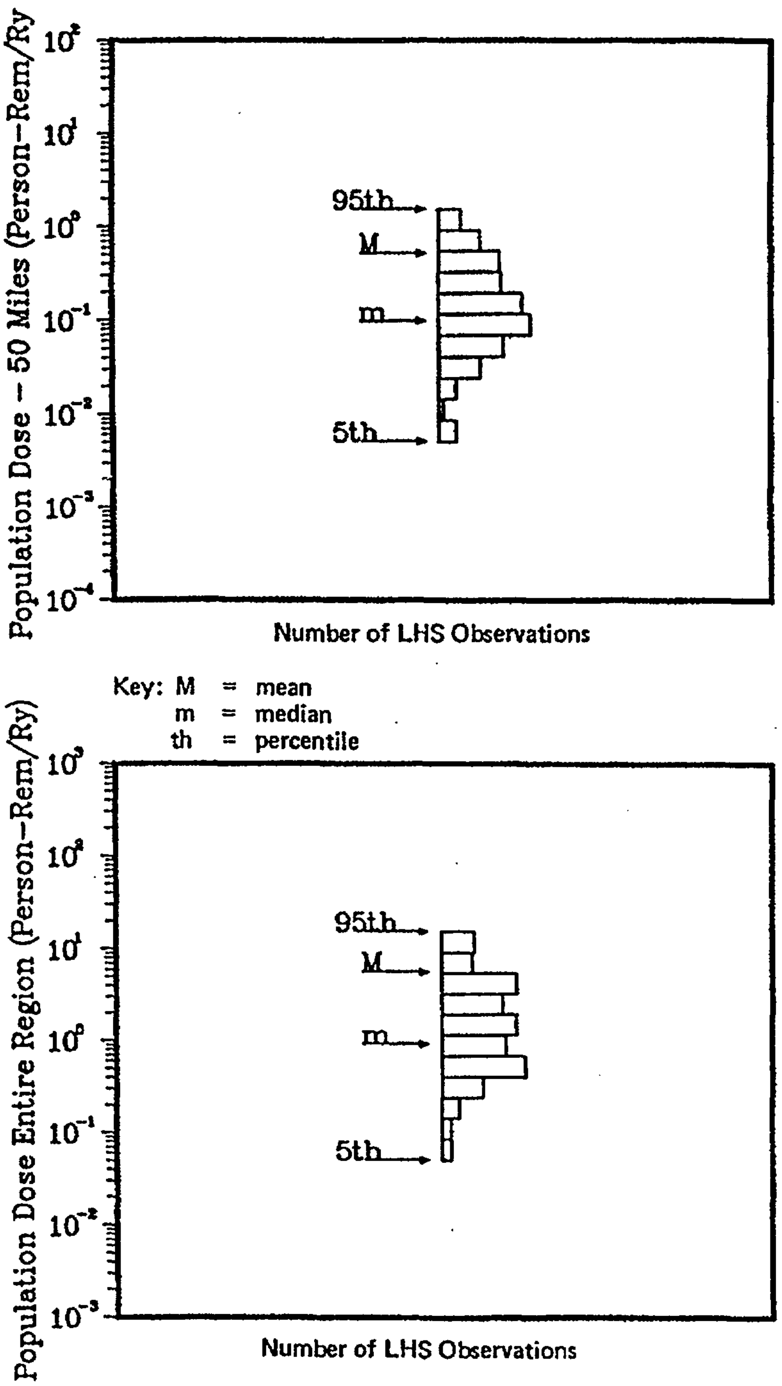
As mentioned before, risk to the public from the operation of the Grand Gulf plant is lower than the other four plants in this study. Some of the plant features that contribute to these low risk estimates are described below.

- The very low early fatality risk at Grand Gulf is due to a combination of low core damage



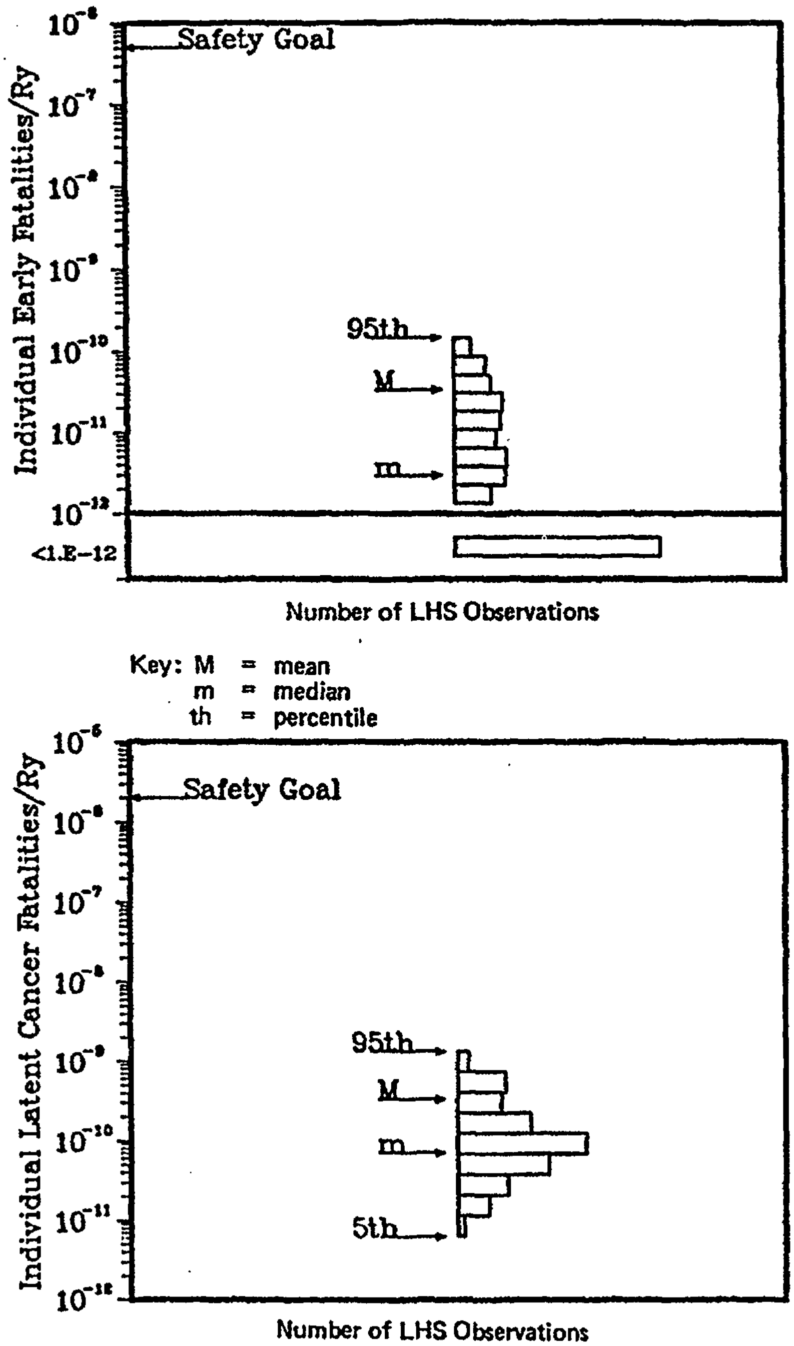
Note: As discussed in Reference 6.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 6.9 Early and latent cancer fatality risks at Grand Gulf (internal initiators).



Note: As discussed in Reference 6.4, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 6.10 Population dose risks at Grand Gulf (internal initiators).



Note: As discussed in Reference 6.4, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 6.11 Individual early and latent cancer fatality risks at Grand Gulf (internal initiators).

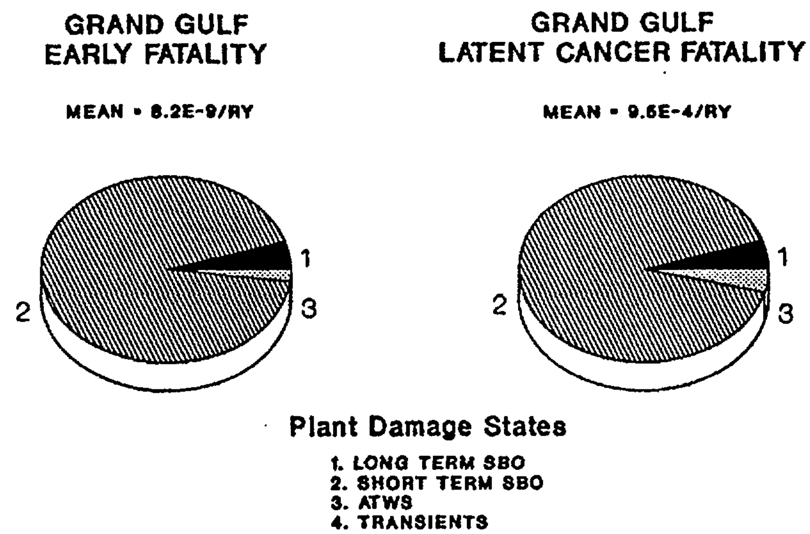


Figure 6.12 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Grand Gulf (internal initiators).

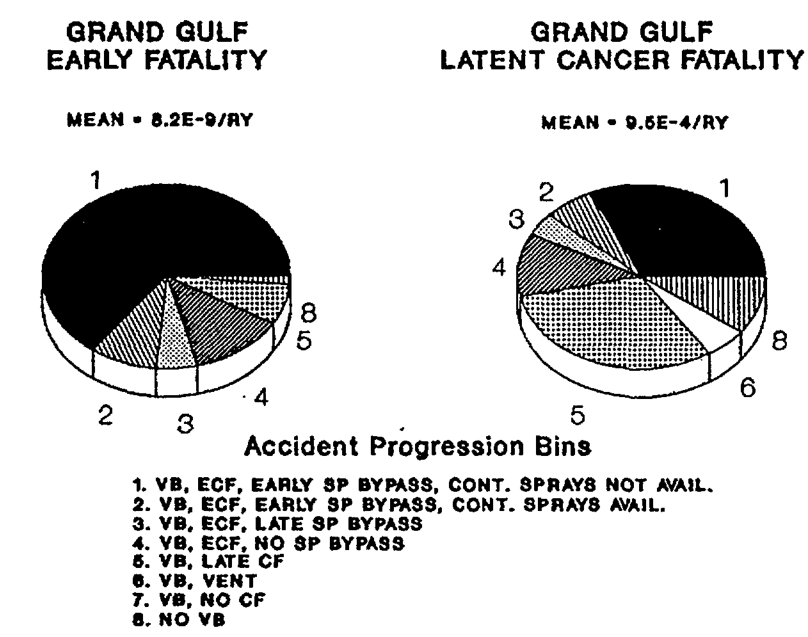


Figure 6.13 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Grand Gulf (internal initiators).

frequency, reduced source terms (as a result of suppression pool scrubbing), and low population density around the plant. The latter leads to short evacuation delays and fast evacuation speeds. Timing is not as important for latent cancer fatalities.

- Although the Grand Gulf plant has relatively high probability of early containment failure, caused mainly by hydrogen deflagration, the probability of early drywell failure, which may lead to a large source term, is about half of the probability of early containment failure. Furthermore, in most cases, in-vessel releases pass through the suppression pool.

- There is a high probability of having water in the reactor cavity following vessel breach. Thus, there is a high probability that core debris would be coolable. Even when any core-concrete interaction may occur, it is generally under water, and, therefore, the resulting releases are scrubbed by overlaying water (if not by the suppression pool).

REFERENCES FOR CHAPTER 6

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

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

6.3 S. W. Hatch et al., "Reactor Safety Study Methodology Applications Program: Grand Gulf No. 1 BWR Power Plant," Sandia National Laboratories and Battelle Columbus Laboratories, NUREG/CR-1659/4 of 4, SAND80-1897/4 of 4, November 1981.

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

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

6.6 Mississippi Power & Light Company, "Evacuation Time Estimates for the Grand Gulf Nuclear Station Plume Exposure Pathway Emergency Planning Zone," Revision 4, March 1986.

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

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

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

7. ZION PLANT RESULTS

7.1 Summary Design Information

The Zion Nuclear Plant is a two-unit site. Each unit is a four-loop Westinghouse nuclear steam supply system rated at 1100 MWe and is housed in a large, prestressed concrete, steel-lined dry containment. The balance of plant systems were engineered by Sargent & Lundy. Located on the shore of Lake Michigan, about 40 miles north of Chicago, Illinois, Zion 1 started commercial operation in December 1973. Some important design features of the Zion plant are described in Table 7.1. A general plant schematic is provided in Figure 7.1.

This chapter provides a summary of the results provided in the risk analyses underlying this report (Refs. 7.1 and 7.2). A discussion of perspectives with respect to these results is provided in Chapters 8 through 12.

7.2 Core Damage Frequency Estimates

7.2.1 Summary of Core Damage Frequency Estimates*

The core damage frequency and risk analyses performed for this study considered accidents initiated only by internal events (Ref. 7.1); no external-event analyses were performed. The core damage frequency results obtained are provided in tabular form in Table 7.2. This study calculated a total median core damage frequency from internal events of 2.4E–4 per year.

7.2.1.1 Zion Analysis Approach

The Zion plant was previously analyzed in the Zion Probabilistic Safety Study (ZPSS), performed by the Commonwealth Edison Company, and in the review and evaluation of the ZPSS (Ref. 7.3), commonly called the Zion Review prepared by Sandia National Laboratories.

Since previous analyses of Zion already existed, it was decided to perform an update of the previous analyses rather than perform a complete reanalysis. Therefore, this analysis of Zion represents a limited rebaseline and extension of the dominant accident sequences from the ZPSS in light of the Zion Review comments, although incorporating some methods and issues (such as common-cause failure treatment, electric power recovery, and reactor coolant pump seal LOCA modeling) used in the other four plant studies.

The objective of this study was to perform an analysis that updated the previous Zion analyses and cast the model in a manner more consistent with the other accident frequency analyses. The models were not completely reconstructed in the small-event-tree, large-fault-tree modeling method used in the study of the other NUREG–1150 plants. Instead, the small-fault-tree, large-event-tree models from the original ZPSS were used as the basis for the update. These models were then revised according to the comments from Reference 7.3 and were enhanced to address risk issues using methods employed by the other plant studies.

This study incorporated specific issues into the systems and accident sequence models of the ZPSS. These issues reflect both changes in the Zion plant and general PRA assumptions that have arisen since the ZPSS was performed. New dominant accident sequences were determined by modifying and requantifying the event tree models developed for ZPSS. The major changes reflect the need for component cooling water and service water for emergency core cooling equipment and reactor coolant pump seal integrity. The original set of plant-specific data used in the ZPSS and Zion Review was verified as still valid and was used for this study. Additional discussion of the Zion methods is provided in Appendix A.

7.2.1.2 Internally Initiated Accident Sequences

A detailed description of accident sequences important at the Zion plant is provided in Reference 7.1. For this summary report, the accident sequences described in that reference have been grouped into six summary plant damage states. These are:

- Station blackout,
- Loss-of-coolant accident (LOCA),
- Component cooling water and service water induced reactor coolant pump seal LOCAs,
- Anticipated transients without scram (ATWS),

*In general, the results and perspectives provided here do not reflect recent modifications to the Zion plant. The benefit of the changes is noted, however, in specific places in the text (and discussed in more detail in Section 15 of Appendix C).

Table 7.1 Summary of design features: Zion Unit 1.

1. High-Pressure Injection	a. Two centrifugal charging pumps. b. Two 1500-psig safety injection pumps. c. Charging pumps inject through boron injection tank. d. Provides seal injection flow. e. Requires component cooling water.
2. Low-Pressure Injection	a. Two RHR pumps deliver flow when RCS is below about 170 psig. b. Heat exchangers downstream of pumps provide recirculation heat removal. c. Recirculation mode takes suction on containment sump and discharges to the RCS, HPI suction, and/or containment spray pump suction. d. Pumps and heat exchangers require component cooling water.
3. Auxiliary Feedwater	a. Two 50 percent motor-driven pumps and one 100 percent turbine-driven pump. b. Pumps take suction from own unit condensate storage tank (CST) but can be manually crosstied to the other unit's CST.
4. Emergency Power System	a. Each unit consists of three 4160 VAC class 1E buses, each feeding one 480 VAC class 1E bus and motor control center. b. For the two units there are 5 diesel generators, with one being a swing diesel generator shared by both units. c. Three trains of dc power are supplied from the inverters and 3 unit batteries.
5. Component Cooling Water	a. Shared system between both units. b. Consists of 5 pumps, 3 heat exchangers, and 2 surge tanks. c. Cools RHR heat exchangers, RCP motors and thermal barriers, RHR pumps, SI pumps, and charging pumps. d. One of 5 pumps can provide sufficient flow.
6. Service Water	a. Shared system between both units. b. Consists of 6 pumps and 2 supply headers. c. Cools component cooling heat exchangers, containment fan coolers, diesel generator coolers, auxiliary feedwater pumps. d. Two of 6 pumps can supply sufficient flow.
7. Containment Structure	a Large, dry, prestressed concrete. b. 2.6 million cubic foot volume. c. 49 psig design pressure.
8. Containment Spray	a. Two motor-driven pumps and 1 independent diesel-driven pump. b. No train crossties. c. Water supplied by refueling water storage tank.
9. Containment Fan Coolers	a. Five fan cooler units, a minimum of 3 needed for post-accident heat removal. b. Fan units shift to low speed on SI signal. c. Coolers require service water.

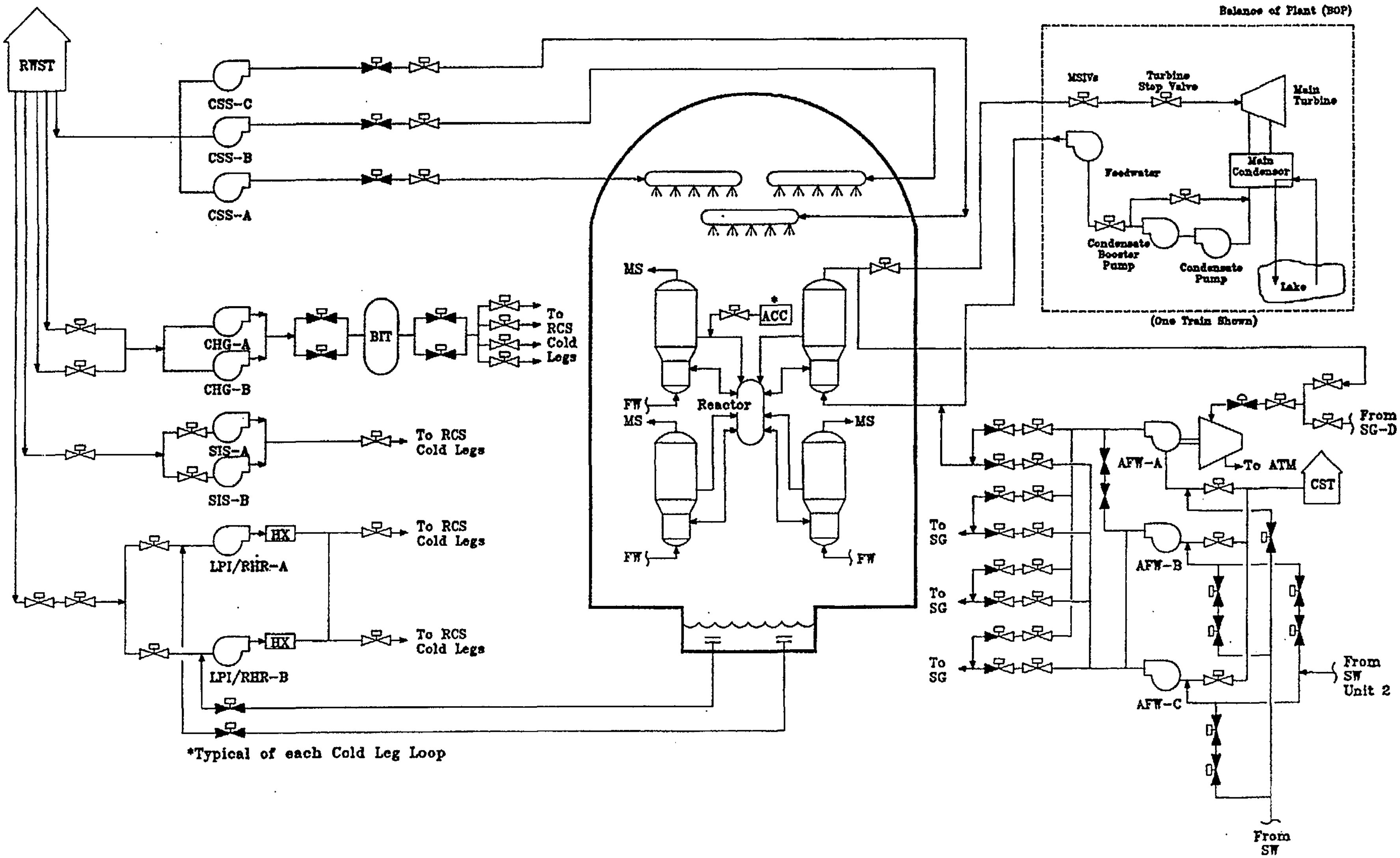


Figure 7.1 Zion plant schematic.

Table 7.2 Summary of core damage frequency results: Zion.

	5%	Median	Mean	95%
Internal Events	1.1E–4	2.4E–4	3.4E–4*	8.4E–4

*See text (Section 7.2.1) for benefit of recent modifications.

- Interfacing-system LOCA and steam generator tube rupture (SGTR), and
- Transients other than station blackout and ATWS.

The relative contribution of the accident types to mean core damage frequency at Zion is shown in Figure 7.2. It is seen that the dominating contributors to the core damage frequency are the loss of component cooling water and loss of service water. The more probable combinations of failures are:

- Reactor coolant pump seals fail because of the loss of cooling and injection. Core damage occurs because of failure to recover the service water/component cooling water systems in time to reestablish reactor coolant system inventory control. In cases with failure of the service water system, containment fan coolers are also failed.
- Reactor coolant pump seals fail because of the loss of cooling and injection. The cooling system is recovered in time to provide injection from the refueling water storage tank (RWST). Recirculation cooling fails to continue to provide long-term inventory control.

To address the issue of the importance of component cooling water system failures, Commonwealth Edison (the Zion licensee) committed in 1989 to perform the following actions (Ref. 7.4):

- Provide an auxiliary water supply to each charging pump's oil cooler via either the service water system or fire protection system. Hoses, fittings, and tools will be maintained locally at each unit's charging pump area allowing for immediate hookup to existing taps on the oil coolers, if required. As an interim measure, a standing order in the control room will instruct operators as to how and when to hook up auxiliary water to the oil coolers.
- Formal procedures, including a 10 CFR 50.59 review addressing the loss of component cooling water system scenario, will be fully implemented within 60 days (of the date of Ref. 7.4) to supersede the standing order.
- When new heat-resistant reactor coolant pump seal o-rings are made available by Westinghouse, the existing o-rings will be changed when each pump is disassembled for routine scheduled seal maintenance.

These actions provide a backup water source to the Zion station charging pump oil coolers.

As of October 1990, Commonwealth Edison had performed some of the noted actions (Ref. 7.5). Sensitivity studies have been performed to assess the benefit of the modifications made to date. These studies, discussed in more detail in Section C.15 of Appendix C, indicate that the Zion estimated mean core damage frequency has been reduced from 3.4E–4 per year to approximately 6E–5 per year.

7.2.2 Important Plant Characteristics (Core Damage Frequency)

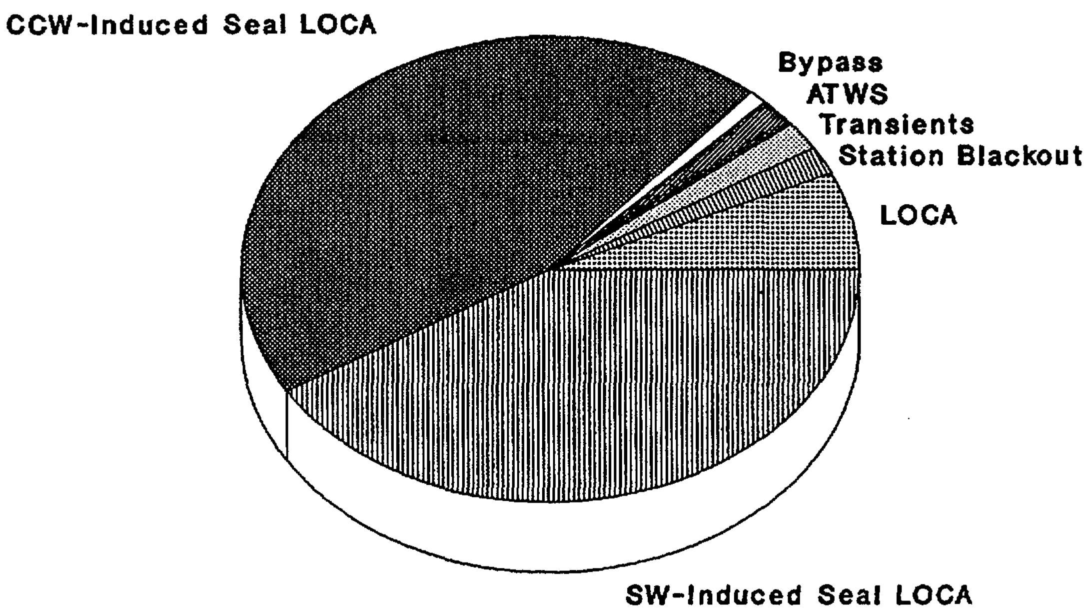
Characteristics of the Zion plant design and operation that have been found to be important in the analysis of the core damage frequency include:

1. **Shared Systems Between Units**

   The Zion nuclear station shares the service water and component cooling water (CCW) systems between the two units. Power is supplied to these systems from all five onsite diesel generators.

2. **Crossties Between Units**

   Crossties between units exist for the condensate storage tanks to provide water supply for the auxiliary feedwater system. Crossties also exist between Unit 1 and Unit 2 ac power systems, as well as between Unit 1 and Unit 2 dc power systems.



Total Mean Core Damage Frequency: 3.4E-4

Note: See text (Section 7.2.1) for benefit of recent modifications.

Figure 7.2 Contributors to mean core damage frequency from internal events at Zion.

3. **Diesel Generators**

Zion is a two-unit site with five emergency diesel generators. One diesel generator is a swing diesel that can be lined up to supply either unit. This differs from a number of other two-unit sites that have only four diesel generators on site. The Zion diesel generators are dependent on a common service water system for sustained operation.

4. **Support System Dependencies**

The component cooling water system supplies cooling water for the reactor coolant pump thermal barriers and for the charging pumps that supply seal injection. Failure of the component cooling water system results in a major challenge to reactor coolant pump seal integrity. In addition, failure of the component cooling water support systems (service water and ac power) also leads to loss of reactor coolant pump seal integrity. In contrast, some other PWRs do not have a common dependency for both seal cooling and seal injection; therefore, at other PWRs, seal LOCAs are only important in station blackout cases. As indicated above, the licensee has committed to and implemented plant changes to reduce this dependency.

5. **Battery Depletion Time**

The battery depletion time following a complete loss of all ac power was estimated at 6 hours, somewhat longer than that found at some other plants. The additional time tends to reduce the significance of the station blackout sequences as contributors to the core damage frequency.

6. **Reactor Coolant Pump Seal Performance**

   The inability of the reactor coolant pump seals to survive loss of cooling and injection without developing significant leakage dominates the core damage frequency. As noted above, the licensee has committed to replacing present seals with a new model.

7.2.3 Important Operator Actions

Several operator actions and recovery actions are important to the analysis of the core damage frequency. While the analysis included a wide range of operator actions from test and maintenance errors before an initiating event to recovery actions well into an accident sequence, the following actions surface as the most important:

- Successful switchover to recirculation

  The operator must recognize that switchover should be initiated, take action to open the proper set of motor-operated valves depending on reactor coolant system conditions, and verify that recirculation flow is proper.

- Successful execution of feed and bleed cooling

  The operator must recognize that secondary cooling is lost, establish sufficient injection flow, open both power-operated relief valves (and their block valves, if necessary), and verify that adequate heat removal is taking place.

- Recovery of the component cooling water and service water systems

  The operator must recognize that the failure of equipment or rising equipment operating temperatures are due to failure of the service water or component cooling water systems, determine the cause of system failure, and take appropriate action to isolate ruptures, restart pumps, and provide alternative cooling paths as required by the situation.

- Actions to refill the RWST in the event of recirculation failure

  This action requires that the operator recognize the failure of recirculation cooling in sufficient time that refill can begin before core damage occurs. The operator must then carry out the procedure for emergency refill of the RWST. This action is not adequate for inventory control in the case of larger LOCAs because of the limitations of the refilling equipment.

Switchover to recirculation cooling and initiation of feed and bleed cooling were included in the original Zion Probabilistic Safety Study and have been given close scrutiny by the licensee. Each one of these actions is present in the emergency procedures. Appropriate consideration of the procedures, scenarios, timing, and training went into the determination of the human error probabilities associated with these actions. Because of the importance and uncertainty associated with several of these actions, they were addressed in the sensitivity analyses. However, the refilling of the RWST in the event of recirculation failure and recovery of CCW and service water were not included in the original Zion Probabilistic Safety Study. Appropriate consideration of the procedures, scenarios, timing, and training went into the determination of the human error probabilities associated with these actions. Because of the importance and uncertainty associated with several of these actions, they were addressed in the sensitivity analyses.

7.3 Containment Performance Analysis

7.3.1 Results of Containment Performance Analysis

The Zion containment consists of a large, dry containment building that houses the reactor pressure vessel, reactor coolant system piping, and the secondary system's steam generators. The containment building is a prestressed concrete structure with a steel liner. This building has a volume of 2.6 million cubic feet with a design pressure of 49 psig and an estimated mean failure pressure of 150 psia. The principal threats to containment integrity from potential severe accident sequences are steam explosions, overpressurization from direct containment heating effects, bypass events, and isolation failures. As previously discussed in Chapter 2, the methods used to estimate loads and containment structural response for Zion made extensive use of expert judgment to interpret and supplement the limited data (Ref. 7.2).

The results of the Zion containment analysis are summarized in Figures 7.3 and 7.4. Figure 7.3 displays information in which the conditional probabilities of four accident progression bins, e.g., early containment failure, are presented for each of five plant damage states, e.g., LOCA. This information indicates that, on a plant damage

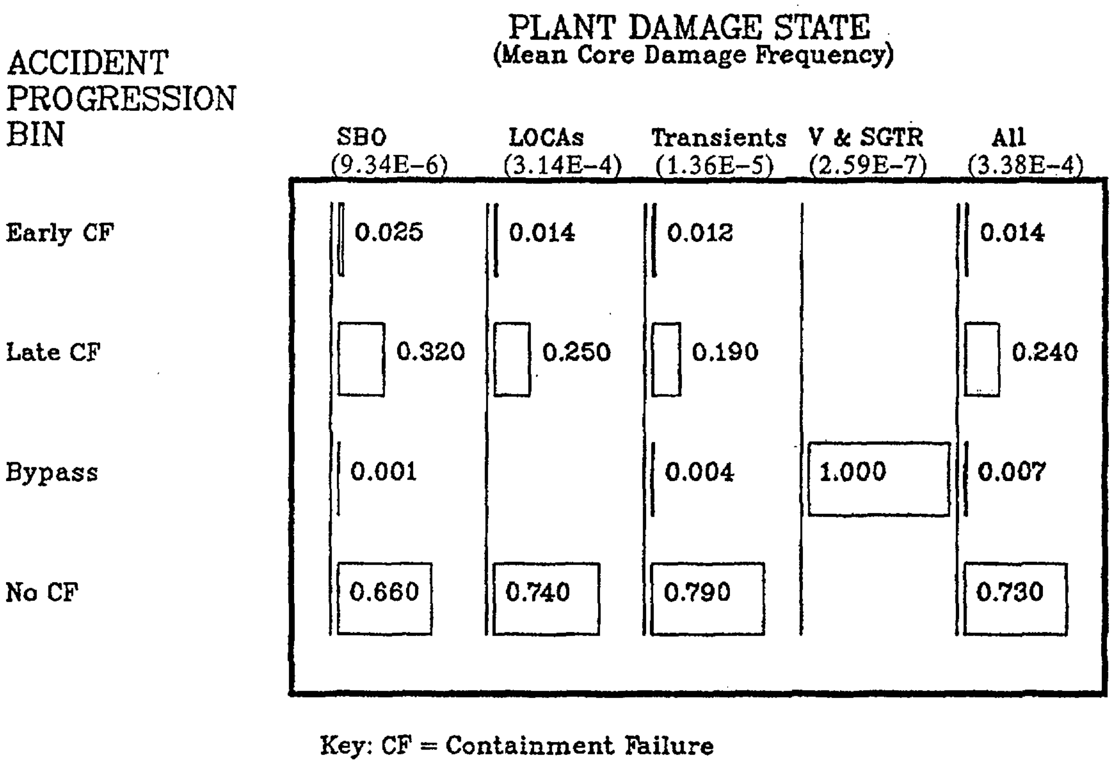


Figure 7.3 Conditional probability of accident progression bins at Zion.

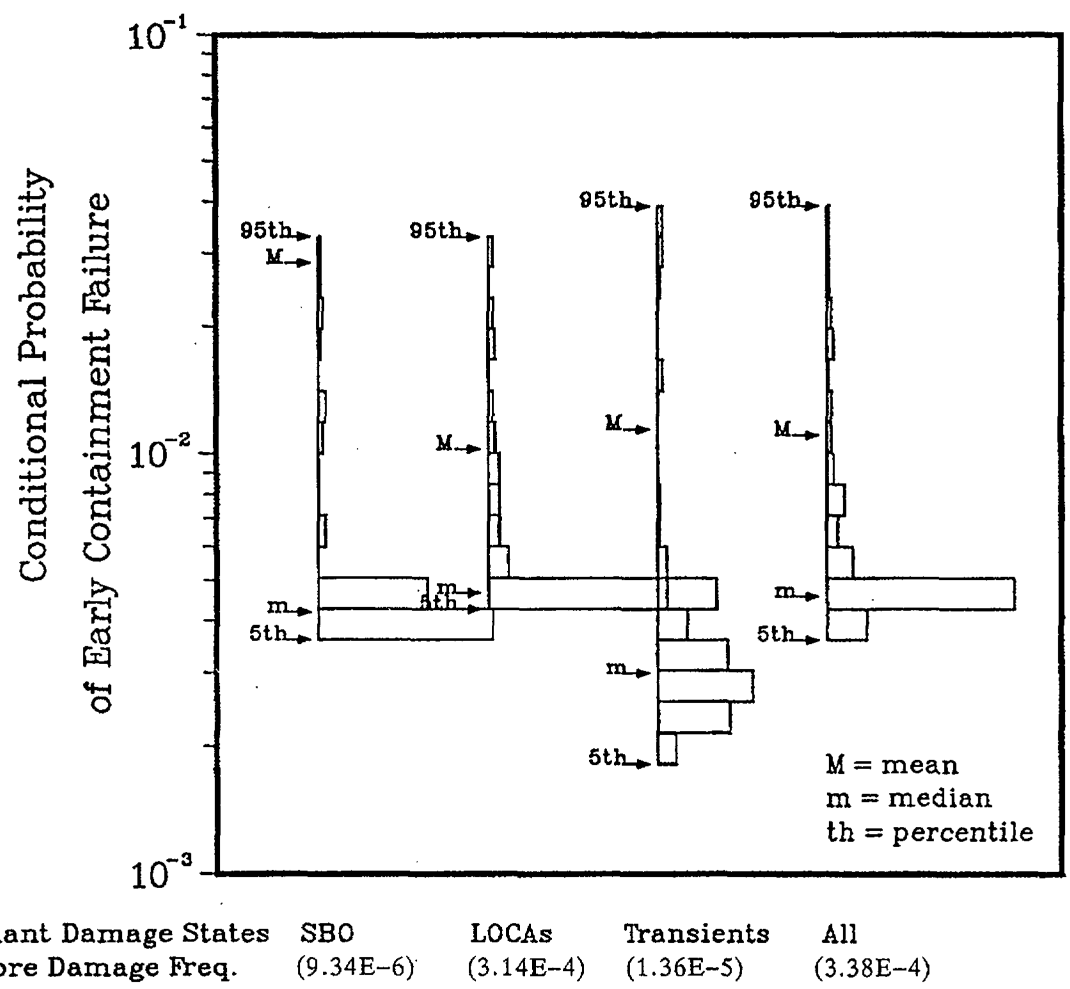


Figure 7.4 Conditional probability distributions for early containment failure at Zion.

state frequency-weighted average,* the mean conditional probabilities from internal events of (1) early containment failure from a combination of in-vessel steam explosions, overpressurization, and containment isolation failures is 0.014, (2) late containment failure, mainly from basemat meltthrough is 0.24, (3) containment bypass from interfacing-system LOCA and induced steam generator tube rupture (SGTR) is 0.006, and (4) probability of no containment failure is 0.73. Figure 7.4 further displays the conditional probability distributions of early containment failure for the plant damage states, thereby providing the estimated range of uncertainties in these containment failure predictions. The principal conclusion to be drawn from the information in Figures 7.3 and 7.4 is that the probability of early containment failure for Zion is low, i.e., 1 to 2 percent.

Additional discussion on containment performance is provided in Chapter 9.

7.3.2 Important Plant Characteristics (Containment Performance)

Characteristics of the Zion design and operation that are important to containment performance include:

1. **Containment Volume and Pressure Capability**

   The combined magnitude of Zion's containment volume and estimated failure pressure provide considerable capability to withstand severe accident threats.

2. **Reactor Cavity Geometry**

   The Zion containment design arrangement has a large cavity directly beneath the reactor pressure vessel that communicates to the lower containment by means of an instrument tunnel. Provided the contents of the refueling water storage tank have been injected prior to vessel breach, this arrangement should provide a mechanism for quenching the molten core for some severe accidents (although there remains some uncertainties with respect to the coolability of molten core debris in such circumstances).

*Each value in the column in Figure 7.3 labeled "All" is a frequency-weighted average obtained by calculating the products of individual accident progression bin conditional probabilities for each plant damage state and the ratio of the frequency of that plant damage state to the total core damage frequency.

7.4 Source Term Analysis

7.4.1 Results of Source Term Analysis

The containment performance results for the Zion (large, dry containment) plant and the Surry (subatmospheric containment) plant are quite similar. The source terms for analogous accident progression bins are also quite similar. Figure 7.5 illustrates the source term for early containment failure. As at Surry, the source terms for early failure are somewhat less than those for containment bypass. Within the range of the uncertainty band, however, the source terms from early containment failure are potentially large enough to result in some early fatalities.

The most likely outcome of a severe accident at the Zion plant is that the containment would not fail. Figure 7.6 illustrates the range of source terms for the no containment failure accident progression bin. Other than for the noble gas and iodine radionuclide groups, the entire range of source terms is below a release fraction of 10E-5.

Additional discussion on source term perspectives is provided in Chapter 10.

7.4.2 Important Plant Characteristics (Source Term)

1. **Containment Spray System**

   The containment spray system at the Zion plant is not required to operate to provide long-term cooling to the containment, in contrast to the Surry plant. Operation of the spray system is very effective, however, in reducing the airborne concentration of aerosols. Other than the release of noble gases and some iodine evolution, the release of radioactive material to the atmosphere resulting from late containment leakage or basemat meltthrough in which sprays have operated for an extended time would be very small. The source terms for the late containment failure accident progression bin are slightly higher than, but similar to, those of the no containment failure bin illustrated in Figure 7.6.

2. **Cavity Configuration**

   The Zion cavity is referred to as a wet cavity, in that the accumulation of a relatively small amount of water on the containment floor will lead to overflow into the cavity. As a result, there is a substantial likelihood of eliminating by forming a coolable debris bed or

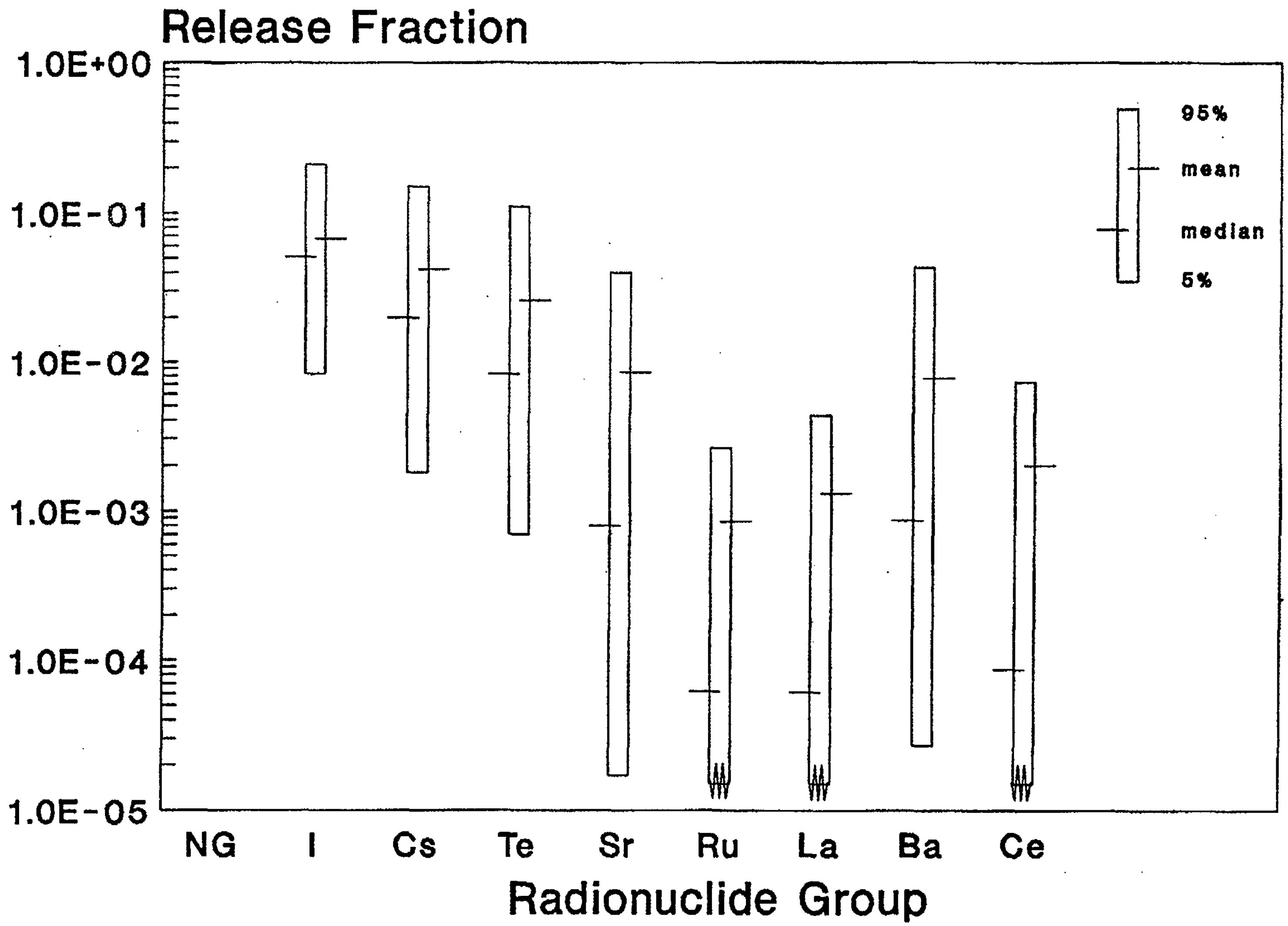


Figure 7.5 Source term distributions for early containment failure at Zion.

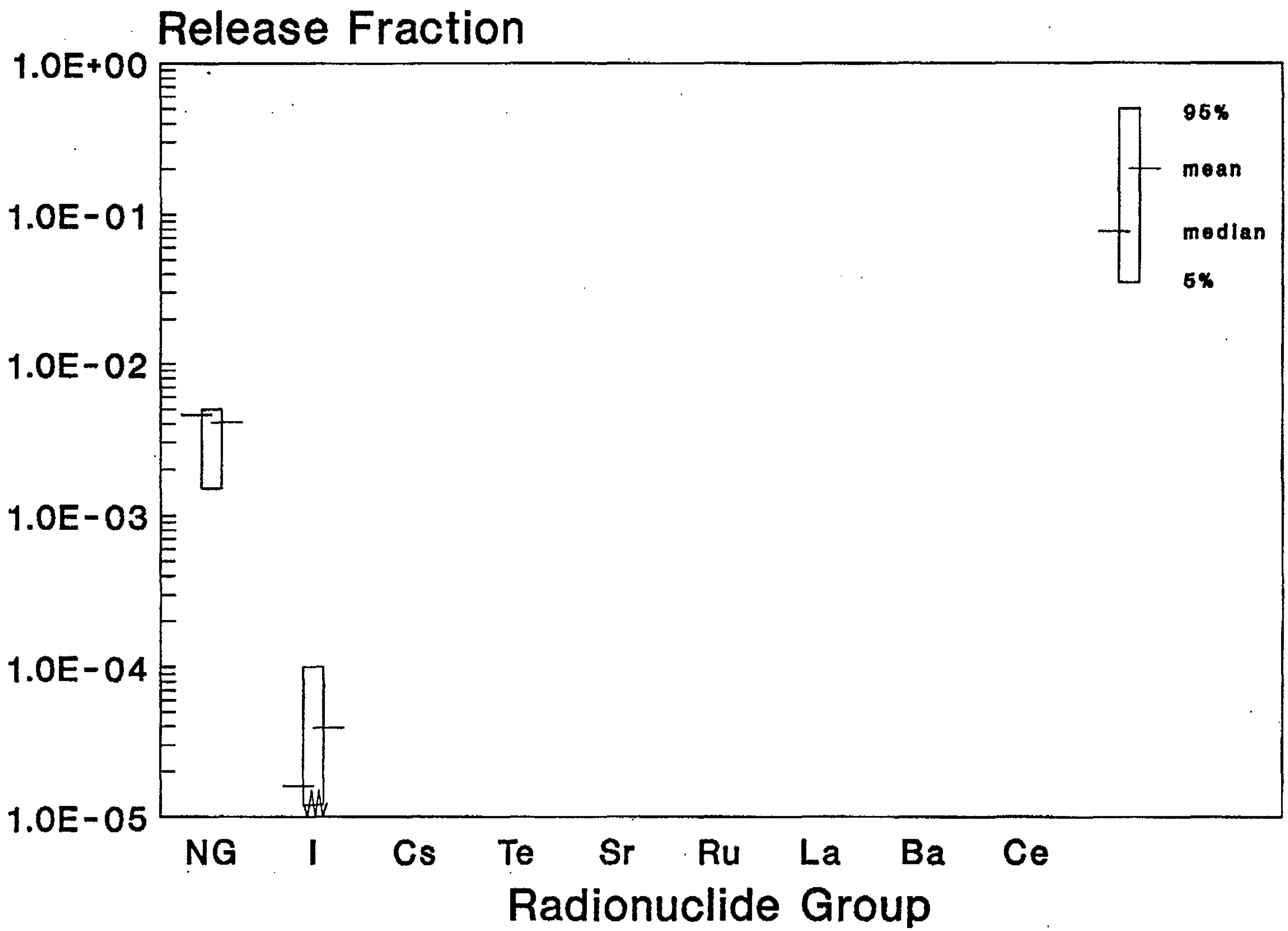


Figure 7.6 Source term distributions for no containment failure at Zion.

mitigating by the presence of an overlaying pool of water the release of radionuclides from core-concrete interactions.

7.5 Offsite Consequence Results

Figure 7.7 displays the frequency distributions in the form of graphical plots of the complementary cumulative distribution functions (CCDFs) of four offsite consequence measures—early fatalities, latent cancer fatalities, and the 50-mile region and entire site region population exposures (in person-rems). These CCDFs include contributions from all source terms associated with reactor accidents caused by internal initiating events. Four CCDFs, namely, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs are shown for each consequence measure.

Zion plant-specific and site-specific parameters were used in the consequence analysis for these CCDFs. The plant-specific parameters included source terms and their frequencies, the licensed thermal power (3250 MWt) of the reactor, and the approximate physical dimensions of the power plant building complex. The site-specific parameters included exclusion area radius (400 meters), meteorological data for 1 full year collected at the site meteorological tower, the site region population distribution based on the 1980 census data, topography (fraction of the area which is land—the remaining fraction is assumed to be water), land use, agricultural practice and productivity, and other economic data for up to 1,000 miles from the Zion plant.

The consequence estimates displayed in these figures have incorporated the benefits of the following protective measures: (1) evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), (2) early relocation of the remaining population only from the heavily contaminated areas both within and outside the 10-mile EPZ, and (3) decontamination, temporary interdiction, or condemnation of land, property, and foods contaminated above acceptable levels.

The population density within the Zion 10-mile EPZ is about 1360 persons per square mile. About 45 percent of the 10-mile EPZ is water. The average delay time before evacuation (after a warning prior to radionuclide release) from the 10-mile EPZ and average effective evacuation speed used in the analyses were derived from information contained in a utility-sponsored Zion evacuation time estimate study (Ref. 7.7) and in an independent analysis by the Federal Emergency Management Agency (Ref. 7.8) and the NRC requirements for emergency planning.

The results displayed in Figure 7.7 are discussed in Chapter 11.

7.6 Public Risk Estimates

7.6.1 Results of Public Risk Estimates*

A detailed description of the results of the Zion risk analysis is provided in Reference 7.2. For this summary report, results are provided for the following measures of public risk:

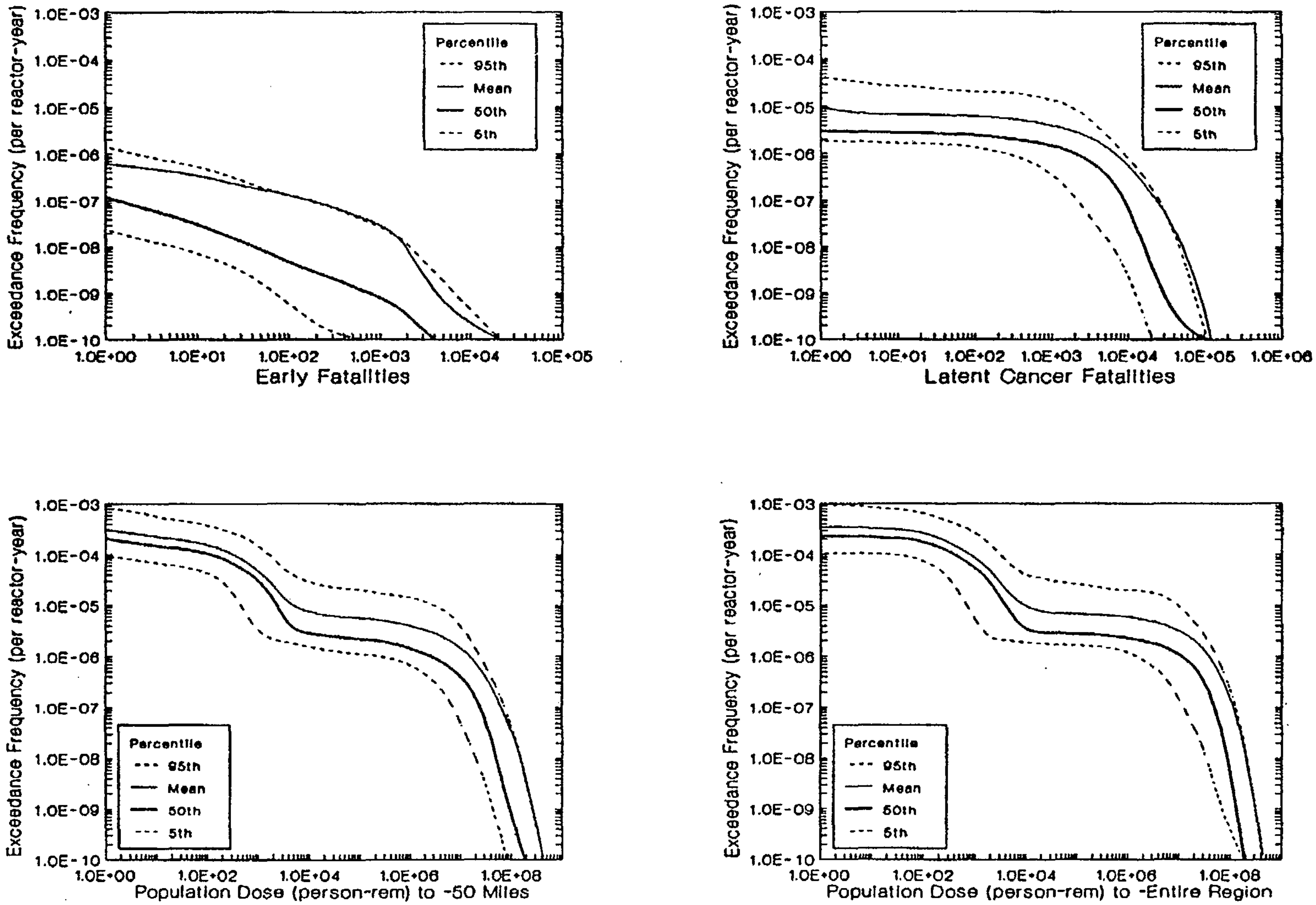
- Early fatality risk,
- Latent cancer fatality risk,
- Population dose within 50 miles of the site,
- Population dose within the entire site region,
- Individual early fatality risk in the population within 1 mile of the Zion exclusion area boundary, and
- Individual latent cancer fatality risk in the population within 10 miles of the Zion site.

The first four of the above measures are commonly used measures in nuclear plant risk studies. The last two are those used to compare with the NRC safety goals (Ref. 7.9).

The results of the Zion risk analyses are shown in Figures 7.8 through 7.10. The figures display variabilities in mean risks estimated from the meteorology-based conditional mean values of the consequence measures. The risk estimates are slightly higher than those of the other two PWR plants (Surry and Sequoyah) in this study. Individual early and latent cancer fatality risks are well below the NRC safety goals. Detailed comparisons of results are given in Chapter 12.

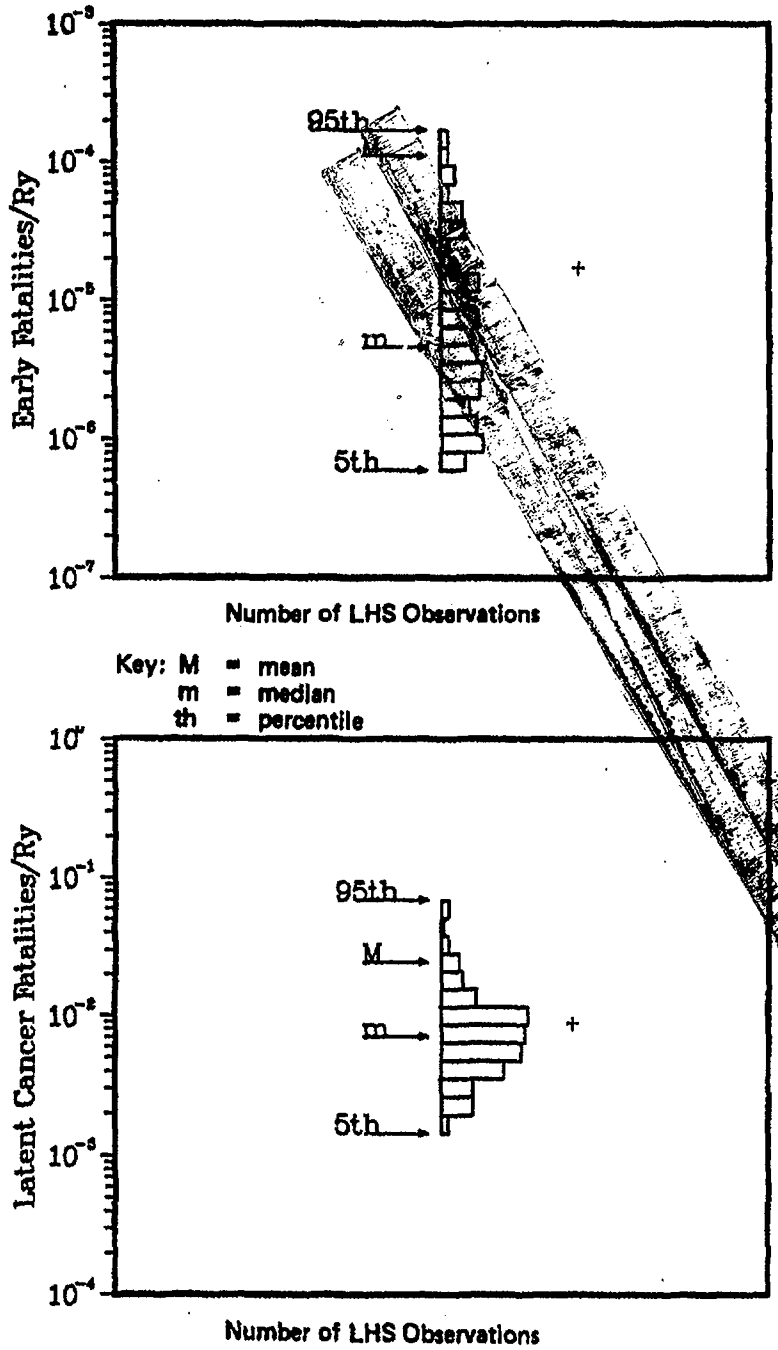
The risk results shown in Figure 7.8 have been analyzed to identify the principal contributors (accident sequences and containment failure modes) to plant risk. These results are presented in Figures 7.11 and 7.12. As may be seen, both for early and latent cancer fatality risks, the dominant plant damage state is loss-of-coolant-accident (LOCA) sequences, which have the highest relative frequency and relatively high release fractions. Zion plant risks are dominated by early containment failure (alpha-mode failure, containment isolation failure, and overpressurization

*As noted in Section 7.2, sensitivity studies have been performed to reflect recent modifications in the Zion plant. The impact on risk is displayed on the figures in this section. More detailed discussion on the sensitivity studies may be found in Section C.15 of Appendix C.



Note: As discussed in Reference 7.6, 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.

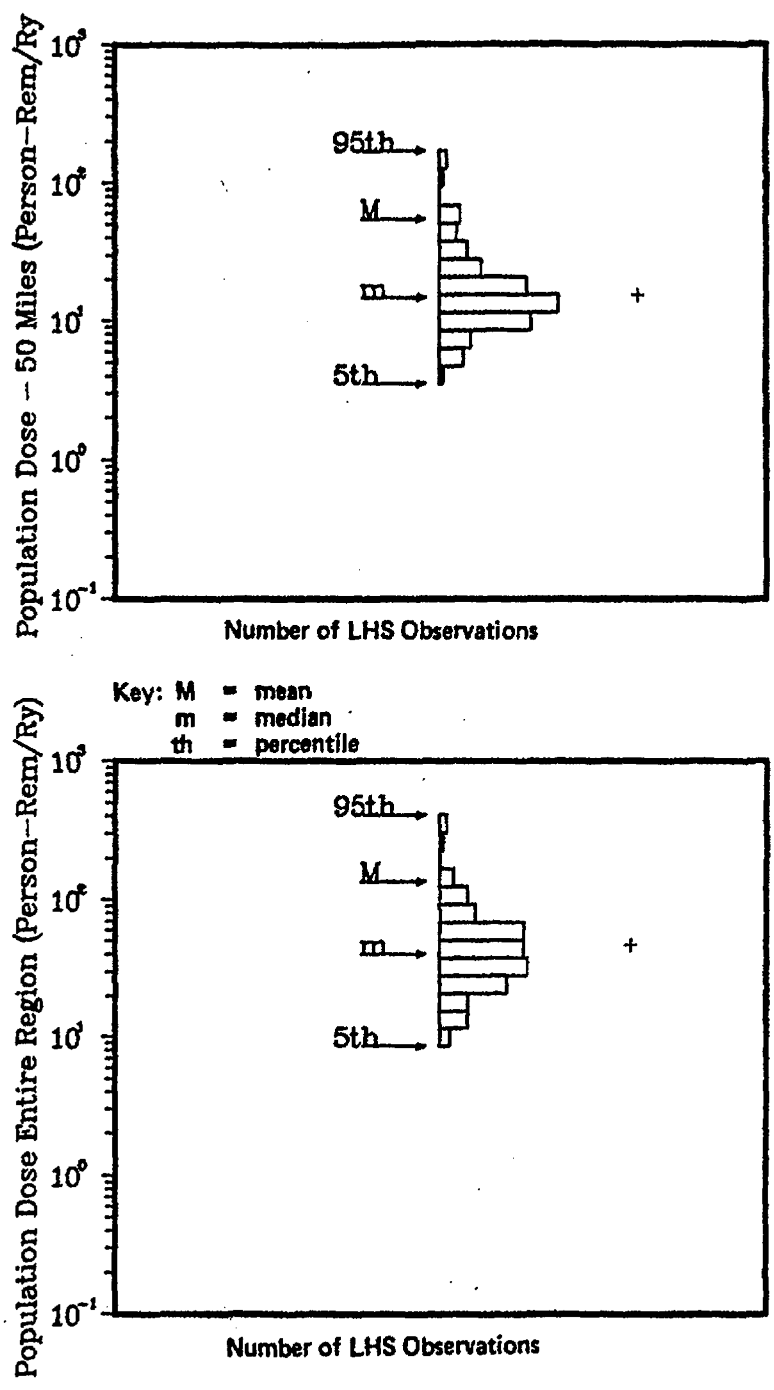
Figure 7.7 Frequency distributions of offsite consequence measures at Zion (internal initiators).



Notes As discussed in Reference 7.6, estimated risks at or below 1E-7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

"+" shows recalculated mean value based on plant modifications discussed in Section 7.2.1.

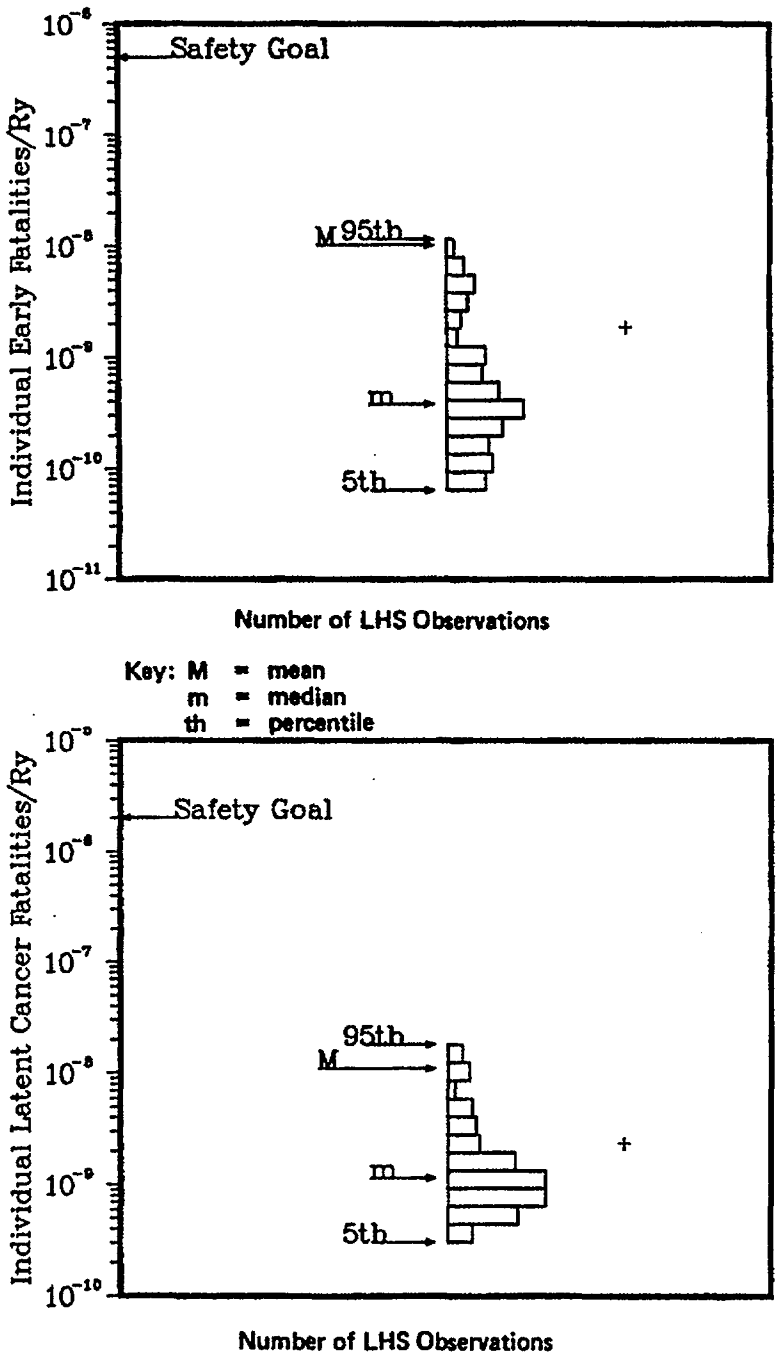
Figure 7.8 Early and latent cancer fatality risks at Zion (internal initiators).



Notes: As discussed in Reference 7.6, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

"+"shows recalculated mean value based on plant modifications discussed in Section 7.2.1.

Figure 7.9 Population dose risks at Zion (internal initiators).



Notes: As discussed in Reference 7.6, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of other health effects not studied in the risk analyses.

"+"shows recalculated mean value based on plant modifications discussed in Section 7.2.1.

Figure 7.10 Individual early and latent cancer fatality risks at Zion (internal initiators).

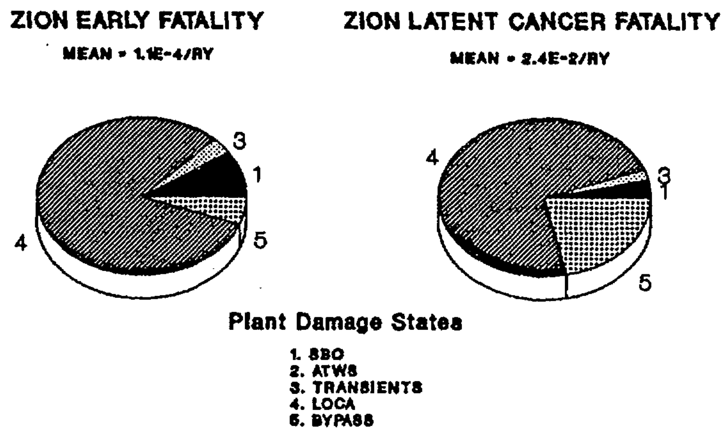


Figure 7.11 Major contributors (plant damage states) to mean early and latent cancer fatality risks at Zion (internal initiators).

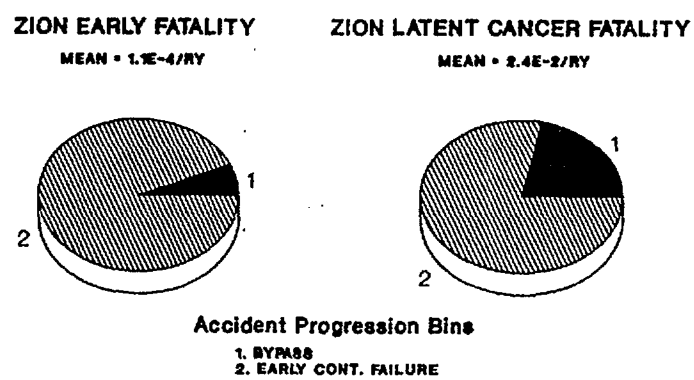


Figure 7.12 Major contributors (accident progression bins) to mean early and latent cancer fatality risks at Zion (internal initiators).

failure). This occurs because, although the conditional probability of early failure is low, other failure modes have even lower probabilities.

7.6.2 Important Plant Characteristics (Risk)

- As discussed before, the dominant risk contributor for the Zion plant is early containment failure. The accident progression bin for early containment failure contains several failure modes such as the alpha-mode, containment isolation, and overpressurization failures.
- The containment structure at Zion is robust, with a low probability of failure. This has led to the low risk estimates from the Zion plant. (In comparison with other plants studied in this report, risks from Zion are relatively high; but, in the absolute sense, the risks are very low and well below the NRC safety goals.)

REFERENCES FOR CHAPTER 7

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

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

7.3 D. L. Berry et al., "Review and Evaluation of the Zion Probabilistic Safety Study: Plant Analysis," Sandia National Laboratories, NUREG/CR-3300, Vol. 1, SAND83-1118, May 1984.

7.4 Cordell Reed, Commonwealth Edison Co. (CECo), "Zion Station Units 1 and 2. Commitment to Provide a Backup Water Source to the Charging Oil Coolers," NRC Docket Nos. 50-295 and 50-304, March 13, 1989.

7.5 R. A. Chrzanowski, CECo, "March 13, 1989 Letter from Cordell Reed to T. E. Murley," NRC, NRC Docket Nos. 50-295 and 50-304, August 24, 1990.

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

7.7 Stone & Webster Engineering Corporation, "Preliminary Evacuation Time Study of the 10-Mile Emergency Planning Zone at the Zion Station," prepared for Commonwealth Edison Company, January 1980.

7.8 Federal Emergency Management Agency, "Dynamic Evacuation Analyses: Independent Assessments of Evacuation Times from the Plume Exposure Pathway Emergency Planning Zones of Twelve Nuclear Power Stations," December 1980.

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

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

PART III

Perspectives and Uses

8. PERSPECTIVES ON FREQUENCY OF CORE DAMAGE

8.1 Introduction

Chapters 3 through 7 have summarized the core damage frequencies individually for the five plants assessed in this study. Significant differences among the plants can be seen in the results, both in terms of the core damage frequencies and the particular events that contribute most to those frequencies. These differences are due to plant-specific differences in the plant designs and operational practices. Despite the plant-specific nature of the study, it is possible to obtain important perspectives that may have implications for a larger number of plants and also to describe the types of plant-specific features that are likely to be important at other plants. This chapter provides some of these perspectives.

8.2 Summary of Results

As discussed in Chapter 2, the core damage frequency is not a value that can be calculated with absolute certainty and thus is best characterized by a probability distribution. It is therefore discussed in this report in terms of the mean, median, and various percentile values. The internal-event core damage frequencies are illustrated graphically in Figure 8.1 (Refs. 8.1 through 8.5). The figure does not include the contributions of external events, which are discussed in Section 8.4.

In Figure 8.1 the lower and upper extremities of the bars represent the 5th and 95th percentiles of the distributions, with the mean and median of each distribution also shown. Thus, the bars include the central 90 percent of the distributions (it should be remembered that the distributions are not uniform within these bars). These figures show that the range between the 5th and 95th percentiles covers from one to two orders of magnitude for the five plants. There is also significant overlap among the distributions, as discussed below. The reader should refer to References 8.1 through 8.5 for detailed discussion of the distributions.

Figures 8.2 and 8.3 show the contributions of the principal types of accidents to the mean core damage frequency for each plant. Figure 8.4 also presents this breakdown, but on a relative scale. These figures show that some types of accidents, such as station blackouts, contribute to the core damage frequencies for all the plants; however, there is substantial plant-to-plant variability among important accident sequences.

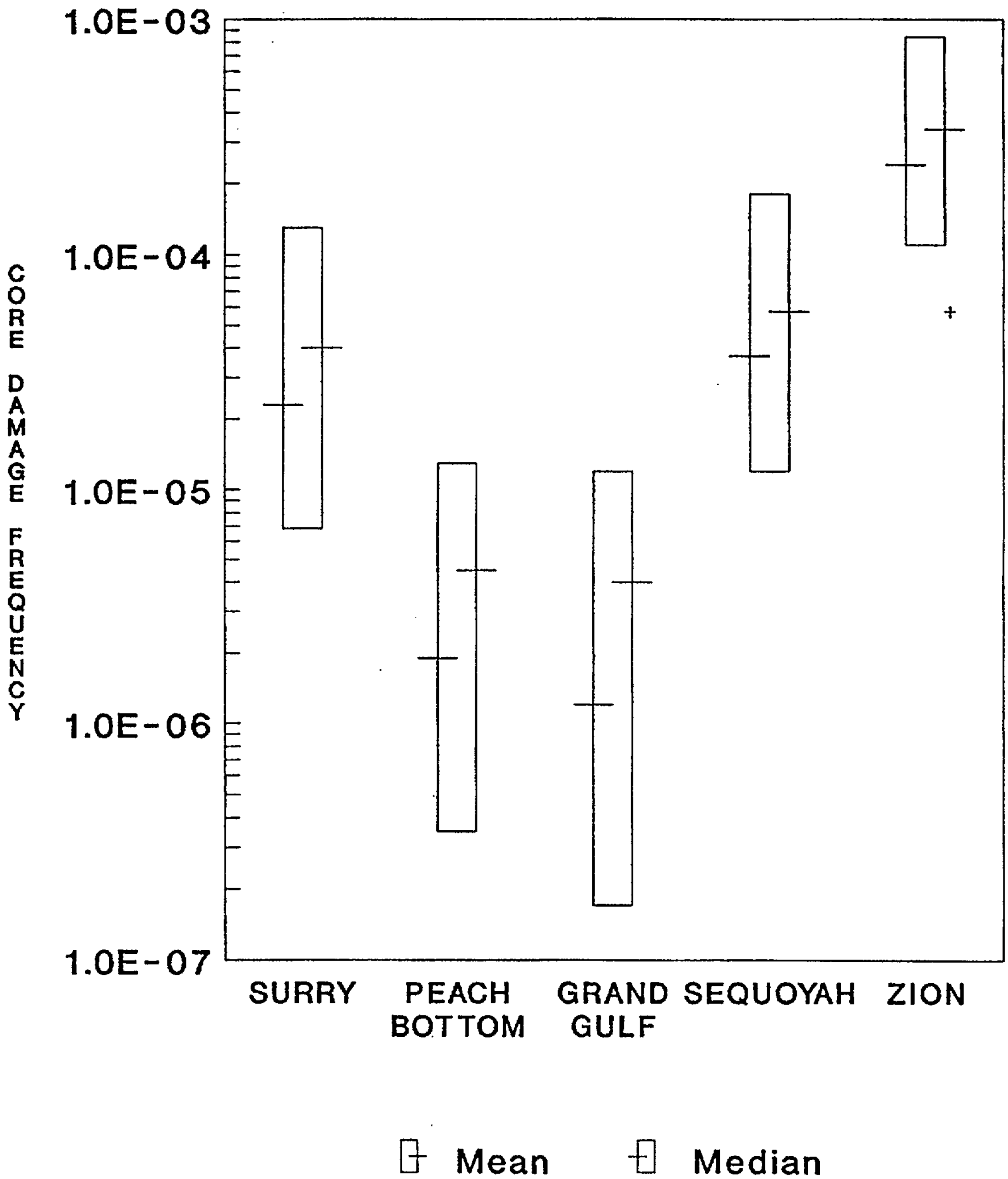
Figures 8.5 through 8.8 provide the results of the external-event analyses, and Figures 8.9 through 8.12 give the breakdown of these analyses according to the principal types of accident sequences.

8.3 Comparison with Reactor Safety Study

Figures 8.13 and 8.14 show the internal core damage frequency distributions calculated in this present study for Surry and Peach Bottom along with distributions synthesized from the Reactor Safety Study (Ref. 8.6), which also analyzed Surry and Peach Bottom. The Reactor Safety Study presented results in terms of medians but not means. It can be seen that the medians are lower in the present work, although observation of the overlap of the ranges shows that the change is more significant for Peach Bottom than for Surry.

There are two important reasons for the differences between the new figures and those of the Reactor Safety Study. The first is the fact that probabilistic risk analyses (PRAs) are snapshots in time. In these cases, the snapshots are taken about 15 years apart. Both plants have implemented hardware modifications and procedural improvements with the stated purpose of increasing safety, which drives core damage frequencies downward.

The second reason is that the state of the art in applying probabilistic analysis in nuclear power plant applications has advanced significantly since the Reactor Safety Study was performed. Computational techniques are now more sophisticated, computing power has increased enormously, and consequently the level of detail in modeling has increased. In some cases, these new methods have reduced or eliminated previous analytical conservatisms. However, new types of failures have also been discovered. For example, the years of experience with probabilistic analyses and plant operation have uncovered the reactor coolant pump seal failure scenario as well as intersystem dependencies, common-mode failure mechanisms, and other items that were less well recognized at the time of the Reactor Safety Study. Of course, this same experience has also uncovered new ways in which recovery can be achieved during the course of a possible core damage scenario (except for the



Notes: As discussed in Reference 8.7, core damage frequencies below 1E–5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

"+" indicates recalculated Zion mean core damage frequency based on recent plant modifications (see Section 7.2.1).

Figure 8.1 Internal core damage frequency ranges (5th to 95th percentiles).

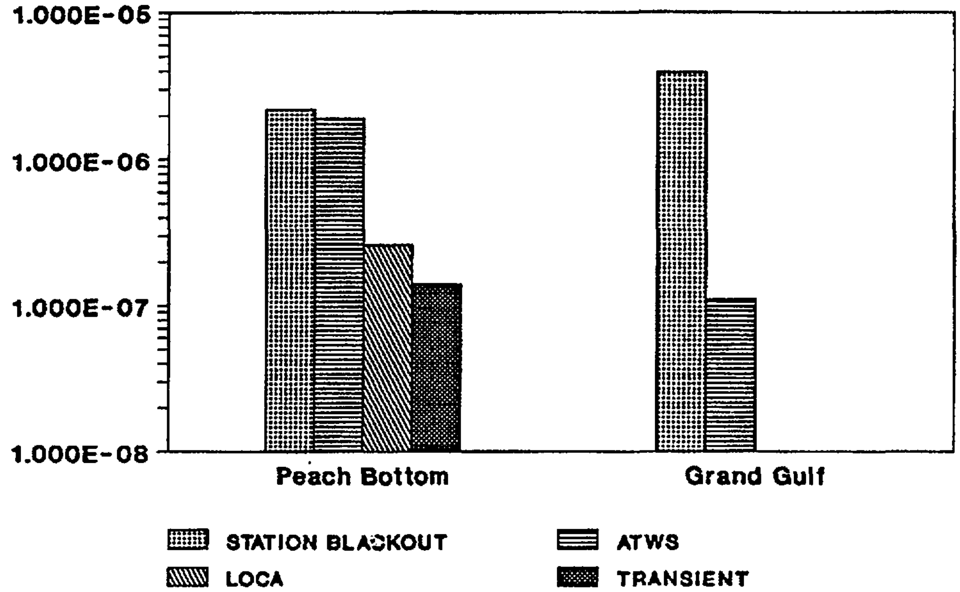
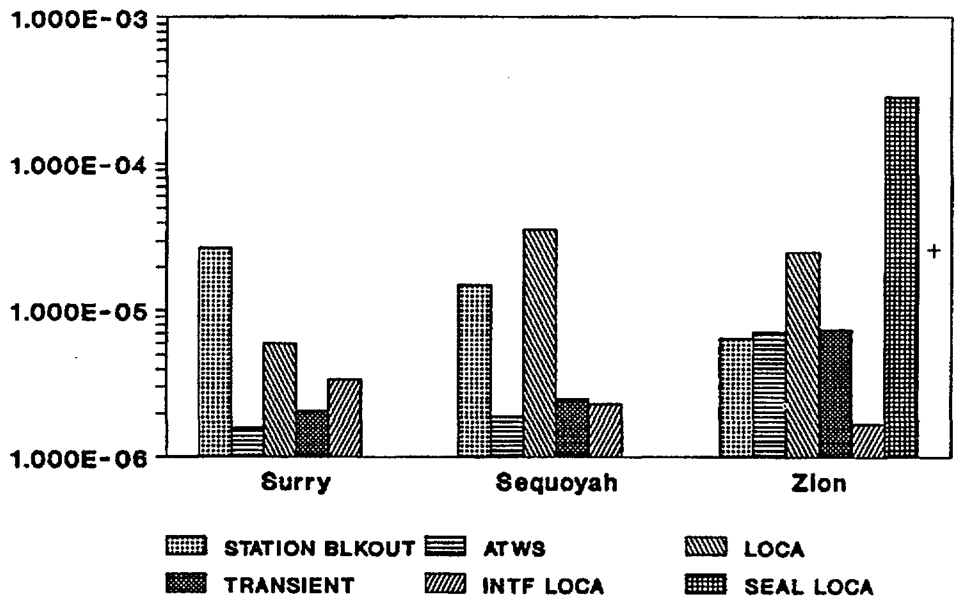


Figure 8.2 BWR principal contributors to internal core damage frequencies.



Notes: As discussed in Reference 8.7, core damage frequencies below 1E–5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

"+" indicates recalculated mean seal LOCA plant damage state frequency based on recent plant modifications (see Section 7.2.1).

Figure 8.3 PWR principal contributors to internal core damage frequencies.

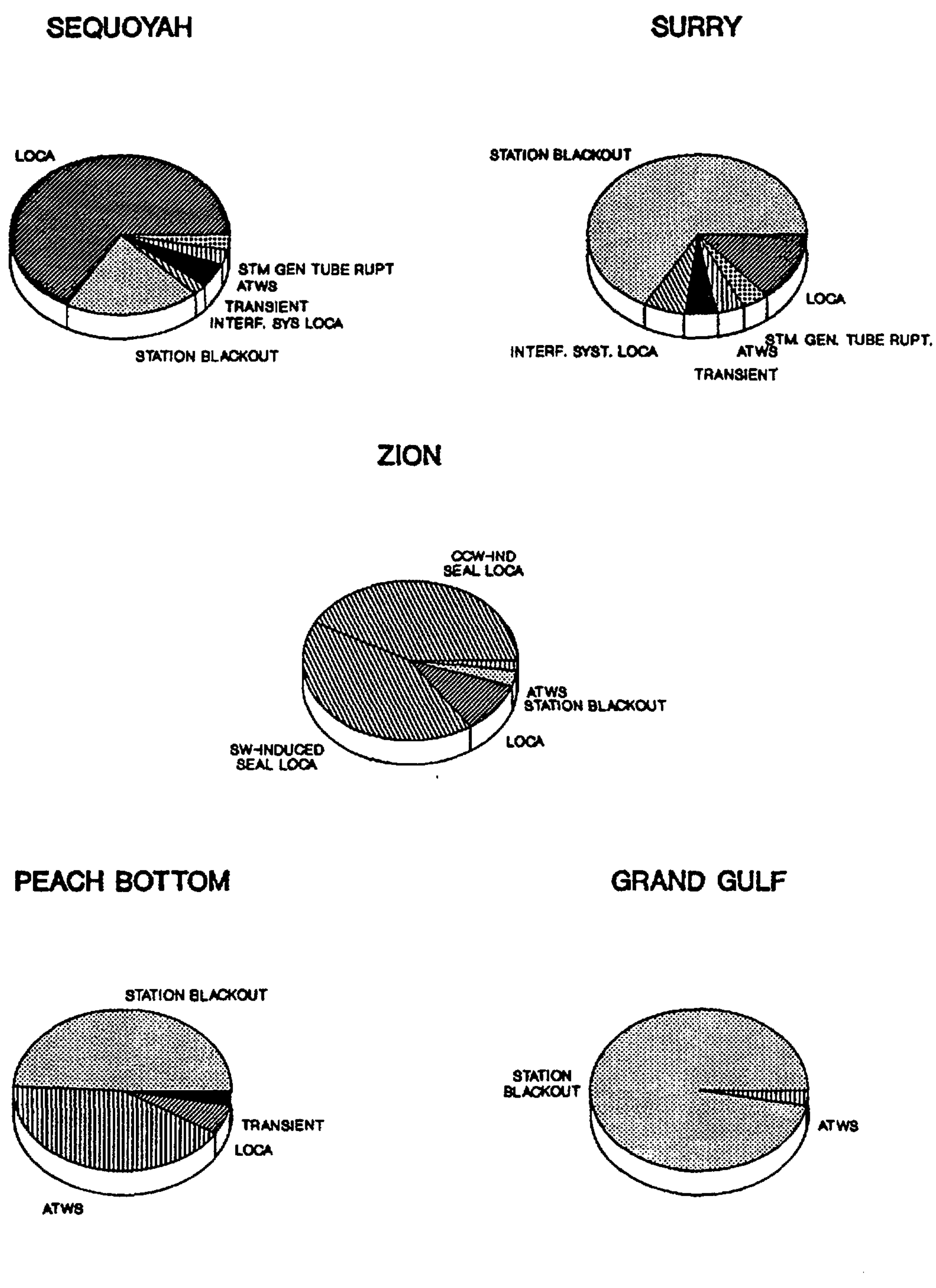


Figure 8.4 Principal contributors to internal core damage frequencies.

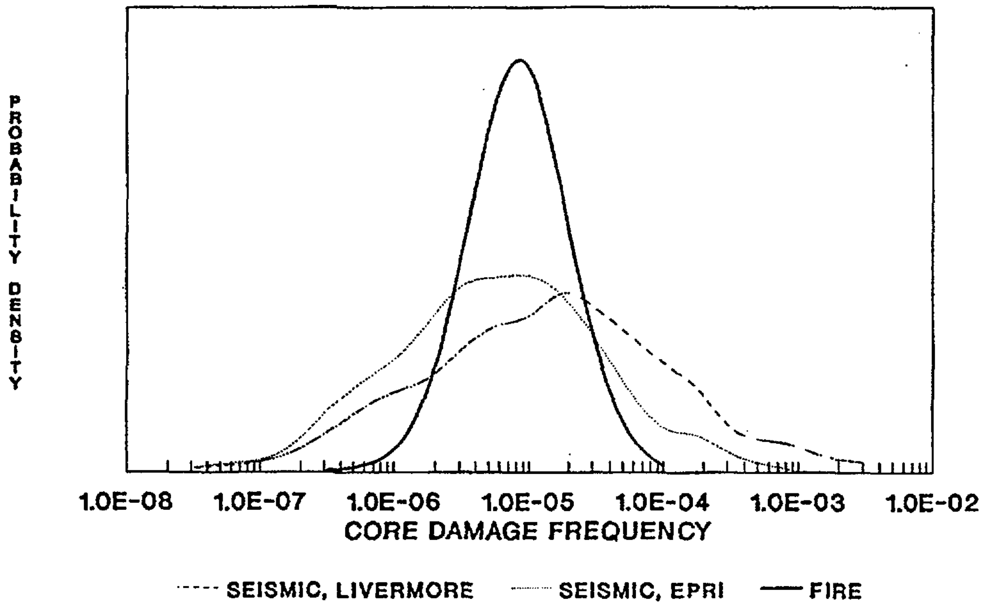
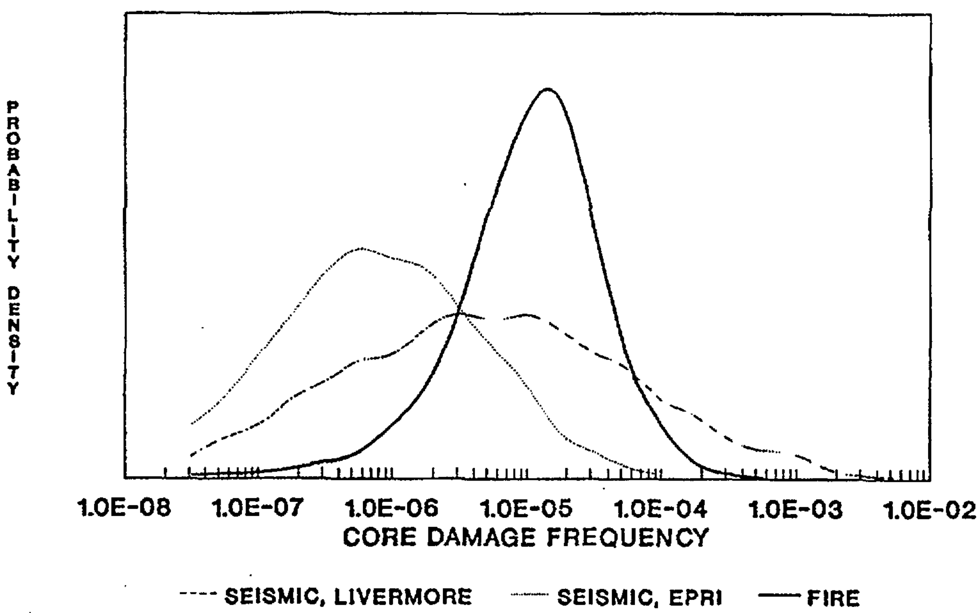


Figure 8.5 Surry external-event core damage frequency distributions.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.6 Peach Bottom external-event core damage frequency distributions.

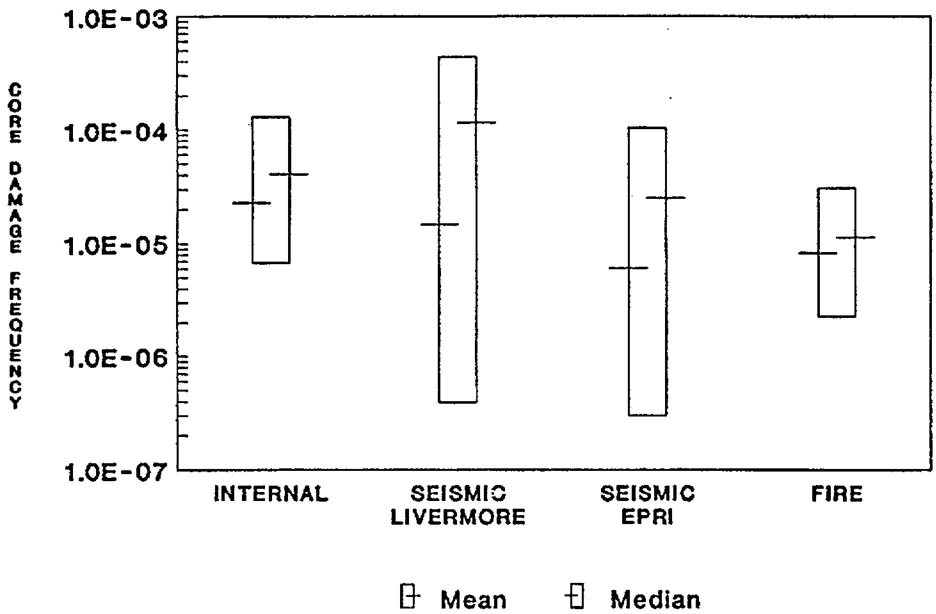
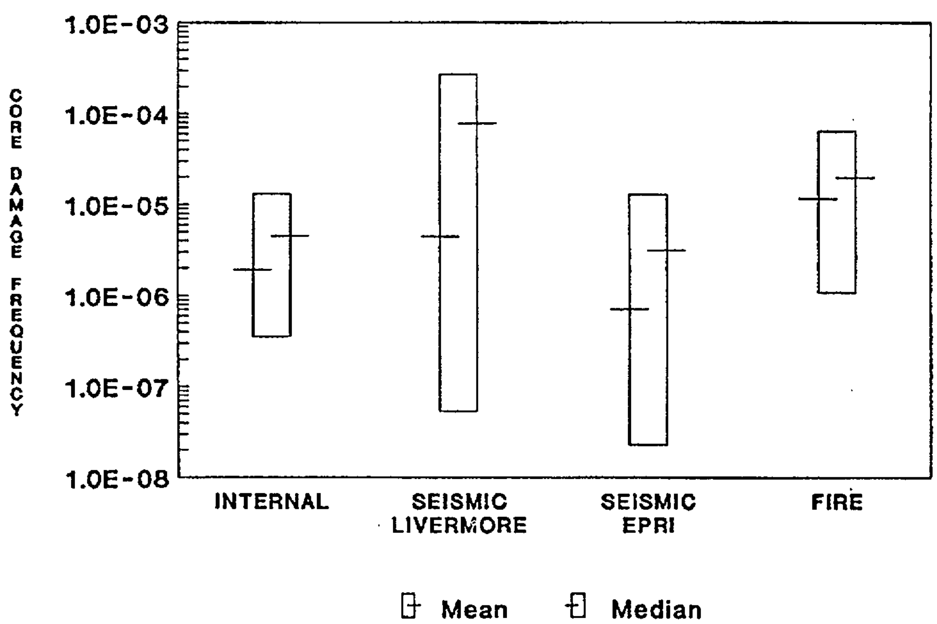


Figure 8.7 Surry internal- and external-event core damage frequency ranges.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.8 Peach Bottom internal- and external-event core damage frequency ranges.

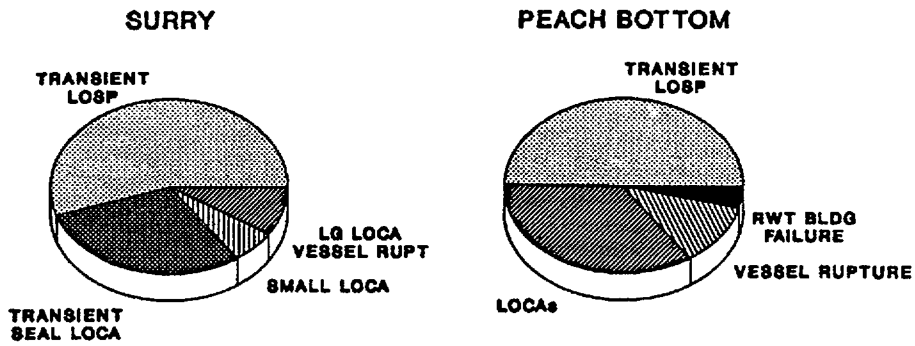


Figure 8.9 Principal contributors to seismic core damage frequencies.

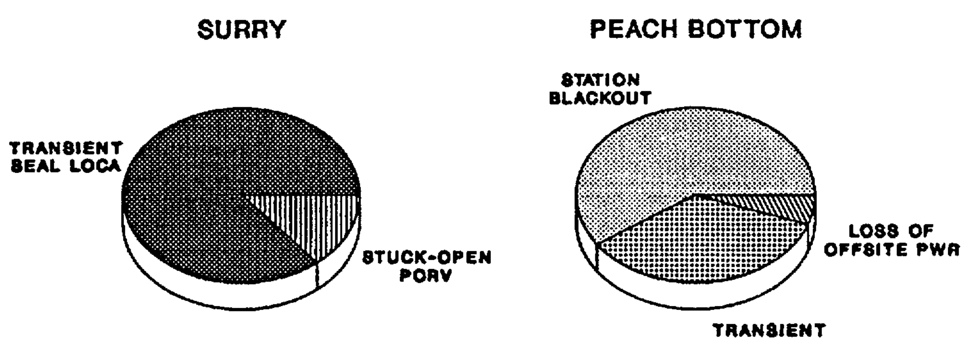


Figure 8.10 Principal contributors to fire core damage frequencies.

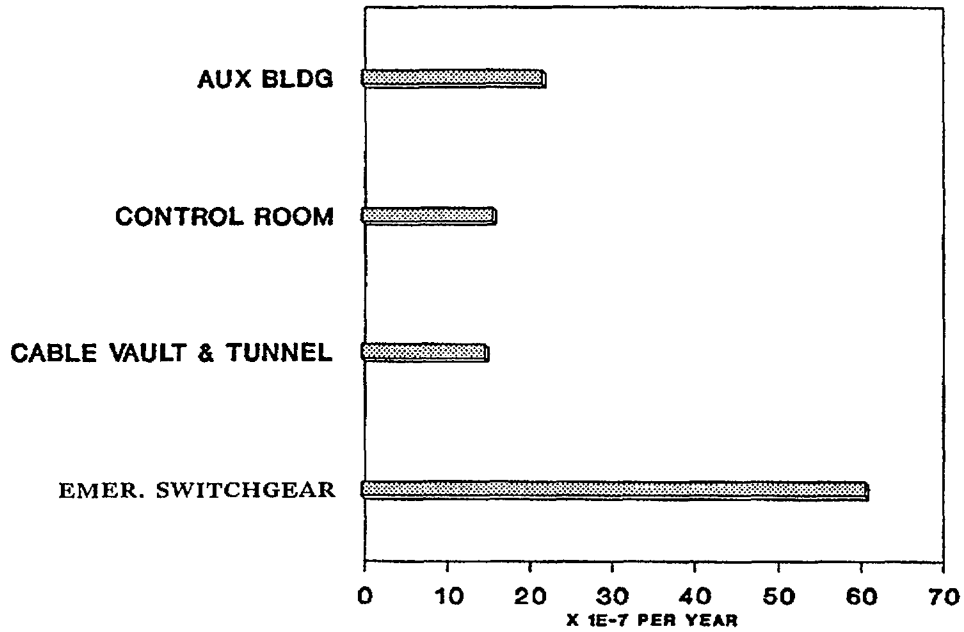
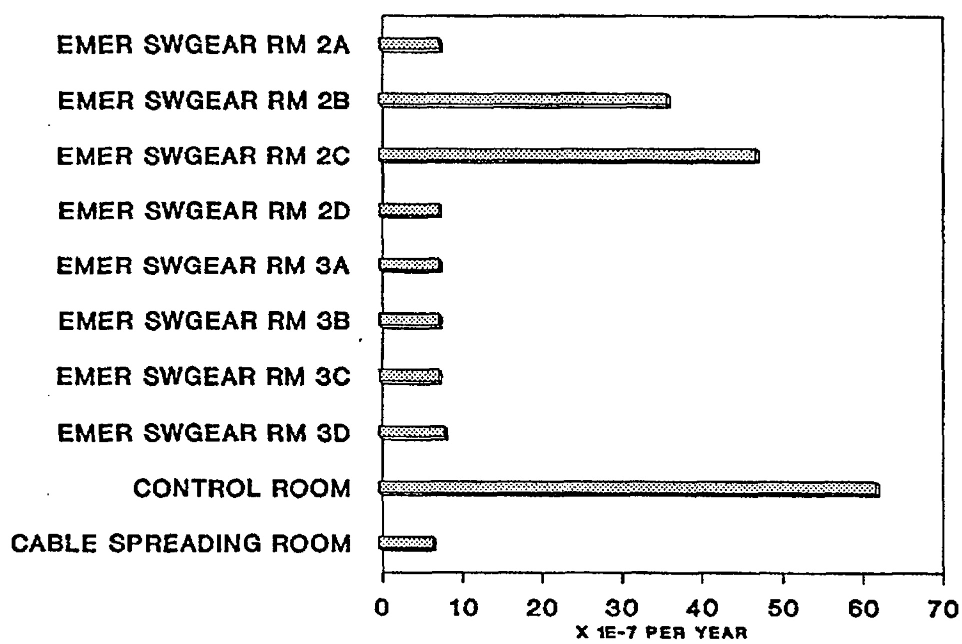


Figure 8.11 Surry mean fire core damage frequency by fire area.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.12 Peach Bottom mean fire core damage frequency by fire area.

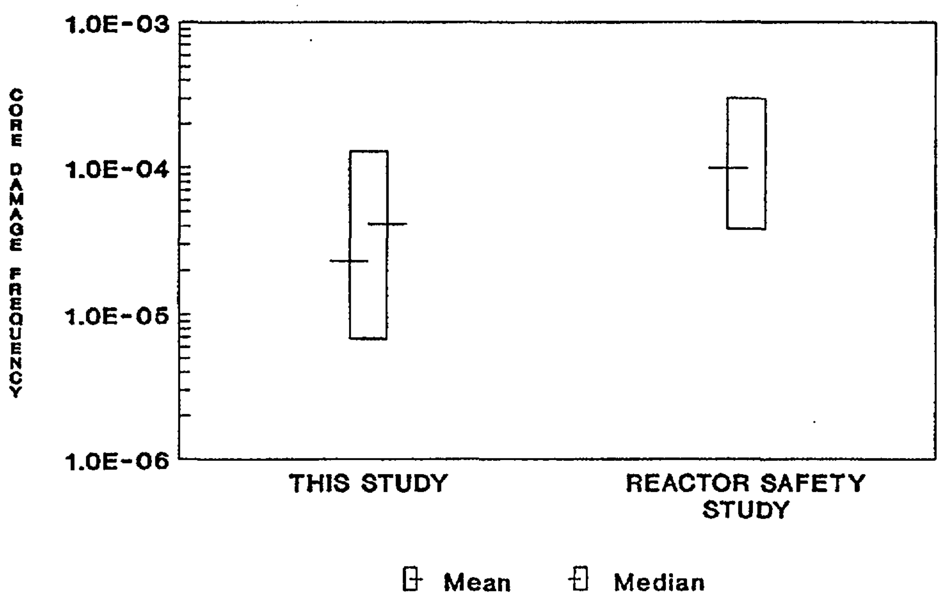
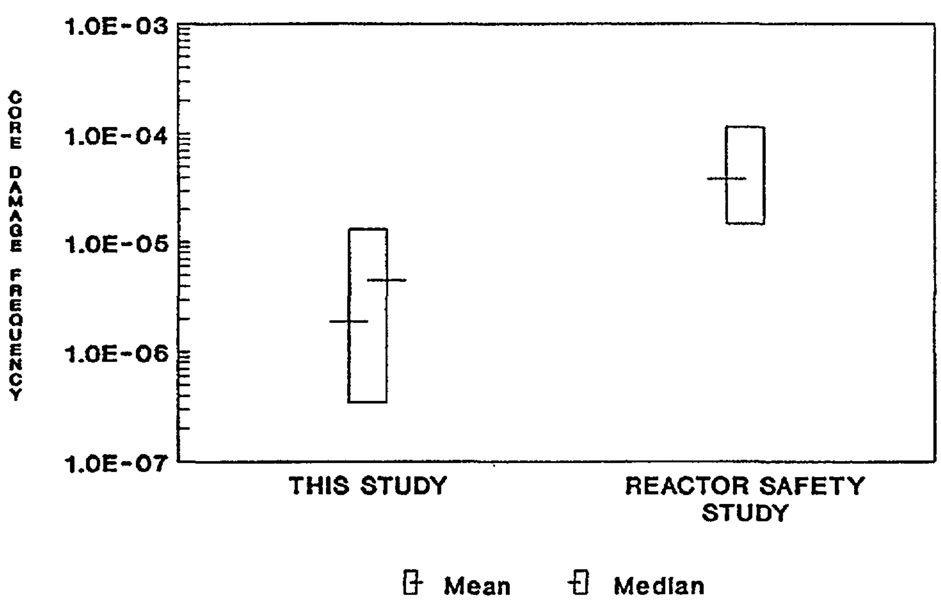


Figure 8.13 Comparison of Surry internal core damage frequency with Reactor Safety Study.



Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Figure 8.14 Comparison of Peach Bottom internal core damage frequency with Reactor Safety Study.

recovery of ac power, the Reactor Safety Study did not consider recovery actions). Thus, the net effect of including these new techniques and experience is plant specific and can shift core damage frequencies in either higher or lower directions.

In the case of the Surry analysis, the Reactor Safety Study found the core damage frequency to be dominated by loss-of-coolant accidents (LOCAs). For the present study, station blackout accidents are dominant, while the LOCA-induced core damage frequency is substantially reduced from that of the Reactor Safety Study, particularly for the small LOCA events. This occurred in spite of a tenfold increase in the small LOCA initiating event frequency estimates, which was a result of the inclusion of reactor coolant pump seal failures. One reason for the reduction lies in plant modifications made since the Reactor Safety Study was completed. These modifications allow for the crossconnection of the high-pressure safety injection systems, auxiliary feedwater systems, and refueling water storage tanks between the two units at the Surry site. These crossties provide a reliable alternative for recovery of system failures. Thus, the plant modifications (the crossconnections) have driven the core damage frequencies downward, but new PRA information (the higher small LOCA frequency) has driven them upward. In this case, the net effect is an overall reduction in the core damage frequency for internal events.

In the case of Peach Bottom, the Reactor Safety Study found the core damage frequency to be comprised primarily of ATWS accident sequences and of transients with long-term failure of decay heat removal. The present study concludes that station blackout scenarios are dominant. The possibility of containment venting and allowing for some probability of core cooling after containment failure has considerably reduced the significance of the long-term loss of decay heat removal accidents. In addition, the plant has implemented some ATWS improvements, although ATWS events remain among the dominant accident sequence types. Moreover, more modern neutronic and thermal-hydraulic simulations of the ATWS sequences have calculated lower core power levels during the event, allowing more opportunity for mitigation such as through the use of low-pressure injection systems. Thus, for Peach Bottom, both advances in PRA methodology and plant modifications have contributed to a reduction in the estimated core damage frequency from internal events.

In summary, there have been reductions in the core damage frequencies for both plants since the Reactor Safety Study. The reduction in core damage frequency for Peach Bottom is more significant than for Surry; however, there is still considerable overlap of the uncertainty ranges of the two studies. The conclusion to be drawn is that the hardware and procedural changes made since the Reactor Safety Study appear to have reduced the core damage frequency at these two plants, even when accounting for more accurate failure data and reflecting new sequences not identified in the Reactor Safety Study (e.g., the reactor coolant pump seal LOCA).

8.4 Perspectives

8.4.1 Internal-Event Core Damage Probability Distributions

The core damage frequencies produced by all PRAs inherently have large uncertainties. Therefore, comparisons of frequencies between PRAs or with absolute limits or goals are not simply a matter of comparing two numbers. It is more appropriate to observe how much of the probability distribution lies below a given point, which translates into a measure of the probability that the point has not been exceeded. For example, if the median were exactly equal to the point in question, half of the distribution would lie above and half below the point, and there would be a 50 percent probability that the point had not been exceeded.

Similarly, when comparing core damage frequencies calculated for two or more plants, it is not sufficient to simply compare the mean values of the probability distributions. Instead, one must compare the entire distribution. If one plant's distribution were almost entirely below that of another, then there would be a high probability that the first plant had a lower core damage frequency than the second. Seldom is this the case, however. Usually, the distributions have considerable overlap, and the probability that one plant has a higher or lower core damage frequency than another must be calculated. References 8.1 through 8.5 contain more detailed information on the distributions that would support such calculations.

Although the distributions are not compared in detail here, the overlap of such core damage frequency distributions is clearly shown in Figure 8.1. For example, one can have relatively high confidence that the internal-event core damage frequency for Grand Gulf is lower than that of Sequoyah or Surry. Conversely, it can readily be seen that the differences in core damage frequency between Surry and Sequoyah are not very significant.

Interpretation of extremely low median or mean core damage frequencies (<1E-5) is somewhat difficult. As discussed in Section 1.3 and in Reference 8.7, there are limitations in the scope of the study that could lead to actual core damage frequencies higher than those estimated. In addition, the uncertainties in the sequences included in the study tend to become more important on a relative scale as the frequency decreases. A very low core damage frequency is evident for Grand Gulf with the median of the distribution in the range of 1E-6 per reactor year. However, it is incomplete to simply state that the core damage frequency for this plant is that low since the 95th percentile exceeds 1E-5 per reactor year. Thus, although the central tendency of the calculation is very low, there is still a finite probability of a higher core damage frequency, particularly when considering that the scope of the study does not include certain types of accidents as discussed in Section 1.3.

8.4.2 Principal Contributors to Uncertainty in Core Damage Frequency

In Section 8.4.3, analyses are discussed concerning some of the issues and events that contribute to the magnitude of the core damage frequency. Generally, for the accident frequency analysis, the issues that contribute most to the magnitude of the frequency are also the issues that contribute most to the estimated uncertainty. More detail concerning the contributions of various parameters to the uncertainty in core damage frequency may be found in References 8.1 through 8.5. Perspectives on the contributions of accident frequency issues to the uncertainty in risk may be found in Chapter 12.

8.4.3 Dominant Accident Sequence Types

The various accident sequences that contribute to the total core damage frequency can be grouped by common factors into categories. Older PRAs generally did this in terms of the initiating event, e.g., transient, small LOCA, large LOCA. Current practice also uses categories, such as ATWS, seal LOCA, and station blackout. Generally, these categories are not equal contributors to the total core damage frequency. In practice, four or five sequence categories, sometimes fewer, usually contribute almost all the core damage frequency. These will be referred to below as the dominant plant damage states (PDSs).

It should be noted that the selection of categories is not unique in a mathematical sense, but instead is a convenient way to group the results. If the core damage frequency is to be changed, changing something common to the dominant PDS will have the most effect. Thus, if a particular plant had a relatively high core damage frequency and a particular group of sequences were high, a valuable insight into that plant's safety profile would be obtained.

It should also be noted that the importance of the highest frequency accident sequences should be considered in relationship to the total core damage frequency. The existence of a highly dominant accident sequence or PDS does not of itself imply that a safety problem exists. For example, if a plant already had an extremely low estimated core damage frequency, the existence of a single, dominant PDS would have little significance. Similarly, if a plant were modified such that the dominant PDS were eliminated entirely, the next highest PDS would become the most dominant contributor.

Nevertheless, it is the study of the dominant PDS and the important failures that contribute to those sequences that provides understanding of why the core damage frequency is high or low relative to other plants and desired goals. This qualitative understanding of the core damage frequency is necessary to make practical use of the PRA results and improve the plants, if necessary.

Given this background, the dominant PDSs for the five studies are illustrated in Figures 8.2, 8.3, and 8.4. Additional discussion of these PDSs can be found in Chapters 3 through 7. Several observations on these PDSs and their effects on the core damage frequency can be made, as discussed below.

Boiling Water Reactor versus Pressurized Water Reactor

It is evident from Figure 8.1 that the two particular BWRs in this study have internal-event core damage frequency distributions that are substantially lower than those of the three PWRs. While it would be inappropriate to conclude that all BWRs have lower core damage frequencies than PWRs, it is useful to consider why the core damage frequencies are lower for these particular BWRs.

The LOCA sequences, often dominant in the PWR core damage frequencies, are minor contributors in the case of the BWRs. This is not surprising in view of the fact that most BWRs have many more systems than PWRs for injecting water directly into the reactor coolant system to provide makeup. For BWRs, this includes two low-pressure emergency core cooling (ECC) systems (low-pressure coolant injection and low-pressure core spray), each of which is multitrain; two high-pressure injection systems (reactor core isolation cooling and either high-pressure coolant injection or high-pressure core spray); and usually several other alternative injection systems, such as the control rod drive hydraulic system, condensate, service water, firewater, etc. In contrast, PWRs generally have one high-pressure and one low-pressure ECC system (both multitrain), plus a set of accumulators. The PWR ECCS does have considerable redundancy, but not as much as that of most BWRs.

For many types of transient events, the above arguments also hold. BWRs tend to have more systems that can provide decay heat removal than PWRs. For transient events that lead to loss of water inventory due to stuck-open relief valves or primary system leakage, BWRs have numerous systems to provide makeup. ATWS events and station blackout events, as discussed below, affect both PWRs and BWRs.

BWRs have historically been considered more subject than PWRs to ATWS events. This perception was partly due to the fact that some ATWS events in a BWR involve an insertion of positive reactivity. Except for the infrequent occurrence of an unfavorable moderator temperature coefficient, an ATWS event in a PWR is slower, allowing more time for mitigative action.

In spite of this historical perspective for ATWS, it is evident from Figures 8.2 and 8.3 that the ATWS frequencies for the two BWRs are not dramatically higher than for the PWRs. There are several reasons for this. First, plant procedures for dealing with ATWS events have been modified over the past several years, and operator training specifically for these events has improved significantly. Second, the ability to model and analyze ATWS events has improved. More modern neutronic and thermal-hydraulic simulations of the ATWS sequences have calculated lower core power levels during the event than predicted in the past. Further, these calculations indicate that low-pressure injection systems can be used without resulting in significant power oscillations, thus allowing more opportunity for mitigation. Note that for both BWRs and PWRs the frequency of reactor protection system failure remains highly uncertain. Therefore, all comparisons concerning ATWS should be made with caution.

Station blackout accidents contribute a high percentage of the core damage frequency for the BWRs. However, when viewed on an absolute scale, station blackout has a higher frequency at the PWRs than at the BWRs. To some extent this is due to design differences between BWRs and PWRs leading to different susceptibilities. For example, in station blackout accidents, PWRs are potentially vulnerable to reactor coolant pump seal LOCAs following loss of seal cooling, leading to loss of inventory with no method for providing makeup. BWRs, on the other hand, have at least one injection system that does not require ac power. While important, it would be incorrect to imply that the differences noted above are the only considerations that drive the variations in the core damage frequency. Probably more important is the electric power system design at each plant, which is largely independent of the plant type. The station blackout frequency is low at Peach Bottom because of the presence of four diesels that can be shared between units and a maintenance program that led to an order of magnitude reduction in the diesel generator failure rates. Grand Gulf has essentially three trains of emergency ac power for one unit, with one of the trains being both diverse and independent from the other two. These characteristics of the electric power system design tend to dominate any differences in the reactor design. Therefore, a BWR with a below average electric power system reliability could be expected to have a higher station blackout-induced core damage frequency than a PWR with an above average electric power system.

For both BWRs and PWRs, the analyses indicate that, along with electric power, other support systems, such as service water, are quite important. Because these systems vary considerably among plants, caution must be exercised when making statements about generic classes of plants, such as PWRs versus BWRs. Once significant plant-specific vulnerabilities are removed, support-system-driven sequences will probably dominate the core damage frequency of both types of plants. Both types of plants have sufficient redundancy and diversity so as to make multiple independent failures unlikely. Support system failures introduce dependencies among the systems and thus can become dominant.

Boiling Water Reactor Observations

As shown in Figure 8.1, the internal-event core damage frequencies for Peach Bottom and Grand Gulf are extremely low. Therefore, even though dominant plant damage states and contributing failure events can be identified, these items should not be considered as safety problems for the two plants. In fact, these dominating factors should not be overemphasized because, for core damage frequencies below 1E-5, it is possible that other events outside the scope of these internal-event analyses are the ones that actually dominate. In the cases of these two plants, the real perspectives come not from understanding why particular sequences dominate, but rather why all types of sequences considered in the study have low frequencies for these plants.

Previously it was noted that LOCA sequences can be expected to have low frequencies at BWRs because of the numerous systems available to provide coolant injection. While low for both plants, the frequency of LOCAs is higher for Peach Bottom than for Grand Gulf. This is primarily because Grand Gulf is a BWR-6 design with a motor-driven high-pressure core spray system, rather than a steam-driven high-pressure coolant injection system as is Peach Bottom. Motor-driven systems are typically more reliable than steam-driven systems and, more importantly, can operate over the entire range of pressures experienced in a LOCA sequence.

It is evident from Figures 8.2 and 8.4 that station blackout plays a major role in the internal-event core damage frequencies for Peach Bottom and Grand Gulf. Each of these plants has features that tend to reduce the station blackout frequency, some of which would not be present at other BWRs.

Grand Gulf, like all BWR-6 plants, is equipped with an extra diesel generator dedicated to the high-pressure core spray system. While effectively providing a third train of redundant emergency ac power for decay heat removal, the extra diesel also provides diversity, based on a different diesel design and plant location relative to the other two diesels. Because of the aspect of diversity, the analysis neglected common-cause failures affecting all three diesel generators. The net effect is a highly reliable emergency ac power capability. In those unlikely cases where all three diesel generators fail, Grand Gulf relies on a steam-driven coolant injection system that can function until the station batteries are depleted. At Grand Gulf the batteries are sized to last for many hours prior to depletion so that there is a high probability of recovering ac power prior to core damage. In addition, there is a diesel-driven firewater system available that can be used to provide coolant injection in some sequences involving the loss of ac power.

Peach Bottom is an older model BWR that does not have a diverse diesel generator for the high-pressure core spray system. However, other factors contribute to a low station blackout frequency at Peach Bottom. Peach Bottom is a two-unit site, with four diesel generators available. Any one of the four diesels can provide sufficient capacity to power both units in the event of a loss of offsite power, given that appropriate crossties or load swapping between Units 2 and 3 are used. This high level of redundancy is somewhat offset by a less redundant service water system that provides cooling to the diesel generators. Subtleties in the design are such that if a certain combination of diesel generators fails, the service water system will fail, causing the other diesels to fail. In addition, station dc power is needed to start the diesels. (Some emergency diesel generator systems, such as those at Surry, have a separate dedicated dc power system just for starting purposes.) In spite of these factors, the redundancy in the Peach Bottom emergency ac power system is considerable.

While there is redundancy in the ac power system design at Peach Bottom, the most significant factor in the low estimated station blackout frequency relates to the plant-specific data analysis. The plant-specific analysis determined that, because of a high-quality maintenance program, the diesel generators at Peach Bottom had approximately an order of magnitude greater reliability than at an average plant. This factor directly influences the frequency.

Finally, Peach Bottom, like Grand Gulf, has station batteries that are sized to last several hours in the event that the diesel generators do fail. With two steam-driven systems to provide coolant injection and several hours to recover ac power prior to battery depletion, the station blackout frequency is further reduced.

Unlike most PWRs, the response of containment is often a key in determining the core damage frequency for BWRs. For example, at Peach Bottom, there are a number of ways in which containment conditions can affect coolant injection systems. High pressure in containment can lead to closure of primary system relief valves, thus failing low-pressure injection systems, and can also lead to failure of steam-driven high-pressure injection systems due to high turbine exhaust backpressure. High suppression pool temperatures can also lead to the failure of systems that are recirculating water from the suppression pool to the reactor coolant system. If the containment ultimately fails, certain systems can fail because of the loss of net positive suction head in the suppression pool, and also the reactor building is subjected to a harsh steam environment that can lead to failure of equipment located there.

Despite the concerns described in the previous paragraph, the core damage frequency for Peach Bottom is relatively low, compared to the PWRs. There are two major reasons for this. First, Peach Bottom has the ability to vent the wetwell through a 6-inch diameter steel pipe, thus reducing the containment pressure without subjecting the reactor building to steam. While this vent cannot be used to mitigate ATWS and station blackout sequences, it is valuable in reducing the frequency of many other sequences. The second important feature at Peach Bottom is the presence of the control rod drive system, which is not affected by either high pressure in containment or containment failure. Other plants of the BWR-4 design may be more susceptible to containment-related problems if they do not have similar features. For example, some plants have ducting, as opposed to hard piping available for venting. Venting through ductwork may lead to harsh steam environments and equipment failures in the reactor building.*

The Grand Gulf design is generally much less susceptible to containment-related problems than Peach Bottom. The containment design and equipment locations are such that containment rupture will not result in discharge of steam into the building containing the safety systems. Further, the high-pressure core spray system is designed to function with a saturated suppression pool so that it is not affected by containment failure. Finally, there are other systems that can provide coolant injection using water sources other than the suppression pool. Thus, containment failure is relatively benign as far as system operation is concerned, and there is no obvious need for containment venting.

Pressurized Water Reactor Observations

The three PWRs examined in this study reflect much more variety in terms of dominant plant damage states than the BWRs. While the sequence frequencies are generally low for most of the plant damage states, it is useful to understand why the variations among the plants occurred.

For LOCA sequences, the frequency is significantly lower at Surry than at the other two PWRs. A major portion of this difference is directly tied to the additional redundancy available in the injection systems. In addition to the normal high-pressure injection capability, Surry can crosstie to the other unit at the site for an additional source of high-pressure injection. This reduces the core damage frequency due to LOCAs and also certain groups of transients involving stuck-open relief valves.

In addition, at Sequoyah there is a particularly noteworthy emergency core cooling interaction with containment engineered safety features in loss-of-coolant accidents. In this (ice condenser) containment design, the containment sprays are automatically actuated at a very low pressure setpoint, which would be exceeded for virtually all small LOCA events. This spray actuation, if not terminated by the operator can lead to a rapid depletion of the refueling water storage tank at Sequoyah. Thus, an early need to switch to recirculation cooling may occur. Portions of this switchover process are manual at Sequoyah and, because of the timing and possible stressful conditions, leads to a significant human error probability. Thus, LOCA-type sequences are the dominant accident sequence type at Sequoyah.

Station blackout-type sequences have relatively similar frequencies at all three PWRs. Station blackout sequences can have very different characteristics at PWRs than at BWRs. One of the most important findings of the study is the importance of reactor coolant pump seal failures. During station blackout, all cooling to the seals is lost and there is a significant probability that they will ultimately fail, leading to an induced LOCA and loss of inventory. Because PWRs do not have systems capable of providing coolant makeup without ac power, core damage will result if power is not restored. The seal LOCA reduces the time available to restore power and thus increases the station blackout-induced core damage frequency. New seals have been proposed for Westinghouse PWRs and could reduce the core damage frequency if implemented, although they might also increase the likelihood that any resulting accidents would occur at high pressure, which has implications for the accident progression analysis. (See Section C.14 of Appendix C for a more detailed discussion of reactor coolant seal performance.)

Apart from the generic reactor coolant pump seal question, station blackout frequencies at PWRs are determined by the plant-specific electric power system design and the design of other support systems. Battery depletion times for the three PWRs were projected to be shorter than for the two BWRs. A particular characteristic of the

*The staff is presently undertaking regulatory action to require hard pipe vents in all BWR Mark I plants.

Surry plant is a gravity-fed service water system with a canal that may drain during station blackout, thus failing containment heat removal. When power is restored, the canal must be refilled before containment heat removal can be restored.

The dominant accident sequence type at Zion is not a station blackout, but it has many similar characteristics. Component cooling water is needed for operation of the charging pumps and high-pressure safety injection pumps at Zion. Loss of component cooling water (or loss of service water, which will also render component cooling water inoperable) will result in loss of these high-pressure systems. This in turn leads to a loss of reactor coolant pump seal injection. Simultaneously, loss of component cooling water will also result in loss of cooling to the thermal barrier heat exchangers for the reactor coolant pump seals. Thus, the reactor coolant pump seals will lose both forms of cooling. As with station blackout, loss of component cooling water or service water can both cause a small LOCA (by seal failure) and disable the systems needed to mitigate it. The importance of this scenario is increased further by the fact that the component cooling water system at Zion, although it uses redundant pumps and valves, delivers its flow through a common header. The licensee for the Zion plant has made procedural changes and is also considering both the use of new seal materials and the installation of modifications to the cooling water systems. These measures, which are discussed in more detail in Chapter 7, reduce the importance of this contributor.

ATWS frequencies are generally low at all three of the PWRs. This is due to the assessed reliability of the shutdown systems and the likelihood that only slow-acting, low-power-level events will result.

While of low frequency, it is worth noting that interfacing-system LOCA (V) and steam generator tube rupture (SGTR) events do contribute significantly to risk for the PWRs. This is because they involve a direct path for fission products to bypass containment. There are large uncertainties in the analyses of these two accident types, but these events can be important to risk even at frequencies that may be one or two orders of magnitude lower than other sequence types.

During the past few years, most Westinghouse PWRs have developed procedures for using feed and bleed cooling and secondary system blowdown to cope with loss of all feedwater. These procedures have led to substantial reductions in the frequencies of transient sequences involving the loss of main and auxiliary feedwater. Appropriate credit for these actions was given in these analyses. However, there are plant-specific features that will affect the success rate of such actions. For example, the loss of certain power sources (possibly only one bus) or other support systems can fail power-operated relief valves (PORVs) or atmospheric dump valves or their block valves at some plants, precluding the use of feed and bleed or secondary system blowdown. Plants with PORVs that tend to leak may operate for significant periods of time with the block valves closed, thus making feed and bleed less reliable. On the other hand, if certain power failures are such that open block valves cannot be closed, then they cannot be used to mitigate stuck-open PORVs. Thus, both the system design and plant operating practices can be important to the reliability assessment of actions such as feed and bleed cooling.

8.4.4 External Events

The frequency of core damage initiated by external events has been analyzed for two of the plants in this study, Surry and Peach Bottom (Ref. 8.1 (Part 3) and Ref. 8.2 (Part 3)). The analysis examined a broad range of external events, e.g., lightning, aircraft impact, tornados, and volcanic activity (Ref. 8.8). Most of these events were assessed to be insignificant contributors by means of bounding analyses. However, seismic events and fires were found to be potentially major contributors and thus were analyzed in detail.

Figures 8.7 and 8.8 show the results of the core damage frequency analysis for seismic- and fire-initiated accidents, as well as internally initiated accidents, for Surry and Peach Bottom, respectively. Examination of these figures shows that the core damage frequency distributions of the external events are comparable to those of the internal events. It is evident that the external events are significant in the total safety profile of these plants.

Seismic Analysis Observations

The analysis of the seismically induced core damage frequency begins with the estimation of the seismic hazard, that is, the likelihood of exceeding different earthquake ground-motion levels at the plant site. This is a difficult, highly judgmental issue, with little data to provide verification of the various proposed geologic and seismologic models.

The sciences of geology and seismology have not yet produced a model or group of models upon which all experts agree. This study did not itself produce seismic hazard curves, but instead made use of seismic hazard curves for Peach Bottom and Surry that were part of an NRC-funded Lawrence Livermore National Laboratory project that resulted in seismic hazard curves for all nuclear power plant sites east of the Rocky Mountains (Ref. 8.9).

In addition, the Electric Power Research Institute (EPRI) developed a separate set of models (Ref. 8.10). For purposes of completeness and comparison, the seismically induced core damage frequencies were also calculated based upon the EPRI methods. Both sets of results, which are presented in Figures 8.5 through 8.8, were used in this study. More detailed discussion of methods used in the seismic analysis is provided in Appendix A; Section C.11 of Appendix C provides more detailed perspectives on the seismic issue as well.

As can be seen in Figures 8.5 and 8.6, the shapes of the seismically induced core damage probability distributions are considerably different from those of the internally initiated and fire-initiated events. In particular, the 5th to 95th percentile range is much larger for the seismic events. In addition, as can be seen in Figures 8.7 and 8.8, the wide disparity between the mean and the median and the location of the mean relatively high in the distribution indicate a wide distribution with a tail at the high end but peaked much lower down. (This is a result of the uncertainty in the seismic hazard curve.)

It can be clearly seen that the difference between the mean and median is an important distinction. The mean is the parameter quoted most often, but the bulk of the distribution is well below the mean. Thus, although the mean is the "center of gravity" of the distribution (when viewed on a linear rather than logarithmic scale), it is not very representative of the distribution as a whole. Instead, it is the lower values that are more probable. The higher values are estimated to have low probability, but, because of their great distance from the bulk of the distribution, the mean is "pulled up" to a relatively high value. In a case such as this, it is particularly evident that the entire distribution, not just a single parameter such as the mean or the median, must be considered when discussing the results of the analysis.

1. Surry Seismic Analysis

The core damage frequency probability distributions, as calculated using the Livermore and EPRI methods, have a large degree of overlap, and the differences between the means and medians of the two resulting distributions are not very meaningful because of the large widths of the two distributions.

The breakdown of the Surry seismic analysis into principal contributors is reasonably similar to the results of other seismic PRAs for other PWRs. The total core damage frequency is dominated by loss of offsite power transients resulting from seismically induced failures of the ceramic insulators in the switchyard. This dominant contribution of ceramic insulator failures has been found in virtually all seismic PRAs to date.

A site-specific but significant contributor to the core damage frequency at Surry is failure of the anchorage welds of the 4 kV buses. These buses play a vital role in providing emergency ac electrical power since offsite power as well as emergency onsite power passes through these buses. Although these welded anchorages have more than adequate capacity at the safe shutdown earthquake (SSE) level, they do not have sufficient margin to withstand (with high reliability) earthquakes in the range of four times the SSE, which are contributing to the overall seismic core damage frequency results.

Similarly, a substantial contribution is associated with failures of the diesel generators and associated load center anchorage failures. These anchorages also may not have sufficient capacity to withstand earthquakes at levels of four times the SSE.

Another area of generic interest is the contribution due to vertical flat-bottomed storage tanks, e.g., refueling water storage tanks and condensate storage tanks. Because of the nature of their configuration and field erection practices, such tanks have often been calculated to have relatively smaller margin over the SSE than most components in commercial nuclear power plants. Given that all PWRs in the United States use the refueling water storage tank as the primary source of emergency injection water (and usually the sole source until the recirculation phase of ECCS begins), failure of the refueling water storage tank can be expected to be a substantial contributor to the seismically induced core damage frequency.

2. Peach Bottom Seismic Analysis

As can be seen in Figure 8.9, the dominant contributor in the seismic core damage frequency analysis is a transient sequence brought about by loss of offsite power. The loss of offsite power is due to seismically induced failures of onsite ac power. Peach Bottom has four emergency diesel generators, all shared between the two units, and four station batteries per unit. Thus, there is a high degree of redundancy. However, all diesels require cooling provided by the emergency service water system, and failure to provide this cooling will result in failure of all four diesels.

There is a variety of seismically induced equipment failures that can fail the emergency service water system and result in a station blackout. These include failure of the emergency cooling tower, failures of the 4 kV buses (in the same manner as was found at Surry), and failures of the emergency service water pumps or the emergency diesel generators themselves. The various combinations of these failures result in a large number of potential failure modes and give rise to a relatively high frequency of core damage based on station blackout. None of these equipment failure probabilities is substantially greater than would be implied by the generic fragility data available. However, the high probability of exceedance of larger earthquakes (as prescribed by the hazard curves for this site) results in significant contributions of these components to the seismic risk.

Fire Analysis Observations

The core damage likelihood due to a fire in any particular area of the plant depends upon the frequency of ignition of a fire in the area, the amount and nature of combustible material in that area, the nature and efficacy of the fire-suppression systems in that area, and the importance of the equipment located in that area, as expressed in the potential of the loss of that equipment to cause a core damage accident sequence. The methods used in the fire analysis are described in Appendix A and in Reference 8.7; Section C.12 of Appendix C provides additional perspectives on the fire analysis.

1. Surry Fire Analysis

Figure 8.10 shows the dominant contributors to core damage frequency resulting from the Surry fire analysis. The dominant contributor is a transient resulting in a reactor coolant pump seal LOCA, which can lead to core damage. The scenario consists of a fire in the emergency switchgear room that damages power or control cables for the high-pressure injection and component cooling water pumps. No additional random failures are required for this scenario to lead to core damage. It should be noted that credit was given for existing fire-suppression systems and for recovery by crossconnecting high-pressure injection from the other unit. The importance of this scenario is evident in Figure 8.11, which breaks down the fire-induced core damage frequency by location in the plant. The most significant physical location is the emergency switchgear room. In this room, cable trays for the two redundant power trains were run one on top of the other with approximately 8 inches of vertical separation in a number of plant areas, which gives rise to the common vulnerability of these two systems due to fire. In addition, the Halon fire-suppression system in this room is manually actuated.

The other principal contributor is a spuriously actuated pressurizer PORV. In this scenario, fire-related component damage in the control room includes control power for a number of safety systems. Full credit was given for independence of the remote shutdown panel from the control room except in the case of PORV block valves; discussions with utility personnel indicated that control power for these valves was not independently routed.

2. Peach Bottom Fire Analysis

Figure 8.10 shows the mechanisms by which fire leads to core damage in the Peach Bottom analysis. Station blackout accidents are the dominant contributor, with substantial contributions also coming from fire-induced transients and losses of offsite power. The relative importance of the various physical locations is shown in Figure 8.12.

It is evident from Figure 8.12 that control room fires are of considerable significance in the fire analysis of this plant. Fires in the control room were divided into two scenarios, one for fires initiating in the reactor core isolation cooling (RCIC) system cabinet and one for all others. Credit was given for automatic cycling of the RCIC system unless the fire initiated within its control panel. Because of the cabinet configuration within the control room, the fire was assumed not to spread and damage any components outside the cabinet where the fire initiated. The analysis gave credit for the possibility of quick extinguishing of the fire within the applicable cabinet since the control room is continuously occupied. However, should these efforts fail, even with high ventilation rates, these scenarios postulate forced abandonment of the control room due to smoke from the fire and subsequent plant control from the remote shutdown panel.

The cable spreading room below the control room is significant but not dominant in the fire analysis. The scenario of interest is a fire-induced transient coupled with fire-related failures of the control power for the high-pressure coolant injection system, the reactor core isolation cooling system, the automatic depressurization system, and the control rod drive hydraulic system. The analysis gave credit to the automatic CO_2 fire-suppression system in this area.

The remaining physical areas of significance are the emergency switchgear rooms. The fire-induced core damage frequency is dominated by fire damage to the emergency service water system in conjunction with random failures coupled with fire-induced loss of offsite power. In all eight emergency switchgear rooms (four shared between the two units), both trains of offsite power are routed. It was noted that in each of these areas there are breaker cubicles for the 4 kV switchgear with a penetration at the top that has many small cables routed through it. These penetrations were inadequately sealed, which would allow a fire to spread to cabling that was directly above the switchgear room. This cabling was a sufficient fuel source for the fire to cause a rapid formation of a hot gas layer that would then lead to a loss of offsite power. Since both offsite power and the emergency service water systems are lost, a station blackout would occur.

Perspectives: General Observations on Fire Analysis

Figures 8.7 and 8.8 clearly indicate that fire-initiated core damage sequences are significant in the total probabilistic analysis of the two plants analyzed. Moreover, these analyses already include credit for the fire protection programs required by Appendix R to 10 CFR Part 50.

Although the two plants are of completely different design, with completely different fire-initiated core damage scenarios, the possibility of fires in the emergency switchgear areas is important in both plants. The importance of the emergency switchgear room at Surry is particularly high because of the seal LOCA scenario. Further, the importance of the control room at Surry is comparable to that of the control room at Peach Bottom.

This is not surprising in view of the potential for simultaneous failure of several systems by fires in these areas. Thus, in the past such areas have generally received particular attention in fire protection programs. It should also be noted that the significance of various areas also depends upon the scenario that leads to core damage. For example, the importance of the emergency switchgear room at Surry could be altered (if desired) not only by more fire protection programs but also by changes in the probability of the reactor coolant pump seal failure.

REFERENCES FOR CHAPTER 8

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

8.2 A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

8.3 R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

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

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

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

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

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

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

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

9. PERSPECTIVES ON ACCIDENT PROGRESSION AND CONTAINMENT PERFORMANCE

9.1 Introduction

The consequences of severe reactor accidents depend greatly on containment safety features and containment performance in retaining radioactive material. The early failure of the containment structures at the Chernobyl power plant contributed to the size of the environmental release of radioactive material in that accident. In contrast, the radiological consequences of the Three Mile Island Unit 2 (TMI-2) accident were minor because overall containment integrity was maintained and bypass was small. Normally three barriers (the fuel rod cladding, the reactor coolant system pressure boundary, and the containment pressure boundary) protect the public from the release of radioactive material generated in nuclear fuel. In most core meltdown scenarios, the first two barriers would be progressively breached, and the containment boundary represents the final barrier to release of radioactivity to the environment. Maintaining the integrity of the containment can affect the source term by orders of magnitude. The NRC's 1986 reassessment of source term issues reaffirmed that containment performance "is a major factor affecting source terms" (Ref. 9.1).

In most severe accident sequences, the ability of a containment boundary to maintain integrity is determined by two factors: (1) the magnitude of the loads, and (2) the response to those loads of the containment structure and the penetrations through the containment boundary. Although there is no universally accepted definition of containment failure, it does not necessarily imply gross structural failure. For risk purposes, containment is considered to have failed to perform its function when the leak rate of radionuclides to the environment is substantial. Thus, failure could occur as the result of a structural failure of the containment, tearing of the containment liner, or a high rate of a leakage through a penetration. Finally, valves that are open during normal operation may not close properly when the accident occurs. Failure of the containment isolation system can result in leakage of radioactive material to a secondary building or directly to the environment.

In some accidents, the containment building is completely bypassed. In interfacing-system loss-of-coolant accidents (LOCAs), check valves isolating low-pressure piping fail, and the piping connected to the reactor coolant system fails outside the containment. The radionuclides can escape to secondary buildings through the reactor coolant system piping without passing through the containment. A similar bypass can occur in a core meltdown sequence initiated by the rupture of a steam generator tube in which release is through relief valves on the steam line from the failed steam generators.

Although the five plants analyzed in the present study were selected to span the basic types of containment design used in the United States, it cannot be assumed that the containment performance results obtained are characteristic of a class of plants. The loads in an accident sequence, the relative frequencies of specific accident sequences, and the load level at which the containment fails can all be influenced by design details that vary among reactors within a class of containments. (Additional discussion of the extrapolability of PRA results is provided in Chapter 13.)

9.2 Summary of Results

If the containment function is maintained in a severe accident, the radiological consequences will be minor. If the containment function does fail, the timing of failure can be very important. The longer the containment remains intact relative to the time of core melting and radionuclide release from the reactor coolant system, the more time is available to remove radioactive material from the containment atmosphere by engineered safety features or natural deposition processes. Delay in containment failure or containment bypass also provides time for protective action, a very important consideration in the assessment of possible early health effects. Thus, in evaluating the performance of a containment, it is convenient to consider no failure, late failure, bypass, and early failure of containment as separate categories characterizing different degrees of severity. For those plants in which intentional venting is an option, this is also represented as a separate category.

Not all accident sequences that involve core damage would necessarily progress to vessel failure, as illustrated by the TMI-2 accident. The operator may recover a critical system (such as by the return of offsite power) or the state of the plant may change (for example, the system pressure may fall to a point where low-pressure emergency coolant systems can be activated) allowing the core to be recovered and the accident to be terminated. The likelihood of containment failure in terminated accidents is typically less than in accidents involving vessel failure, and the radiological consequences are usually very small.

9.2.1 Internal Events

The probability of early containment failure and vessel breach conditional on the indicated class of sequence (and the mean frequency of the class) is illustrated in Figure 9.1 for three classes of accident sequences in the pressurized water reactors (PWRs) analyzed in this study and in Figure 9.2 for three classes of accident sequences in the boiling water reactors (BWRs) analyzed (Refs. 9.2 through 9.6). Containment bypass scenarios are not included in these figures, and the results are for internally initiated accidents. For different plant designs, the nature of the loads and the response of the containment are different, even for the same accident class.

The predicted likelihoods of early containment failure in the Zion (large, dry design) plant and the Surry (subatmospheric design) plant are quite small (mean value of about 1 percent). The principal mechanisms leading to these failures are loads resulting from high-pressure melt ejection in accident sequences with high reactor coolant system (RCS) pressures (at time of vessel breach) and in-vessel steam explosions in sequences with low RCS pressures at vessel breach. Both phenomena involve substantial uncertainties.

The principal reason that the probability of early containment failure from loads at vessel breach is so small in the Surry and Zion analyses is that the reactor coolant system is not likely to be at high pressure when vessel meltthrough occurs. Some of the mechanisms that were found to be effective in depressurizing the vessel are hot leg or surge line failure at elevated temperature, failure of a reactor coolant pump seal, or a stuck-open relief valve. If an extreme case at Surry is selected, which is a large core fraction ejected, a dry cavity, no sprays, a large hole in the vessel, and high reactor coolant system pressure, the conditional probability of containment failure is approximately 30 percent. However, this is a very unlikely case. For cases with small holes in the reactor vessel and a small or intermediate fraction of the core ejected, which are much more likely, the probability of containment failure is a few percent or less.

For accident sequences at Surry and Zion in which core uncovery is initiated with the reactor coolant system at high pressure, the probability of overheating and rupturing steam generator tubes after the onset of core damage, with subsequent bypass of the containment, is of the same magnitude as the probability of early containment failure from high-pressure ejection of core debris with direct containment heating. In Figure 9.1, the smaller spread in uncertainty in the downward direction for the Zion plant is due to the higher frequency of containment isolation failure, which establishes a lower bound for the distribution.

The results for the Sequoyah plant indicate that early containment failure is somewhat more likely for ice condenser designs than for large, high-pressure containments. The mean likelihood of early failure is approximately 12 percent (8 percent includes vessel breach, 4 percent does not). Early containment failure is primarily the result of loads at vessel failure. For scenarios in which the vessel is at high pressure at the time of vessel breach, early failure results from overpressurization (including the pressure load from hydrogen burning) or from direct attack of the containment by hot debris following failure of the seal table. If the vessel is at low pressure at vessel breach, the principal failure mechanism is overpressurization.

The predicted probability of early failure of the Peach Bottom and Grand Gulf pressure-suppression containments is substantially higher than for the PWR containment designs. For Grand Gulf, the mean probability of early failure is approximately 50 percent while at Peach Bottom the mean probability of early failure is about 56 percent.

In the Peach Bottom (Mark I design) plant, failure is predicted to occur primarily in the drywell as a result of direct attack by molten core debris. Drywell rupture due to pedestal failure or rapid overpressurization (more quickly than the water columns in the vent lines can be cleared) is also an important contributor to early containment failure. If failure occurs in the drywell, releases of radionuclides from fuel after vessel failure will not pass through the suppression pool. Late failure of containment is also most likely to occur in the drywell but in the form of prolonged leakage past the drywell head.

At Grand Gulf, early containment failure in station blackout is dominated by hydrogen deflagrations. Hydrogen detonations are also small contributors to early failure. For short-term station blackouts (the dominant plant damage state groups), the conditional probability of early

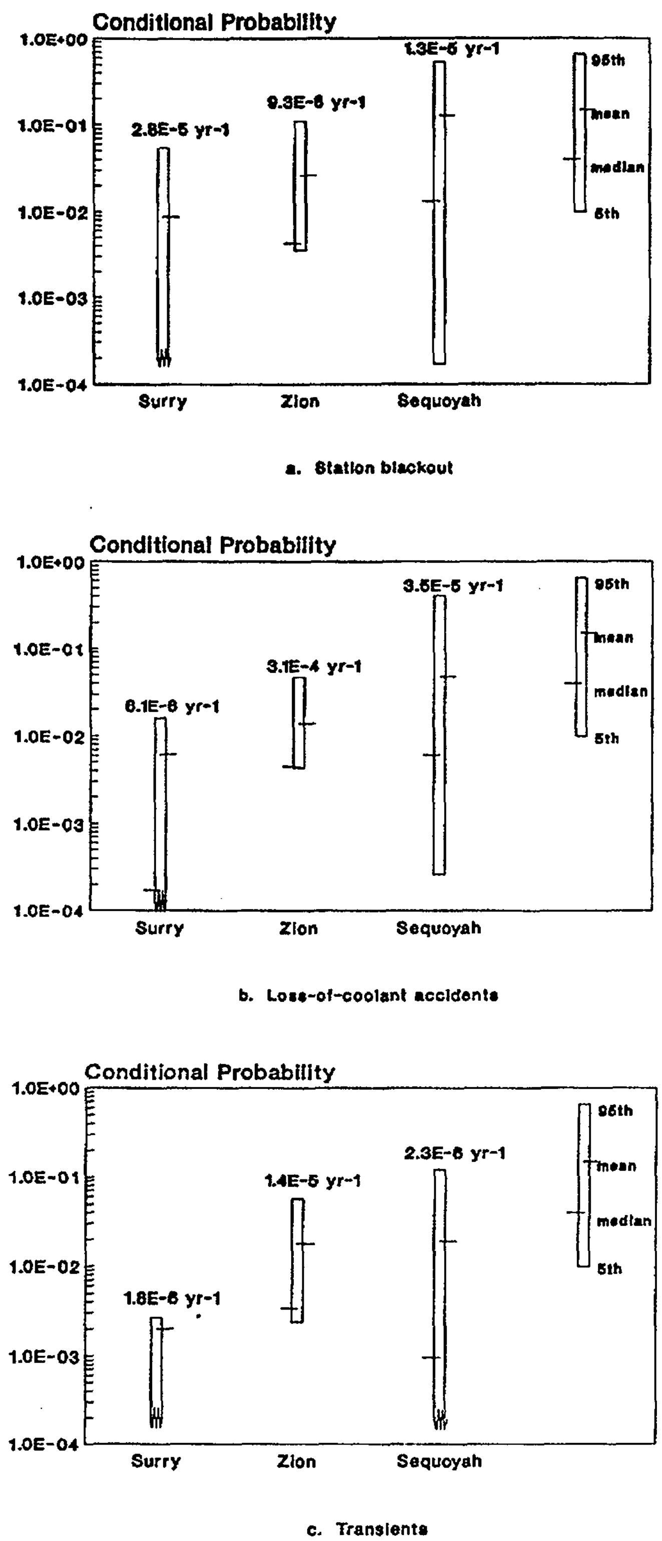


Figure 9.1 Conditional probability of early containment failure for key plant damage states (PWRs).

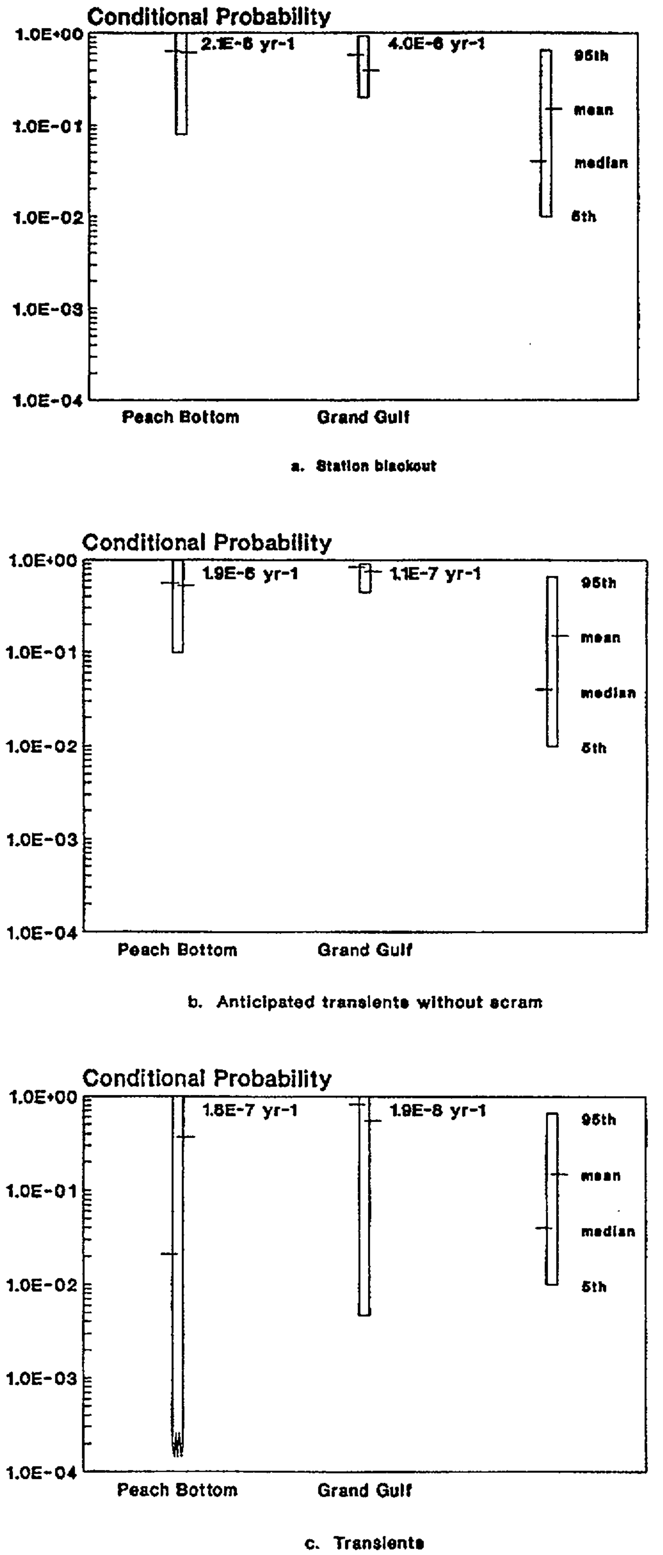


Figure 9.2 Conditional probability of early containment failure for key plant damage states (BWRs).

containment failure is 50 percent. About half of the early containment failures occur before vessel breach, and the other half occur at or shortly after vessel breach. For the long-term station blackouts, the mean conditional probability of early containment failure is 85 percent.

The probability of drywell failure at Grand Gulf is somewhat less than that of containment failure and occurs in approximately one-half the early containment failures. Drywell failures before vessel breach result from rapid hydrogen deflagrations in the wetwell. At the time of vessel breach, however, drywell failures are primarily from drywell pressurization loads at vessel breach (steam blowdown, direct containment heating, ex-vessel steam explosions, and hydrogen combustion). Failure of the drywell is more likely when vessel breach occurs with the vessel at high pressure.

Intentional venting of the containment was considered to prevent overpressurization failure of the containment for both Peach Bottom and Grand Gulf. The mean probability of sequences in which containment venting occurs and no containment failure occurs is approximately 10 percent for Peach Bottom station blackout sequences and 4 percent for Grand Gulf. The values are small, mostly because of the high probability of early failure mechanisms for which venting is ineffective. Furthermore, for the short-term station blackout plant damage state that dominates the core melt frequency at Grand Gulf, ac power is not available initially to permit venting.

Figure 9.3 illustrates the frequency of early failure or bypass of containment (the two types of failure with the potential for a large release of radionuclides) for internally initiated accidents in each of the five plants. (Peach Bottom scenarios in which the containment has been vented but subsequent early containment failure has occurred are categorized as early containment failures.) Note that, on a basis of absolute frequency, early containment failure or bypass for the BWR designs analyzed is similar to that of the PWRs because of the lower predicted frequency of core damage in the BWRs.

The relative probabilities of early containment failure, bypass, late failure, venting, and no containment failure are illustrated in Figure 9.4 for each of the plants. For the Surry plant, the likelihood of bypass, an interfacing-system LOCA, or steam generator tube rupture is somewhat greater than that of early failure from severe accident loads. In Figure 9.4, the capability of the Zion plant to avoid a large early release of radioactive material appears to be particularly good because of the small fraction of failures that result in either early failure or bypass.

It should be noted that the averaging of containment failure mode probabilities for different plant damage states can be misleading. To a large degree, the relative probability of bypass at Zion is substantially smaller than at Surry because the frequency of plant damage states, other than the interfacing-system LOCA, is higher. On an absolute frequency scale, as shown in Figure 9.3, the performances of the Surry and Zion containments in severe accidents are quite similar. In Sequoyah, the probability of early failure is somewhat larger than for the other PWRs analyzed and on a frequency-weighted mean basis is essentially the same as for bypass. The most likely outcome for these plants is that the containment will not fail.

Using early containment failure or containment bypass as a measure for comparison, the performance of the two BWR containments analyzed does not appear as good as the performance of the PWR containments. It is important to recognize that early containment failure or bypass is a prerequisite for a large release of radionuclides, but that mitigative features within the plant can substantially limit the release that occurs. This is particularly true for the pressure-suppression containment designs, where the suppression pool or ice condenser can retain radionuclides even if the containment has failed. (The BWR frequency of bypass is assessed to be quite small. Therefore, only early failures (with the potential for some radionuclide scrubbing by the suppression pool) are important.) The frequency of release of different quantities of radionuclides is discussed in Chapter 10.

9.2.2 External Events

Plant damage states that result from external events are quite similar to those that arise from internally initiated accidents except that their relative frequencies differ substantially. In addition, containment status may be affected by the initiating event. Figure 9.5 illustrates the relative probabilities of early containment failure, bypass, late failure, venting, and no failure (no vessel breach or vessel breach with no containment failure) for the two plants for which external-event analyses were performed. The results for internal initiators, fire, and seismic are compared in the figure. The importance of early containment failure relative to the importance of bypass is reversed in the Surry

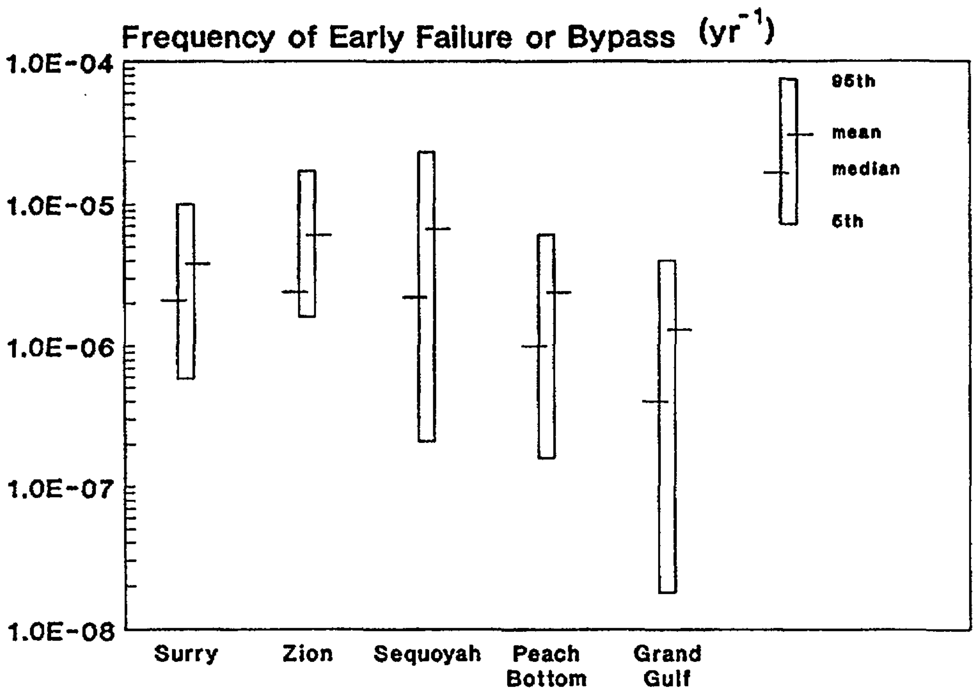


Figure 9.3 Frequency of early containment failure or bypass (all plants).

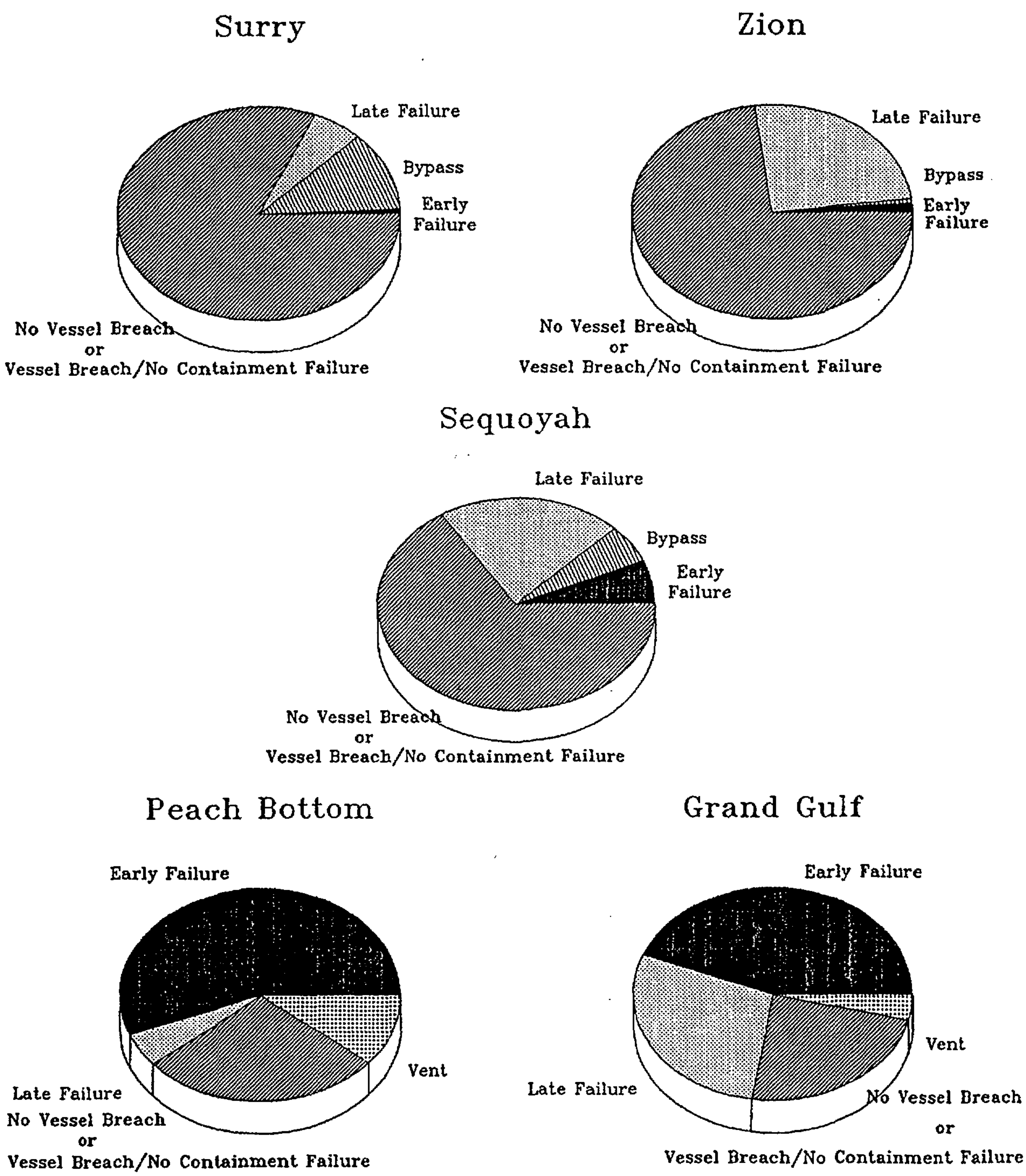


Figure 9.4 Relative probability of containment failure modes (internal events).

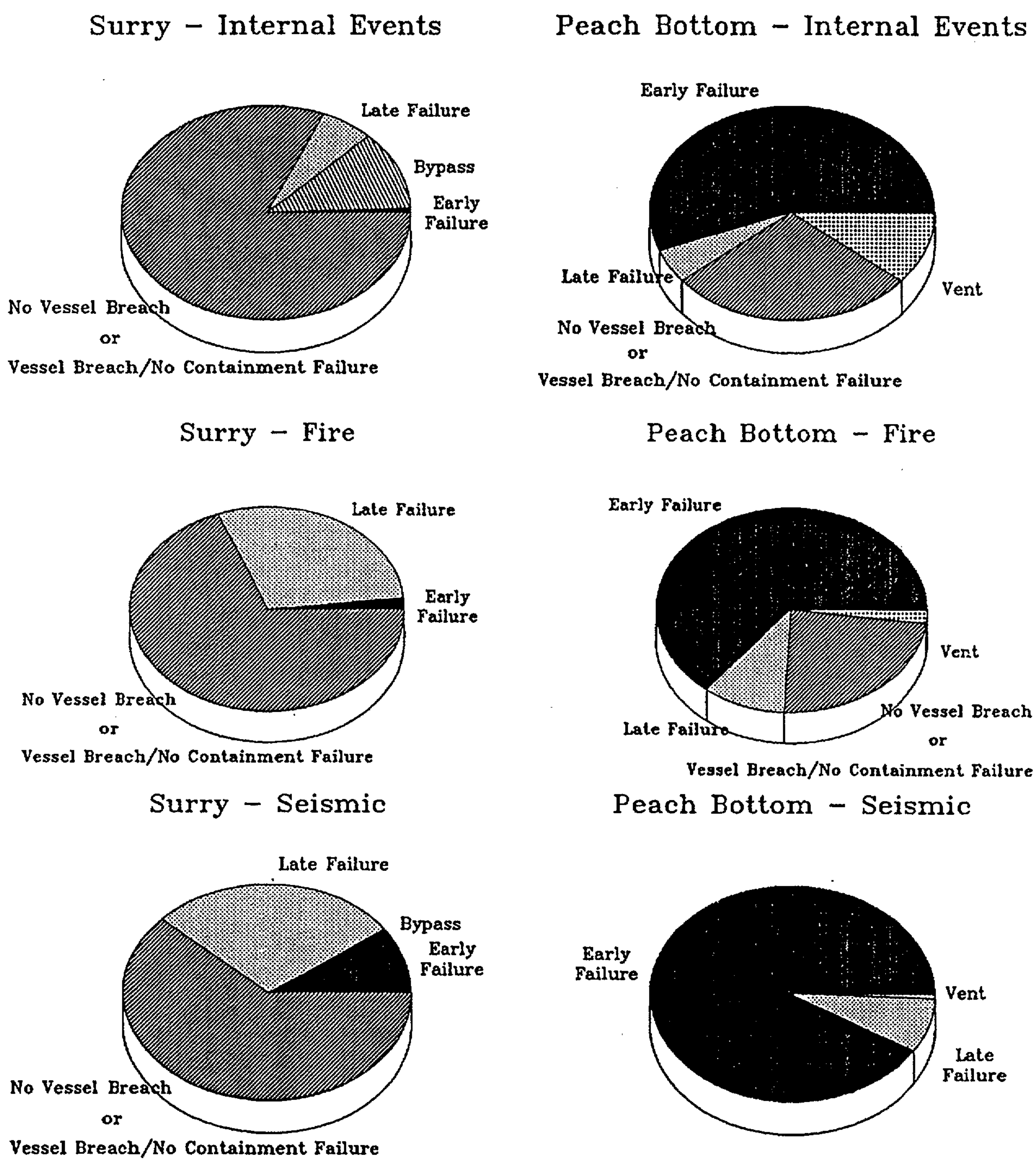


Figure 9.5 Relative probability of containment failure modes (internal and external events, Surry and Peach Bottom).

external-event analysis compared to the internal analysis. In the seismic analysis, the conditional probability of early failure is predicted to increase significantly (to approximately 8 percent). The increased failure likelihood is associated with substantial motion of the reactor coolant system components in an earthquake and resulting damage to the containment. In the fire analysis, there are no externally initiated bypass accidents, the likelihood of bypass induced by overheating of steam generator tubes is assessed to be negligible, and there is only a very slight increase in early containment failure.

Perspectives on the differences between external-event and internal-event containment performance for the Peach Bottom plant are similar to those described for Surry. In the fire analysis, some increase in early containment failure is predicted. In the fire sequences, there is a reduced potential for the recovery of ac power, which results in a reduced probability of injection recovery and an increased likelihood of drywell shell meltthrough.

In the BWR seismic analysis, the probability of containment survival in a severe accident is small; the increased likelihood of early containment failure is the result of substantial motion of the reactor vessel and subsequent damage to the containment during a major earthquake (well beyond the plant's design level) and a reduced recovery potential that increases the likelihood of containment failure as described for the fire sequences.

9.2.3 Additional Summary Results

Based on the results of the five-plant risk analyses summarized in Chapters 3 through 7, and discussed in detail in References 9.2 through 9.6, the following perspectives on containment performance in severe accidents can be drawn.

Zion and Surry Plants (Large, Dry and Subatmospheric Designs)

- Large, dry and subatmospheric containment designs appear to be quite robust in their ability to contain severe accident loads. This study shows a high likelihood of maintaining integrity throughout the early phases of severe accidents in which the potential for a large release of radionuclides is greatest. The uncertainties in describing the magnitude of severe accident loads at vessel breach for pressurized scenarios and the likelihood of depressurization prior to lower head failure are large, however.

- Containment bypass sequences (severe accidents initiated by steam generator tube ruptures, tube ruptures induced by hot circulating gases, or interfacing-system LOCAs) represent a substantial fraction of high-consequence accidents. The absolute frequency of these types of failure is small, however.

- The potential exists for the arrest of core degradation in a significant fraction of core damage scenarios within the reactor vessel as the result of recovery procedures (such as in the TMI-2 accident). The likelihood of containment failure is very small in these scenarios.

- A substantial likelihood exists that the containment will remain intact even if the accident progresses beyond the point of lower head failure.

- The likelihood of early containment failure in seismic events is higher than for internally initiated accidents.

Sequoyah Plant (Ice Condenser Design)

- The likelihood of early failure in a severe accident for the Sequoyah plant is higher than for the large, dry and subatmospheric designs but is less than for the BWRs analyzed. Early failure is primarily associated with loads imposed at the time of vessel breach (from a number of mechanisms, including direct containment heating and hydrogen combustion).

- Containment rupture from high overpressure loads at the time of vessel breach is likely to result in significant damage to the containment wall and effective bypass of the ice bed.

- Containment bypass is potentially an important contributor to the frequency of a large early release of radioactive material.

- The high likelihood of a deeply flooded reactor cavity plays an important role in mitigating severe accident consequences at Sequoyah. The deeply flooded cavity assists in reducing the loads at vessel breach, in preventing direct attack of molten fuel debris on the containment wall, and in avoiding molten core-concrete interactions.

- There is substantial potential for the arrest of core damage prior to vessel failure. There is, however, some likelihood of containment failure from hydrogen combustion events.

- A substantial likelihood exists for containment integrity to be preserved throughout a severe accident, even if the accident progresses beyond vessel breach.

Peach Bottom Plant (Mark I Design)

- The analyses indicate a substantial likelihood for early drywell failure in severe accident scenarios, primarily as the result of direct attack of the drywell shell by molten core debris.

- Considerable uncertainty exists regarding the likelihood of failure of the drywell as the result of direct attack by core debris. Although this is the dominant failure mechanism in the analyses, other loads on the drywell can lead to early drywell failure, such as rapid overpressurization of the drywell. A sensitivity study was performed in which the drywell meltthrough mechanism of failure was eliminated. The resulting reduction in mean early containment failure probability was from 0.56 to 0.2 (Ref. 9.3).

- The principal benefit of wetwell venting indicated by the study is in the reduction of the core damage frequency. Although venting is not effective in eliminating some early drywell failure mechanisms, venting could eliminate other sequences that would result in overpressure failure of the containment.

- There is substantial potential for the arrest of core damage prior to vessel failure. The likelihood of containment failure in arrested scenarios is small.

- The likelihood of early containment failure is higher for fire and seismic events than internally initiated accidents because of the decreased likelihood of ac and dc recovery resulting in higher drywell shell meltthrough probabilities.

Grand Gulf Plant (Mark III Design)

- Grand Gulf containment was predicted to fail at or before vessel breach in a substantial fraction of severe accident sequences. Hydrogen deflagration is the principal mechanism for early containment failure.

- Failure of the integrity of the drywell is predicted to accompany containment failure in approximately one-half the sequences involving early containment failure (resulting in bypass of the suppression pool for radionuclides released after vessel breach). Drywell failure is primarily the result of loads from rapid combustion events prior to reactor vessel breach and loads at vessel breach associated with overpressurization by direct containment heating, ex-vessel steam explosions, and hydrogen combustion in the wetwell region. Scrubbing of releases occurring before vessel breach can still occur in sequences in which the drywell fails and the suppression pool is eventually bypassed.

- There is a large potential for the arrest of core damage prior to vessel failure. If large quantities of hydrogen are produced in the process of recovery, hydrogen combustion could result in containment failure.

- Venting was not found to be particularly effective in preventing containment failure for accident scenarios involving core damage. Furthermore, venting was not as effective in reducing core damage frequency in Grand Gulf as it was in Peach Bottom.

9.3 Comparison with Reactor Safety Study

Prior to the time the Reactor Safety Study (RSS) (Ref. 9.7) analyses were undertaken, there had been no relevant experimentation or modeling of either the loads produced in a severe accident or the response of a containment to loads exceeding the design basis. As a result, the characterization of containment performance in the RSS is simplistic in comparison to the present study.

Containment Failure Modes

Figure 9.6 compares estimates for the present study with those of the RSS for the cumulative failure probability as a function of internal pressure for the Surry plant. The current study indicates that the Surry containment is substantially stronger than did the RSS characterization. In the RSS analyses, failure was assumed to involve rupture of the containment with substantial leakage to the environment. The current study subdivides failure into different degrees of leakage. Failure at the low-pressure end of the range would most

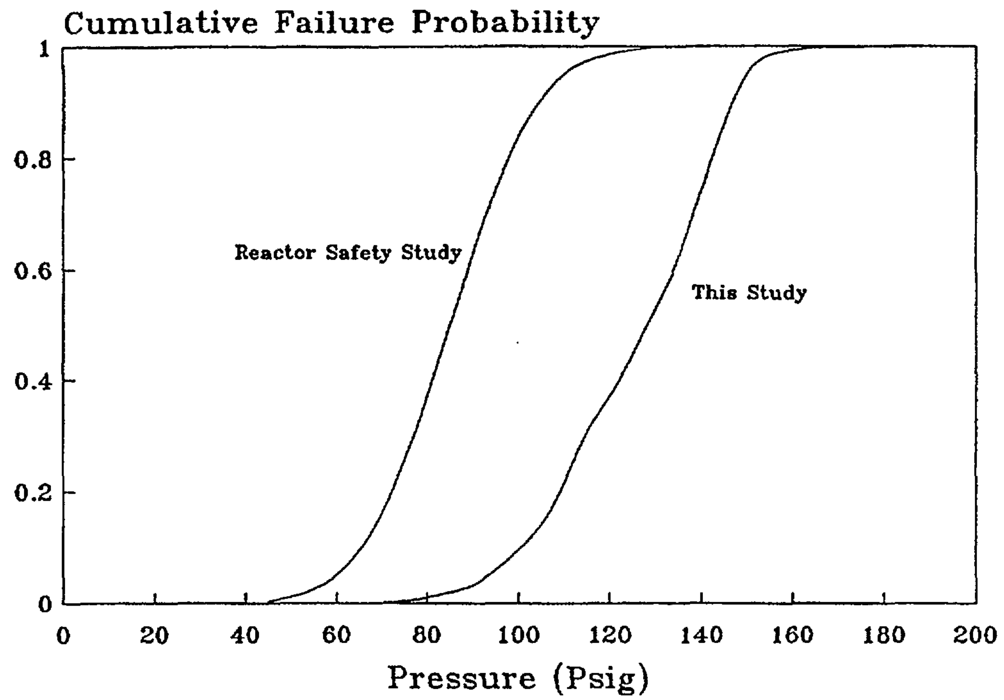


Figure 9.6 Comparison of containment failure pressure with Reactor Safety Study (Surry).

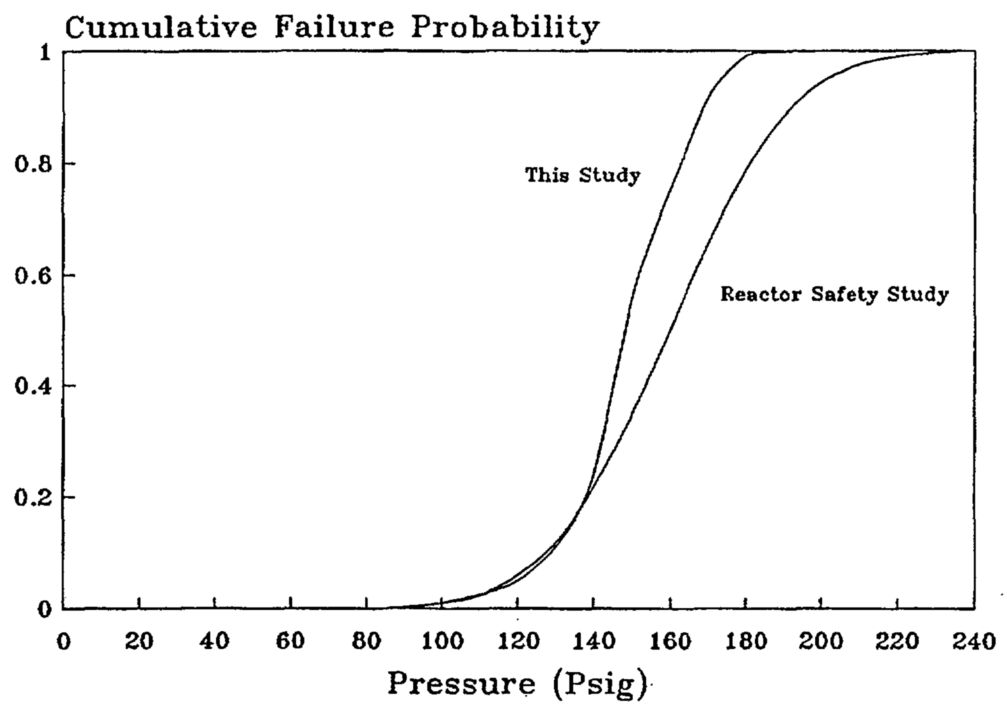


Figure 9.7 Comparison of containment failure pressure with Reactor Safety Study (Peach Bottom).

likely be the result of limited leakage, such as failure at a penetration rather than a substantial rupture of the containment wall. As the failure pressure increases, the likelihood of rupture versus leakage also increases. At pressures close to the ultimate strength of the shell, the potential for gross rupture of the containment exists but was found to be unlikely.

Figure 9.7 compares the current study with RSS estimates for cumulative failure probability as a function of pressure for the Peach Bottom plant (Mark I design). The curves are quite similar, with the current perspective being of a slightly less strong containment than the RSS representation. The curve presented from the current study is representative of a cool drywell (less than 500° F). Cumulative distributions were also developed in the current study for higher drywell temperatures. At 1200° F the median failure pressure was assessed to be 45 psig as opposed to 150 psig at low temperatures.

Failure location in the Mark I design can be as important as failure time. In the RSS, the most likely failure location was assessed to be at the upper portion of the toroidal suppression pool. It was assumed that, following containment failure, the pool would no longer be effective in scrubbing radioactive material. In the current analyses, other mechanisms of containment failure, such as direct attack of the drywell wall by molten core debris, were found to be more important than overpressure failure. The dominant location of overpressure failure is assessed to be the lifting of the drywell head by stretching the head bolts. Gases leaking past the head enter the refueling bay where limited radionuclide retention is expected rather than into the reactor building where more extensive retention could occur. (However, the leakage into the reactor building can also result in severe environments that can cause equipment failure.) Another structural failure from overpressure identified as likely in this study is at the bellows in the downcomer, which would result in leakage from the wetwell vapor space to the reactor building. Thus, although the estimated failure pressures identified in this study and in the RSS are quite similar, the modes and locations of failure are quite different.

Comparison of Surry Results

Risk in the RSS is dominated by a few key sequences for each plant. Containment performance in these sequences was a major aspect of their risk significance. The three key sequences for Surry were station blackout, an interfacing-system LOCA, and the failure of an instrumentation line penetrating the lower head. Figure 9.8 illustrates the range of early failure probability for station blackout in the current analyses and provides the point estimate from the RSS as a comparison. The RSS estimate of early failure likelihood is substantially higher than the present analysis even though the phenomenon of direct containment heating had not been identified at the time of the RSS. In addition to the lower assumed failure pressure of the containment, the RSS prediction of the rate of containment pressurization was unrealistically high.

The current perspective on the behavior of the interfacing-system LOCA in which the break occurs outside the containment resulting in bypass is essentially the same as in the RSS. The RSS did not identify the potential for rupture of a steam generator tube as a potentially important initiator of a severe accident.

The third important sequence in the RSS, involving an instrumentation line rupture, is no longer considered a core meltdown sequence. In the RSS analyses, if the containment spray injection pumps were to fail, damage was assumed to occur to the spray recirculation pumps resulting in loss of containment heat removal, containment failure, and consequent loss of emergency coolant makeup water to the vessel. More detailed analyses (Ref. 9.8) indicate, however, that condensed steam would provide sufficient water in the containment sump to prevent damage to the recirculation spray pumps, avoiding conditions resulting in containment failure and core meltdown.

Comparison of Peach Bottom Results

In the RSS analyses for the Peach Bottom plant, two sequences dominated the risk: a transient event with loss of long-term heat removal from the suppression pool and an anticipated transient without scram (ATWS). Loss of long-term heat removal is an extended accident in which heating of the suppression pool leads to overpressure failure of the containment and consequent loss of makeup water to the vessel. With the procedures now available to vent the Peach Bottom containment to outside the reactor building, the likelihood of loss of long-term heat removal leading to core meltdown has been reduced to the point where it is no longer a substantial contributor to core damage frequency or risk.

In the RSS analyses, early containment failure was considered a certainty in the ATWS sequence.

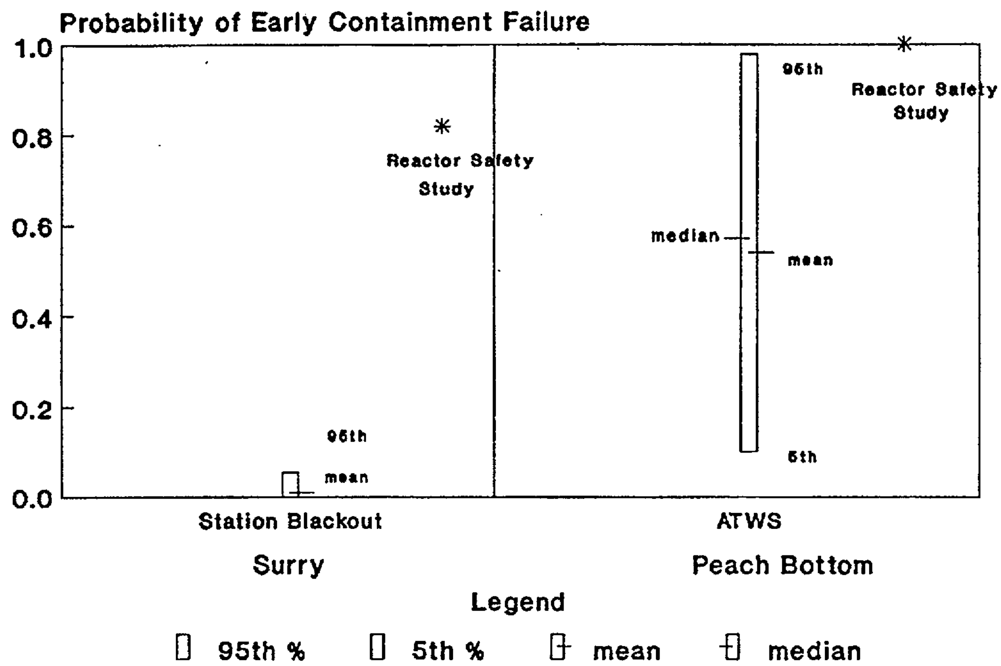


Figure 9.8 Comparison of containment performance results with Reactor Safety Study (Surry and Peach Bottom).

Figure 9.8 indicates that early failure is still considered quite likely for this sequence. The mechanisms resulting in failure and location of failure are different, however.

In summary, changes have occurred in predicting containment performance for the two plants analyzed in the RSS. There have been substantial improvements in the ability to model severe accident phenomena and system behavior in severe accidents. For Surry, the high likelihood of maintaining containment integrity indicated in the present study is the most significant difference in perspective between the two studies.

9.4 Perspectives

9.4.1 State of Analysis Methods

The analysis of severe accident loads and containment response involves substantial uncertainty because of the complexity of core meltdown processes. After a decade of research into severe accident phenomena subsequent to the TMI-2 accident, methods of analysis have been developed that are capable of addressing nearly every aspect of containment loads, including hydrogen deflagration and detonation and core-concrete interactions. In some instances, such as direct attack of the Mark I containment shell by molten material and direct containment heating, research is still being pursued (Ref. 9.9). Although the residual uncertainties are in some instances great, the methods are adequate to support meaningful Level 2 PRA analyses.

The accident progression event tree analysis techniques developed for this study involve a very detailed consideration of threats to containment integrity. A number of large computer analyses were required to support the quantification of event probabilities at each branch of the event tree. The analysis team for this study had the considerable advantage of access to researchers involved in the development and application of computer codes used in the analysis of core melt progression, core-concrete attack, containment behavior, radionuclide release and transport, and hydrogen combustion.

Computer analyses cannot, in general, be used directly and alone to calculate branching probabilities in the accident progression event tree. Since the greatest source of uncertainty is typically associated with the modeling of severe accident phenomena, the results of a single computer run (which uses a specific model) do not characterize the branching uncertainty. It is therefore necessary to use sensitivity studies, uncertainty studies, and expert judgment to characterize the likelihood of alternative events that affect the course of an accident. The effort undertaken in this study to elicit expert opinion was substantial. The expense of the overall accident progression analysis techniques (expert elicitation and computer analysis to support event tree quantification) employed in this study is currently a drawback to their widespread use. However, methods to apply the models, the distributions, and the computer codes to other plants at a reasonable cost are under study.

9.4.2 Important Mechanisms That Defeat Containment Function During Severe Accidents

The challenges to containment integrity that would occur in a severe accident depend on the nature of the accident sequence, as well as the design of the plant. The various containment designs analyzed in this study responded differently to different severe accident challenges.

Containment Bypass and Isolation Failure

When an accident occurs, a number of valves must close to isolate the containment from the environment. On the basis of absolute frequency, failure to isolate the containment was not found to be a likely source of containment failure for any of the plants analyzed. Primarily because of the low frequency of early containment failure by other means, containment isolation failure is a relatively important contributor to early failure at Zion. The subatmospheric containment and nitrogen-inerted Mark I containments are particularly reliable in this regard since it is highly likely that leakage would be identified during operation.

Containment bypass is an important contributor to large early releases of radionuclides for the Surry (subatmospheric), Sequoyah (ice condenser), and Zion (large, dry) containment designs. The principal contributors are accidents initiated by interfacing-system LOCAs and by steam generator tube ruptures. The predicted frequency of these events is quite small, however, and their dominance of risk is the result of the relatively lower frequency of other means to obtain large early releases.

Gas Combustion

Hydrogen and carbon monoxide are the two combustible gases potentially produced in large quantities in severe accidents. The principal source of hydrogen is the reduction of steam by chemical reaction of metals, particularly zirconium and iron. Carbon monoxide would only be produced in the later stages of an accident involving the attack of concrete by molten core debris. Because of the timing of carbon monoxide release, its production does not represent a threat of early failure to the containment but can contribute to delayed failure.

Rapid gas combustion was not found to be a substantial threat to containment for the Surry (subatmospheric), Zion (large, dry), or Peach Bottom (Mark I) containments. The Surry and Zion designs are sufficiently robust to survive deflagrations (rapid burning). At Surry and Zion, the likelihood of global detonations that could fail the containment (by impulsive loads) was assessed to be small. The contribution of hydrogen combustion to the pressure rise in the containment at the time of vessel failure in the event of high-pressure melt ejection of molten fuel was considered, but the likelihood of early failure of containment was also assessed to be small.

Hydrogen combustion is not a threat to the Mark I design because it normally operates with a nitrogen-inerted containment and thus has insufficient oxygen concentration to support combustion.

Hydrogen combustion was found to be a substantial threat to the integrity of the Sequoyah (ice condenser) and Grand Gulf (Mark III) designs. A very small contribution, about 1 percent, to early failure from hydrogen combustion prior to vessel breach is predicted for the station blackout sequences in Sequoyah. In arrested sequences, the containment failure probability is increased 5 percent because of ignition sources from the recovery of ac power. Approximately 12 percent mean early containment failure probability arises at the time of vessel breach, largely as the result of hydrogen combustion.

For the Grand Gulf plant, there is a substantial likelihood of containment failure before vessel breach in the short-term station blackout sequence because of the unavailability of igniters. At the time of vessel breach, hydrogen combustion loads can again occur, which can fail the containment (the percentages of containment failure before and at vessel breach are similar). Two additional reasons combine to make hydrogen events extremely important at Grand Gulf: (1) the BWR core contains an extremely large amount of zirconium that is available for hydrogen production, and (2) the suppression pool is subcooled in the short-term station blackout sequences resulting in condensation of the steam from the drywell or the vessel and leading to hydrogen-rich mixtures in the containment that are readily ignited.

Loads at Vessel Failure

The increase in containment pressure that could occur at vessel failure represents an important challenge to containment for each of the five designs (see Appendix C). In the Zion (large, dry) and Surry (subatmospheric) designs, loads at vessel breach from high-pressure melt ejections (rapid transfer of heat from dispersed core debris accompanied by chemical reactions with unoxidized metals in the debris) represent a mechanism that can result in containment loads high enough to fail containment. The predicted likelihood of failure for these scenarios in the Surry and Zion designs was found to be small, in part because most high-pressure sequences were predicted to depressurize by one or more means prior to vessel failure and because the overlap between the containment load distribution and the containment failure distribution was small.

Although loads at vessel breach have been studied more extensively for PWR containments, they were found to be an important contributor to early containment failure in the Sequoyah (ice condenser) and Peach Bottom (Mark I) plants and to early drywell failure in Grand Gulf (Mark III). In the Sequoyah and Grand Gulf analyses, hydrogen combustion is also a principal contributor to early containment failure from the loads at vessel breach. At Grand Gulf, pedestal failure, due to dynamic loads from ex-vessel steam explosions or subcompartment pressure differential, can also result in drywell failure at this stage of the accident.

Direct attack of the drywell shell is the dominant failure mechanism at vessel breach in the Peach Bottom plant. Overpressurization can also lead to leakage failure in the drywell by lifting the drywell head or to failure in the wetwell.

Direct Attack by Molten Debris

Direct attack of the drywell wall by molten debris in the Peach Bottom (Mark I) design has been the subject of considerable controversy among severe accident experts (see Section C.7 of Appendix C). Essentially half the experts whose opinions were elicited believed that containment failure would occur, and half believed that it would not occur. The numerical aggregation of these diverse views led to a mean likelihood of failure in the present analysis of approximately 30 percent when the pedestal region is wet and 80 percent when the pedestal region is dry (Ref. 9.3).

Molten debris attack was also predicted to be a threat to the Sequoyah (ice condenser containment) in high-pressure sequences in which molten debris could be dispersed into the seal table room, which is outside the crane wall and adjacent to the steel wall of the containment. The likelihood of failure was considerably less than for Peach Bottom, however.

Steam Explosions

When molten core material contacts water, the potential exists for rapid transfer of heat, production of steam, and transfer of thermal energy to mechanical work. Considerable research has been undertaken to determine the conditions under which steam explosions can occur and their energetics. At pressures near atmospheric, it is generally concluded that steam explosions would be likely if molten core material drops into a pool of water. However, the energetics and coherence of the molten fuel-coolant interaction are very uncertain. At high steam pressure, steam explosions are found to be more difficult to initiate.

Steam explosions represent a variety of potential challenges to the containment. If the interaction were to occur in the reactor vessel at the time when molten core material slumps into the lower plenum, the possibility exists of tearing loose the upper head of the vessel, which could impact and fail the containment (this has been called the "alpha mode" of containment failure since the issuance of the RSS). The analyses in this study indicate that the potential for this type of event to result in early containment failure is less than 1 percent for each of the plants. For Surry and Zion, steam explosions represent a significant fraction of the early failure probability, but only because the overall likelihood of early failure is small.

When molten core material drops into water outside the vessel, the potential failure mechanisms are different. In the Grand Gulf plant, a shock wave could propagate through water and impact the concrete structure that provides support to the reactor vessel. Substantial motion of the vessel could then lead to the tearout of penetrations through the drywell wall. Because of the shallow water pool at Peach Bottom, dynamic loads from steam explosions do not represent a similar mechanism for failures.

In addition to potentially producing missiles and shock waves, steam explosions can also rapidly generate large quantities of steam and hydrogen. The steam produced from molten fuel-coolant interactions ex-vessel following vessel breach is an important contributor to the static drywell overpressure failure in the Grand Gulf and Peach Bottom plants.

Gradual Overpressurization

Figure 9.9 illustrates the assessed pressure capability for the five plants analyzed. The ability of a containment to withstand the production of gases in a severe accident depends on the volume of the containment as well as its failure pressure. One of the principal sources of pressurization in a severe accident is steam production. In each plant design, however, engineered safety features are present to condense steam in the form of suppression pools, ice beds, sprays, air coolers, or in some designs, combinations of these systems. Steam pressurization is only a major contributor to the total pressure if, in the scenario being analyzed, the heat removal system has become inoperative; e.g., the spray system has failed, the suppression pool has become saturated, or the ice has melted.

Large quantities of hydrogen are predicted to be released in severe accidents, both in-vessel during the melting phase and ex-vessel during core-concrete attack, debris bed quenching, or high-pressure melt ejection. If the hydrogen does not burn, it will contribute to the containment pressure. Carbon monoxide and carbon dioxide produced during core-concrete attack also contribute to containment pressurization.

Because of its relatively small volume, the Peach Bottom (Mark I) design is more vulnerable to overpressurization failure by noncondensible gas generation. If the accident progression proceeds to vessel penetration and the molten core attacks the concrete, it is unlikely that containment integrity can be maintained in the long term unless other factors mitigate gas production.

Overheating

The effect of high temperature in the drywell on containment failure probability and mode was considered in the Peach Bottom analysis. Although very high gas temperatures can be achieved as the result of hydrogen combustion in the other plant designs, the structure temperatures are not predicted to reach temperatures at which the strength of the structure would be substantially reduced or sealant materials would be degraded. The Peach Bottom drywell, however, is relatively small. Substantial convective and radiative heat transfer from hot core debris could result in very high drywell wall temperatures. Failure could result from the combination of high pressure in the drywell and decreased strength of the steel containment wall. Overheating the drywell is only a contributor to scenarios in which the drywell spray is inoperative. If the sprays are operational, the drywell temperature will be much lower than for the dry case.

Drywell heating in the Peach Bottom plant represents a delayed containment failure mechanism. Since the likelihood of early failure by other mechanisms is high, drywell overtemperature failure is not a substantial contributor to risk.

Loss of Vessel Support

In the earlier section on steam explosions, a mechanism was described for drywell failure in the BWR designs in which structural failure of the reactor pedestal results in vessel motion (tipping or falling) and the tearout of piping penetrations through the drywell wall. Quasistatic pressurization of the pedestal region can result in the same phenomenon. Erosion of the pedestal by molten core attack of the concrete can also lead to the same effect. In this event, however, considerable time is required for the erosion to occur, and the failure would be late and the importance to risk is diminished. The likelihood of this mechanism of failure is generally small for the BWRs analyzed, in part because other mechanisms are likely to result in failure earlier in the accident.

Basemat Meltthrough

For each of the five plants analyzed, some potential exists for core debris to be quenched as a particulate debris bed and cooled in the reactor cavity or pedestal region if a continuous source of water is available. A significant likelihood exists, however, that, even if a replenishable water supply is available, molten core debris will attack the concrete basemat. If the core-concrete interaction does occur, the presence or absence of an overlaying water pool is not expected to have much effect on the downward progression of the melt front.

The depth of the basemat of the Peach Bottom containment, directly under the vessel, is so great that it is unlikely that the basemat would be penetrated before the occurrence of other failure modes. For the other plants, basemat penetration is possible, but the projected consequences are minor in comparison with those of aboveground failures.

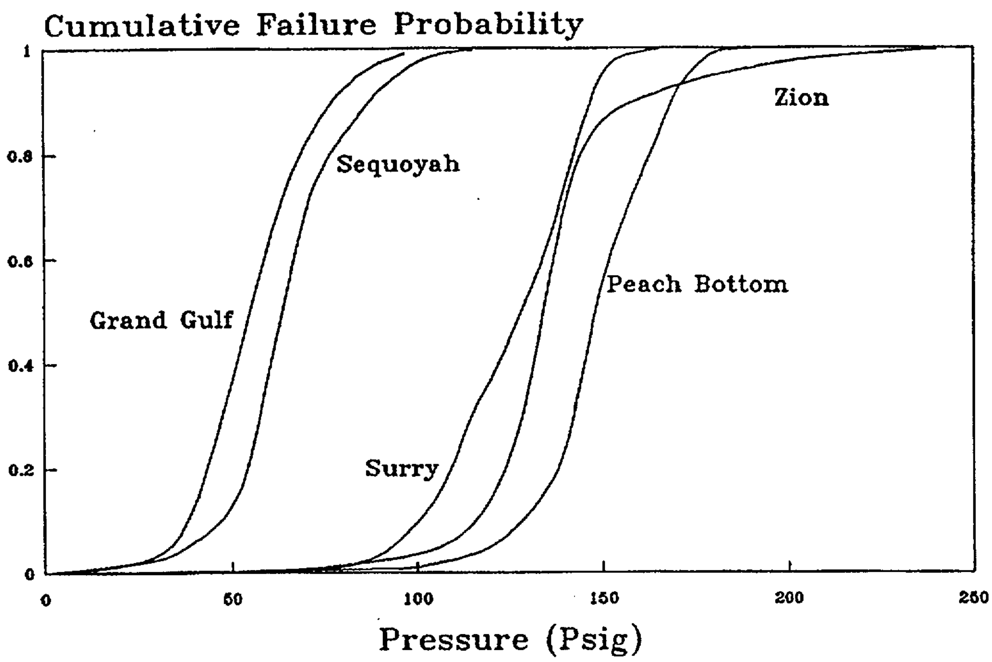


Figure 9.9 Cumulative containment failure probability distribution for static pressurization (all plants).

9.4.3 Major Sources of Uncertainty

The perspectives on the major sources of uncertainty described in this section come from four sources:

- Regression analysis-based sensitivity analyses for the mean values for risk. Simple linear regression models were used to represent the complex risk models, and adequate results were obtained. Better results would require more complex regression models. Insights for this section are deduced from the risk regression studies (regression analyses for conditional containment failure probabilities required for more detailed accident progression insights were not performed). Results of these studies are presented in References 9.2 through 9.6.

- Partial rank correlation analyses for the risk complementary cumulative distribution functions. Results of these studies are presented in References 9.2 through 9.6.

- Sensitivity studies in which separate analyses were performed with certain parameter values set to a specific value. Sensitivity studies were performed on the Mark I drywell shell meltthrough issue and the PWR RCS depressurization scenarios. These studies were only performed for the accident progression analysis; no source term or consequence insights are available.

- The subjective judgment of the analysts performing the plant-specific studies.

Importance of Accident Progression Analysis Variables to Rank Regression Analyses for Annual Risk

The majority of the variables important to the rank regression analyses performed for Surry were the initiating event frequencies of the containment bypass events and the source term variables. The only accident progression event tree variable that was demonstrated to be important to the uncertainty in risk for internal events was the probability of vessel and containment breach by an in-vessel steam explosion; this variable was moderately important to the uncertainty in total early fatality risk (Ref. 9.2).

The regression analyses performed for Sequoyah showed the containment failure pressure and loads at vessel breach to be accident progression variables somewhat important to the uncertainty in both total early fatality risk and total latent cancer fatality risk (Ref. 9.4).

The probability of drywell meltthrough was the only accident progression variable that was at all important to uncertainty in the early fatality risk or the latent cancer fatality risk for the internal regression analysis for Peach Bottom (Ref. 9.3).

The amount of hydrogen produced in-vessel, the probability of drywell failure following pedestal failure, the pressure load in the drywell at vessel breach, and the amount of hydrogen produced and released at and shortly after vessel breach were accident progression variables that were found to be important to the uncertainty in early fatality risk by the Grand Gulf regression analyses. The probability of drywell failure following pedestal failure and the pressure load in the drywell at vessel breach were found to be important to the uncertainty in latent cancer fatality risk (Ref. 9.5).

The majority of variables important to the rank regression analyses performed for Zion were related to failure or recovery of the component cooling water (CCW) system and the source term variables. The only accident progression event tree variable that was demonstrated to be important to the uncertainty in risk was the probability of vessel and containment breach by an in-vessel steam explosion. This result was also obtained from the Surry regression analyses. The probability of a steam explosion failure was found to be important to the uncertainty in both early and latent health risk measures at Zion. The importance of seal LOCA failure to risk uncertainty was expected, given the large contribution of these events to the core damage frequency. Upgrades to the Zion service water and CCW systems have the potential to reduce the importance of these events as discussed in Appendix C (Section C.15) (Ref. 9.6).

Direct Attack of Drywell Shell in Peach Bottom

The divergence of opinion of the panel of containment performance experts, in itself, is an indicator of the uncertainty in the associated phenomena. A sensitivity study was performed to determine the impact on containment performance of eliminating this failure mechanism. The mean early failure probability (averaged over all sequences) was reduced from 56 percent to 20 percent (Ref. 9.3).

High-Pressure Melt Ejection and Vessel Depressurization

For the Surry and Zion plants, early containment failure resulting from loads at vessel breach is assessed to have low probability, on the order of 1 percent. Sensitivity studies were performed to determine the dependence of this result on expert judgments made about various reactor coolant system depressurization mechanisms prior to vessel breach. A sensitivity study was performed for Surry (Ref. 9.2), which removed depressurization by temperature-induced breaks. This study indicated that removal of only temperature-induced failures for depressurization does not result in a significant increase in the likelihood of early containment failure (from roughly 1 percent to roughly 2 percent). This probability study, therefore, implies that other depressurization mechanisms, such as the failure of reactor coolant pump seals and stuck-open relief valves, are also important. However, a sensitivity study was also performed for Zion (Ref. 9.6) in which all depressurization mechanisms were removed. The result of this study was a relatively small increase in the likelihood of early containment failure. For accidents initiated by LOCAs (which dominate the estimated core damage frequency), this change resulted in essentially no change in the conditional probability of early containment failure. The probability of early failure increased by a factor of 5 for accidents initiated by transients (from roughly 0.01 to 0.06) and by a factor of 2 for accidents initiated by station blackout (from roughly 0.03 to 0.06). The reason for the relatively small impact of removing all depressurization mechanisms on the probability of early containment failure is that the Zion containment is expected to withstand high-pressure melt ejection loads (even at the upper end of the uncertainty range) with very high confidence (refer to Section C.5 of Appendix C for a more detailed discussion). Also, at these small probability levels, in-vessel steam explosions contribute to the likelihood of early containment failure. If the reactor coolant system pressure remains high, the likelihood of triggering a steam explosion is decreased. Thus, the slightly higher probability of early containment failure resulting from high-pressure melt ejection loads will be offset to some degree by the lower probability of containment failure from in-vessel steam explosions.

Uncertainties associated with high-pressure melt ejection also affect the early containment failure likelihood for the other three plants. The significance of this issue is greatest for the Sequoyah and Grand Gulf plants, which have lower overpressure capacity and which are vulnerable to the hydrogen produced in the oxidation of dispersed core debris by steam.

Containment Failure by Steam Explosions

The production of missiles by in-vessel steam explosions only appears as a significant contributor to early failure or bypass in the Zion analyses. The contribution of alpha-mode containment failure is the result of the very low probability of other modes of early failure or bypass and is itself a low value. Quasistatic and shock loading from an ex-vessel steam explosion is indicated to be a potentially important contributor to drywell failure for Grand Gulf. Ex-vessel steam explosions also contribute to quasistatic overpressurization failure in the Peach Bottom plant.

Core Melt Progression

Many of the uncertain phenomena that have the potential to lead to early containment failure (e.g., high-pressure melt ejection, drywell shell attack, steam explosions, and hydrogen generation) are sensitive to the details of core melt progression, particularly the later stages of progression in which molten core material enters the lower head of the vessel. The mass of material potentially available for dispersal at head failure, the composition of this material, the timing of head failure, and the mode of head failure have a substantial indirect impact on the likelihood of early containment failure through their effects on early failure mechanisms.

Containment Bypass

The containment bypass sequences have been discussed throughout this report as special scenarios (in which the containment function has failed) and will be briefly mentioned here. The containment bypass initiating event frequencies, transmission factors, and decontamination factors were demonstrated to be the variables most important to the uncertainty in all risk measures in both the Surry and Sequoyah rank regression analyses.

REFERENCES FOR CHAPTER 9

9.1 M. Silberberg et al., "Reassessment of the Technical Bases for Estimating Source Terms," United States Nuclear Regulatory Commission (USNRC) Report NUREG-0956, July 1986.

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

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

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

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

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

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

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

9.8 R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios—PWR, Subatmospheric Containment Design," Battelle Columbus Division, NUREG/CR-4624, Vol. 3, BMI-2139, July 1986.

9.9 USNRC, "Revised Severe Accident Research Program Plan: Fiscal Year 1990-1992," NUREG-1365, August 1989.

10. PERSPECTIVES ON SEVERE ACCIDENT SOURCE TERMS

10.1 Introduction

Shortly after the accident at Three Mile Island, the NRC initiated a program to review the adequacy of the methods available for predicting the magnitude of source terms for severe reactor accidents. After considerable effort and extensive peer review, the NRC published a report entitled "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956 (Ref. 10.1). The report recommended that a set of integrated computer codes, the Source Term Code Package (STCP) (Ref. 10.2), be used as the state-of-the-art methodology for source term analysis provided that uncertainties were considered. The STCP methodology provided a starting point for source term estimates in this study. In addition, the characterization of source term uncertainties was supported by calculations with other system codes such as MELCOR (Ref. 10.3) and MAAP (Ref. 10.4), detailed special purpose codes such as CONTAIN (Ref. 10.5), as well as small codes written for this project to examine specific source term phenomena. Because it was impractical to perform an STCP calculation for each source term required and the STCP does not contain models for all potentially important phenomena, simplified methods of analysis were developed with adjustable parameters that could be benchmarked against the more detailed codes. Probability distributions, which had been developed from the elicitations of the source term panel of experts, were provided for many of the parameters in the simplified computer codes. A large number of source term estimates were generated for each plant by sampling from the probability distributions in the simplified codes.

Source terms are typically characterized by the fractions of the core inventory of radionuclides that are released to the environment, as well as the time and duration of the release, the size distribution of the aerosols released, the elevation of the release, the warning time for evacuation, and the energy released with the radioactive material. All these parameters are required for input to the MACCS (Ref. 10.6) consequence code. Although the illustrations and comparisons of source terms in this chapter emphasize the magnitude of estimated release, it is important to recognize that the other characteristics of the source term noted above, such as the timing of release, can also have an important effect on the ultimate consequences.

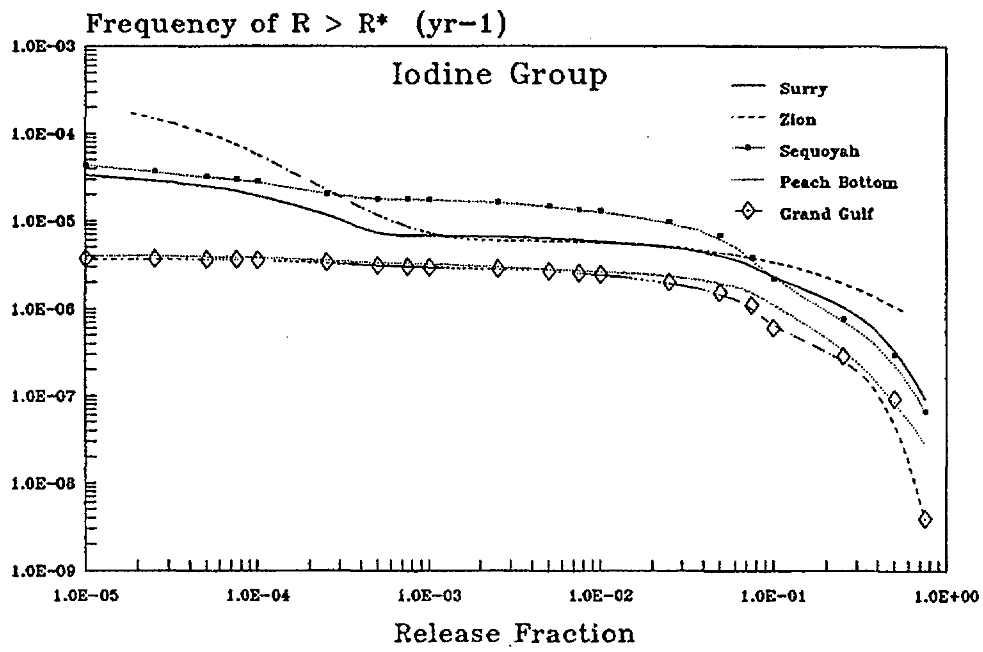
It is widely believed that the approximate treatment of source term phenomena in the Reactor Safety Study (RSS) (Ref. 10.7) analyses led to a substantial overestimation of severe accident consequences and risk. The current risk analyses provide a basis for understanding the differences that exist in source terms calculated using the new methods relative to those calculated using the RSS methods and the impact of these differences on estimated risk.

10.2 Summary of Results

Some examples of source terms (fractions of the core inventory of groups of radionuclides released to the environment) were provided for accident progression bins for each of the analyzed plants in Chapters 3 through 7. As expected, the magnitude of the source term varies between different accident progression bins depending on whether or not containment fails, when it fails, and the effectiveness of engineered safety features (e.g., BWR suppression pool) in mitigating the release. However, within an accident progression bin, which represents a specific set of accident progression events, the uncertainty in predicting severe accident phenomena is great.

In Figure 10.1, the predicted frequency of radioactive releases is compared among the five plants. In this figure, the mean distribution is presented, allowing differences in plant behavior to be illustrated. The y-coordinate in the figure represents the predicted frequency with which a given magnitude of release (the x-coordinate) would be exceeded. The location of the exceedance curve is determined by the frequencies of accident sequences in addition to the spectrum of possible source terms for those sequences.

It is not obvious in examining a radionuclide source term what the potential health impact would be to the public from a specified magnitude of release. Based on the compilation of a number of consequence analyses, however, one method (Ref. 10.8) has been developed that provides an approximate relationship for the minimum fractions of radionuclides released that result in early fatalities or early injuries. For the release of iodine, for example, the thresholds for early fatalities and early injuries occur at release fractions of the core inventory of approximately 0.1 and 0.01, respectively. Figure 10.1 does not indicate major differences in the exceedance curves for the five plant analyses. For the iodine group,



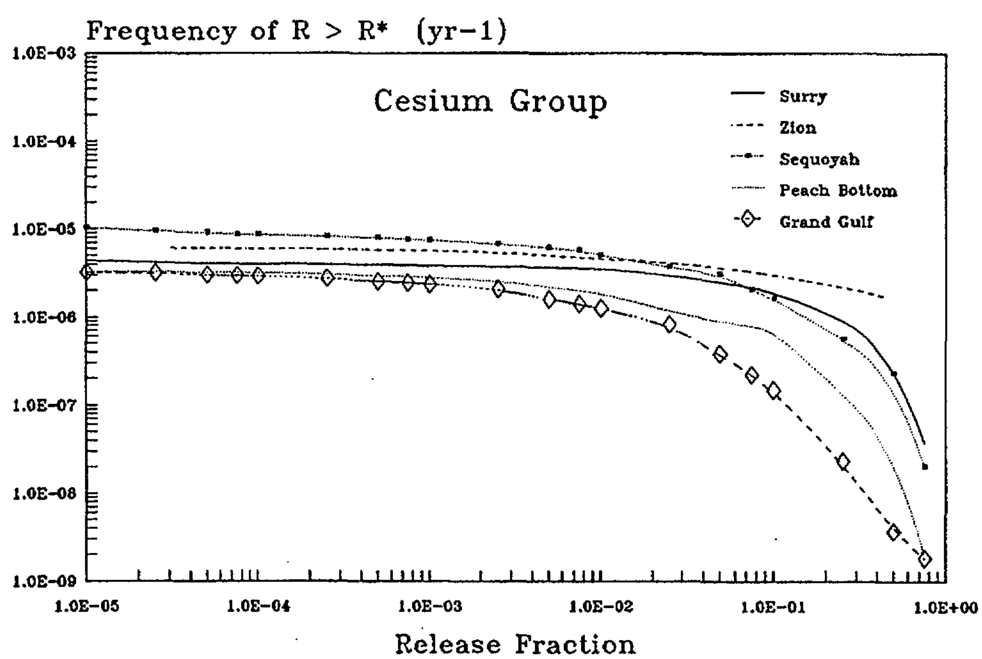
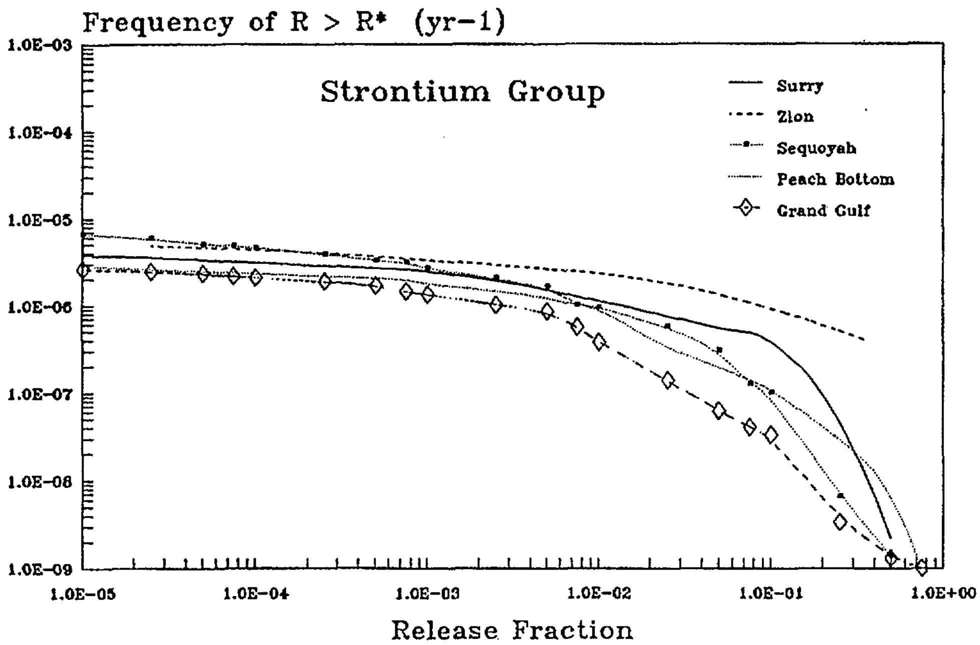


Figure 10.1 Frequency of release for key radionuclide groups.



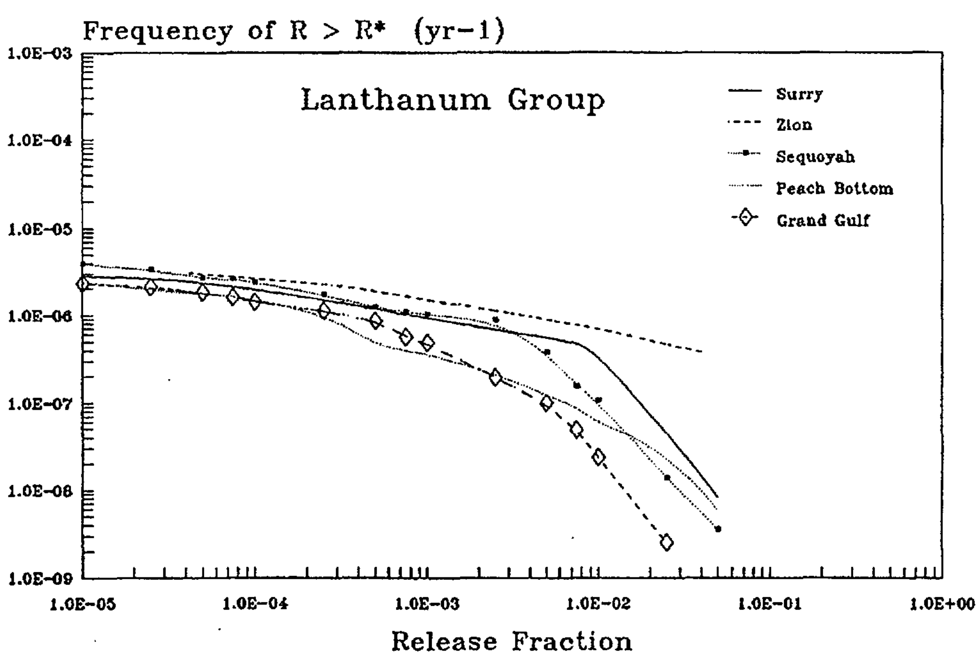


Figure 10.1 (Continued)

the frequency of exceeding a release fraction of 0.1 ranges from 1E–6 to 5E–6 per reactor year for the five plants. Similarly, for a release fraction of 0.01, the exceedance curves range from 2E–6 to 1E–5 per reactor year. The most outstanding feature of these curves is their relative flatness over a wide range of release fractions. For the iodine, cesium, and strontium groups, the curves decrease only slightly over the range of release fractions from 1E–5 to 1E–1 and then fall rapidly from 0.1 to 1. For the lanthanum group, the rapid decrease in the curve occurs at a release fraction that is approximately a decade lower. As a result of the flatness of the exceedance curves, the frequency of accidents with source terms that are marginally capable of resulting in early fatalities is only slightly less than the frequency of accidents covering a very broad spectrum of health consequences up to the occurrence of fatalities. However, the frequency of source terms with the potential for multiple early fatalities falls rapidly with increased release.

Based on the results of the source term analyses for the five plants, a number of general perspectives on severe accident source terms can be drawn:

- The uncertainty in radionuclide source terms is large and represents a significant contribution to the uncertainty in the absolute value of risk. The relative significance of source term uncertainties depends on the plant damage state.

- Source terms for bypass sequences, such as accidents initiated by steam generator tube rupture (SGTR), can be quite large, potentially comparable to the largest Reactor Safety Study source terms.

- Early containment failure by itself is not a reliable indicator of the severity of severe accident source terms. Substantial retention of radionuclides is predicted to occur in many of the early containment failure scenarios in the BWR pressure-suppression designs, particularly for the in-vessel period of release during which radionuclides are transported to the suppression pool. Containment spray system and ice condenser decontamination can also substantially mitigate accident source terms.

- Flooding of reactor cavities or pedestals can eliminate the core-concrete release of radionuclides, if a coolable debris bed is formed, or can significantly attenuate the release from the molten core-concrete interaction by scrubbing in the overlaying pool of water.

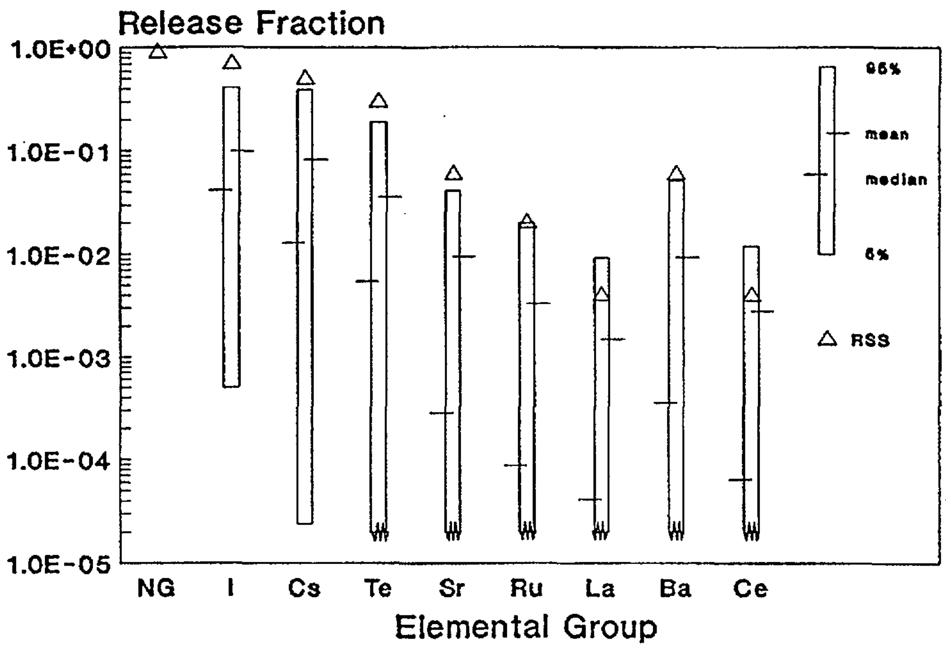
10.3 Comparison with Reactor Safety Study

In the Reactor Safety Study (RSS) (Ref. 10.7), source terms were developed for nine release categories ("PWR1" to "PWR9") for the Surry plant and five release categories for the Peach Bottom plant ("BWR1" to "BWR5"). The RSS release categories are directly analogous to the accident progression bins in the current study in that they are characterized by aspects of accident progression and containment performance that affect the source term. For example, the PWR1 release category represented early containment failure resulting from an in-vessel steam explosion with containment sprays inoperative. A point estimate for release fractions (fraction of the core inventory of an elemental group released to the environment) for seven elemental groups (in the current study, the number of elemental groups has been expanded to nine) was then used to represent this type of release.

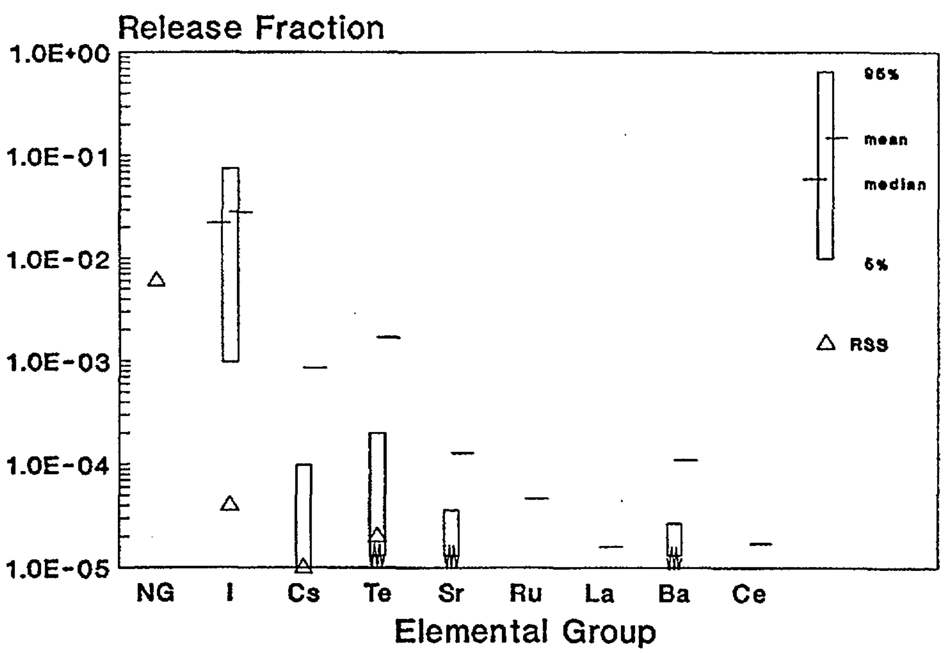
In the current study, source terms were developed for a much larger number of accident progression bins. A distribution of release fractions was also obtained for each of the elemental groups corresponding to the individual sample members of the uncertainty analysis.

In order to simplify the presentation in this report, the results of similar accident progression bins have been aggregated to a level that is comparable to that used in the RSS. Figure 10.2 provides a comparison of an important large release category (PWR2) from the RSS for Surry with a comparable aggregation of accident progression bins (early containment failure, high reactor coolant system pressure) from the current study.* Also shown in Figure 10.2 is a low release category from the RSS (PWR7) with a comparable aggregation of accident progression bins from the current study (late failure). No range is shown for the noble gas release for this study because no permanent retention mechanisms were assumed to affect these gases. The point estimates of the release of radionuclides in the RSS early containment failure bin are more representative of the upper bounds

*Because of the aggregation of accident progression bins, some of the range of the source terms represents variation in accident progression as well as modeling uncertainty. The distribution was developed from all of the sample members within the aggregated bins without consideration of the relative frequencies of these bins.



a. Comparison with Bin PWR2



b. Comparison with Bin PWR7

Figure 10.2 Comparison of source terms with Reactor Safety Study (Surry).

of the range in the current study than the mean or the median. For the late failure comparison, the results for this study are somewhat higher than those obtained for the RSS. The difference is related to the types of failures in the late failure bin. In the RSS, the PWR7 source terms were based on a release associated with meltthrough of the basemat in scenarios with containment sprays operable. The late failure bin in the current study also includes overpressure failure cases with a direct release from the plant to the atmosphere. Of particular significance is the nontrivial release of iodine that is associated with late release mechanisms, which were not considered in the RSS.

Figure 10.3 compares release fractions for an aggregation of early drywell failure accident progression bins from the current study with the BWR2 and BWR3 release categories. In the current study, a range of reactor building decontamination factors is considered depending on the mode of drywell failure and variations in thermal-hydraulic conditions in the building. The BWR2 release fractions are at the upper bounds of the ranges in the current study, and the BWR3 releases are near the mean values.

The second example compares results for an isolation failure in the wetwell region from the RSS, release category BWR4, with the venting accident progression bin from the current study. The RSS results are very similar to the mean release terms for the venting bin, with the exception of the iodine group, which is higher because of the late release mechanisms (reevolution from the suppression pool and the reactor vessel) considered in the current study.

Overall, the comparison indicates that the source terms in the RSS were in some instances higher and in other instances lower than those in the current study. For the early containment failure accident progression bins that have the greatest impact on risk, however, the RSS source terms appear to be larger than the mean values of the current study and are typically at the upper bound of the uncertainty range.*

10.4 Perspectives

10.4.1 State of Methods

The use of parametric source term methods, in which the parameters are fit to reproduce the results of more mechanistic codes, was found to be a practical necessity in performing a PRA that includes a complete treatment of phenomenological uncertainties. Research is in progress in some of the key areas of uncertainty that influence source term results. In a number of cases, the STCP did not have models that represent potentially important phenomena, such as revaporization from reactor coolant system surfaces and reevolution of iodine from water pools. Later codes, such as MELCOR (Ref. 10.3), which have at least rudimentary models for these processes, should provide greater assurance of consistency in the analysis. These advanced codes may not, however, remove the need for parametric codes capable of performing a large number of analyses inexpensively.

Improvement in Understanding

Since the Reactor Safety Study (RSS), substantial improvements have been made in understanding severe accident processes and source term phenomena. A major shortcoming of the RSS was the limited treatment of the uncertainties in severe accident source terms. In the intervening years, particularly subsequent to the Three Mile Island accident, major experimental and code development efforts have broadly explored severe accident behavior. In this study, care has been taken to display the assessed uncertainties associated with the analysis of accident source terms. Many of the severe accident issues that are now recognized as the greatest sources of uncertainty were completely unknown to the RSS analysts 15 years ago.

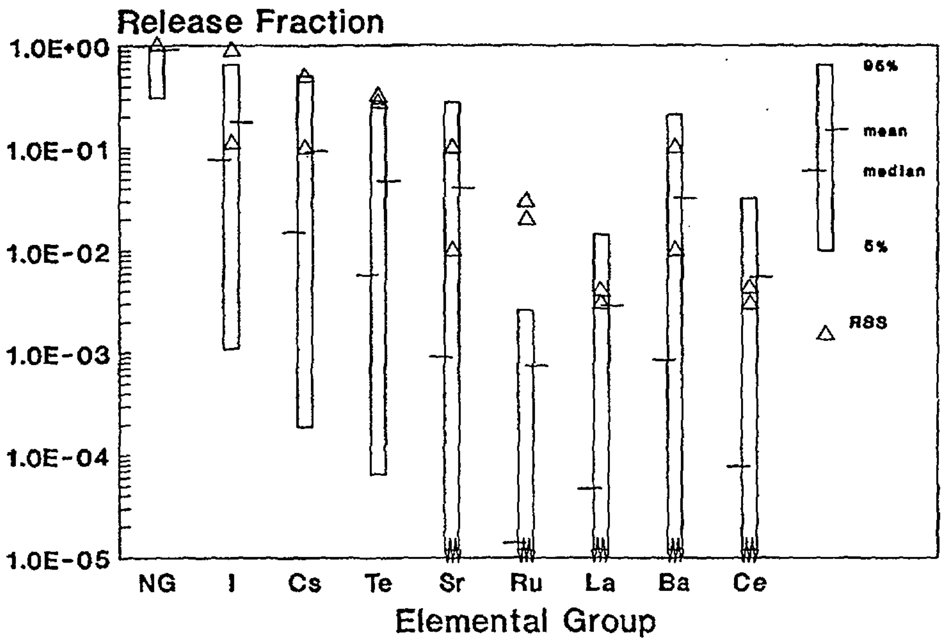
10.4.2 Important Design Features

In Chapter 9, performance of the containments of the five plants was described with respect to the timing of the onset of containment failure and the magnitude of leakage to the environment. In particular, the likelihood of early containment failure was used as a measure of containment performance. Environmental source terms are affected by more than just the mode and timing of containment failure, however. The following paragraphs describe the effect of different safety systems and plant features on the magnitude of source terms.

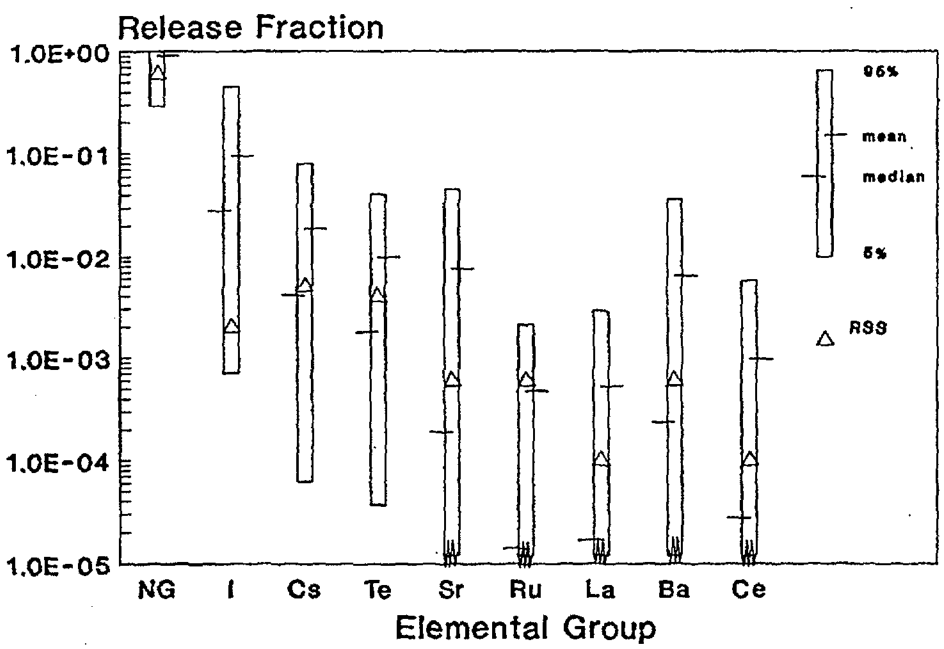
Suppression Pools

Suppression pools can be very effective in the removal of radionuclides in the form of aerosols or

*Additional comparisons with the Reactor Safety Study may be found in Reference 10.9.



a. Comparison with Bins BWR2 and BWR3



b. Comparison with Bin BWR4

Figure 10.3 Comparison of source terms with Reactor Safety Study (Peach Bottom).

soluble vapors. Some of the most important radionuclides, such as isotopes of iodine, cesium, and tellurium, are primarily released from fuel during the in-vessel release period. Because risk-dominant accident sequences in BWRs typically involve transient sequences rather than pipe breaks, the in-vessel release is directed to the suppression pool rather than being released to the drywell. As a result, the in-vessel release is subjected to scrubbing in the suppression pool, even if containment failure has already occurred. For the Peach Bottom plant, decontamination factors used in this study for scrubbing the in-vessel component ranged from approximately 1.2 to 4000, with a median value of 80. Since the early release of volatile radioactive material is typically the major contributor to early health effects, the effect of the suppression pool in depressing this component of the release is one of the reasons the likelihood of early fatalities is so low for the BWR designs analyzed.

Depending on the timing and location of containment failure, the suppression pool may also be effective in scrubbing the release occurring during core-concrete attack or reevolved from the reactor coolant system after vessel failure. In the Peach Bottom analyses, containment failure was found to be likely to occur in the drywell early in the accident. Thus, in many scenarios the suppression pool was not effective in mitigating the delayed release of radioactive material. Similarly, in the Grand Gulf design, drywell failure accompanied containment failure in approximately one-half the early containment failure scenarios analyzed. As a result, the suppression pool was found to be ineffective in mitigating ex-vessel releases in a substantial fraction of the scenarios for both BWR plants analyzed.

Although the decontamination factors for suppression pools are typically large, radioactive iodine captured in the pool will not necessarily remain there. Reevolution of iodine was found to be important in accident scenarios in which the containment has failed and the suppression pool is boiling.

Containment Sprays

If given adequate time, containment sprays can also be effective in reducing airborne concentrations of radioactive aerosols and vapors. In the Surry (subatmospheric) and Zion (large, dry) designs, approximately 20 percent of core meltdown sequences were predicted to eventually result in delayed failure or basemat meltthrough. The effect of sprays, in those scenarios in which they are operational for an extended time, is to reduce the concentration of radioactive aerosols airborne in the containment to negligible levels in comparison with non-aerosol radionuclides (e.g., noble gases) with respect to potential radiological effects. For shorter periods of operation, sprays would be less effective but can still have a substantial mitigative effect on the release.

The Sequoyah (ice condenser) design has containment sprays for the purpose of condensing steam that might bypass the ice bed, as well as for use after the ice has melted. The effects of the sprays and ice beds in removing radioactive material are not completely independent since they both tend to remove larger aerosols preferentially.

In the Peach Bottom plant, drywell sprays can be operated in sequences in which ac power is available. Scrubbing of radioactive material released from fuel during core-concrete attack can be accomplished by a water layer developed on the drywell floor, as well as by the spray droplets. Containment spray operation in Grand Gulf is most important for scenarios in which both the containment and drywell have failed. In the short-term station blackout plant damage state, power recovery that is too late to arrest core damage can still be important for the operation of containment sprays and the mitigation of the extended period of ex-vessel release from fuel.

Ice Condenser

The ice beds in an ice condenser containment remove radioactive material from the air by processes that are very similar to those in the BWR pressure-suppression pools. The decontamination factor is very sensitive to the volume fraction of steam in the flowing gas, which in turn depends on whether the air-return fans are operational. For a typical case with the air-return fans on, the magnitude of the decontamination factors was assessed to be in the range from 1.2 to 20, with a median value of 3. Thus, the effectiveness of the ice bed in mitigating the release of radioactive material is likely to be substantially less than for a BWR suppression pool.

Drywell-Wetwell Configuration

The Mark III design has the apparent advantage, relative to the Mark I and Mark II designs, of the wetwell boundary completely enclosing the drywell, in effect providing a double barrier to radioactive material release. As long as the drywell remains intact, any release of radioactive material from the fuel would be subject to decontamination by the suppression pool. For this reason, failure of the Mark III containment is not as important to severe accident risk as the potential for containment failure in combination with drywell failure. Figures 6.5 and 6.6 illustrate the difference in the environmental source terms for the early containment failure bins with and without drywell failure. With the drywell intact, the environmental source term is reduced to a level at which early fatalities would not be expected to occur, even for early failure of the outer containment. The potential advantages of the drywell-wetwell configuration were found to be limited in this study by the significant probability of drywell failure in an accident.

Cavity Flooding

The configuration of PWR reactor cavity or BWR pedestal regions affects the likelihood of water accumulation and water depth below the reactor vessel. The Surry reactor cavity is not connected by a flowpath to the containment floor. If the spray system is not operating, the cavity will be dry at vessel failure. In the Peach Bottom (Mark I) design, there is a maximum water depth of approximately 2 feet on the pedestal and drywell floor before water would overflow into the downcomer. The other three designs investigated have substantially greater potential for water accumulation in the pedestal or cavity region. In the Sequoyah design, the water depth could be as much as 40 feet.

If a coolable debris bed is formed in the cavity or pedestal and makeup water is continuously supplied, core-concrete release of radioactive material would be avoided. Even if molten core-concrete interaction occurs, a continuous overlaying pool of water can substantially reduce the release of radioactive material to the containment.

Reactor Building/Auxiliary Building Retention

Radionuclide retention was evaluated for the Peach Bottom reactor building, but an evaluation was not made for the portion of the reactor building that surrounds the Grand Gulf containment, which was assessed to have little potential for retention. The range of decontamination factors for aerosols for the Peach Bottom reactor building subsequent to drywell rupture was 1.1 to 80 with a median value of 2.6. The location of drywell failure affects the potential for reactor building decontamination. Leakage past the drywell head to the refueling building was assumed to result in very little decontamination. Failure of the drywell by meltthrough resulted in a release that was subjected to a decontamination factor of 1.3 to 90 with a median value of 4.

In the interfacing LOCA sequences in the PWRs, some retention of radionuclides was assumed in the auxiliary building (in addition to water pool decontamination for submerged releases). In the Sequoyah analyses, retention was enhanced by the actuation of the fire spray system.

Containment Venting

In the Peach Bottom (Mark I) and Grand Gulf (Mark III) designs, procedures have been implemented to intentionally vent the containment to avoid overpressure failure. By venting from the wetwell air space (in Peach Bottom) and from the containment (in Grand Gulf), assurance is provided that, subsequent to core damage, the release of radionuclides through the vent line will have been subjected to decontamination by the suppression pool.

As discussed in Chapter 8, containment venting to the outside can substantially improve the likelihood of recovery from a loss of decay heat removal plant damage state and, as a result, reduce the frequency of severe accidents. The results of this study indicate, however, only limited benefits in consequence mitigation for the existing procedures and hardware for venting. Uncertainties in the decontamination factor for the suppression pool and for the ex-vessel release and in the reevolution of iodine from the suppression pool are quite broad. As a result, the consequences of a vented release are not necessarily minor. Furthermore, the effectiveness of venting in the two plant designs is limited by the high likelihood of mechanisms leading to early containment failure, which would result in bypass of the vent.

10.4.3 Important Phenomenological Uncertainties

In order to identify the principal sources of uncertainties in the estimated risk, regression analyses were performed for each of the plant types in this study. In general, in these regression analyses, the dependent variable is risk expressed in terms of consequences per year (e.g., early fatalities per year or latent cancer fatalities per year). For the Surry plant (Ref. 10.10), however, additional regression analyses were performed in which the dependent variable is the quantity of release per year for each of the radionuclide groups. These analyses are particularly useful in investigating how uncertainties in source term variables affect the releases of different radionuclides. Also determined were partial correlation coefficients that represent the importance of uncertain variables as a function of the magnitude of the environmental release.

Relative Importance of Source Term Variables

The results of these regression analyses indicate that uncertainties in source term variables are important contributors to the uncertainties in risk but are often not the largest contributors. The relative contribution of uncertainties in source term variables depends on the characteristics of each plant damage state as illustrated in the Peach Bottom and Sequoyah regression analyses (Refs. 10.11 and 10.12). In general, the five plant analyses indicate that the importance of the aggregate of variables that affect release frequencies (accident frequency variables and accident progression variables) is similar to or greater than the importance of the aggregate of variables that affect source term magnitude.

Source term variables tend to have less importance to the uncertainty in latent cancer fatality (or population dose) risk than to the risk of early fatalities. Because of the threshold nature of early fatalities, these risk results are particularly sensitive to pessimistic values of source term variables.

Importance of Source Term Variables to Uncertainty in Environmental Release

Based on analyses performed for the Surry plant (Ref. 10.10), the importance of source term variables is seen to be different for different groups of radionuclides. The uncertainty in the release of noble gases is dominated by the uncertainty in accident frequency variables. The relative uncertainties in release fractions for the noble gases and in retention mechanisms (only volumetric holdup is assumed) are small.

The character of the risk-dominant accident sequences at Surry plays an important role in determining the importance of the source term variables for the other radionuclide groups. The steam generator tube rupture (SGTR) accident and the interfacing-system LOCA sequences (the risk-dominant sequences) involve bypass routes in which radionuclides released from the core transport to the environment without being subjected to containment deposition processes. As a result, steam generator retention and the release of radionuclides from the fuel during in-vessel melt progression are the largest contributors to uncertainty for the volatile radionuclides, iodine and cesium, and for the semivolatile radionuclides, tellurium, barium, strontium, and ruthenium. For the involatile radionuclides, lanthanum and cerium, the release of radionuclides during core-concrete interactions is also an important contributor.

The Surry analyses also indicate that the uncertainties in source term variables tend to have relatively more importance for large releases. For small releases of radionuclides, the uncertainties are dominated by the uncertainties associated with the accident frequencies.

Plant-Specific Importance of Source Term Variables to Uncertainty in Risk

Consistent with the discussion in the previous section, the largest contributors to uncertainty in early fatality risk for the Surry plant (Ref. 10.10) are the frequency of the interfacing-system LOCA sequence and two source term variables, retention in the steam generator (in an SGTR accident) and release from the fuel during in-vessel melt progression. For latent cancer fatality risk, the frequency of SGTR accidents becomes of higher importance and the frequency of interfacing-system LOCAs of reduced importance. Steam generator retention and in-vessel release of radionuclides are of comparable importance to the accident frequency variables.

The Zion results (Ref. 10.13) are similar to those for Surry but reflect a reduced significance of the interfacing-system LOCA sequence and an increased importance of steam explosions as a mode of early containment failure (this results from a much lower frequency of interfacing-system LOCA in Zion). Release of radionuclides from fuel in-vessel, steam generator retention (in an SGTR accident), and containment retention of material released prior to vessel breach (as applied in a steam explosion scenario) are the most important source term contributors to the uncertainty in early fatality risk. For latent cancer fatality risk, containment failure from a steam explosion is of reduced significance and, as a result, containment retention is not an important contributor to risk uncertainty.

For early fatality risk at Sequoyah (Ref. 10.12), the frequency of the interfacing-system LOCA is the most important contributor to uncertainty. Containment failure by overpressurization is a more likely early failure mechanism for Sequoyah than for the large, high-pressure containments at Zion and Surry. As a result, accident progression mechanisms such as pressure rise at vessel breach and containment failure pressure are also important contributors to risk uncertainty for the

Sequoyah design. The most significant source term variables are in-vessel retention fraction, containment retention fraction for the in-vessel release, and steam generator deposition (in an SGTR accident). For latent cancer fatality risk, the frequency of the SGTR accident is the most important contributor to uncertainty; none of the source term variables is significant.

Regression results were obtained for internal initiators, fire events, and seismic events for the Peach Bottom plant (Ref. 10.11). For early fatality risk from internal initiators, release from fuel in-vessel, release during core-concrete interactions, and fractional release from containment of the core-concrete source terms are the most important contributors to uncertainty. The containment building decontamination factor, late release of iodine, reactor coolant system retention, and revaporization also contribute at a level similar to the contribution from the frequencies of the accident sequences. For fire initiators, the contributions from the various source term variables are similar but slightly reduced consistent with greater uncertainty in the initiator frequency.

For latent cancer fatality risk at Peach Bottom, the important source term variables are the same as for the early fatality risk but are relatively less important than the contribution from uncertainties in the accident frequencies.

In the Grand Gulf analyses (Ref. 10.14), the source term variables were indicated to be less important than the accident sequence and accident progression variables. The most significant source term variable was indicated to be the release fraction from containment following vessel failure. The decontamination factor for the suppression pool, spray decontamination factor, in-vessel release of radioactive material, and in-vessel retention of radioactive material were also identified as moderate contributors to the uncertainty in risk.

REFERENCES FOR CHAPTER 10

10.1 M. Silberberg et al., "Reassessment of the Technical Bases for Estimating Source Terms," U.S. Nuclear Regulatory Commission (USNRC) Report NUREG-0956, July 1986.

10.2 J. A. Gieseke et al., "Source Term Code Package, A User's Guide (Mod. 1)," Battelle Columbus Division, NUREG/CR-4587, BMI-2138, July 1986.

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

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

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

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

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

10.8 G. D. Kaiser, "The Implications of Reduced Source Terms for Ex-Plant Consequence Modeling," Executive Conference on the Ramifications of the Source Term (Charleston, SC), March 12, 1985.

10.9 L. LeSage et al., "Report of the Special Committee on NUREG-1150, The NRC's Study of Severe Accident Risks," American Nuclear Society, June 1990.

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

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

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

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

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

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

11. PERSPECTIVES ON OFFSITE CONSEQUENCES

11.1 Introduction

Frequency distributions, in the form of complementary cumulative distribution functions (CCDFs), of four selected offsite consequence measures of the atmospheric releases of radionuclides in reactor accidents (with all source terms contributing) have been presented in Chapters 3 through 7 for the five plants* covered in this study. For each consequence measure, the 5th percentile, 50th percentile (median), 95th percentile, and the mean CCDFs were shown. This chapter provides some perspectives on the offsite consequence results for these plants.

Section 11.2 provides a discussion on the basis of the CCDFs. Section 11.3 discusses, summarizes, and compares the consequence results for the five plants displayed in the mean and the median CCDFs. Section 11.4 compares the results from the mean and median CCDFs with those of the Reactor Safety Study (Ref. 11.1). Sections 11.5 and 11.6, respectively, provide discussions on potential sources of uncertainty in consequence analysis and on sensitivities of the mean CCDFs to the assumptions on the offsite protective measures to mitigate the consequences.

Some of the perspectives provided in this chapter relate to the effectiveness of various methods of offsite emergency response. For these five plants, it appears that evacuation is the most effective emergency response for the risk-dominant accident sequences. However, as discussed below, the calculated effectiveness of a response is sensitive to assumptions on the timing of warnings to people offsite before radioactive release, the estimated delay before evacuation and the effective speed of evacuating populations, and the energy of the release. In this chapter, the results of sensitivity studies on some of these factors are discussed. The reader should not infer that these results signal a modification to NRC's emergency response guidance. Rather, they provide a glimpse of the type of technical assessment that would be required in NRC's reevaluation of emergency response.

11.2 Discussion of Consequence CCDFs

As discussed in the earlier chapters, a large number of source terms, each with its own frequency, were initially developed for each of the five plants. They spanned a wide spectrum of plant damage states, phenomenological scenarios, and source term uncertainties for each plant that led to radionuclide releases to the atmosphere. However, for the purpose of the manageability of the offsite consequence analysis, such large numbers of source terms for each plant were reduced to a much smaller number (about 30 to 60) of representative source term groups.

Each source term group was treated as a single source term in the offsite consequence analysis code, MACCS (Ref. 11.2). The MACCS analyses incorporated the mitigating effects of the offsite protective actions. The magnitudes of the selected consequence measures and their meteorology-based probabilities were calculated by MACCS for each source term group and were used to generate the meteorology-based CCDFs. These conditional CCDFs of the consequence measures for all individual source term groups served as the basic data set for further analysis. When the conditional CCDFs of a consequence measure were weighted by the frequencies of the source term groups, the 5th percentile, 50th percentile (median), 95th percentile, and the mean values of the frequencies at various magnitude levels of the consequence measure were obtained and displayed as CCDFs in Chapters 3 through 7.

Thus, in this procedure, both the frequencies of the source term groups and the probabilities of the site meteorology (which in combination with the source term groups lead to the various consequence magnitude levels) have been used in generating the percentile and mean CCDFs. (The construction of these CCDFs is discussed in Section A.9 of Appendix A.)

11.3 Discussion, Summary, and Interplant Comparison of Offsite Consequence Results

The various percentile and the mean CCDFs of the consequence measures shown in Chapters 3 through 7 display the uncertainties in the offsite consequences stemming from the in-plant uncertainties up to the source terms and their frequencies and the ex-plant uncertainties due to the variability of the site meteorology. The 5th and 95th percentile CCDFs provide a reasonable display of the bounds of the offsite consequences frequency distributions for the five plants.

*See Figures 3.9, 3.10; 4.9, 4.10; 5.8; 6.8; and 7.7, respectively, for Surry, Peach Bottom, Sequoyah, Grand Gulf, and Zion.

Tables 11.1 and 11.2 present the information contained in the mean and the median CCDFs in tabular form. Entries in these tables are the exceedance frequency levels of 10^{-5}, 10^{-6}, 10^{-7}, 10^{-8}, and 10^{-9} per reactor year and the magnitudes of the consequences that will be exceeded at these frequencies for the five plants.

As stated in Chapters 3 through 7, the CCDFs of the consequence measures presented in those chapters (and, therefore, the results shown in Tables 11.1 and 11.2) incorporate the benefits of evacuation of 99.5 percent of the population within the 10-mile plume exposure pathway emergency planning zone (EPZ), early relocation of the remaining population from the heavily contaminated areas both within and outside the 10-mile EPZ, and other protective measures. Details of the assumptions on the protective measures are presented in Table 11.3.

The results shown in Tables 11.1 and 11.2 for the five plants are discussed below.

Early Fatality Magnitudes

The early fatality magnitudes (persons) at various exceedance frequencies for a plant are driven by the core damage frequency and the radionuclide release parameters of the source term groups for the plant; the site meteorology and the population distribution in the close-in site region; and the effectiveness of the emergency response. These factors are different for the five plants. Therefore, different values of early fatality magnitudes are shown for equal levels of exceedance frequencies.

Some of the plant/site features contributing to the differences between the early fatality CCDFs of the five plants are discussed below:

- Core damage frequencies for the internal initiators for Peach Bottom and Grand Gulf are lower than those for the other three plants. Therefore, the early fatality CCDFs for Peach Bottom and Grand Gulf are associated with relatively low exceedance frequencies.

- Quantities of radionuclides associated with the early phase of the release* in the source term groups for Peach Bottom and Grand Gulf are typically smaller than those for the other three plants because of suppression pool scrubbing. This lowered the early fatality magnitudes for these two plants.

- Several source term groups for Surry and Sequoyah with large quantities of radionuclides associated with the early release phase are also associated with large thermal energy in this phase. This resulted in vertical rise of the plume in several meteorological scenarios, reducing the potential for large early fatality magnitudes.

- The time of warning before the start of the radionuclide release strongly influences the effectiveness of the emergency response, particularly the evacuation. The source term groups for Peach Bottom and Grand Gulf with potential for early fatalities, unless mitigated by emergency response, are also associated with warning times that are well in advance of the release compared to those for the other three plants because the most important accident sequences for the BWRs develop more slowly than those for the PWRs of this study. In contrast, warning times are close to the start of the release (about 40 minutes before the release) for the source term groups containing the fast-developing interfacing-system LOCA accident sequences for Surry and Sequoyah, which also have large quantities of radionuclides in the release.

- The Zion site has the highest population density within the 10-mile EPZ among the five plants (although about half of the area in this zone for Zion is water). It is followed by Surry, Sequoyah, Peach Bottom, and Grand Gulf.

- For Zion, Surry, and Sequoyah, relatively long evacuation delay times after the warnings and slow effective evacuation speeds were calculated. For Peach Bottom and Grand Gulf, relatively short evacuation delay times and fast effective evacuation speeds were calculated. Values of these parameters were based on the utility-sponsored plant-specific studies and the NRC requirements for emergency planning. The utility-sponsored evacuation time estimate studies, however, were not evaluated in terms of how well they realistically represent the sites.

In the MACCS calculations, early warnings before the radionuclide release and short evacuation

*Virtually all source term groups developed for this study have two release phases—an early release phase and a later release phase. Early fatalities are essentially due to the early release. This is because the wind direction may change before the later release, so that the later release would not always add to the radiation dose of the same people who were affected by the early release, and evacuation or relocation would likely be completed before the later release would occur.

Table 11.1 Summaries of mean and median CCDFs of offsite consequences—fatalities.

Exceedance Frequency (ry^{-1})	Early Fatalities (persons)[a]							Latent Cancer Fatalities (persons)[a]						
	1*	2*	3*	4*	5*	6*	7*	1*	2*	3*	4*	5*	6*	7*
10^{-5}														
Int.[b]	0	0	0	0	0	-	-	0	0	6(1)[c]	0	0	-	-
	0	0	0	0	0	0	0	0	0	2(1)	0	0	7(2)	1(3)
Fire	0	0	-	-	-	-	-	0	6(2)	-	-	-	-	-
	0	0	-	-	-	-	-	0	0	-	-	-	-	-
10^{-6}														
Int.	0	0	0	0	0	-	-	1(3)	1(3)	4(3)	3(2)	8(3)	-	-
	0	0	0	0	0	0	0	4(2)	2(2)	1(3)	0	2(3)	5(3)	5(3)
Fire	0	0	-	-	-	-	-	1(1)	8(3)	-	-	-	-	-
	0	0	-	-	-	-	-	7(0)	3(3)	-	-	-	-	-
10^{-7}														
Int.	3(0)	0	5(1)	0	2(2)	-	-	8(3)	8(3)	9(3)	1(3)	3(4)	-	-
	0	0	2(0)	0	2(0)	2(2)	2(0)	4(3)	3(3)	6(3)	6(2)	1(4)	2(4)	2(4)
Fire	0	0	-	-	-	-	-	4(2)	2(4)	-	-	-	-	-
	0	0	-	-	-	-	-	2(1)	1(4)	-	-	-	-	-
10^{-8}														
Int.	4(1)	0	4(2)	0	3(3)	-	-	2(4)	2(4)	2(4)	3(3)	8(4)	-	-
	0	0	5(1)	0	5(1)	1(3)	3(2)	9(3)	1(4)	1(4)	2(3)	2(4)	3(4)	3(4)
Fire	0	1(0)	-	-	-	-	-	5(3)	4(4)	-	-	-	-	-
	0	0	-	-	-	-	-	6(1)	2(4)	-	-	-	-	-
10^{-9}														
Int.	1(2)	1(0)	2(3)	0	4(3)	-	-	4(4)	4(4)	2(4)	6(3)	1(5)	-	-
	8(0)	0	2(2)	0	8(2)	4(3)	2(3)	2(4)	2(4)	2(4)	3(3)	4(4)	4(4)	5(4)
Fire	1(1)	3(0)	-	-	-	-	-	2(4)	5(4)	-	-	-	-	-
	0	0	-	-	-	-	-	1(3)	4(4)	-	-	-	-	-

*Plant Names: 1 = Surry; 2 = Peach Bottom; 3 = Sequoyah; 4 = Grand Gulf; 5 = Zion; 6 = RSS-PWR; 7 = RSS-BWR
a. First line of entries corresponds to mean CCDF; second line corresponds to median CCDF.
b. Int. = Internal initiating events
c. 6(1) = 6×10^1 = 60

Table 11.2 Summaries of mean and median CCDFs of offsite consequences—population exposures.

Exceedance Frequency (ry^{-1})	50-Mile Region Population Exposure (person-rem)[a]					Entire Site Region Population Exposure (person-rem)[a]				
	1*	2*	3*	4*	5*	1*	2*	3*	4*	5*
10^{-5}										
Int.[b]	7(2)[c]	0	1(5)	0	5(3)	2(3)	0	4(5)	0	9(3)
	2(2)	0	4(4)	0	3(3)	3(2)	0	1(5)	0	4(3)
Fire	5(1)	1(6)	-	-	-	1(2)	3(6)	-	-	-
	0	2(3)	-	-	-	0	3(3)	-	-	-
10^{-6}										
Int.	1(6)	3(6)	3(6)	2(5)	2(7)	8(6)	7(6)	2(7)	2(6)	5(7)
	6(5)	6(5)	1(6)	1(2)	3(6)	2(6)	1(6)	7(6)	2(2)	1(7)
Fire	3(4)	1(7)	-	-	-	1(5)	5(7)	-	-	-
	2(4)	6(6)	-	-	-	6(4)	2(7)	-	-	-
10^{-7}										
Int.	8(6)	1(7)	8(6)	6(5)	8(7)	5(7)	5(7)	6(7)	9(6)	2(8)
	5(6)	6(6)	4(6)	3(5)	3(7)	2(7)	2(7)	3(7)	3(6)	7(7)
Fire	6(5)	3(7)	-	-	-	2(6)	1(8)	-	-	-
	1(5)	1(7)	-	-	-	2(5)	7(7)	-	-	-
10^{-8}										
Int.	2(7)	2(7)	2(7)	1(6)	2(8)	1(8)	1(8)	9(7)	2(7)	3(8)
	9(6)	1(7)	7(6)	6(5)	7(7)	6(7)	8(7)	6(7)	9(6)	1(8)
Fire	6(6)	5(7)	-	-	-	3(7)	2(8)	-	-	-
	5(5)	3(7)	-	-	-	6(5)	1(8)	-	-	-
10^{-9}										
Int.	3(7)	4(7)	4(7)	2(6)	4(8)	2(8)	2(8)	1(8)	3(7)	4(8)
	1(7)	2(7)	1(7)	1(6)	1(8)	1(8)	1(8)	1(8)	2(7)	2(8)
Fire	2(7)	6(7)	-	-	-	9(7)	3(8)	-	-	-
	1(6)	4(7)	-	-	-	8(6)	2(8)	-	-	-

*Plant Names: 1 = Surry; 2 = Peach Bottom; 3 = Sequoyah; 4 = Grand Gulf; 5 = Zion

a. First line of entries corresponds to mean CCDF; second line corresponds to median CCDF.

b. Int. = Internal initiating events

c. 7(2) = 7 X 10^2 = 700

Table 11.3 Offsite protective measures assumptions.

1. **Emergency Response Assumptions**

   a. Within 10-mile plume exposure pathway emergency planning zone (EPZ):

   Evacuation of people after a delay* following the warning given by the reactor operator on the imminent radionuclide release.

   Average evacuation delay times (hr): Surry 2.0, Peach Bottom 1.5, Sequoyah 2.3, Grand Gulf 1.25, Zion 2.3.

   Average effective radial evacuation speeds (mile/hr): Surry 4.0, Peach Bottom 10.7, Sequoyah 3.1, Grand Gulf 8.3, Zion 2.5.

   b. Outside of 10-mile EPZ:

   Early relocation of people: within 12 hours/24 hours after plume passage from areas where the projected lifetime effective whole body dose equivalent (EDE), as defined in ICRP Publications 26 and 30, from a 7-day occupancy would exceed 50 rems/25 rems.

   Note: These assumptions are also extended inward up to the plant site boundary for the nonevacuating or nonsheltering people.

2. **Protective Action Guides (PAGs) for Long-Term Countermeasures**

   a. FDA "emergency" PAG for directly contaminated foods and animal feeds—dose not to exceed 5-rem EDE and 15-rem thyroid (Ref. 11.3).

   b. EPA's proposed PAGs for continuation of living in contaminated environment—dose not to exceed:

   - 2-rem EDE in the first year
   - 0.5-rem EDE in the second year

   from groundshine and inhalation of resuspended radionuclides.

   Note: EPA's criteria (Ref. 11.4) are approximated in MACCS as dose not to exceed 4-rem EDE in 5 years.

   c. In absence of any Federal agency criteria for ingestion dose to an individual from foods grown on contaminated soil via root uptakes, MACCS assumes a PAG of 0.5-rem EDE and 1.5-rem thyroid for this pathway, which is similar to FDA's "preventive" PAG for directly contaminated food and animal feeds (Ref. 11.3).

*Time steps involved during the delay are: (1) notification of the offsite authorities, (2) evaluation and decision by the authorities, (3) public notification advising evacuation, and (4) people's preparation for evacuation.

delay times for Peach Bottom and Grand Gulf enabled the evacuees to have a substantial head start on the plume. This, coupled with relatively fast effective evacuation speeds, enabled the evacuees to almost always avoid the trailing radioactive plumes. Thus, the relatively lower core damage frequencies, lower magnitudes of source term groups in the early phase of release, early warnings, lower population densities, lower evacuation delays, and higher evacuation speeds made the Peach Bottom and Grand Gulf early fatality CCDFs in Figures 4.9 and 6.8 lie in the low frequency and low magnitude regions, and early fatality magnitude entries in Table 11.1 small or nil.

Surry and Sequoyah fit between Peach Bottom/ Grand Gulf and Zion. For Surry and Sequoyah, warnings close to release in the interfacing-system LOCA accident sequences made evacuation less effective for these sequences. Also, evacuation was less effective in the plume rise scenarios for those source terms for which early release phases were associated with large quantities of radionuclides and large amounts of thermal energy (sequences with early containment failure at vessel breach). With the plume rise, the highest air and ground radionuclide concentrations occur at some distance farther from the reactor (instead of occurring close to the reactor without plume rise). In such cases, the late starting evacuees from the close-in regions moving away from the reactor in the downwind direction encounter higher concentrations and receive higher doses.

Latent Cancer Fatality Magnitudes

The estimates of latent cancer fatality magnitude at various exceedance frequencies include the benefits of the protective measures discussed above. Contributions from radiation doses down to very low levels have been included. If future research concludes that it is appropriate to truncate the individual dose at a *de minimis* level, reduced latent cancer fatality estimates would be obtained.

Variations of the latent cancer fatality magnitude for the five plants at equal exceedance frequency levels primarily arise because of differences in the source term groups and their frequencies, site meteorologies, and differences in the site demography, topography, land use, agricultural practice and productivity, and distribution of fresh water bodies up to 50 to 100 miles from the plants.

Emergency response in the close-in regions has only a limited beneficial impact on delayed cancer fatality magnitude and does not contribute substantially to the differences in the cancer fatality CCDFs for the five plants. The long-term protective measures, such as temporary interdiction, condemnation, and decontamination of land, property, and foods contaminated above acceptable levels are based on the same protective action guides (PAGs) for all plants. Further, the site differences for the five plants are not large enough beyond the distances of 50 to 100 miles to contribute substantially to the differences in the latent cancer fatality CCDFs.

Population Exposure Magnitudes

Population exposure magnitudes (person-rem*) at various exceedance frequencies include the contributions from the early and chronic exposures. These magnitudes reflect the dose-saving actions of the protective measures and, therefore, are the residual magnitudes.

Variations of the population exposure magnitudes for the five plants at equal exceedance frequency levels were similar to those of the cancer fatality magnitudes discussed earlier.

The relative contributions of the exposure pathways to the population dose for a given plant are highly source term dependent. Examples of relative contributions of early and chronic exposure pathways (see Chapter 2 and Appendix A) to the meteorology-averaged mean estimates of the 50-mile and entire region population dose for selected source term groups for the five plants are shown in Table 11.4. For brevity of presentation, only four source term groups that are the top contributors to the risks of the population dose for the five plants are selected. These source term groups are designated only by their identification numbers in Table 11.4. The chronic exposure pathway is shown subdivided in terms of direct (groundshine and inhalation of resuspended radionuclides) and ingestion (food and drinking water) pathways.

For a qualitative understanding of the results shown in Table 11.4, it should be noted that:

- All radionuclides contribute to the early exposure pathway; all nonnoble gas radionuclides contribute to the chronic direct exposure pathway; and only the radionuclides of iodine, strontium, and cesium contribute to the chronic ingestion exposure pathway.

*Effective dose equivalent (EDE) (as defined in ICRP Publications 26 and 30) in the unit of rem is used in the definition of person-rem.

Table 11.4 Exposure pathways relative contributions (percent) to meteorology-averaged conditional mean estimates of population dose for selected source term groups.

Plant Name	Source Term Group Identification Number	50-Mile Region*			Entire Region*		
		Early Exposure	Chronic Exposure		Early Exposure	Chronic Exposure	
			Direct	Ingestion		Direct	Ingestion
Surry	9	28	68	2	10	69	20
	33	51	41	3	14	74	12
	37	33	58	5	9	79	12
	49	13	80	7	9	58	33
Peach Bottom	28	28	66	2	15	77	7
	34	42	47	5	24	68	5
	37	38	52	5	20	72	6
	40	23	70	3	10	81	8
Sequoyah	32	49	36	8	11	68	20
	35	42	47	6	8	59	32
	43	49	28	19	11	73	15
	44	59	29	9	12	75	13
Grand Gulf	19	24	62	12	17	46	42
	25	16	65	16	4	54	41
	28	10	72	16	3	41	57
	32	41	39	17	12	62	25
Zion	139	50	46	1	27	56	16
	175	71	21	2	49	39	8
	142	24	73	1	23	60	15
	136	44	49	2	12	67	20

*The difference between 100 percent and the sum of the pathway contributions is the relative population dose to the decontamination workers.

- Early exposure pathway population dose estimated is largely unmitigated, except for the evacuated and relocated people. In addition to cloudshine and cloud inhalation during plume passage, it includes the groundshine and inhalation of resuspended radionuclides for a period of 7 days after the radionuclide release.

- Chronic exposure pathway involves dose integration from 7 days to all future times (i.e., the sum total of the dose over time).

- In the MACCS analysis, the protective actions to mitigate the chronic exposure pathways are largely confined to the 50-mile region of the site. Outside the 50-mile region, the mitigative actions (based on the PAGs) are generally not triggered in MACCS because of the relatively low levels of contamination (however, sometimes they are triggered depending on the meteorology and the source term magnitudes).

- Protective actions are not assumed for water ingestion.

Except for Grand Gulf, Table 11.4 shows that in the 50-mile region the early exposure pathway population dose and the chronic direct exposure pathway population dose are roughly similar; the chronic ingestion pathway makes smaller contributions. For the entire region, the chronic direct exposure pathway has increased contributions relative to the early exposure pathway. This is because at longer distances the early exposure pathway has weakened as a result of low air and ground concentrations and the short (i.e., 7 days) integration time for ground exposure. Relative contributions of the chronic ingestion exposure pathway are also higher for the entire region. This is because the chronic direct exposure is dependent on population size and the chronic ingestion exposure is dependent on farmland and water body surface area. An increase in the population size with distance from a plant generally occurs less rapidly compared to the increase in the area with distance.

For Grand Gulf, generally the contributions from the early exposure pathway are lower than the chronic direct exposure pathway in the 50-mile region relative to the other four plants and are due to the characteristics of the selected source term groups. For the entire region, the relative contributions of the early exposure pathway and chronic direct exposure pathway are similar to the other plants. However, the ingestion exposure pathway has higher contributions both in the 50-mile and entire region compared to the other plants. This is because the Grand Gulf site region has a smaller population size and a larger area devoted to farming than the other four sites of this study.

11.4 Comparison with Reactor Safety Study

The mean and the median CCDFs of two of the selected consequence measures, namely, early fatalities and latent cancer fatalities, displayed in Chapters 3 through 7 for the internal initiators of the reactor accidents and summarized in Table 11.1, may be compared with the CCDFs displayed in the Reactor Safety Study (RSS). However, the RSS CCDFs are the results of superpositions of the meteorology-based conditional CCDFs for the RSS "release categories"* after being weighted by the median frequencies of the release categories. The CCDFs shown in Chapters 3 through 7 are calculated in a different way from the RSS CCDFs. Thus, they are not strictly comparable.

The RSS CCDFs of early fatalities and latent cancer fatalities are shown in the RSS Figures 5-3 and 5-5, respectively. The magnitudes of delayed cancer fatalities shown in the RSS CCDFs are actually the magnitudes of their projected uniform annual rates of occurrence over a 30-year period. Thus, these RSS rate magnitudes need to be multiplied by a factor of 30 to derive their total magnitudes. After performing this step, the RSS results have been entered in Table 11.1 for comparison with the results of this study.

Table 11.1 shows that, for one or more early fatality magnitudes, the mean and median frequencies for the three PWRs of this study (Surry, Sequoyah, and Zion) and the median frequency for the RSS-PWR are similar and are less than 10^{-6} per reactor year. However, Table 11.1 also shows that these frequencies for the two BWRs of this study (Peach Bottom and Grand Gulf) are significantly lower than that for the RSS-BWR. For one or more early fatality magnitude, the median frequency is less than 10^{-6} per reactor year for the RSS-BWR; whereas, the mean and median frequencies are less than 10^{-8} per reactor year for Peach Bottom and less than 10^{-9} per reactor year for Grand Gulf.

Further, the comparison of the early fatality magnitudes in the median exceedance frequency

*RSS "release categories" are analogous to the source term groups in the present study but were developed by different procedures.

range of 10^{-9} to 10^{-7} per reactor year shows that the RSS estimates are significantly higher than the estimates for the five plants of this study.

Table 11.1 shows that for the one or more latent cancer fatality magnitudes, the mean and median frequencies of only one plant (Sequoyah) of this study and the median frequencies for the RSS-PWR and RSS-BWR are similar and are less than 10^{-4} per reactor year. However, these frequencies for the other four plants of this study are an order of magnitude lower than that for the RSS; i.e., less than 10^{-5} per reactor year.

The RSS estimates of latent cancer fatality magnitudes for the median exceedance frequency range of 10^{-9} to 10^{-5} per reactor year are higher (in some instances significantly higher) than those for the five plants of this study—except for Zion at the median exceedance frequency of 10^{-9} per reactor year where they are about equal.

There are several factors contributing to the differences in the frequency distributions of the offsite consequences for this study and the RSS. Some of these factors are mentioned below:

- Accident sequence frequency differences.
- Source term characterization difference. Most of the source terms of this study have two releases—an early release and a later release. Early fatalities from a source term are mostly the consequences of the early release. Cancer fatalities are the consequences of both early and later releases. On the other hand, the RSS source terms did not have such a breakdown in terms of early or later release. Therefore, the early fatalities from an RSS source term were the consequences of the entire release, as were the latent cancer fatalities.
- Consequence analyses for this study are site specific, using data for the site features described in Chapters 3 through 7. The RSS consequence analysis was generic; it used composite offsite data by averaging over 68 different sites.
- In the present study, evacuation to a distance of 10 miles is assumed; whereas, in the RSS, evacuation to a distance of 25 miles was assumed.
- Health effect models of this study are different from those of the RSS.
- Protective action guide dose levels for controlling the long-term exposure are different.
- There are other miscellaneous differences between the accident consequence models and input data used in this study and the RSS.
- Different procedures were used for constructing the CCDFs.

11.5 Uncertainties and Sensitivities

There are uncertainties in the CCDFs of the offsite consequence measures. Some of these uncertainties are inherited from the uncertainties in the source term group specifications and frequencies. However, even after disregarding the source term group uncertainties, there are significant uncertainties in the CCDFs of the consequence measures due to uncertainties in the modeling of atmospheric dispersion, deposition, and transport of the radionuclides; transfer of radionuclides in the terrestrial exposure pathways; emergency response and long-term countermeasures; dosimetry, shielding, and health effects; and uncertainties in the input data for the model parameters.

Because of time constraints, uncertainty analyses for the offsite consequences, except for the uncertainties due to variability of the site meteorology, have not been performed for this report. They are planned for future studies. For this study, only best estimate values of the parameters for representation of the natural processes have been used in MACCS. An analysis of sensitivity of the CCDFs to the alternative protective measure assumptions is provided in the following section.

11.6 Sensitivity of Consequence Measure CCDFs to Protective Measure Assumptions

Emergency response, such as evacuation, sheltering, and early relocation of people, has its greatest beneficial impact on the early fatality frequency distributions. The long-term protective measures, such as decontamination, temporary interdiction, and condemnation of contaminated land, property, and foods in accordance with various radiological protective action guides (PAGs), have their largest beneficial impact on the latent cancer fatality and population exposure frequency distributions.

11.6.1 Sensitivity of Early Fatality CCDFs to Emergency Response

Four alternative emergency response modes within the 10-mile EPZ, as characterized in Table

11.5, are assumed in order to show the sensitivity of early fatality CCDFs to these response modes.

Table 11.6 summarizes the early fatality mean CCDFs in tabular form for Surry, Peach Bottom, Sequoyah, and Grand Gulf for two alternative emergency response modes, and Zion for all four alternative emergency response modes. Several inferences are drawn later in this section regarding the effectiveness of these alternative emergency response modes for the five plants based on these data. However, more analysis is needed to support these inferences for emergency response and to provide detailed insight into the underlying competing processes involved that diminish or enhance the effectiveness of any emergency response mode.

In particular, the effectiveness of evacuation is very site specific and source term specific. It is largely determined by two site parameters, namely, evacuation delay time and effective evacuation speed, and two source term parameters—warning time before release and energy associated with the release (which, during some meteorological conditions, could cause the radioactive plume to rise while being transported downwind). Therefore, it cannot be extrapolated across the source terms for a plant or across the plants for similar source terms.

The CCDFs discussed here include contributions from many source term groups. The effectiveness of any emergency response mode judged from the sensitivity of the early fatality mean CCDF for a plant is essentially the effectiveness for the dominant source terms in specific frequency intervals included in the CCDF. With these caveats, the inferences based on the data shown in Table 11.6 are as follows:

Zion

1. Evacuation from the 0-to-5 mile EPZ combined with sheltering in the 5-to-10 mile EPZ is as effective as evacuation from the entire 10-mile EPZ. Effectiveness of evacuation in close-in regions of radius less than 5 miles and sheltering in the outer regions will be evaluated in future studies. (See Chapter 13.)

2. Sheltering, due to better shielding protection indoors, is more effective than early relocation from the state of normal activity. (See Tables 11.3 and 11.5 for distinctions between evacuation, early relocation, and sheltering modes of response assumed in this study.)

Sequoyah

1. Evacuation is more effective than relocation for exceedance frequencies higher than 10^{-8} per reactor year.

2. In the low frequency region (i.e., 10^{-8} per reactor year or less), the early relocation mode is more effective than evacuation. This "crossover" of the early fatality mean CCDFs for the two response modes is likely because of the dominance of the low frequency large source terms that also have short warning times before release and/or high energy contents and calculated long evacuation delay time and slow effective evacuation speed. Because of the short warning time before release and a long delay between the warning and the start of evacuation, many evacuees become vulnerable to the radiation exposures from the passing plume and contaminated ground rather than escape these exposures. Because of the plume-rise effect (for the hot plumes), the peak values of the air and ground radionuclide concentrations occur at some distance farther from the plant. In such a case, the evacuees from close-in regions moving in the downwind direction move from areas of lower concentrations to areas of higher concentrations and receive a higher dose. It should be noted that, while evacuating, the people are out in the open and have minimal shielding protection. For the above situations, the sheltering mode also would show the same crossover effect.

   However, the crossover effect showing that relocation or sheltering may be more effective than evacuation may not be realistic because of uncertainties in the consequence analysis.

Peach Bottom, Grand Gulf

The source terms and features of these two low population density sites make evacuation a very effective mode of offsite response.

Surry

Although entries in Table 11.6 show that evacuation is more effective than relocation from the state of normal activity, some low probability accident sequences for Surry are similar to those of Sequoyah (short warning times of the interfacing-system LOCA accident sequences and large

Table 11.5 Assumptions on alternative emergency response modes within 10-mile plume exposure pathway EPZ for sensitivity analysis.

a. Evacuation (see Table 11.3).

b. Early relocation in lieu of evacuation or shelter: Extends the assumptions for relocation outside the 10-mile EPZ (see Table 11.3) inward up to the plant site boundary.

c. Sheltering* (getting to and remaining indoors) in lieu of evacuation, followed by fast relocation after plume passage.

d. Evacuation for the inner 0-5 mile region and sheltering* in the outer 5-10 mile region followed by fast relocation after plume passage.

*Sheltering assumptions details: After an initial delay of 45 minutes from the reactor operator's warning, people get indoors and remain indoors and are relocated to uncontaminated areas within a maximum of 24 hours of remaining indoors. However, virtually all source terms analyzed in this study have two release phases—an early (first) release and a later (second) release. If there is a sufficient time gap (about 4 hours) between the two release phases, then people from indoors can be relocated to uncontaminated areas during this gap and avoid the exposure from the second release. With this perspective, two cases of relocation earlier than 24 hours are implemented in calculations as follows:

- Relocation within 4 hours after termination of the initial (the first) release, if the second release does not occur within this 4 hours; otherwise,
- Relocation within 4 hours after termination of the second release (provided this relocation time is earlier than 24 hours of indoor occupancy; otherwise, relocation is at 24 hours of indoor occupancy).

The dose for the above extra 4-hour period is assumed to account for the dose during the period of waiting for the plume to leave the area after termination of the release and the dose during people's transit to the relocation areas.

Table 11.6 Sensitivity of mean CCDF of early fatalities to assumptions on offsite emergency response.

Exceedance Frequency (ry^{-1})	10-mile EPZ Emergency Response Mode*	Early Fatalities (persons)				
		Surry	Peach Bottom	Sequoyah	Grand Gulf	Zion
10^{-5}	a. Evacuation	0/0	0/0	0	0	0
	b. Relocation	0/0	0/0	0	0	0
	c. Shelter	**	**	**	**	0
	d. Evac/Shelter	**	**	**	**	0
10^{-6}	a. Evacuation	0/0	0/0	0	0	0
	b. Relocation	0/0	0/2(1)	6(0)	0	6(0)[a]
	c. Shelter	**	**	**	**	0
	d. Evac/Shelter	**	**	**	**	0
10^{-7}	a. Evacuation	0/0	0/0	5(1)	0	2(2)
	b. Relocation	2(1)/0	1(1)/1(2)	7(1)	2(0)	1(3)
	c. Shelter	**	**	**	**	7(2)
	d. Evac/Shelter	**	**	**	**	2(2)
10^{-8}	a. Evacuation	4(1)/0	0/0	4(2)	0	3(3)
	b. Relocation	2(2)/0	7(1)/3(2)	2(2)	2(1)	8(3)
	c. Shelter	**	**	**	**	6(3)
	d. Evac/Shelter	**	**	**	**	3(3)
10^{-9}	a. Evacuation	1(2)/1(1)	0/0	2(3)	0	4(3)
	b. Relocation	9(2)/5(1)	2(2)/5(2)	6(2)	8(1)	2(4)
	c. Shelter	**	**	**	**	9(3)
	d. Evac/Shelter	**	**	**	**	4(3)

Note: Under each plant name, the first entry is for the internal initiators and the second entry is for fire.

*See Table 11.3 for assumptions.
**No data
a. 6(0) = 6×10^0 = 6

thermal energy for the sequences with early containment failure at vessel breach). Analyses of the sensitivity of early fatality CCDFs to sheltering, or a combination of evacuation and sheltering, have not been performed for Surry (nor for Peach Bottom, Sequoyah, and Grand Gulf).

11.6.2 Sensitivity of Latent Cancer Fatality and Population Exposure CCDFs to Radiological Protective Action Guide (PAG) Levels for Long-Term Countermeasures

The potential for latent cancer fatalities and population exposure is assumed to exist down to any low level of radiation dose and, therefore, over the entire site region. Although both early and chronic exposure pathways contribute to these consequence measures, only the chronic exposure pathways are expected to be mitigated by the long-term countermeasures such as decontamination, temporary interdiction, or condemnation of contaminated land, property, and foods based on guidance provided by responsible Federal agencies in terms of PAGs. This implies that, if the radiation dose to an individual from a chronic exposure pathway would be projected to exceed the PAG (or intervention) level for that pathway, countermeasures should be undertaken to reduce the projected dose from the pathway so that it does not exceed the PAG level. Therefore, the latent cancer fatalities and the population exposures stemming from the chronic exposure pathways are expected to be sensitive to the PAG values.

The chronic exposure pathways base case PAGs are shown in Table 11.3. The only alternative PAG used for this sensitivity analysis is the RSS PAG for the groundshine dose to an individual for continuing to live in the contaminated environment. The RSS PAG adopted here is 25-rem EDE from groundshine and inhalation of resuspended radionuclides (instead of the RSS 25-rem whole body dose from groundshine only) in 30 years. This alternative is used to replace the base case PAG of 4-rem EDE in 5 years.

Summaries of the latent cancer fatality and population exposure mean CCDFs for both cases for the five plants for the internal initiating events are shown in Table 11.7.

Table 11.7 shows that there is practically no difference between the consequence magnitudes for the five plants for the two PAGs for continuing to live in the contaminated environment at the exceedance frequency of 10^{-5} per reactor year. This is because the source terms with frequency 10^{-5} per reactor year or higher have low release magnitudes such that the resulting environmental contaminations are below both the EPA and RSS PAG-based trigger levels for protective actions (i.e., no protective actions are needed).

At lower exceedance frequencies, source terms with larger release magnitudes contribute and the two PAGs reduce the consequences to different extents. The RSS PAG is less restrictive than the EPA PAG. Thus, the long-term consequence magnitudes with the RSS PAG are generally higher than those with the EPA PAG at equal exceedance frequencies. However, the economic consequences, discussed in the supporting contractor reports (Refs. 11.5 through 11.9), would show just the opposite behavior, i.e., economic consequences would be higher for the EPA PAG than for the RSS PAG.

Table 11.7 Sensitivity of mean CCDFs of latent cancer fatalities and population exposures to the PAGs for living in contaminated areas—internal initiating events.

Exceedance Frequency (ry-1)	Cancer Fatalities (persons)					50-Mile Pop. Exp. (person-rem)					Entire Region Pop. Exp. (person-rem)				
	1*	2*	3*	4*	5*	1*	2*	3*	4*	5*	1*	2*	3*	4*	5*
10^{-5}															
EPA+	0	0	6(1)[a]	0	0	7(2)	0	1(5)	0	5(3)	2(3)	0	4(5)	0	9(3)
RSS+	0	0	6(1)	0	0	7(2)	0	1(5)	0	5(3)	2(3)	0	4(5)	0	9(3)
10^{-6}															
EPA	1(3)	1(3)	4(3)	3(2)	8(3)	1(6)	3(6)	3(6)	2(5)	2(7)	8(6)	7(6)	2(7)	2(6)	5(7)
RSS	2(3)	2(3)	5(3)	3(2)	1(4)	2(6)	4(6)	5(6)	2(5)	3(7)	1(7)	1(7)	3(7)	2(6)	8(7)
10^{-7}															
EPA	8(3)	8(3)	9(3)	1(3)	3(4)	8(6)	1(7)	8(6)	6(5)	8(7)	5(7)	5(7)	6(7)	9(6)	2(8)
RSS	9(3)	1(4)	1(4)	2(3)	4(4)	1(7)	2(7)	1(7)	1(6)	2(8)	6(7)	7(7)	6(7)	1(7)	2(8)
10^{-8}															
EPA	2(4)	2(4)	2(4)	3(3)	8(4)	2(7)	2(7)	2(7)	1(6)	2(8)	1(8)	1(8)	9(7)	2(7)	3(8)
RSS	2(4)	4(4)	2(4)	4(3)	1(5)	2(7)	4(7)	2(7)	2(6)	3(8)	2(8)	2(8)	1(8)	2(7)	4(8)
10^{-9}															
EPA	4(4)	4(4)	2(4)	6(3)	1(5)	3(7)	4(7)	4(7)	2(6)	4(8)	2(8)	2(8)	1(8)	3(7)	4(8)
RSS	5(4)	4(4)	3(4)	6(3)	-	4(7)	6(7)	4(7)	3(6)	4(8)	3(8)	5(8)	2(8)	4(7)	4(8)

* Plant Names: 1 = Surry; 2 = Peach Bottom; 3 = Sequoyah; 4 = Grand Gulf; 5 = Zion

+ Long-term relocation PAGs:

EPA = 4-rem EDE in 5 years from groundshine—an approximation of EPA-proposed long-term relocation PAG

RSS = 25-rem EDE in 30 years from groundshine—RSS long-term relocation PAG

a. 6(1) = 6×10^1 = 60

REFERENCES FOR CHAPTER 11

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

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

11.3 U.S. Department of Health and Human Services/Food and Drug Administration, "Accidental Radioactive Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies," *Federal Register*, Vol. 47, No. 205, pp. 47073-47083, October 22, 1982.

11.4 U.S. Environmental Protection Agency, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," Draft, 1989.

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

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

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

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

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

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

12. PERSPECTIVES ON PUBLIC RISK

12.1 Introduction

One of the objectives of this study has been to gain and summarize perspectives regarding risk to public health from severe accidents at the five studied commercial nuclear power plants. In this chapter, risk measures for these plants are compared and perspectives drawn from these comparisons.

As discussed in Chapter 2, the quantitative assessment of risk involves combining severe accident sequence frequency data with corresponding containment failure probabilities and offsite consequence effects. An important aspect of the risk estimates in this study is the explicit treatment of uncertainties. The risk information discussed here includes estimates of the mean and the median of the distributions of the risk measures and the 5th percentile and the 95th percentile values. The risk results obtained have been analyzed with respect to major contributing accident sequences, plant-specific design and operational features, and accident phenomena that play important roles.

The assessments of plant risk that support the discussions of this chapter are discussed in detail in References 12.1 through 12.7 and summarized in Chapters 3 through 7 for the five individual plants. Appendix C to this report provides more detailed information on certain technical issues important to the risk studies. This work was performed by Sandia National Laboratories (on the Surry, Sequoyah, Peach Bottom, and Grand Gulf plants) and Idaho National Engineering Laboratory and Brookhaven National Laboratory (on the Zion plant).

12.2 Summary of Results

Estimates of risk presented in Chapters 3 through 7 for the five plants studied are compared in this section. Risk measures that are used for these comparisons are: early fatality, latent cancer fatality, average individual early fatality, and average individual latent cancer fatality risks for internally initiated and externally initiated (fire) events (additional risk measures are provided in Refs. 12.3 through 12.7). For reasons discussed in Chapter 1, seismic risk is not discussed here.

In order to display the variabilities in the noted risk measures, the early fatality and latent cancer fatality risk results of all five plants from internally initiated accidents are plotted together in Figure 12.1. Individual early fatality and latent cancer fatality risks from internally initiated accidents are compared with the NRC safety goals* (Ref. 12.8) in Figure 12.2. Similar risk results from externally initiated (fire) accidents for the Surry and Peach Bottom plants are presented in Figures 12.3 and 12.4. Estimates of the frequencies of a "large release" of radioactive material (using a definition of large as a release that results in one or more early fatalities) are presented in Figure 12.5.

Based on the results of the risk analyses for the five plants, a number of general conclusions can be drawn:

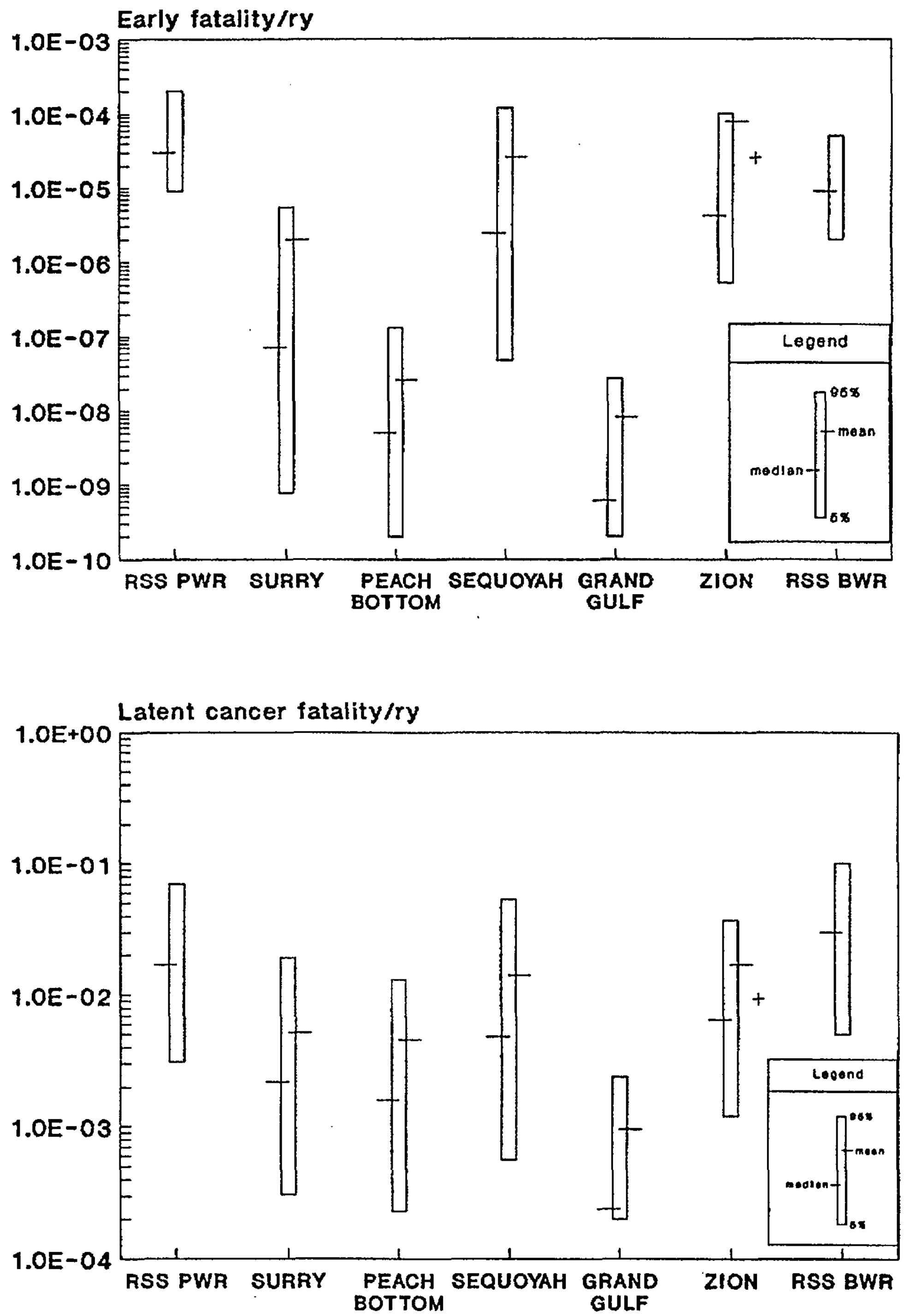
- The risks to the public from operation of the five plants are, in general, lower than the Reactor Safety Study (Ref. 12.10) estimates for two plants in 1975. Among the five plants studied, the two BWRs show lower risks than the three PWRs, principally because of the much lower core damage frequencies estimated for these two plants, as well as the mitigative capabilities of the BWR suppression pools during the early portions of severe accidents.

- Individual early fatality and latent cancer fatality risks from internally initiated events for all of these five plants, and from fire-initiated accidents for Surry and Peach Bottom, are well below the NRC safety goals.

- Fire-initiated accident sequences have relatively minor effects on the Surry plant risk compared to the risks from internal events but have a significant impact on Peach Bottom risk.

- The Surry and Zion plants benefit from their strong and large containments and therefore have lower conditional early containment failure probabilities. The Peach Bottom and Grand Gulf have higher conditional probabilities of early failure, offsetting to some degree the risk benefits of estimated lower core damage frequencies for these plants.

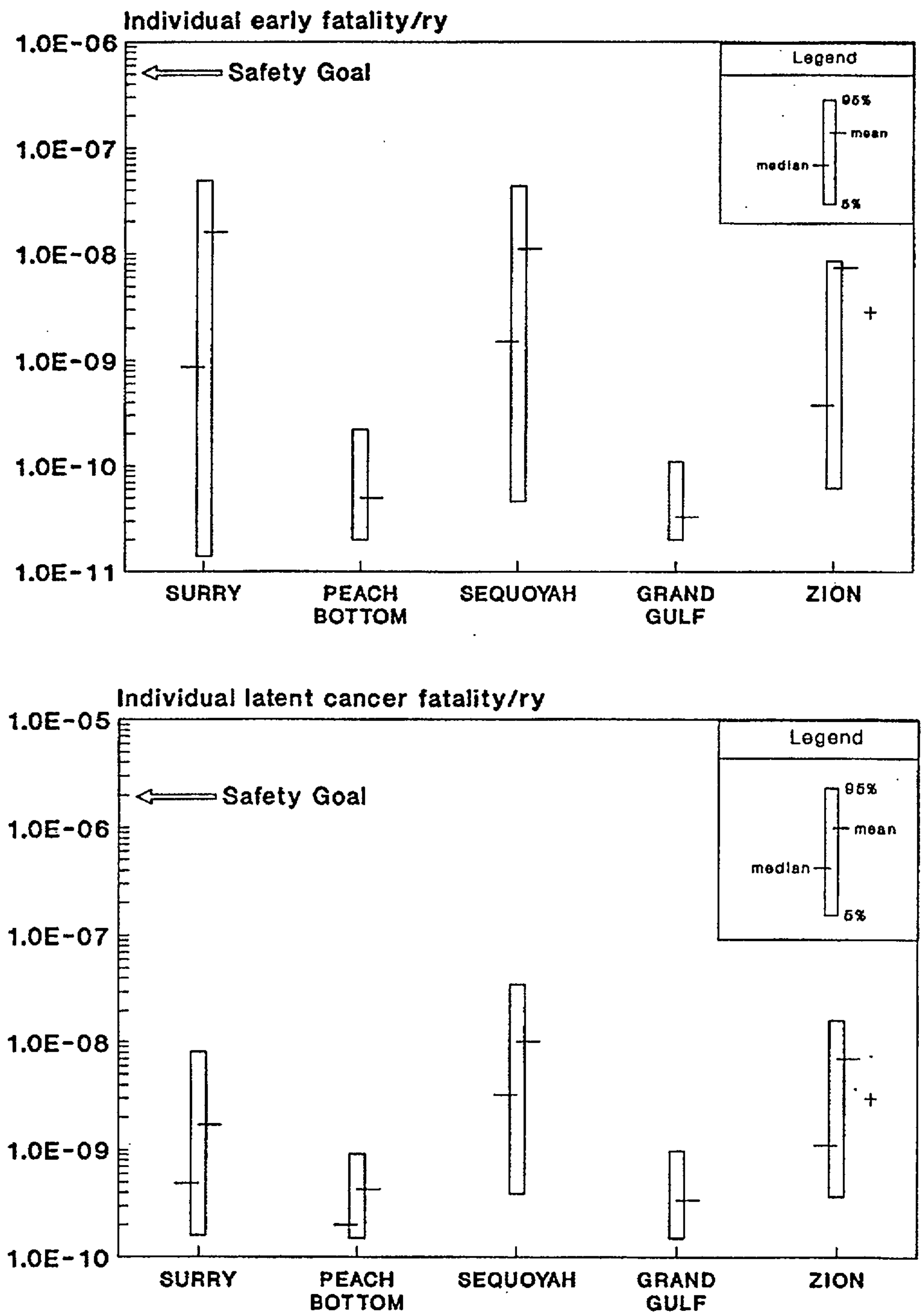
*Throughout this report, discussion of and comparison with the NRC safety goals relates specifically and only to the two quantitative health objectives identified in the Commission's policy statement (Ref. 12.8).



Notes: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

"+" indicates recalculated mean value based on recent modifications to the Zion plant (as discussed in Section C.15).

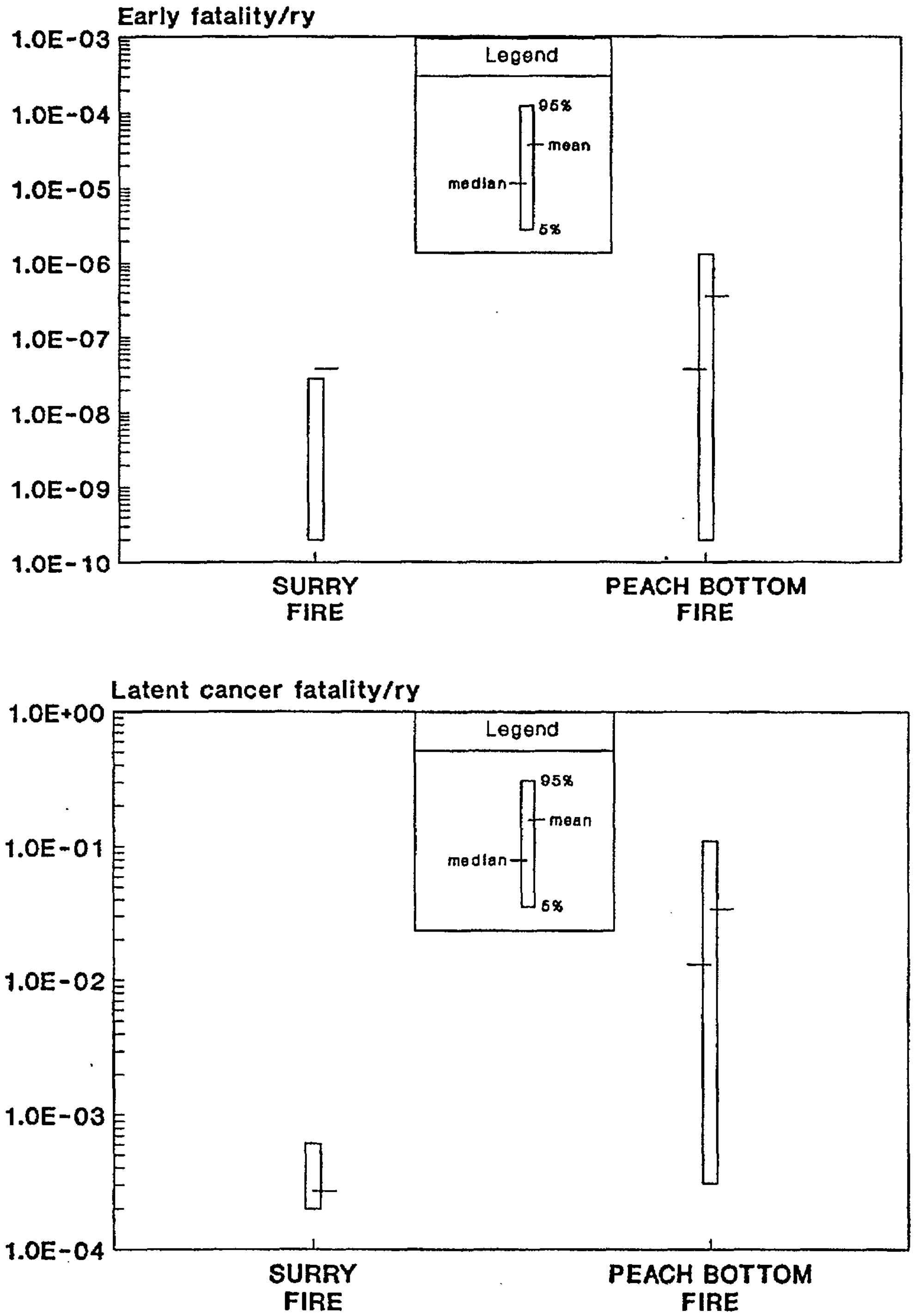
Figure 12.1 Comparison of early and latent cancer fatality risks at all plants (internal events).



Notes: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

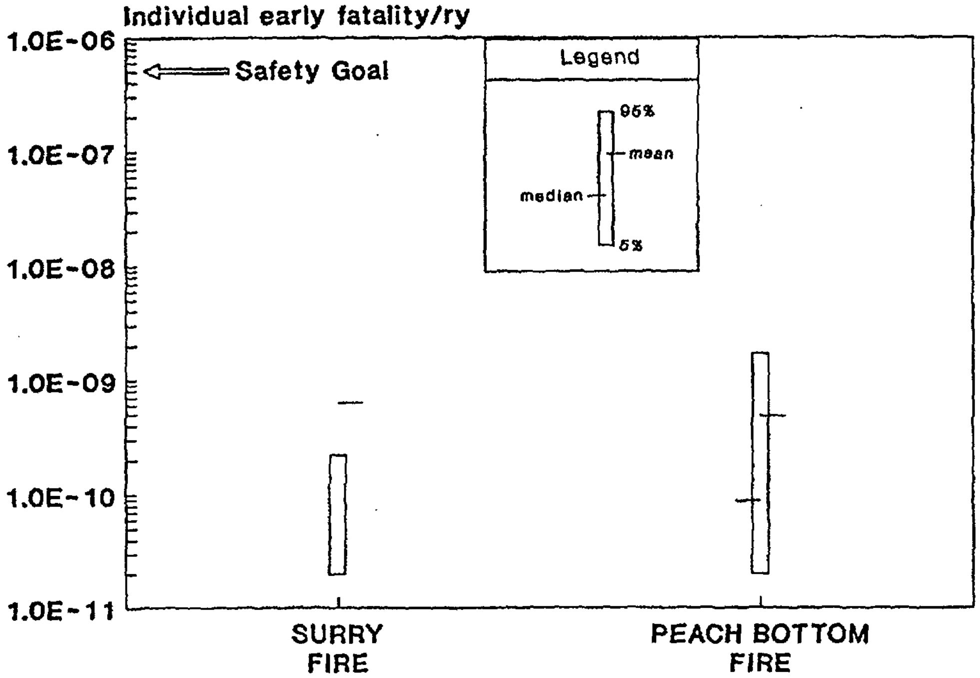
"+" indicates recalculated mean value based on recent modifications to the Zion plant (as discussed in Section C.15).

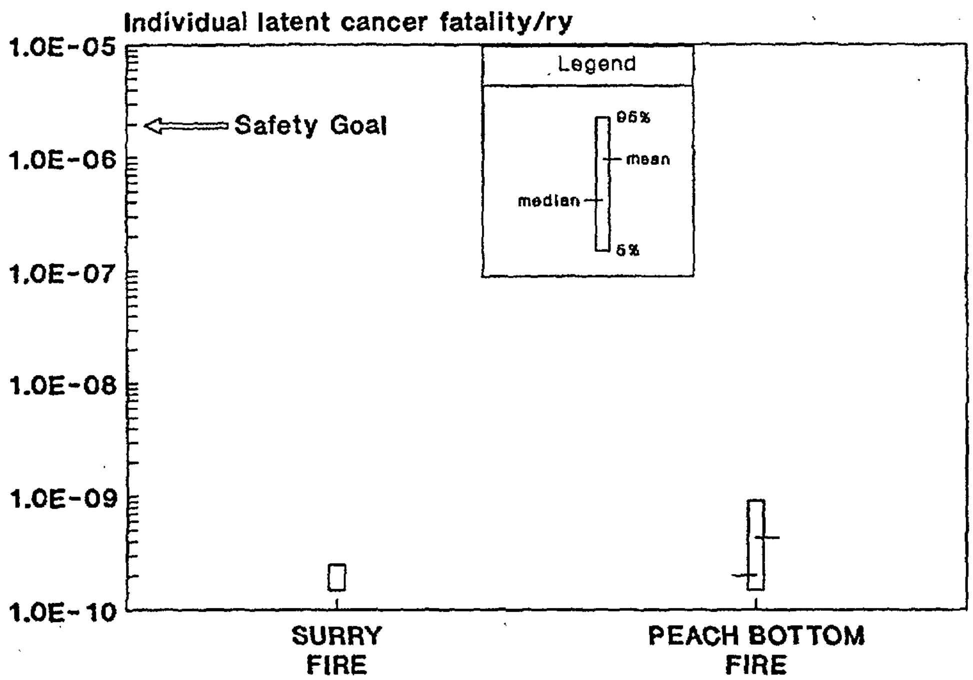
Figure 12.2 Comparison of risk results at all plants with safety goals (internal events).



Note: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

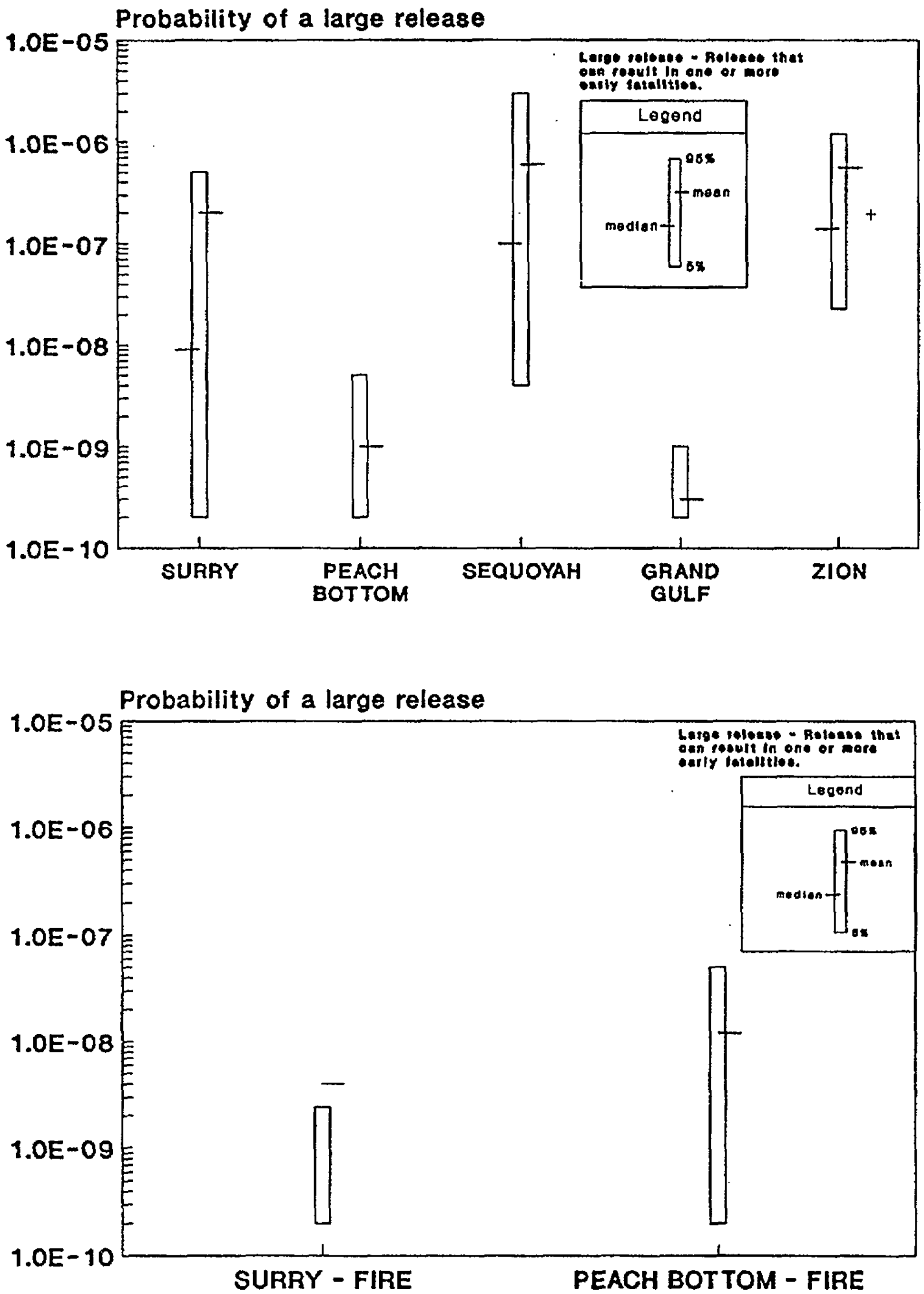
Figure 12.3 Comparison of early and latent cancer fatality risks at Surry and Peach Bottom (fire-initiated accidents).





Note: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 12.4 Comparison of risk results at Surry and Peach Bottom with safety goals (fire-initiated accidents).



Notes: As discussed in Reference 12.9, estimated risks at or below 1E-7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

"+" indicates recalculated mean value based on recent modifications to the Zion plant (as discussed in Section C.15).

Figure 12.5 Frequency of one or more early fatalities at all plants.

- The principal challenges to containment structures vary considerably among the five plants studied. Hydrogen combustion is a significant threat to the Sequoyah and Grand Gulf plants (in part because of the inoperability of ignition systems in some key accident sequences), while direct attack of the containment structure by molten core material is most important in the Peach Bottom plant. Few physical processes were identified that could seriously challenge the Surry and Zion containments.

- Emergency response parameters (warning time, evacuation speed, etc.) appear to have a significant impact on early fatality risk but almost no effect on latent cancer fatality risk.

12.3 Comparison with Reactor Safety Study

Results of the present study (for internal initiators) are compared with the Surry and Peach Bottom results in the Reactor Safety Study (RSS) in Figure 12.1. In general, for the early fatality risk measure, the Surry risk estimates in this study are lower than the corresponding RSS PWR values. Similarly, the present Peach Bottom risk estimates are lower than the RSS BWR estimates. For the latent cancer fatality risk measure, the patterns in the results are less clear; the RSS risk estimates for both of the plants lie in the upper portion of the risk estimates of this study.

Focusing on the major contributors to risk, it may be seen that, in the RSS, the Surry risk was dominated by interfacing-system LOCA (the V sequence), station blackout (TMLB'), and small LOCA sequences, with hydrogen burning and overpressure failures of containment. While the estimated risks of the interfacing-system LOCA accident sequence are lower in the present study because of a lower estimated frequency, it is still an important contributor to risk. Also important (because of their large source terms) are containment bypass accidents initiated by steam generator tube rupture, compounded by operator errors (which result in core damage) and subsequent stuck-open safety-relief valves on the secondary side. Early overpressurization containment failure at Surry is much less probable.

In the Peach Bottom analysis of the RSS, risk was dominated by transient-initiated events with loss of heat removal (TW type of sequence) and ATWS accidents with failure of containment prior to vessel breach. Dominant containment failure modes were from steam overpressurization. In the present study, risk is dominated by long-term station blackout and ATWS accident sequences. The dominant containment failure mode is drywell meltthrough.

The RSS did not perform an analysis of accidents initiated by fires. As such, comparisons of the present study's fire risk estimates with the RSS are not possible.

Since the publication of the RSS in 1975, a vast amount of work has been done in all areas of risk analysis, funded by government agencies and the nuclear industry. Major improvements have been made in the understanding of severe accident phenomenology and approaches to quantification of risk, many of which have been used in this study. These efforts have helped in lowering the estimates of overall risk levels in the present study to some extent by reducing the use of conservative and bounding types of analyses. Equally important, some plants have made modifications to plant systems or procedures based on PRAs, lessons learned from the Three Mile Island accident, etc., thus reducing risk. On the other hand, new issues have been raised and the possibility of new phenomena such as direct containment heating and drywell meltthrough has been introduced, which added to the previous estimates of risk. For issues that are not well understood, expert judgments were elicited that frequently showed diverse conclusions. The net effect of this improved understanding is that total plant risk estimates are lower than the RSS estimates, but the distributions of these risk measures are very broad.

12.4 Perspectives

As discussed above, plant-specific features contribute largely to the estimates of risks. In order to compare the variables and characteristics of the three PWR plants (Surry, Sequoyah, and Zion) and two BWR plants (Peach Bottom and Grand Gulf) in this study, the dominant contributors to early and latent cancer fatality risks for the PWRs and BWRs from internally initiated events are shown in Figures 12.6 through 12.10. Dominant contributors to risk from fire-initiated accidents for Surry and Peach Bottom are compared in Figure 12.9. Perspectives on risks for the five plants from these comparisons, supplemented by information in the supporting contractor reports (Refs. 12.1 through 12.7) are discussed below.

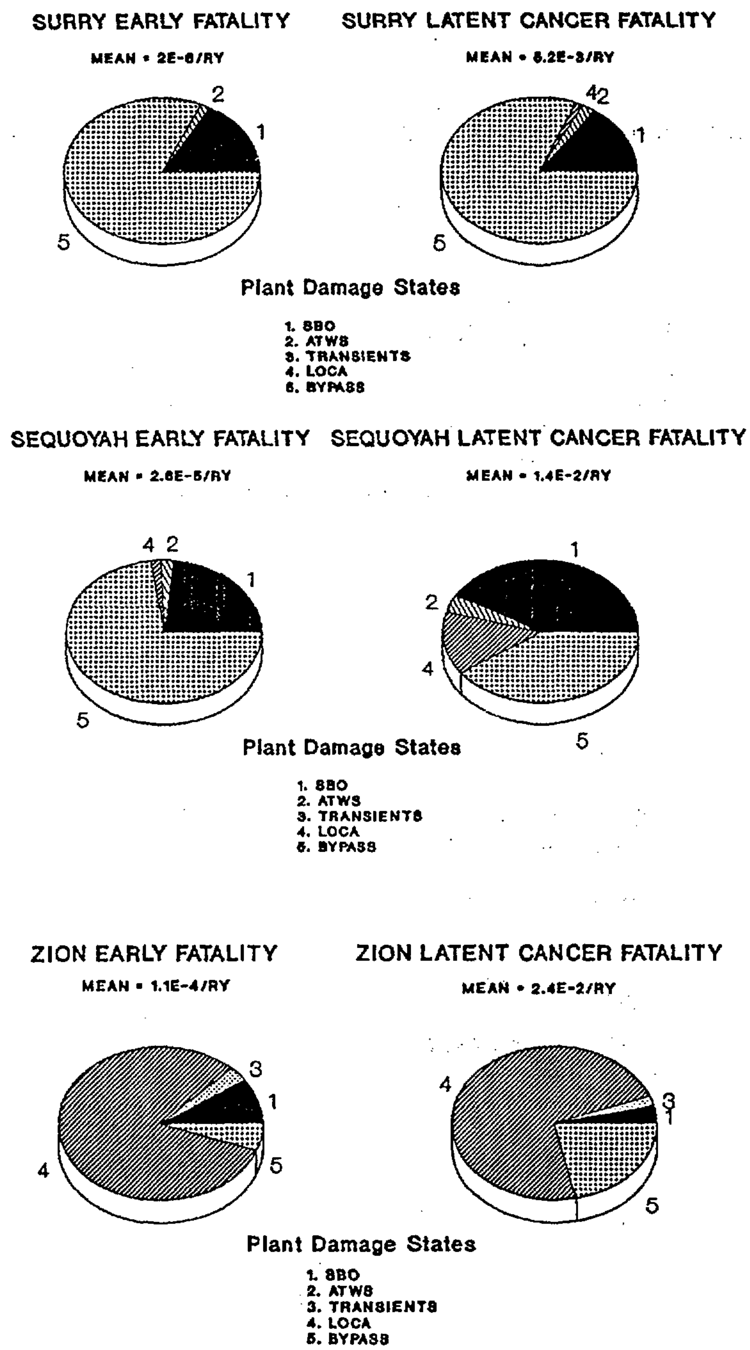


Figure 12.6 Contributions of plant damage states to mean early and latent cancer fatality risks for Surry, Sequoyah, and Zion (internal events).

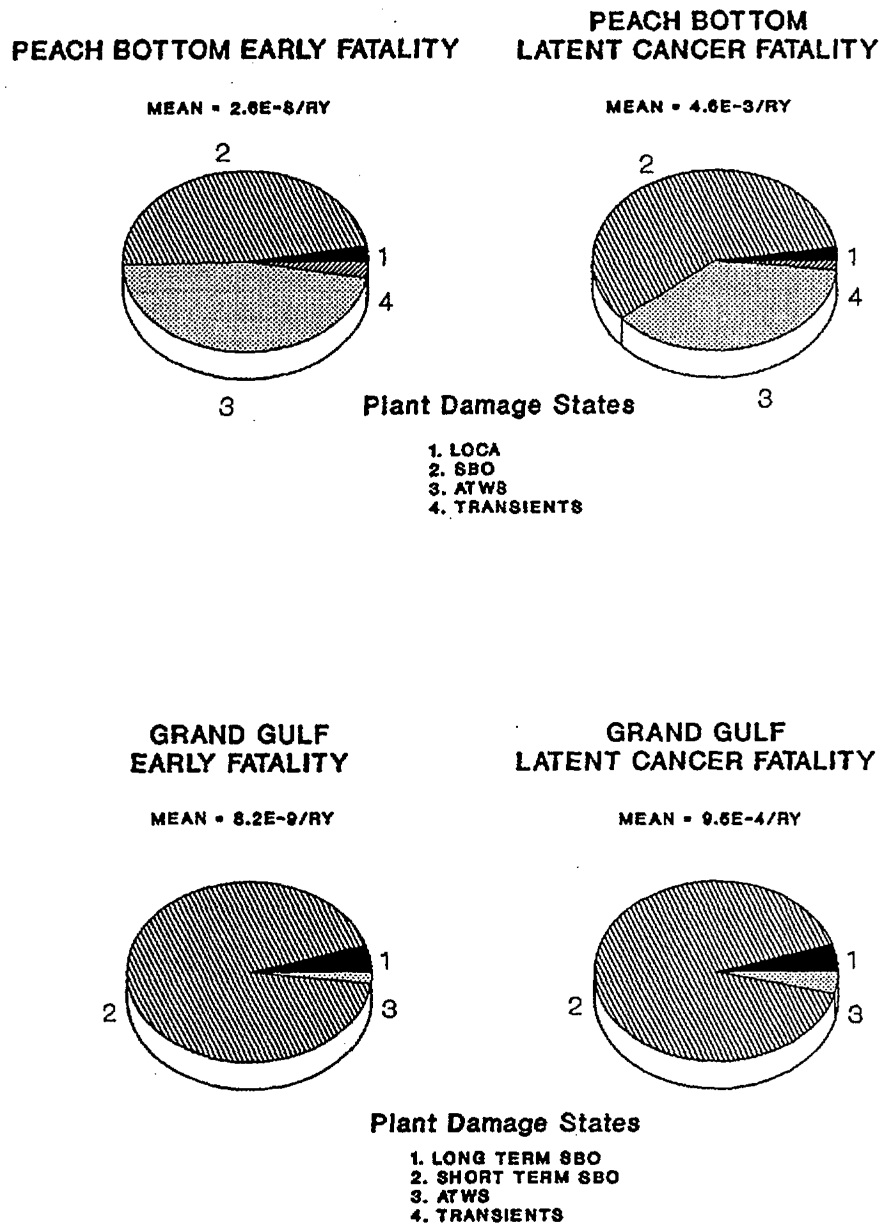


Figure 12.7 Contributions of plant damage states to mean early and latent cancer fatality risks for Peach Bottom and Grand Gulf (internal events).

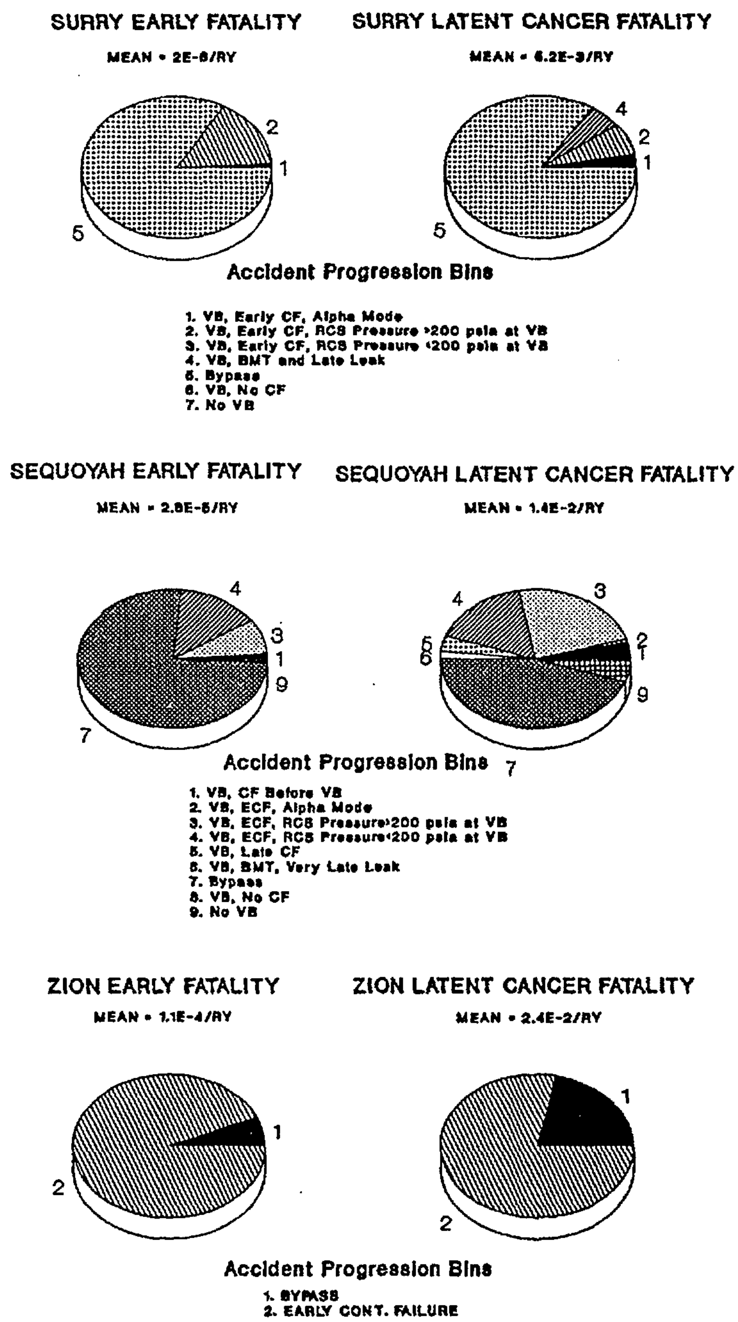


Figure 12.8 Contributions of accident progression bins to mean early and latent cancer fatality risks for Surry, Sequoyah, and Zion (internal events).

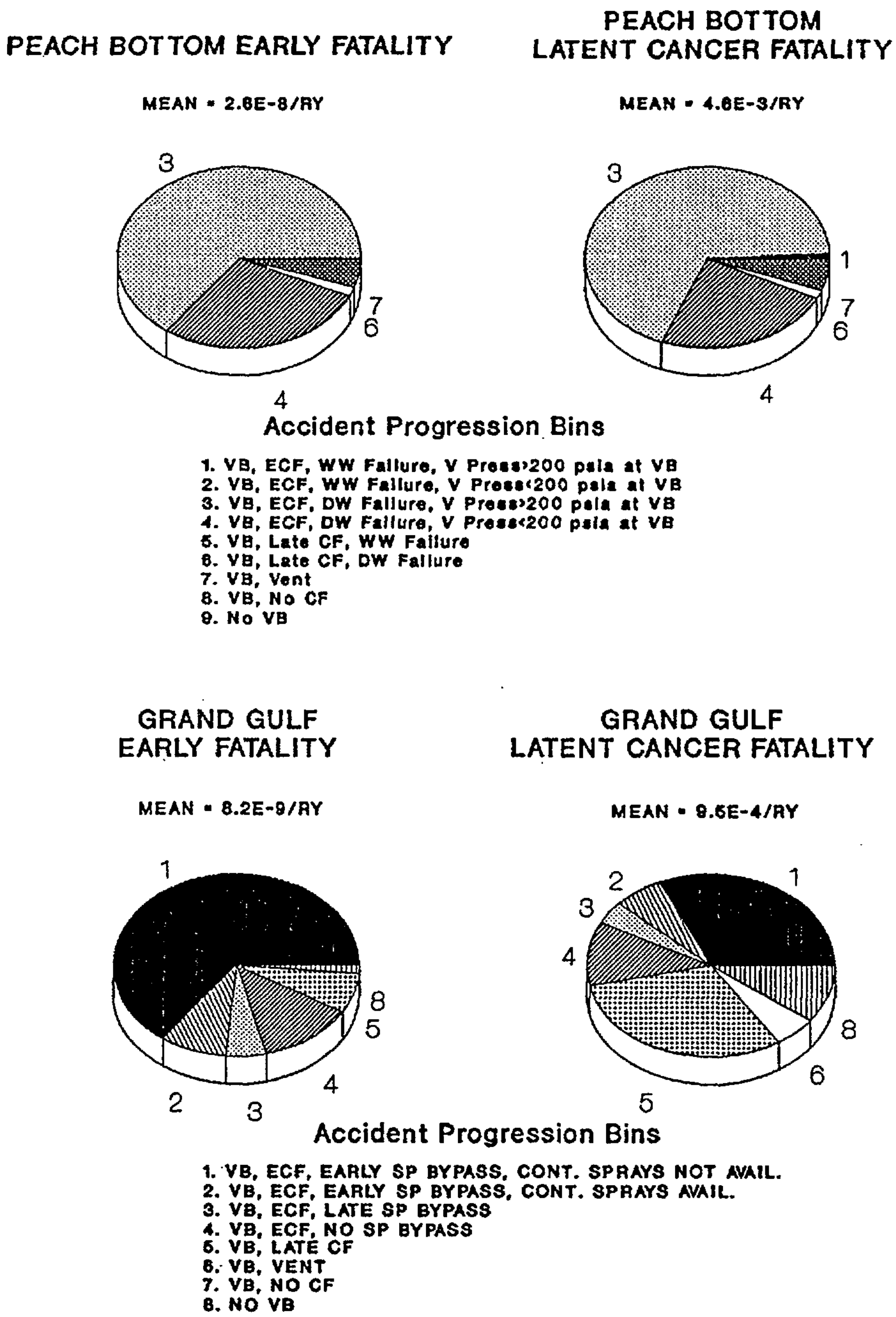


Figure 12.9 Contributions of accident progression bins to mean early and latent cancer fatality risks for Peach Bottom and Grand Gulf (internal events).

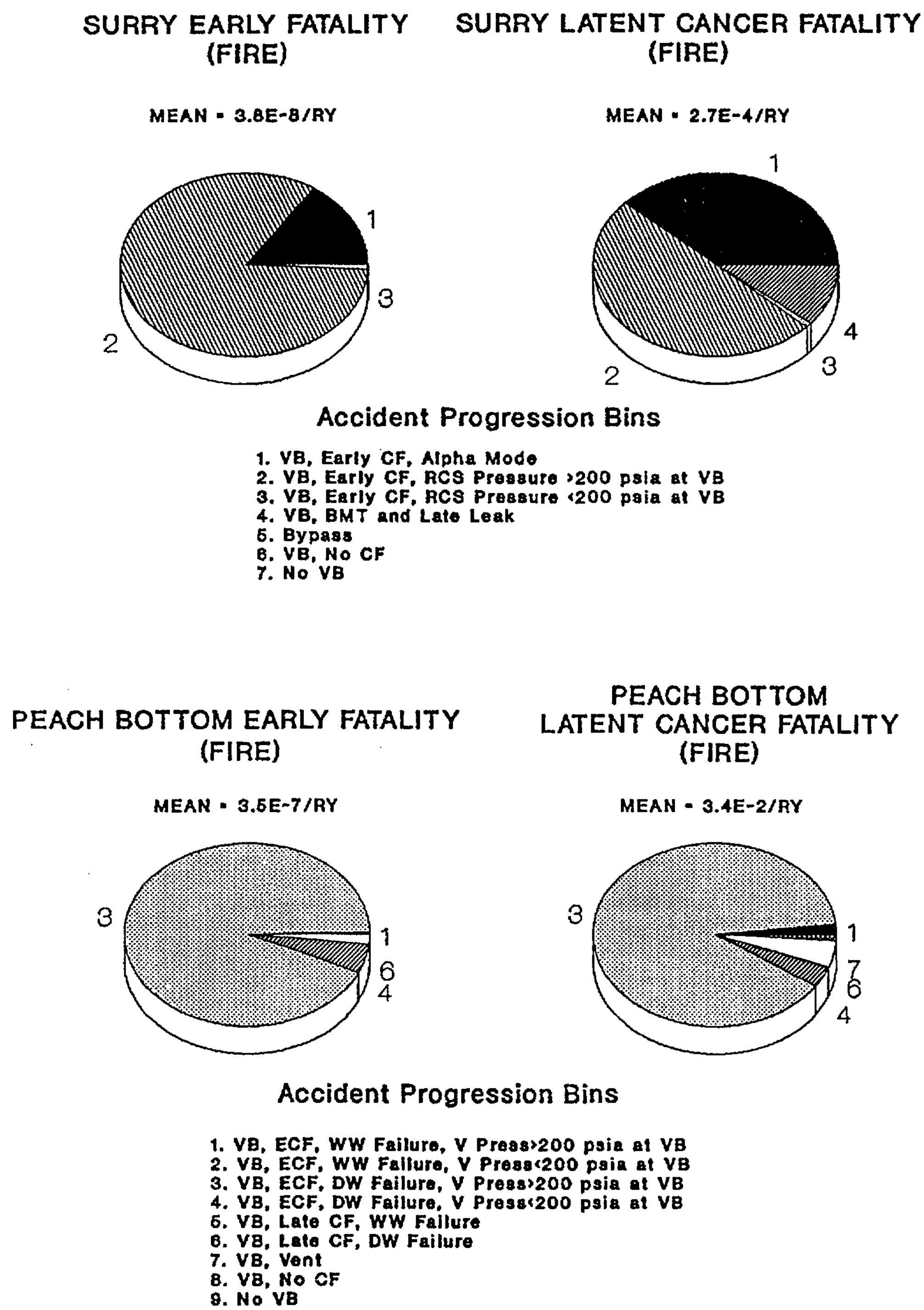


Figure 12.10 Contributions of accident progression bins to mean early and latent cancer fatality risks for Surry and Peach Bottom (fire-initiated accidents).

Accident Sequences Important to Risk

- Mean early fatality risks at Surry and Sequoyah and latent cancer fatality risk at Surry are dominated by bypass accidents (Event V and steam generator tube rupture accidents). Sequoyah latent cancer risk is dominated equally by loss of offsite power sequences and bypass accidents. The risk at Zion is dominated by medium LOCA sequences resulting from the failure of reactor coolant pump seals, induced by failures of the component cooling water system (CCWS) or service water system. Zion has the feature that CCWS (supported by the service water system) cools both the reactor coolant pump seals and high-pressure injection pump oil coolers, thus creating the potential for a common-mode failure. (As discussed in Chapter 7, steps have been taken by the plant licensee to address this dependency.)

- BWR risks are driven by events that fail a multitude of systems (i.e., reduce the redundancy through some common-mode or support system failure) or events that require a small number of systems to fail in order to get to core damage, such as ATWS sequences. The accidents important to both early fatality and latent cancer fatality risk at Peach Bottom are station blackouts and ATWS; the accident most important at Grand Gulf is station blackout.

- For the Peach Bottom plant, the estimated risks from accidents initiated by fires, while low, are greater than those from accidents initiated by internal events. Fire-initiated accidents are similar to station blackout accidents in terms of systems failed and accident progression. As such, the conditional probability of early containment failure is relatively high, principally due to the drywell shell melt-through failure mode (see Chapter 9 for additional discussion) (the conditional probability is somewhat higher because of the lower probability of ac power recovery). For the Surry plant, the fire risks are estimated to be smaller than those from internal events. This is because of two reasons: the frequency of core damage from fire initiators is lower; and fire-initiated accidents result in low conditional probabilities of early containment failure. As noted above, the internal-event risks are dominated by containment bypass accidents.

Containment Failure Issues Important to Risk

- At Surry, containment bypass events (interfacing-system LOCAs and steam generator tube ruptures) are assessed to be most important to risk. Other containment failure modes of less importance are: static failure at the containment spring line from loads at vessel breach (i.e., direct containment heating loads, hydrogen burns, ex-vessel steam explosion loads, and steam blowdown loads); or containment failure from in-vessel steam explosions (the "alpha-mode" failure of the Reactor Safety Study). These failure modes have only a small probability of resulting in early containment failure.

- At Zion, the conditional probability of early containment failure is small, comparable to that of Surry. Those containment failure modes that contribute to this small failure probability include alpha-mode failure, containment isolation failure, and overpressurization failure at vessel breach.

- In previous studies, the potential impact of direct containment heating loads was found to be very important to risk. In this study, the potential impact is less significant for the Surry and Zion plants. Reasons for this reduced importance include:

  - Temperature-induced and other depressurization mechanisms that reduce the probability of reactor vessel breach at high reactor coolant system pressure, either eliminating direct containment heating (DCH) or reducing the pressure rise at vessel breach. These depressurization mechanisms are stuck-open power-operated relief valves, reactor coolant pump seal failures, accident-induced hot leg and surge line failures, and deliberate opening of PORVs by operators; and

  - The size and the strength of the Surry containment (the maximum DCH load has only a conditional probability of 0.3 of failing the containment).

  Additional discussion of the issue of direct containment heating may be found in Section 9.4.3 and Section C.5 of Appendix C.

- At Sequoyah, containment bypass events are assessed to be most important to mean early fatality risk. Another failure important to early fatality risk is early failure of containment. In particular, the catastrophic rupture failure mode dominates early containment failures, which occur as a result of pre-vessel-breach hydrogen events and failures at vessel breach. The failures at vessel breach are the result of a variety of load sources (individually or in some combinations), including direct containment heating loads, hydrogen burns, direct contact of molten debris with the steel containment, alpha-mode failures, or loads from ex-vessel steam explosions. The bypass mode of containment failure and early containment failures dominate the mean latent cancer risk at Sequoyah and contribute about equally to this consequence measure.

- At Peach Bottom, drywell meltthrough is the most important mode of containment failure. Other containment failure modes of importance are: drywell overpressure failure, static failure of the wetwell (above as well as below the level of the suppression pool), and static failure at the drywell head.

- At Grand Gulf, the risk is most affected by containment failures in which both the drywell and the containment fail. As discussed in Chapter 9, roughly one-half the containment failures analyzed in this study also resulted in drywell failure. The principal causes of the combined failures were hydrogen combustion in the containment atmosphere and loads at reactor vessel breach (direct containment heating, ex-vessel steam explosions, or steam blowdown from the reactor vessel).

Source Term and Offsite Consequence Issues Important to Risk

- BWR suppression pools provide a significant benefit in severe accidents in that they effectively trap radioactive material (such as iodine and cesium) released early in the accident (before vessel breach) and, for some containment failure locations, after vessel breach as well.

- Accidents that bypass the containment structure compromise the many mitigative features of these structures and thus can have significant estimated radioactive releases. As noted above, such accidents dominated the risk for the Surry and Sequoyah plants.

- The design of the reactor cavity can significantly influence long-term releases of radioactive material; if large amounts of water can enter the cavity (e.g., as at Sequoyah), releases during core-concrete interactions can be significantly mitigated.

- Site parameters such as population density and evacuation speeds can have a significant effect on some risk measures (e.g., early fatality risk). Other risk measures, such as latent cancer fatality risk and individual early fatality risk, are less sensitive to such parameters. Latent cancer fatality risks are sensitive to the assumed level of interdiction of land and crops. (These issues are discussed in more detail below.)

Factors Important to Uncertainty in Risk

In order to identify the principal sources of uncertainties in the estimated risk, regression analyses have been performed for each of the plants in this study. A stepwise linear model is used, and, in general, the dependent variable is a risk measure (e.g., early fatalities per year) although some study has been done on the Surry plant using frequencies of radionuclide releases (discussed in Section 10.4.3). The independent variables consisted of individual parameters and groups of correlated parameters. Also, the analyses are generally performed for the complete risk model, although in some cases analyses are performed on specific plant damage states. The extent to which this model accounted for the overall uncertainty (the R-square value) varied considerably, from roughly 30 percent in the analysis of latent cancer fatality risk in the Sequoyah plant to roughly 75 percent in the analysis of early fatality risk in the Surry plant.

The results of the regression analyses indicate the following:

- For Surry, the uncertainty in all risk measures is dominated by the uncertainties in parameters determining the frequencies of containment bypass accidents (interfacing-system LOCA and steam generator tube rupture (SGTR)) and the radioactive release magnitudes of these accidents. More specifically, the most important parameters are the initiating event frequencies for these bypass accidents, the fraction of the core radionuclide inventory released into the vessel, and the fraction of material in the vessel in an SGTR-initiated core damage accident that is released to the environment. With the high risk importance of bypass accidents, it is not surprising that uncertainties in bypass accident parameters are important to risk uncertainty, while other parameters such as those relating to source terms in containment, containment strength, etc., are not found to be important.

- For Zion, the regression analyses also indicated that accident frequency and source term parameter uncertainties were most important. More specifically, the most important parameters were the initiating event frequencies for loss of component cooling water (CCW)/service water (SW), the failure to recover CCW/SW, the fraction of the core radionuclide inventory released into the vessel, the radionuclide containment transport fraction at vessel breach, and the fraction of radionuclides released to the environment through the steam generators. The importance of the loss of CCW/SW frequencies is not surprising, given the large contribution of accidents initiated by these events to the core damage frequency. Also, those source term parameters that influence the release fractions for early containment failure and bypass events are not surprisingly important to some risk measures. The only accident progression parameter that was demonstrated to be important to the uncertainty in risk was the probability of vessel and containment breach by an in-vessel steam explosion. This result occurs because the probability of early containment failure from all other causes is extremely low at Zion, so that (at these very low probability levels) uncertainty in the in-vessel steam explosion failure mode becomes more significant. The importance of the steam explosion failure mode is also more significant because the accident progression analysis for Zion indicates that the reactor coolant system (RCS) is not likely to be at high pressure when vessel breach occurs. This means that loads at vessel breach from direct containment heating are likely to be smaller than would have been the case if RCS pressure were high. Also, at low RCS pressure, the probability of triggering an in-vessel steam explosion is increased.

- For Sequoyah, the regression analysis for the complete risk model did not account for a large fraction of the uncertainty. As such, regression analyses were performed for individual plant damage states (PDSs). For the containment bypass PDSs (which dominated the mean risk at Sequoyah), the most important uncertainties related to accident frequency and source term issues. More specifically, for the interfacing-system LOCA PDS, the most important parameter uncertainties were those for the initiating event frequency, the probability that releases will be scrubbed by fire sprays in the vicinity of the break, and the decontamination factor of the fire sprays. For the SGTR-initiated core damage accident, the most important parameters are the initiating event frequency, the fraction of the core radionuclide inventory released into the vessel, and the fraction of material in the vessel that is released to the environment.

  For the station blackout, LOCA, and transient plant damage states, the uncertainty in early fatality risk is accounted for by parameters from the accident frequency, accident progression, and source term analysis, with none of these groups or any small set of parameters dominating. In this circumstance, the parameters relating to the containment failure pressure, the fraction of the core participating in a high-pressure melt ejection, and the pressure rise at vessel breach for low-pressure accident sequences appeared as somewhat important for each of these plant damage states (but, again, did not by themselves or in combination dominate the uncertainty estimation).

- For Peach Bottom, the regression analysis for the complete internal-event model indicated that the risk uncertainty is dominated by uncertainties in radioactive release uncertainties—more specifically, the dominating parameters relating to the fraction of the core radionuclide inventory released into the vessel before vessel breach, the fraction of the radionuclide inventory released during core-concrete interaction that is released from containment, and the fraction of the radionuclide inventory remaining in the core material at the initiation of core-concrete interaction that is released during that interaction.

  The regression analysis on the fire risk model does not show such a clear domination by any parameters. The early fatality risk uncertainty is dominated by radioactive release parameters (the fraction of core radionuclide inventory released to the vessel before vessel breach, the fraction of radionuclide inventory remaining in the core material at the initiation of core-concrete interaction that is released during that interaction, and the fraction of the radionuclide inventory released during core-concrete interaction that is released from containment). The latent cancer fatality risk uncertainty is dominated by accident frequency parameters (fire initiating event frequencies, diesel generator failure-to-run probability).

- For Grand Gulf, the uncertainty in early health effect parameters (early fatalities and individual early fatalities within 1 mile) is not dominated by any small set of parameters. Rather, it is accounted for by a number of parameters that determine the frequencies and radioactive release magnitudes of those events leading to early containment failure, such as the amount of hydrogen generated during the in-vessel portion of the accident progression, and the frequency of loss of offsite power. The uncertainties in the other risk measures are dominated by uncertainties in accident frequency parameters (including loss of offsite power frequency, diesel generator failure-to-start probability, diesel generator failure-to-run probability, and the probability that the batteries fail to deliver power when needed).

Impact of Emergency Response and Protective Action Guide Options

Sensitivity calculations were performed as a part of this study to assess the impacts of different emergency response and protective action guide options on estimates of risks for the five plants.

Emergency Response Options

In order to study the effects of emergency response options under severe accident conditions on public risk, the plants were analyzed using the following assumptions, and changes in the early fatality risk were calculated:

- Base Case: 99.5 percent evacuation from 0 to 10 miles
- Option 1: 100 percent evacuation from 0 to 10 miles
- Option 2: 0 percent evacuation with early relocation from high contamination areas
- Option 3: 100 percent sheltering
- Option 4: 100 percent evacuation from 0 to 5 miles and 100 percent sheltering from 5 to 10 miles

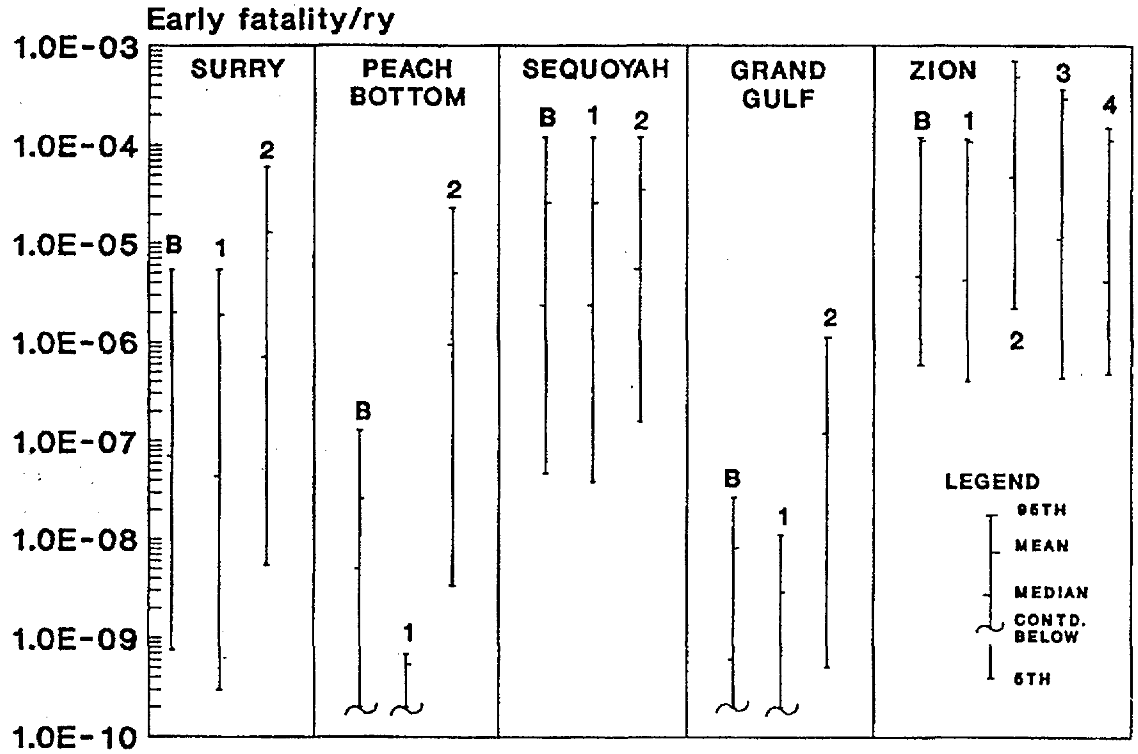
The last two options are used in the Zion plant analysis only. Results of the analyses are presented in Figure 12.11.

As discussed in Section 11.3, radionuclide release magnitudes associated with the early phase of an accident for Peach Bottom and Grand Gulf are typically smaller than those for the other three plants because of the mitigative effects of suppression pool scrubbing. The source term groups for Peach Bottom and Grand Gulf were typically found to have longer warning times than for the PWRs studied because the accident sequences developed more slowly. Further, Peach Bottom and Grand Gulf have very low surrounding population densities, which leads to shorter evacuation delays and higher evacuation speeds. The effect of all these considerations is that, for Peach Bottom and Grand Gulf, evacuation is more effective in reducing early fatality risk than for Surry, Sequoyah, and Zion.

For Surry and Sequoyah, the risk-dominant accident is the interfacing-system LOCA (the V sequence). This accident has a very short warning time, and, consequently, evacuation actions are not very effective. Also for Sequoyah, some high-consequence releases occur from containment failure at vessel breach; these releases are highly energetic and cause plume rise. This reduces early fatality risk, as is indicated in the case of Option 2 for Sequoyah; however, this also reduces the effectiveness of evacuation. Further details on emergency response options are provided in Chapter 11.

Protective Action Options

In this study an interdiction criterion of 4 rems (effective dose equivalent (EDE)) in 5 years has been used for groundshine and inhalation of resuspended radionuclides. Sensitivity calculations have been performed using the equivalent of the Reactor Safety Study (RSS) criterion, i.e., 25-rem EDE in 30 years. The impact of such an alternative criterion on mean latent cancer fatality risk is shown in Figure 12.12. As may be seen, the RSS criterion is less restrictive than the criterion used in this study, and the corresponding latent cancer fatalities using the RSS criterion are higher by 12 percent (for Grand Gulf) to 47 percent (for Peach Bottom).



BASE CASE (B)

99.5% Evacuation from 0 to 10 miles

EMERGENCY RESPONSE OPTIONS (1 TO 4)

1. 100% Evacuation from 0 to 10 miles
2. 0% Evacuation with early relocation from high contamination areas
3. 100% Sheltering
4. 100% Evacuation from 0 to 5 miles, and 100% sheltering from 5 to 10 miles

Note: As discussed in Reference 12.9, estimated risks at or below 1E–7 should be viewed with caution because of the potential impact of events not studied in the risk analyses.

Figure 12.11 Effects of emergency response assumptions on early fatality risks at all plants (internal events).

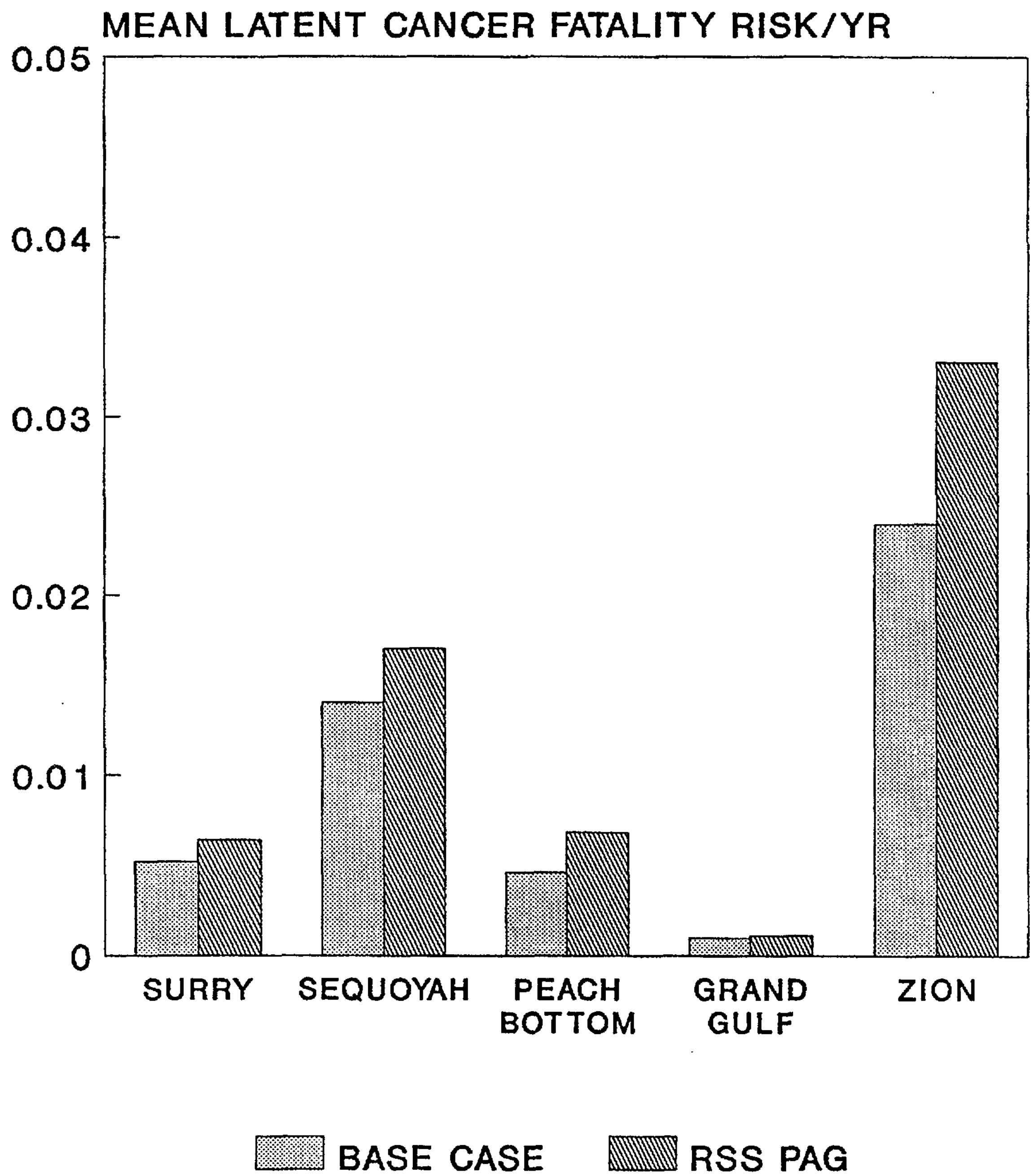


Figure 12.12 Effects of protective action assumptions on mean latent cancer fatality risks at all plants (internal events).

REFERENCES FOR CHAPTER 12

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

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

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

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

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

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

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

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

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

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

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

13. NUREG–1150 AS A RESOURCE DOCUMENT

13.1 Introduction

NUREG–1150 is one element of the NRC's program to address severe accident issues. The entire program was discussed in a staff document entitled "Integration Plan for Closure of Severe Accident Issues" (SECY–88–147) (Ref. 13.1). NUREG–1150 is used to provide a snapshot of the state of the art of probabilistic risk analysis (PRA) technology, incorporating improvements since the issuance of the Reactor Safety Study (Ref. 13.2). This chapter discusses the results of NUREG–1150 (and its supporting contractor studies, Refs. 13.3 through 13.16) as a resource document and examines the extent to which information provided in the document can be applied in regulatory activities. This is accomplished by applying NUREG–1150 results and principles to selected regulatory issues to illustrate how the information and insights described in Chapters 3 through 12 of this document can be used in the regulatory process. The discussion will concentrate on technical issues although it is recognized that there are other issues (e.g., legal, procedural) that must be taken into account when making regulatory decisions.

This report includes an examination of the severe accident frequencies and risks and their associated uncertainties for five licensed nuclear power plants and uses the latest source term information available from both the NRC and its contractors and the nuclear industry. The information in the report provides a valuable resource and insights to the various elements of the severe accident integration plan. The information provided and how it will be used include the following:

- Probabilistic models of the spectrum of possible accident sequences, containment events, and offsite consequences of severe accidents for use in:
  - Development of guidance for the individual plant examinations of internally and externally initiated accidents;
  - Accident management strategies;
  - Analysis of the need and appropriate means for improving containment performance under severe accident conditions;
  - Characterization of the importance of plant operational features and areas potentially requiring improvement;
  - Analysis of alternative safety goal implementation strategies; and
  - Emergency preparedness and consequences.
- Data on the major contributing factors to risk and the uncertainty in risk for use in:
  - Prioritization of research;
  - Prioritization of generic issues; and
  - Use of PRA in inspection.

In the following sections, these uses will be discussed in greater detail, using examples based on the risk analysis results discussed in previous chapters.

13.2 Probabilistic Models of Accident Sequences

NUREG–1150 identifies the dominant accident sequences and plant features contributing significantly to risk at a given plant as well as the plant models used in the study. The plant models and results underlying the report can be used to support the development of staff guidance on licensee-performed studies (individual plant examinations, accident management studies) and staff work in other areas related to severe accidents (e.g., improving containment performance under severe accident conditions). Such uses are discussed in greater detail in the following sections.

13.2.1 Guidance for Individual Plant Examinations

Plant-specific PRAs have yielded valuable perspectives on unique plant vulnerabilities. The NRC and the nuclear industry both have considerable experience with plant-specific PRAs. This experience, coupled with the interactions of NRC and the nuclear industry on severe accident issues, have resulted in the Commission's formulating an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributions (sometimes called "outliers") that might be plant specific and might be missed without a systematic approach. In November 1988, the NRC requested (by generic letter) that each licensed nuclear power plant perform an individual plant examination (IPE) to identify any plant-specific vulnerabilities to severe accidents (Ref. 13.17). The technical data generated in the course of preparing NUREG–1150 on severe accident frequencies, risks, and important uncertainties were used in developing the analysis requirements described in the IPE generic letter and the supplemental guidance on the IPE external-event analysis (Ref. 13.18).* These studies will also aid the staff in evaluating individual submittals, assessing the adequacy of the identification of plant-specific vulnerabilities by the licensee, and evaluating any associated potential plant modifications.

The extent to which NUREG–1150 results are applicable to different classes of reactors or to operating U.S. light-water reactors as a group is illustrated in Table 13.1. The generic insights presented in NUREG–1150 are indicative of items that may be applicable within a class of plants. This includes the identification of possible vulnerabilities that may exist in plants of similar design. These insights cannot be assumed to apply to a given plant without consideration of plant design and operational practices because of the design differences that exist in U.S. plants, particularly those involving ancillary support systems (e.g., ac power, component cooling water) for the engineered safety features and differences in details of containment design.

For some issues, the state of knowledge is very limited, and it is not possible to identify plant-specific features that may influence the issue because sensitivity analyses have not been performed. In other cases, the methodology is broadly applicable, but the results are highly plant specific. In spite of the plant-specific nature of many of the results, much can be learned from one plant that can be applied to another. Example types of generic applicability are presented in Table 13.1.

The NUREG–1150 methods refer not only to the analytical techniques employed but the general structure and framework upon which the analyses were conducted. These methods include the uncertainty analysis, expert elicitation methods, accident progression event tree analysis, and source term modeling. The general approaches adopted in these analysis procedures are not plant specific and are therefore adaptable to other plant analyses.

As noted above, plant-specific PRAs have yielded valuable perspectives on unique plant vulnerabilities. These perspectives are, in general, not directly applicable to other plants, although they provide useful information to the study of plants of similar NSSS (nuclear steam supply system) and containment design. At the present time, the principal contributors to the likelihood of a core damage accident at boiling water reactors (BWRs) include sequences related to station blackout or anticipated transients without scram (ATWS). Accident sequences making important contributions to the frequency of core damage accidents at pressurized water reactors (PWRs) include those initiated by a variety of electrical power system disturbances (loss of a single ac bus, which initiates a transient; loss of offsite portions of the equipment needed to respond to the transient; loss of offsite power; and complete station blackout), small loss-of-coolant accidents (LOCAs), loss of coolant support systems such as the component cooling water system, ATWS, and interfacing-system LOCAs or steam generator tube ruptures in which reactor coolant is released outside the containment boundary. All have the potential for being important at PWRs.

NUREG–1150 provides a wide spectrum of phenomenological and operational data (much of it of a very detailed nature). For example, information on hydrogen generation has been compiled from experimental and calculational results as well as interpretations of these data by experts. This data base provides an important source of information that may be used for NSSS containment types similar to those studied here but is somewhat less applicable for different NSSS containment types. The operational data base includes component failure rates, maintenance times, and initiating-event frequency data. Much of these data are generic in nature and thus applicable for selected classes of plants.

The analyses presented in Chapters 3 through 7, when combined with the information gained from earlier PRA work sponsored by both NRC (e.g., Ref. 13.19) and utilities, make it clear that the quantitative results (core damage frequencies and risk results) calculated for internal and external initiators cannot be considered applicable to another plant, even if the plant has a similar NSSS design and the same architect-engineer was involved in the design of the balance of plant.

*In addition, NUREG–1150 provides extensive and detailed analyses of five nuclear power plants and thus offers licensees of those plants an opportunity to use these studies in developing their IPEs and submitting them on an expedited basis.

Table 13.1 Utility of NUREG-1150 PRA process to other plant studies.

Example Results	Applicability: Class of Plants	Applicability: Plant Population
1. Methods (e.g., uncertainty, elicitation, event tree/fault tree)	high	high
2. General perspectives (e.g., principal contributors to core damage frequency and risk)	medium	low
3. Supporting data base on design features, operational characteristics, and phenomenology (e.g., hydrogen generation in core damage accidents, operational data)	high	medium
4. Quantitative results (e.g., core damage frequency, containment performance, risk)	low	low

Site-specific requirements and differing utility requirements often lead to significant differences in support system designs (e.g., ac power, dc power, service water) that can significantly influence the response of the plant to various potential accident-initiating events. Further, different operational practices, including maintenance activities and techniques for monitoring the operational reliability of components or systems can have a significant influence on the likelihood or severity of an accident.

13.2.2 Guidance for Accident Management Strategies

Certain preparatory and recovery measures can be taken by the plant operating and technical staff that could prevent or significantly mitigate the consequences of a severe accident. Broadly defined, such "accident management" includes the measures taken by the plant staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and finally (4) minimize the consequences of offsite releases. In addition, accident management includes certain measures taken before the occurrence of an event (e.g., improved training for severe accidents, hardware or procedure modifications) to facilitate implementation of accident management strategies. With all these factors taken together, accident management is viewed as an important means of achieving and maintaining a low risk from severe accidents.

Under the staff program, accident management programs will be developed and implemented by licensees. The NRC will focus on developing the regulatory framework under which the industry programs will be developed and implemented, as well as providing an independent assessment of licensee-proposed accident management capabilities and strategies. NUREG-1150 has been used by the NRC staff to support the development of the accident management program. NUREG-1150 methods provide a methodological framework that can be used to evaluate particular strategies, and the current results provide some insights into the efficacy of strategies in place or that might be considered at the NUREG-1150 plants. Thus, the NUREG-1150 methods and results will support a staff review of licensee accident management submittals.

PRA information has been used in the past to influence accident management strategies; however, the methods used in NUREG-1150 can bring added depth and breadth to the process, along with a detailed, explicit treatment of uncertainties. The integrated nature of the methods is particularly important, since actions taken during early parts of an accident can affect later accident progression and offsite consequences. For example, an accident management strategy at a BWR may involve opening a containment vent. This action can affect such things as the system response and core damage frequency, the retention of radioactive material within the containment, and the timing of radionuclide releases (which impacts evacuation strategies). It is possible that actions to reduce the core damage frequency can yield accident sequences of lower frequency but with much higher consequences. All these factors need to be considered in concert when developing appropriate venting strategies. The treatment of uncertainties is another key aspect of accident management. Generally, procedures are developed based on "most likely" or "expected" outcomes. For severe accidents, the outcomes are particularly uncertain. PRA models and results, such as those produced in the accident progression event trees, can identify possible alternative outcomes for important accident sequences. By making this information available to operators and response teams, unexpected events can be recognized when they occur, and a more flexible approach to severe accidents can be developed. The recent trend toward symptom-based, as opposed to event-based, procedures is consistent with this need for flexibility.

To demonstrate the potential benefits of an accident management program, some example calculations were performed, as documented in Reference 13.20. For this initial demonstration, these calculations were limited to the internal-event accident sequence portion of the analysis. Further, the numerical results presented are "point estimates" of the core damage frequency as opposed to mean frequency estimates. Selected examples from the initial analysis are presented below.

Effect of Firewater System at Grand Gulf

The first NUREG–1150 analysis of the Grand Gulf plant (Ref. 13.21) did not credit use of the firewater system for emergency coolant injection because of the unavailability of operating procedures for its use in this mode and the difficulties in physically configuring its operation. However, since that time, the licensee has made significant system and procedural modifications. As a result, the firewater system at Grand Gulf can now be used as a backup source of low-pressure coolant injection to the reactor vessel. The system would be used for long-term accident sequences, i.e., where makeup water was provided by other injection systems for several hours before their subsequent failure. The firewater system primarily aids the plant during station blackout conditions and is considered a last resort effort.

An examination has been made of the benefit of these licensee modifications to the Grand Gulf plant. As shown in Figure 13.1, these analyses showed that the total core damage frequency was reduced from 4E–6 to 2E–6 per reactor year because of these changes.

Effect of Feed and Bleed on Core Damage Frequency at Surry

The NUREG–1150 analysis for Surry includes the use of feed and bleed cooling for those sequences in which all feedwater to the steam generators is lost (thus causing their loss as heat removal systems). Feed and bleed cooling restores heat removal from the core using high-pressure injection (HPI) to inject into the reactor vessel and the power-operated relief valves (PORVs) on the pressurizer to release steam and regulate reactor coolant system pressure.

An examination has been made to determine to what extent feed and bleed cooling decreases core damage frequency at Surry. The current Surry model includes two basic events representing failure modes for feed and bleed cooling in the event of a loss of all feedwater. These modes are: operator failure to initiate high-pressure injection and operator failure to properly operate the PORVs. In order to examine the impact of feed and bleed cooling, both basic events were assumed to always occur. As shown in Figure 13.1, the resulting total core damage frequency for Surry (if feed and bleed cooling were not available) then increases by roughly a factor of 1.3. That is, the availability of the feed and bleed core cooling option in the Surry design and operation is estimated to reduce core damage frequency from 4E–5 to 3E–5 per reactor year.

Gas Turbine Generator Recovery Action at Surry

The present NUREG–1150 modeling and analysis of the Surry plant have not considered the benefits of using onsite gas turbine generators for recovery in the event of station blackout accidents. Both a 25 MW and a 16 MW gas turbine generator are available to provide emergency ac power to safety-related and non-safety-related equipment. These generators were not included in the analysis because, as currently configured, they would not be available to mitigate important accident sequences.

An examination has been made of the effect on core damage frequency at Surry of including the gas turbine generators as a means of recovery from station blackout sequences. To give credit for the addition of one generator for emergency ac power, it is assumed that Surry plant personnel have the authority to start the gas turbines when required and that 1 hour is required to start the gas turbines and energize the safety buses. In the analysis, the gas turbines were assumed to be available 90 percent of the time.

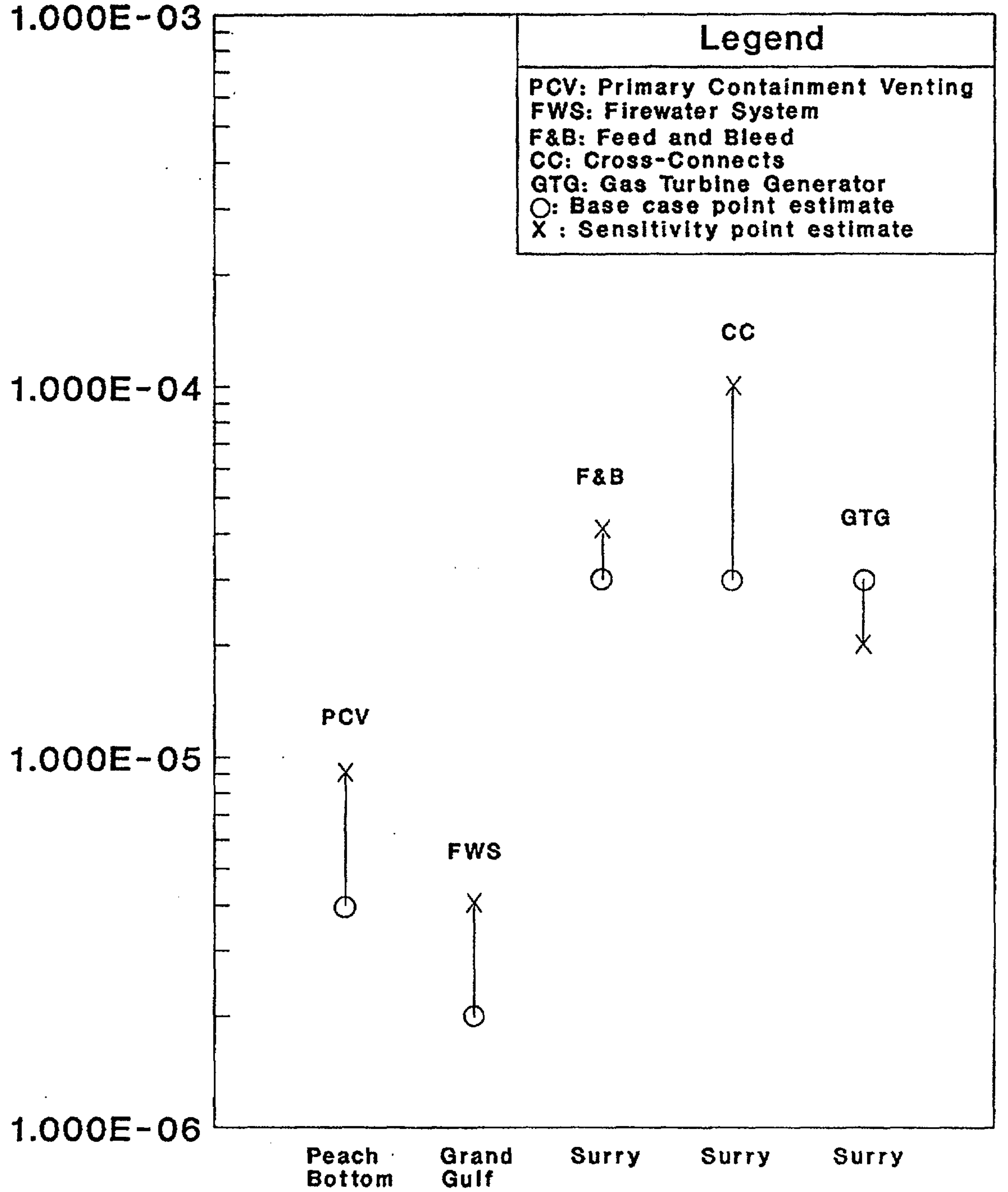


Figure 13.1 Benefits of accident management strategies.

The use of the onsite gas turbine was estimated to reduce core damage frequency from 3E-5 to 2E-5 per reactor year.

High-Pressure Injection and Auxiliary Feedwater Crossconnects at Surry

The Surry Unit 1 plant is configured to recover from loss of either the high-pressure injection (HPI) system or the auxiliary feedwater (AFW) system by operator-initiated crossconnection to the analogous system at Unit 2. While these actions provide added redundancy to these systems, new failure modes (e.g., flow diversion pathways) that were included in the modeling process for Surry have been created. The alignment of the Unit 1 and Unit 2 HPI and AFW systems for crossconnect injection is modeled as a recovery action.

Analysis of the importance of crossconnect injection at Surry includes two parts. First, credit for crossconnect injection was removed from all applicable dominant sequences, which were then requantified. Second, sequences that were previously screened out of the analysis were checked to determine if they would become dominant in the absence of crossconnect injection. As shown in Figure 13.1, the point estimate of the total core damage frequency without crossconnects is 1E-4, compared to the value of 3E-5 for internally initiated events in the base case.

Primary Containment Venting at Peach Bottom

The primary containment venting (PCV) system at Peach Bottom is used to prevent primary containment overpressurization during accident sequences in which all containment heat removal is lost. Most sequences of this type involve failure of the residual heat removal systems. Because of the existence of this venting capability, no such accident sequences appeared as dominant in the NUREG-1150 analysis for Peach Bottom.

The effect of the PCV system on the core damage frequency at Peach Bottom was determined by examining the sequences screened out in the NUREG-1150 analysis that included the PCV system as an event (primarily the sequences involving loss of containment heat removal). Credit for the PCV system was removed from these sequences, which were then summed and added to the current point estimate of the core damage frequency. As shown in Figure 13.1, this results in a point estimate of the Peach Bottom core damage frequency without containment venting of 9E-6, about a factor of 2.6 increase over the NUREG-1150 value of 4E-6.

13.2.3 Improving Containment Performance

The NRC has performed an assessment of the need to improve the capabilities of containment structures to withstand severe accidents (Ref. 13.1). Staff efforts focused initially on BWR plants with a Mark I containment, followed by the review of other containment types. This program was intended to examine potential enhanced plant and containment capabilities and procedures with regard to severe accident mitigation. NUREG-1150 provided information that served to focus attention on areas where potential containment performance improvements might be realized. NUREG-1150 as well as other recent risk studies indicate that BWR Mark I risk is dominated by station blackout and anticipated transient without scram (ATWS) accident sequences. NUREG-1150 further provided a model for and showed the benefit of a hardened vent for Peach Bottom (discussed above and displayed in Figure 13.1). The staff is currently pursuing regulatory actions to require hardened vents in all Mark I plants, using NUREG-1150 and other PRAs in the cost-benefit analysis.

The NUREG-1150 accident progression analysis models were used by the staff and its contractors in the evaluation of possible containment improvements for the PWR ice condenser and BWR Mark III designs. The result of the staff reviews of these designs (and all others except the Mark I) was that potential improvements would best be pursued as part of the individual plant examination process (discussed in Section 13.2.1).

13.2.4 Determining Important Plant Operational Features

NUREG-1150 will provide a source of information for investigating the importance of operational safety issues that may arise during day-to-day plant operations. The NUREG-1150 models, methods, and results have already been used to analyze the importance of venting of the suppression pool, the importance of keeping the PORVs and atmospheric dump valves unblocked, the importance of operational characteristics of the ice condenser containment design, the importance of operator recovery during an accident sequence, and the importance of crossties between systems. These operational and system characteristics, as well as many others, are described in detail in Chapters 3 through 7. For example, characteristics of the Surry plant design and operation that have been found to be important include crossties between units, diesel generators, reactor coolant pump seals, battery capacity, capability for feed and bleed core cooling, subatmospheric containment operation, post-accident heat removal system, and reactor cavity design.

13.2.5 Alternative Safety Goal Implementation Strategies

On August 21, 1986, the Commission published a Policy Statement on Safety Goals for the Operation of Nuclear Power Plants (Ref. 13.22). In this statement, the Commission established two qualitative safety goals supported by two risk-based quantitative objectives that deal with individual and societal risks posed by nuclear power plant operation. The objective of the policy statement was to establish goals that broadly define an acceptable level of radiological risk that might be imposed on the public as a result of nuclear power plant operation.

The Commission recognized that the safety goals could provide a useful tool by which the adequacy of regulations or regulatory decisions regarding changes to the regulations could be judged. Safety goals could be of benefit also in the much more difficult task of assessing whether existing plants that have been designed, constructed, and operated to comply with past and current regulations conform adequately with the intent of the safety goal policy.

The models and results of NUREG–1150 can be used in a number of ways in the NRC staff's analysis and implementation of safety goal policy. For example, the five plants studied for this report have been compared with the two quantitative health objectives, as shown in Figure 13.2 for internal initiators. Figure 13.3 compares Surry and Peach Bottom with the quantitative health objectives for fire initiators. As may be seen, the present risk estimates for these five plants (considering internally initiated accidents) and for the Surry and Peach Bottom plants (considering fire initiators) fall beneath the quantitative health objective risk goals. In addition, however, it may be seen that the risk estimates among the five plants vary considerably. An analysis of the plant design and operational differences that cause this variability can provide valuable information to the staff in its consideration of the balance of the present set of regulations and the areas of regulation that could most benefit from improvement.

The staff has reviewed the NUREG–1150 results at a broad level to determine the causes of the variability among plant risks shown in Figure 13.2. A number of design, operational, and siting factors are important to this measure of plant risk and determine the relative location of a specific plant's risk range in comparison with other plants and with the safety goal. At a general level, core damage frequency, containment and source term performance, and surrounding population demographics all can affect the risk range. Thus, using the Surry plant as an example, the combination of a relatively low core damage frequency, relatively good containment performance, and a low population density act to ensure with a high probability that the risk is below the safety goal.

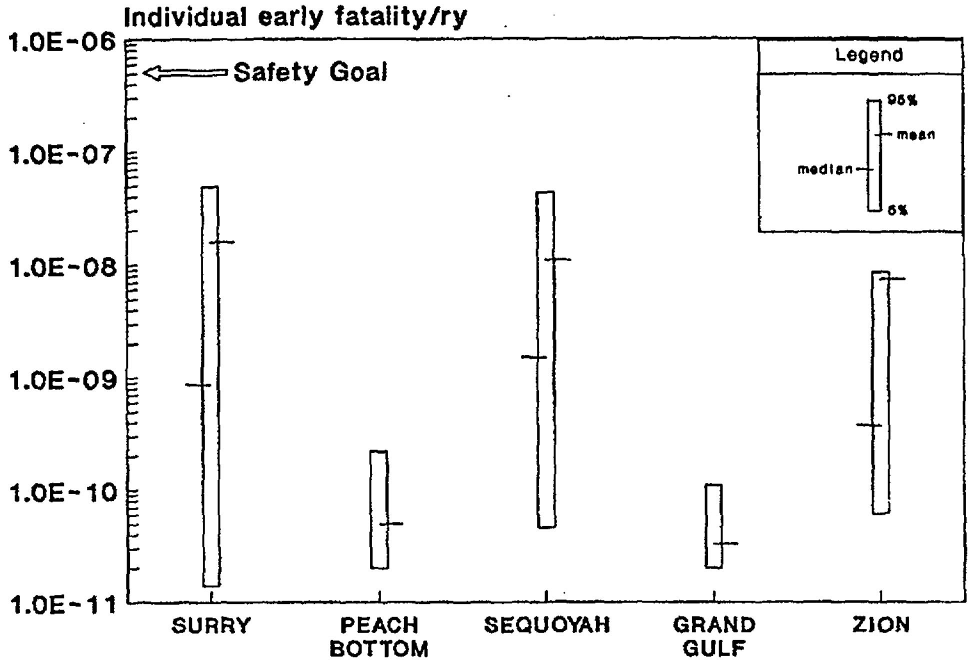
The NUREG–1150 results can also be used to support the analysis of alternative safety goal implementation approaches. One subject of discussion in the staff's work is the need for a supplemental definition of containment performance in severe accidents using the probability of a large release as a measure. An acceptable frequency for such a release was defined as 1E–6 per reactor year. A potential definition of a large release is one that can cause one or more early fatalities.* The present NUREG–1150 risk analyses have been evaluated to provide the frequency of such a release, as shown in Figure 13.4. The mean large release probabilities are below 1E–6 per reactor year. Further staff work in assessing alternative definitions is planned as part of the safety goal implementation program, and it is expected that NUREG–1150 methods and results will be used.

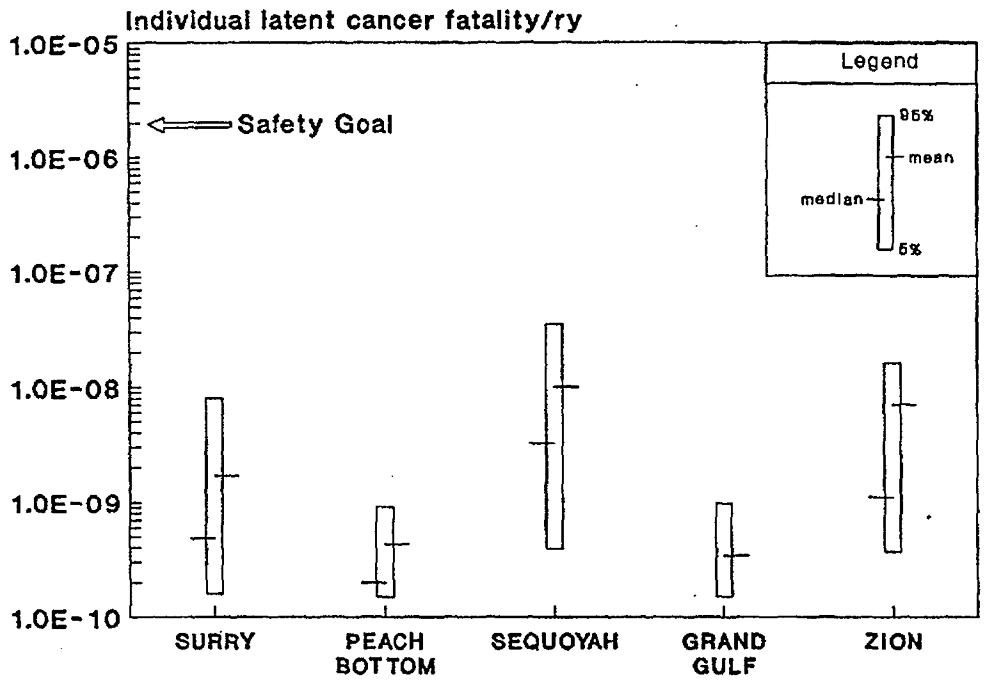
13.2.6 Effect of Emergency Preparedness on Consequence Estimates

NUREG–1150 provides information for developing protective action strategies that could be followed near a nuclear power plant in case of a severe accident. In developing strategies, consideration must be given to several types of protective actions, such as sheltering, evacuation, and relocation and various combinations. These strategies are influenced by the types of severe accidents that might occur at a nuclear power plant, their frequency of occurrence, and the radioactive release expected to result from each accident type as well as by the topography, weather, population density, and other site-specific characteristics.

NUREG–1150 provides assessments of a broad spectrum of potential core damage accidents that could occur at a nuclear power plant. These assessments permit the evaluation of hypothetical

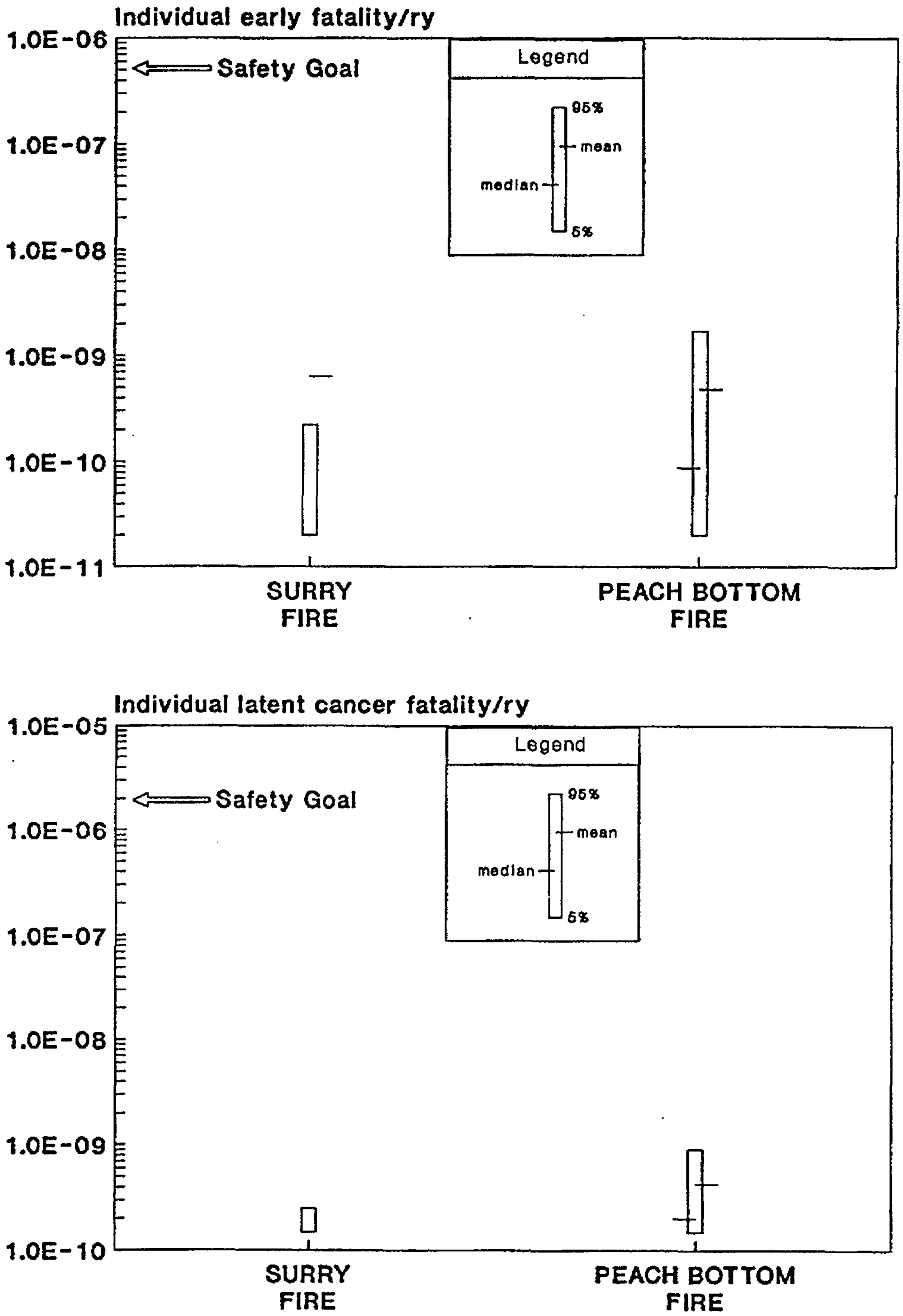
*The Commission has now indicated that this is not an appropriate definition and has asked the staff to review and propose an alternative definition.





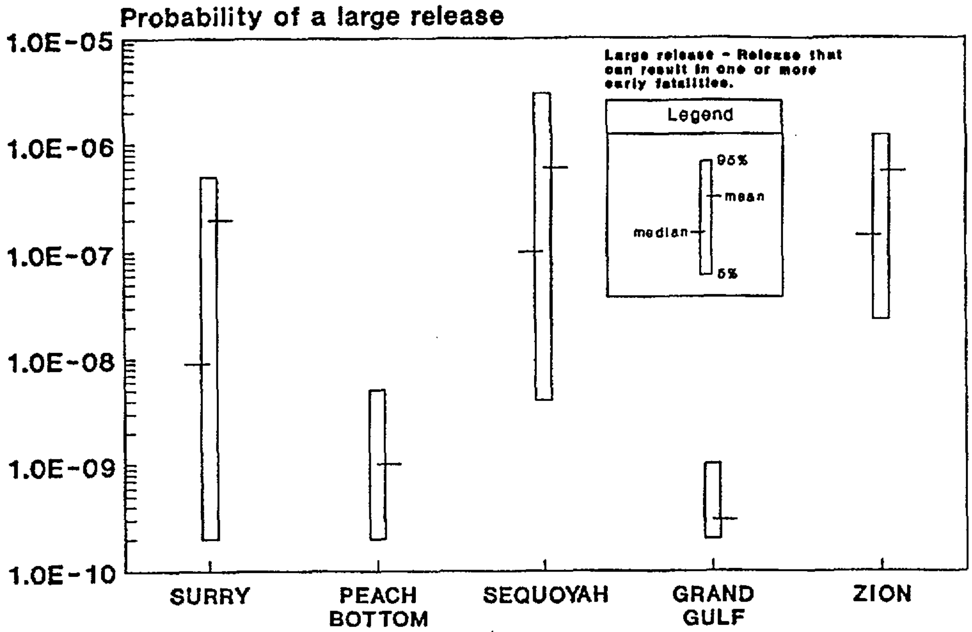
Note: As discussed in Reference 13.23, estimated risks at or below 1E–7 per reactor year should be viewed with caution because of the potential impact of events not studied in the risk analyses.

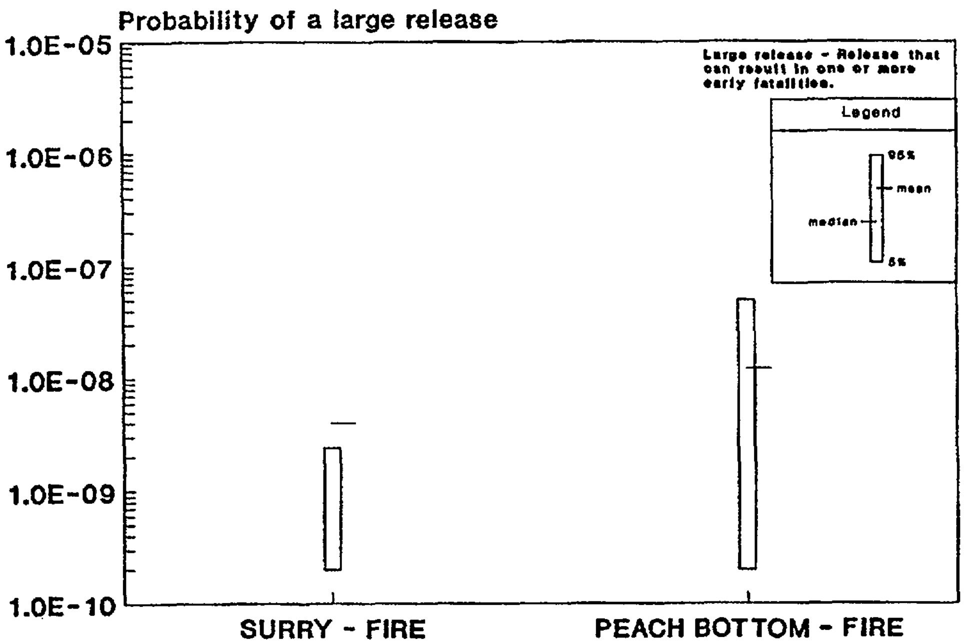
Figure 13.2 Comparison of individual early and latent cancer fatality risks at all plants (internal initiators).



Note: As discussed in Reference 13.23, 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 13.3 Comparison of individual early and latent cancer fatality risks at Surry and Peach Bottom (fire initiators).





Note: As discussed in Reference 13.23, 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 13.4 Frequency of one or more early fatalities.

dose savings for a spectrum of accidents and provide a means for evaluating potential reduction in early severe health effects (injuries and fatalities) in the event of an accident by implementing emergency response strategies.

The most important considerations in establishing emergency preparedness strategies are the warning times before release to initiate the emergency response and magnitude of the release of the radioactive material to the environment. The warning time and magnitude of radioactive release are in turn strongly influenced by the time and size of containment failure or bypass. If the containment fails early, the radioactive release is generally larger and more difficult to predict than if the containment fails late.

To evaluate the effectiveness of various protective actions, the conditional probabilities of acute red bone marrow doses exceeding 200 rems and 50 rems were calculated for several possible actions, using Zion plant source terms as examples. Doses were calculated on the plume centerline for various distances from the plant. The actions evaluated are:

- Normal activity—assumed that no protective actions were taken during the release but assumed that people were relocated within 6 hours of plume arrival.
- Home sheltering—sheltering in a single family home (see Table 11.5 for a definition of sheltering). The penetration fractions for groundshine and cloudshine were representative of masonry houses without basements as well as wood frame houses with basements. Indoor protection for inhalation of radionuclides was assumed. People were relocated from the shelter mode within 6 hours of plume arrival.
- Large building shelter—sheltering in a large building, for example, an office building, hospital, apartment building, or school. Indoor protection for inhalation of radionuclides was assumed. People were relocated from the shelter mode within 6 hours of plume arrival.
- Evacuation—doses were calculated for people starting to travel at the time of release, 1 hour before start of release, and 1 hour after start of release. An evacuation speed of 2.5 mph was assumed.

Figure 13.5 shows the conditional probabilities of exceeding a 50-rem and a 200-rem red bone marrow dose for the various possible response modes assuming an early containment failure at Zion with source term magnitudes varying from low to high. Figure 13.6 shows similar results for a late containment failure at Zion.

Use of the above assumptions indicates that if a large release occurs (Fig. 13.5), there is a large probability of doses exceeding 200 rems within 1 to 2 miles from the reactor. Sheltering does not significantly lower this probability. Thus, if a large release can occur, it is prudent to consider prompt evacuation prior to the start of the release.

At 3 miles and beyond, it is possible to avoid doses exceeding 200 rems by sheltering in large buildings even if a large release were to occur. Thus, people in large buildings such as hospitals would not necessarily have to be immediately evacuated, but could shelter instead. Of course, further reductions in dose are possible by evacuation.

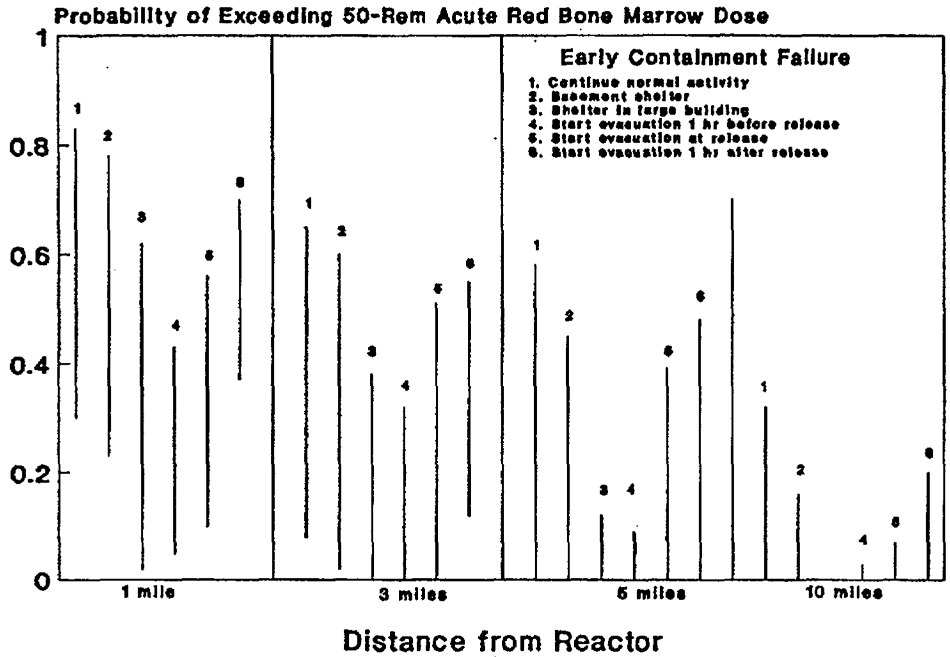
At 10 miles, no protective actions except relocation would be necessary to avoid 200-rem doses. Sheltering in large buildings or evacuation prior to release would probably keep doses below 50 rems.

13.3 Major Factors Contributing to Risk

NUREG-1150 results can be used to identify dominant plant risk contributors and associated uncertainties. A discussion of these dominant risk contributors is found in Chapters 3 through 8 and Chapter 12. This section focuses on the use in guiding research, generic issue resolution, and inspection programs.

Because of its integrated nature, discussion of uncertainties, and reliance on more realistic assessments, PRA-based information found in NUREG-1150 and its supporting documents can be used to guide and focus a wide spectrum of activities designed to improve the state of knowledge regarding the safety of individual nuclear power plants, as well as that of the nuclear industry as a whole. The resources of both the NRC and the industry are limited, and the application of PRA techniques and subsequent insights provides an important tool to aid the decisionmaker in effectively allocating these resources.

The nature of the many decisions necessary to allocate regulatory resources does not require great precision in PRA results. For example, in assigning priorities to research or efforts to resolve generic safety issues, it is sufficient to use broad



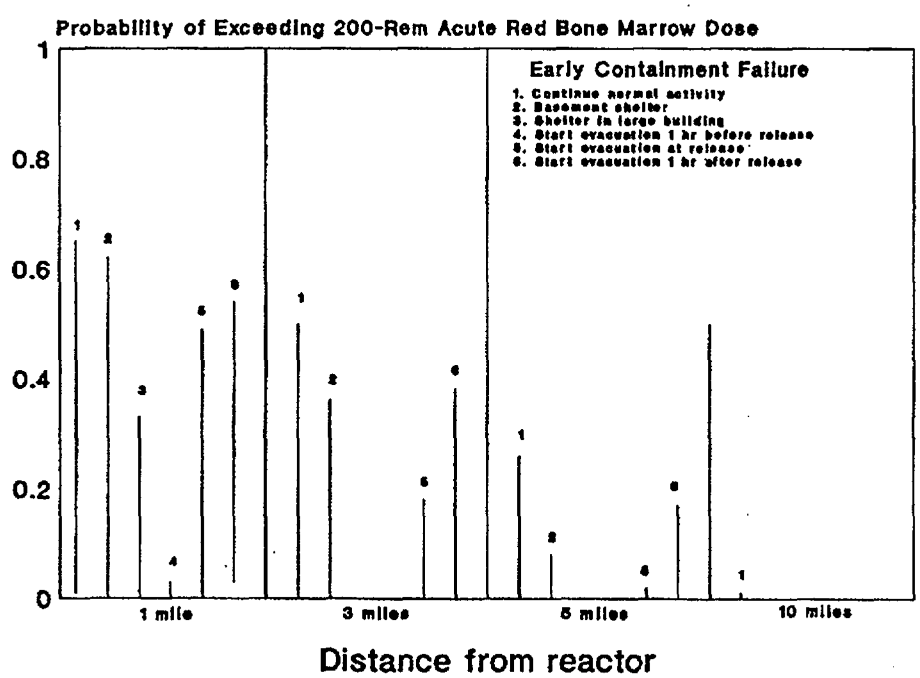
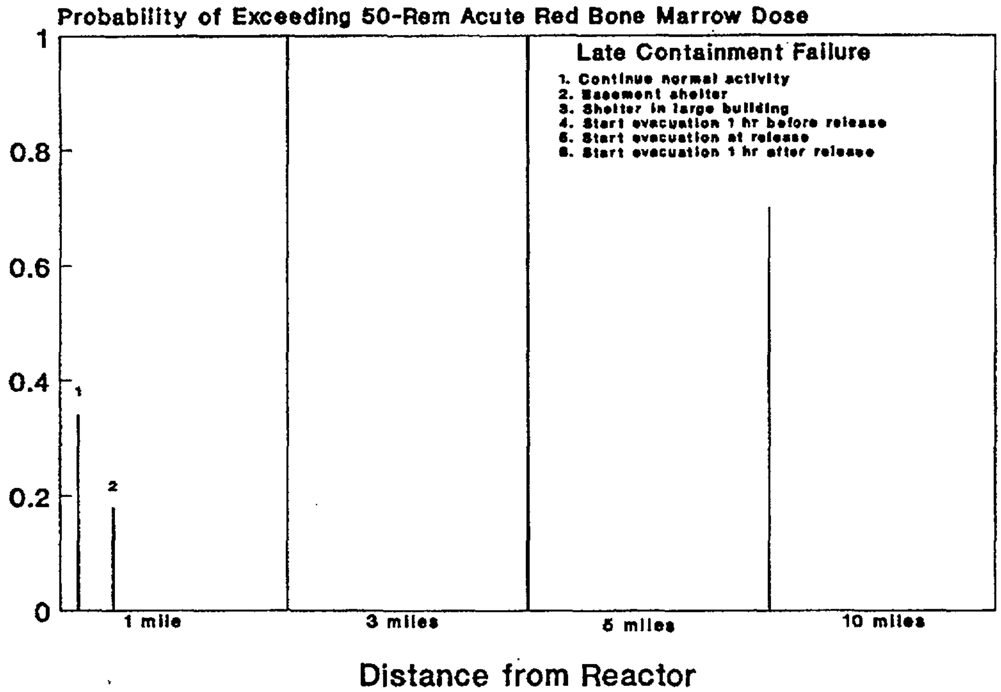


Figure 13.5 Relative effectiveness of emergency response actions assuming early containment failure with high and low source terms.



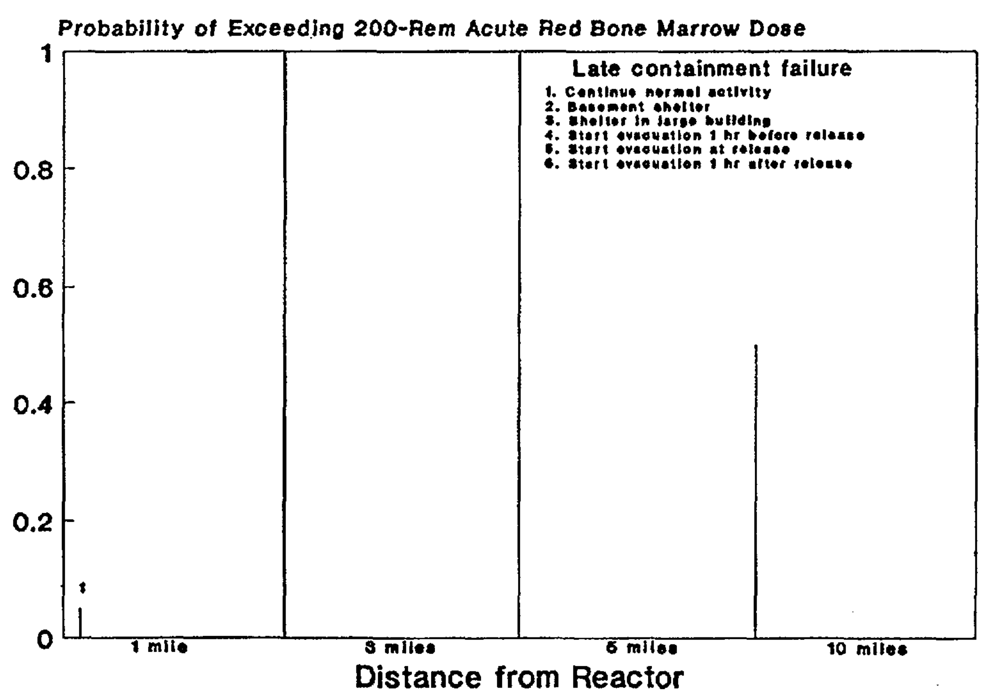


Figure 13.6 Relative effectiveness of emergency response actions assuming late containment failure with high and low source terms.

categories of risk impact (e.g., high, medium, and low) (Ref. 13.24). In a similar manner, information from PRAs can be used to guide the allocation of resources in inspection and enforcement programs (see Section 13.3.3).

13.3.1 Reactor Research

As noted earlier, the nature of the decisions necessary to allocate resources does not require great precision in PRA results. In prioritizing research efforts, it is sufficient to use broad categories of risk impact (e.g., high, medium, and low). A given issue can be evaluated in terms of the number of plants affected, the risk impacts on each plant, the effect of modifications in reducing the risk, and the effect of additional knowledge on improving the prediction of plant risk or severe core damage frequency or on reducing or defining more clearly the associated uncertainties. These generic measures of significance, combined appropriately with other information (e.g., cost of resolving the issue) can be used to evaluate the issue under consideration.

13.3.2 Prioritization of Generic Issues

The NRC has been setting priorities for generic safety issues for several years using PRA as one informational input (Ref. 13.25). In prioritizing efforts to resolve generic safety issues, it is sufficient to use broad categories of risk impact (e.g., high, medium, and low) in which only order-of-magnitude variations are considered important. The reasoning is that a potential safety issue would not be dismissed unless it were clearly of low risk. Thus, one or more completed PRA studies can often be selected as surrogates for the purpose of assigning such priorities, even though they clearly do not fully represent the characteristics of some plants, provided the nature of the difference is reasonably understood and can be qualitatively evaluated.

As with any priority-assignment method, the final results must be tempered with an engineering evaluation of the reasonableness of the assignment, and the PRA-based analysis can serve as only one ingredient of the overall decision.

One of the most important benefits of using PRA as an aid to assigning priorities is the documentation of a comprehensive and disciplined analysis of the issue, which enhances debate on the merits of specific aspects of the issue and reduces reliance on more subjective judgments. Clearly, some issues would be very difficult to quantify with reasonable accuracy, and the assignment of priorities to these issues would have to be based largely on subjective judgment.

PRA is being usefully applied to setting priorities for generic safety issues and to evaluating new issues as they are identified. In this effort, each issue is assessed as to its nature, its probable core damage frequency and public risk, and the cost of one or more conceptual fixes that could resolve the issue. A matrix is developed whereby each issue is characterized as of high, medium, or low probability, or whether the issue should be summarily dropped from further regulatory consideration. This matrix considers both the absolute magnitude of the core damage frequency or risk and the value/impact ratio of conceptual fixes. Risk-reduction estimates are normally made using surrogate PWRs and BWRs, based on existing PRAs.

A principal benefit of PRA-based prioritization, compared to other methods for allocating resources to safety issues, is that important assumptions made in quantifying the risk are displayed and uncertainties in the analyses are estimated. A principal limitation is that some of the issues, such as those dealing with human factors, are only subjectively quantified. Thus, the uncertainties can be large. However, on balance, PRA-based prioritization has been found to be quite useful. Although uncertainties may be large, the process forces attention on these uncertainties to a much higher degree than if the quantification were not attempted. Also, the uncertainties are normally part of the issues themselves and not just an artifact of the PRA analysis.

Since, as discussed above, the prioritization is done on an approximate (order-of-magnitude) basis, the new information developed in NUREG-1150 is not expected to substantially change previously developed priority rankings. However, a sample of key issues will be reexamined to determine whether, based on the updated information in NUREG-1150, changes in dominant accident sequences or performance of mitigative systems could substantially affect the previous rankings.

13.3.3 Use of PRA in Inspections

The importance to NRC of risk-based inspection data is exemplified by the following statement in NRC's 5-Year Plan: "Probabilistic risk assessment techniques will be applied to all phases of the inspection program in order to insure that inspection activities are prioritized and conducted in an integrated fashion." Within NRC, the Risk Applications Branch of the Office of Nuclear Reactor Regulation has the responsibility of directly providing risk-based information to the regional offices and resident inspectors. This ongoing effort has resulted in the development of plant-specific, and in some cases generic, PRA perspectives that help to provide an optimization of inspection resources and a prioritization of inspection resources on the high-risk aspects of a plant. Using draft NUREG–1150 data, team inspection procedures based on plant-specific PRA information have been developed and implemented on such plants as Grand Gulf. Formalization of these inspection activities can be found in a recently issued inspection module entitled "Risk Focused Operation Readiness Inspection Procedures." This module focuses on how to use PRA perspectives and conduct a risk-based team inspection based on risk insights. The spectrum of reactor plant design types addressed in NUREG–1150 provide a broad risk data base that in many instances can be used to assist in inspection-type decisions even for plants without a PRA.

REFERENCES FOR CHAPTER 13

13.1 U.S. Nuclear Regulatory Commission (USNRC), "Integration Plan for Closure of Severe Accident Issues," SECY-88-147, May 25, 1988.

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

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

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

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

13.6 A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

13.7 R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

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

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

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

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

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

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

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

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

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

13.17 USNRC, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR §50.54(f)," Generic Letter 88-20, November 23, 1988.

13.18 USNRC, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, Draft Report for Comment, July 1990.

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

13.19 USNRC, "Probabilistic Risk Assessment Reference Document," NUREG-1050, September 1984.

13.20 A. L. Camp et al., "The Risk Management Implications of NUREG-1150 Methods and Results," Sandia National Laboratories, NUREG/CR-5263, SAND88-3100, September 1989.

13.21 M. T. Drouin et al., "Analysis of Core Damage Frequency from Internal Events: Grand Gulf, Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 6, SAND86-2084, April 1987.

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

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

13.24 USNRC, "Categorization of Reactor Safety Issues from a Risk Perspective," NUREG-1115, March 1985.

13.25 R. Emrit et al., "A Prioritization of Generic Safety Issues," USNRC Report NUREG-0933, July 1984.

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1150
Vol. 1

2. TITLE AND SUBTITLE

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

Final Summary Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1990

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Final Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

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 which provide the detailed discussion of the methods used and results obtained in these risk studies.

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.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

severe accidents
risk analysis
probabilistic risk analysis

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified
(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

NUREG–1150
Vol. 2

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

Appendices A, B, and C

Final Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW, Lower Level, Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the Code of Federal Regulations, and Nuclear Regulatory Commission Issuances.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. Federal Register notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Information Resources Management, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG–1150
Vol. 2

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

Appendices A, B, and C

Final Report

Manuscript Completed: October 1990
Date Published: December 1990

Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ABSTRACT

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

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

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

CONTENTS

Page

APPENDIX B—AN EXAMPLE RISK CALCULATION

APPENDIX C—ISSUES IMPORTANT TO QUANTIFICATION OF RISK

FIGURES

TABLES

APPENDIX A

RISK ANALYSIS METHODS

CONTENTS

FIGURES

TABLE

A.1 Introduction and Overview

A.1.1 Introduction

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

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

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

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

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

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

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

A.1.2 Overview of Risk Analysis Process*

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

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

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

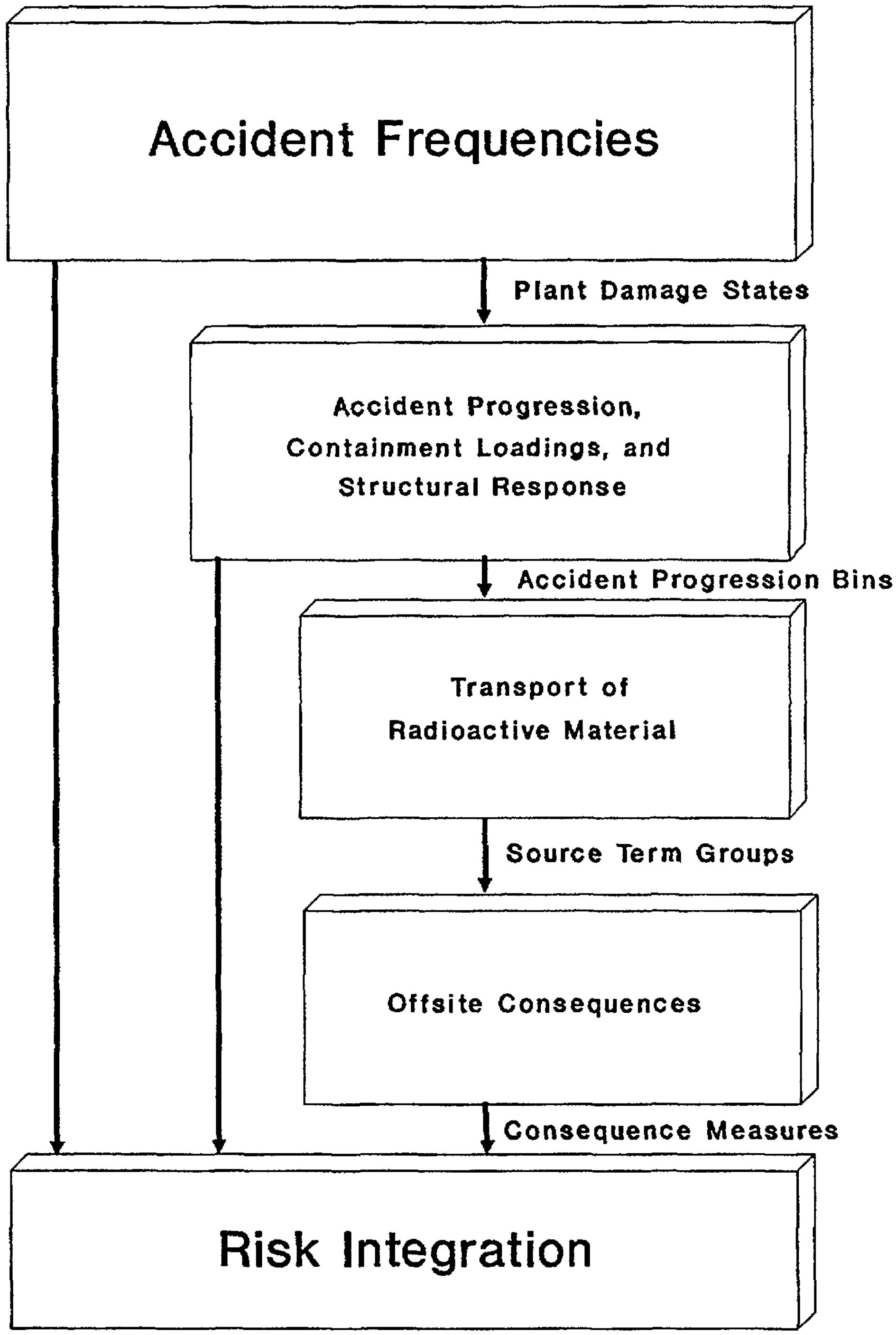


Figure A.1 Principal steps in NUREG-1150 risk analysis process.

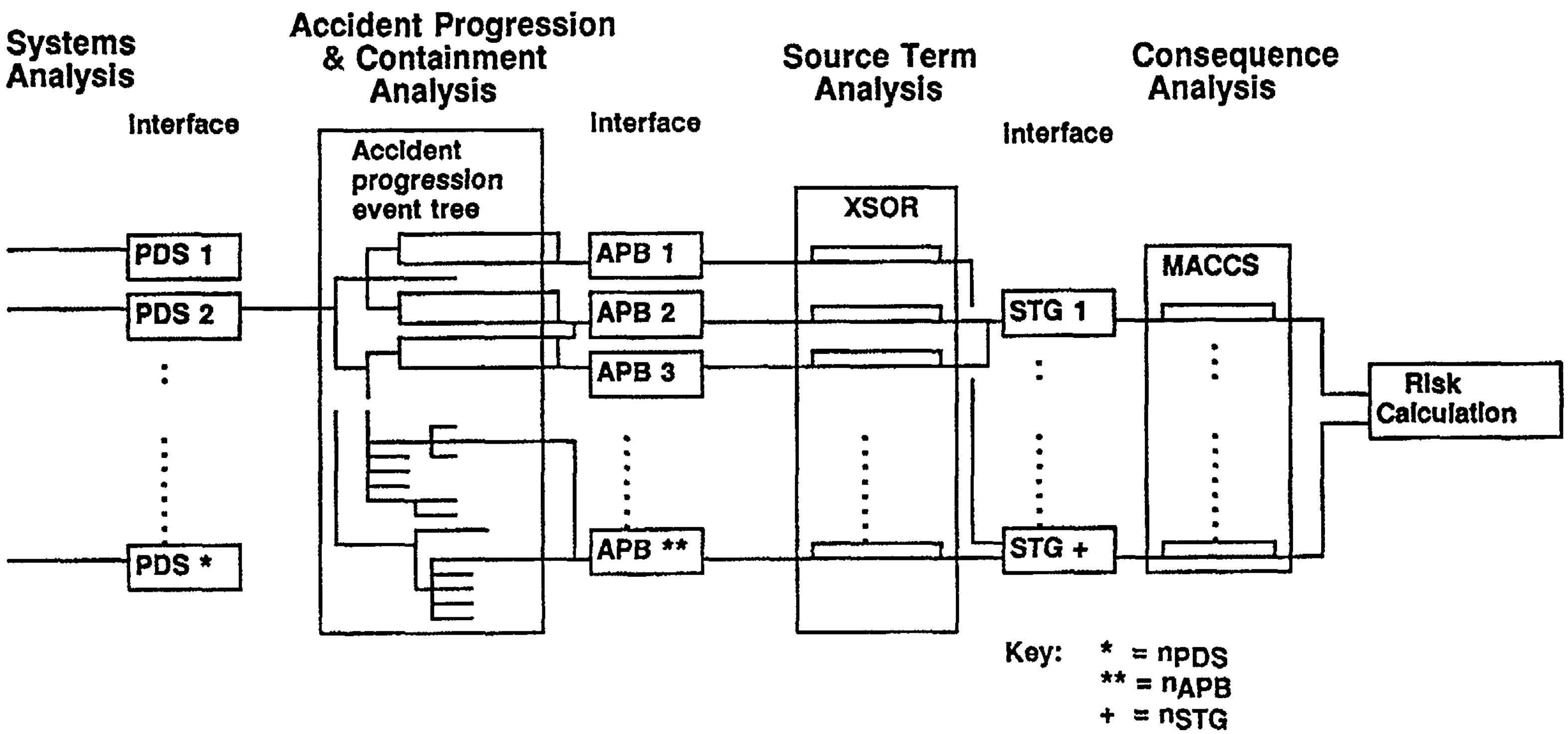


Figure A.2 Interfaces between risk analysis steps.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

where:

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

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

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

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

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

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

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

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

A.2 Accident Frequency Analysis Methods

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

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

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

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

Each of these steps will be discussed below.

Plant Familiarization Analysis

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

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

Accident Sequence Initiating Event Analysis

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

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

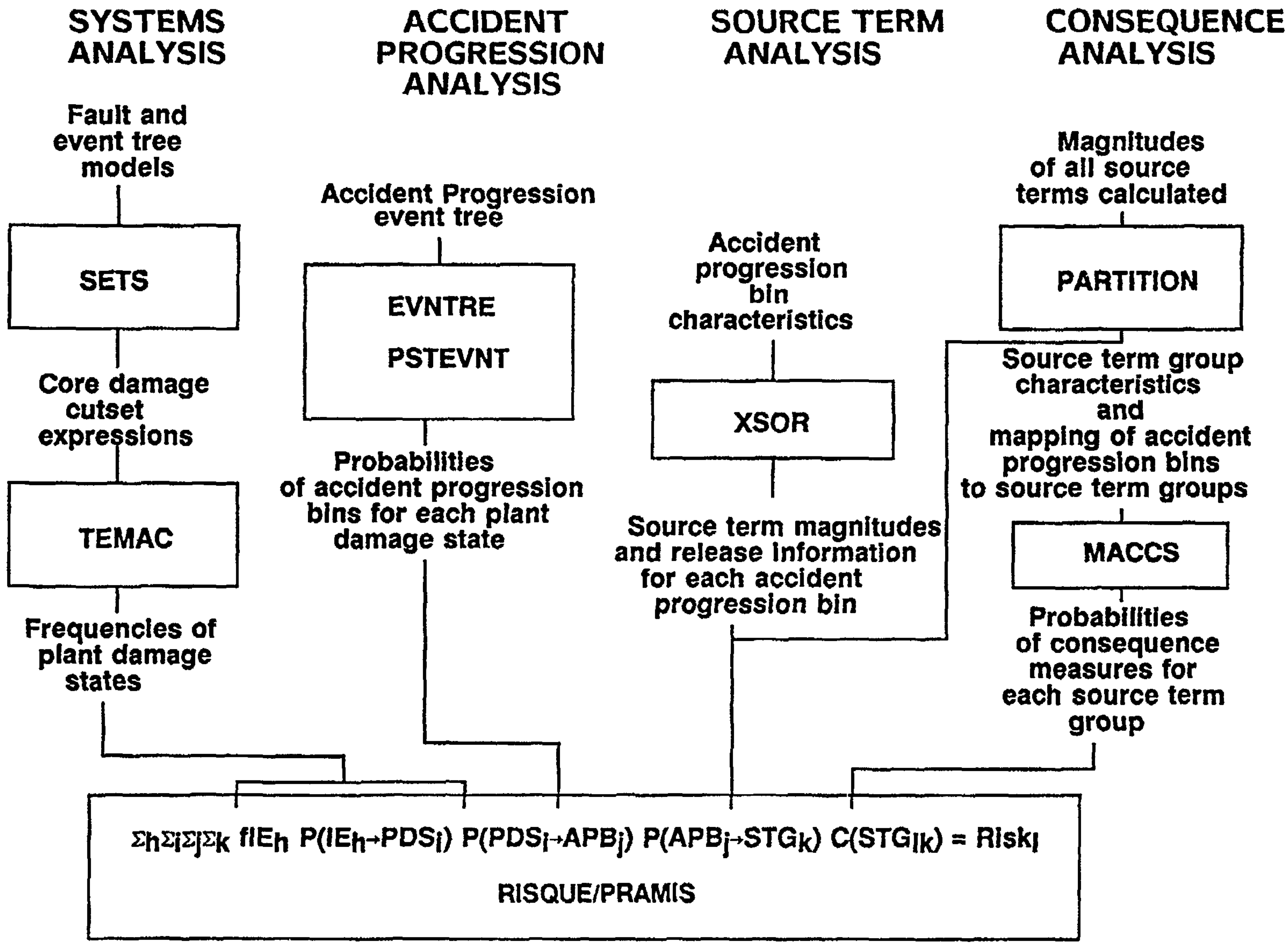


Figure A.3 Models used in calculation of risk.

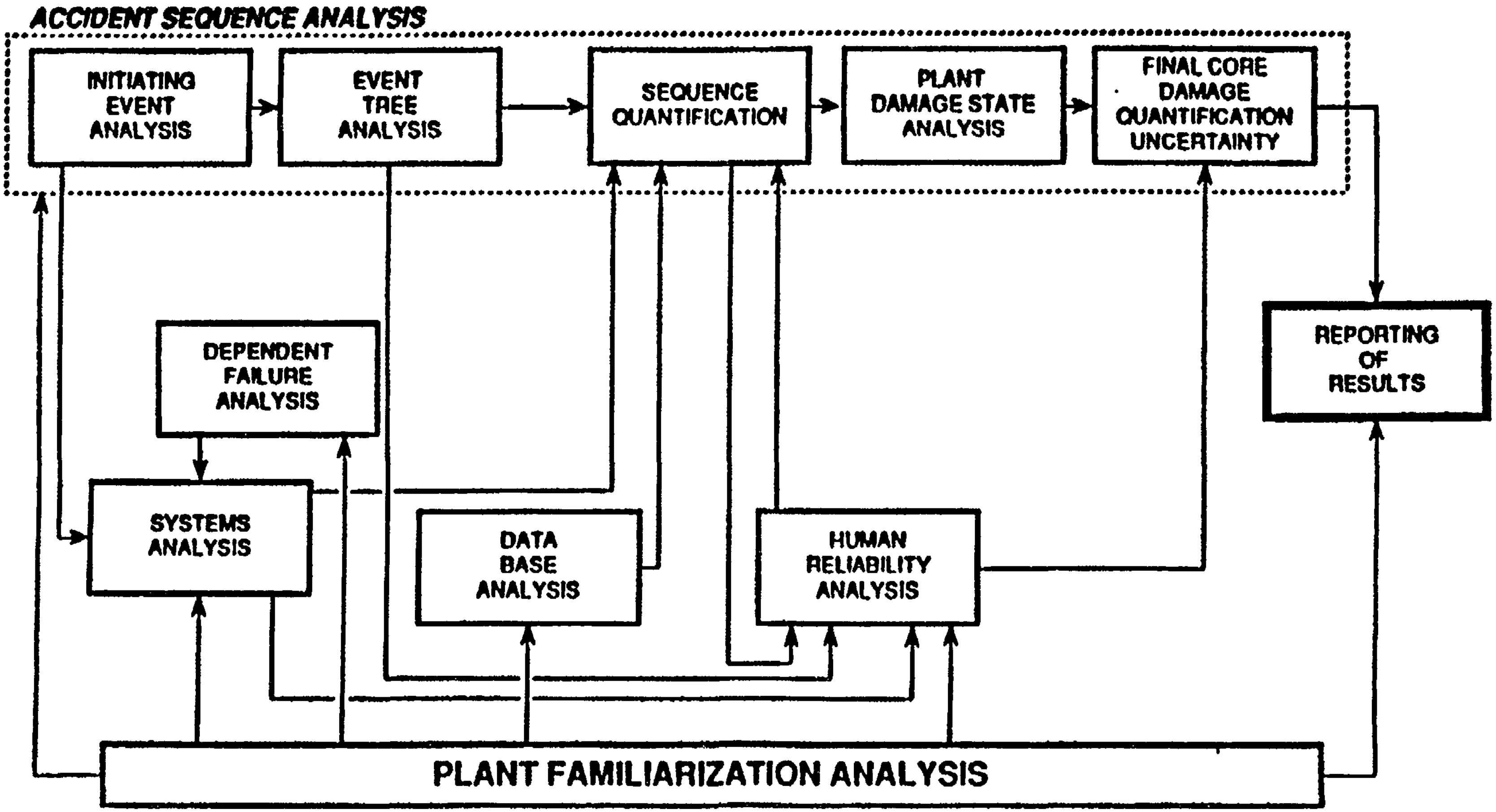


Figure A.4 Steps in accident frequency analysis of Surry, Sequoyah, Peach Bottom, and Grand Gulf.

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

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

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

Accident Sequence Event Tree Analysis

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

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

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

Systems Analysis

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

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

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

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

Dependent and Subtle Failure Analysis

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

The dependent failures included:

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

The subtle failures included:

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

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

Human Reliability Analysis

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

Data Base Analysis

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

Accident Sequence Quantification Analysis

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

Plant Damage State Analysis

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

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

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

Uncertainty Analysis

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

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

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

A.2.2 Internal-Event Methods for Zion*

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

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

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

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

Identification of Initiating Events—Zion

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

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

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

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

Plant Response Modeling—Zion

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

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

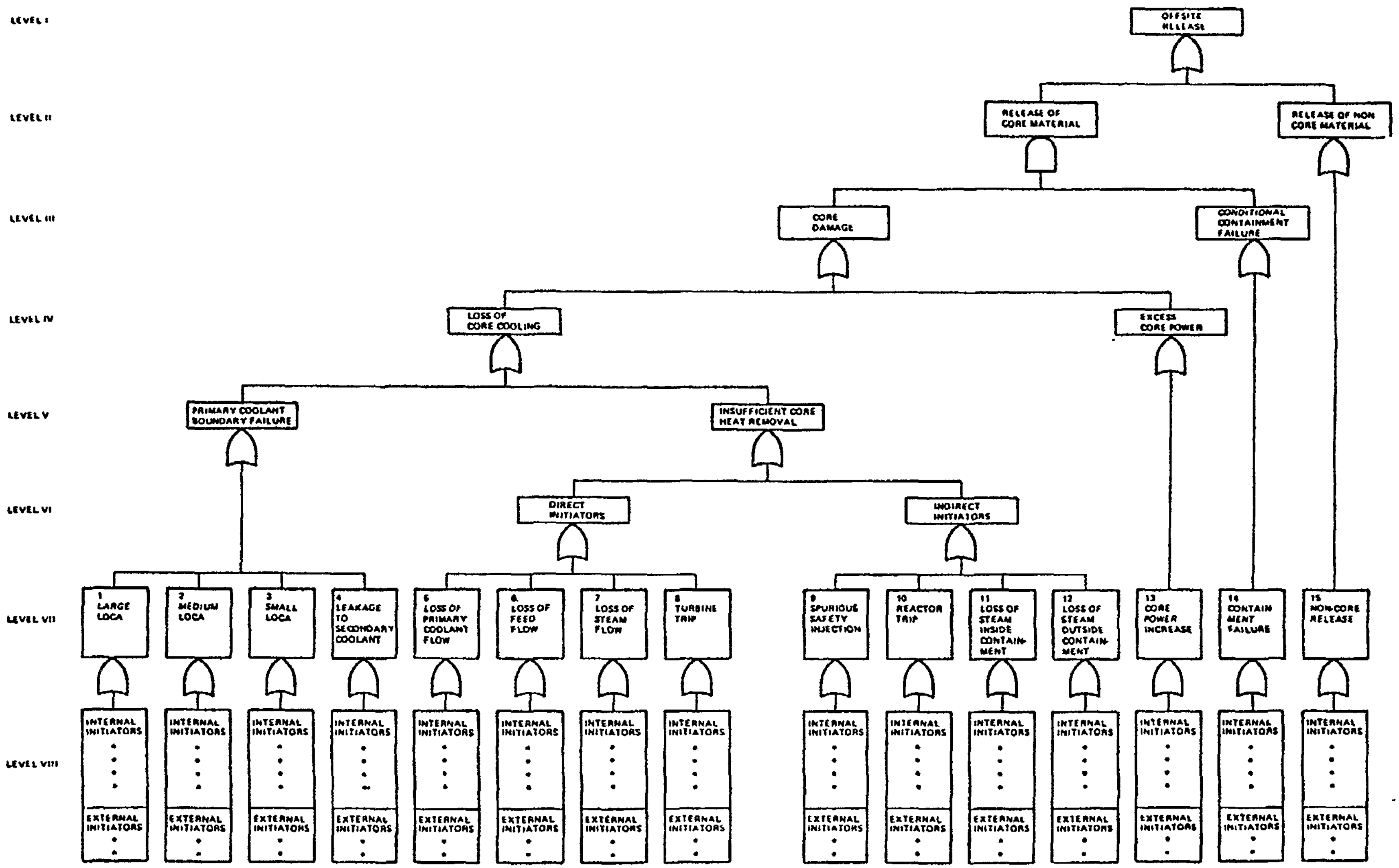


Figure A.5 Zion Probabilistic Safety Study master logic diagram.

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

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

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

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

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

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

Human Reliability Analysis—Zion

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

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

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

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

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

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

Data Analysis—Zion

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

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

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

Quantification—Zion

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

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

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

Sensitivity/Uncertainty Analyses—Zion

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

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

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

Seismic Accident Frequency Analysis Methods

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

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

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

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

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

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

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

Work performed in each of these steps is summarized below.

Determination of Local Earthquake Hazard

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

Identification of Accident Sequences

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

Determination of Failure Modes

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

Determination of Fragilities

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

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

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

Determination of Responses

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

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

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

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

Computation of Frequency of Core Damage

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

Estimation of Uncertainty

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

Fire Accident Frequency Analysis Methods

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

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

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

Each of these steps is summarized below.

Initial Plant Visit

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

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

Screening of Potential Fire Locations

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

The screening analysis was comprised of:

- Identification of relevant fire zones

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

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

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

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

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

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

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

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

Accident Sequence Quantification

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

- Determination of the temperature response in each fire zone

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

- Computation of component fire fragilities

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

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

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

- Performance of recovery analyses

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

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

Bounding Analysis of Other External Events

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

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

A.2.4 Products of Accident Frequency Analysis

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

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

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

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

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

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

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

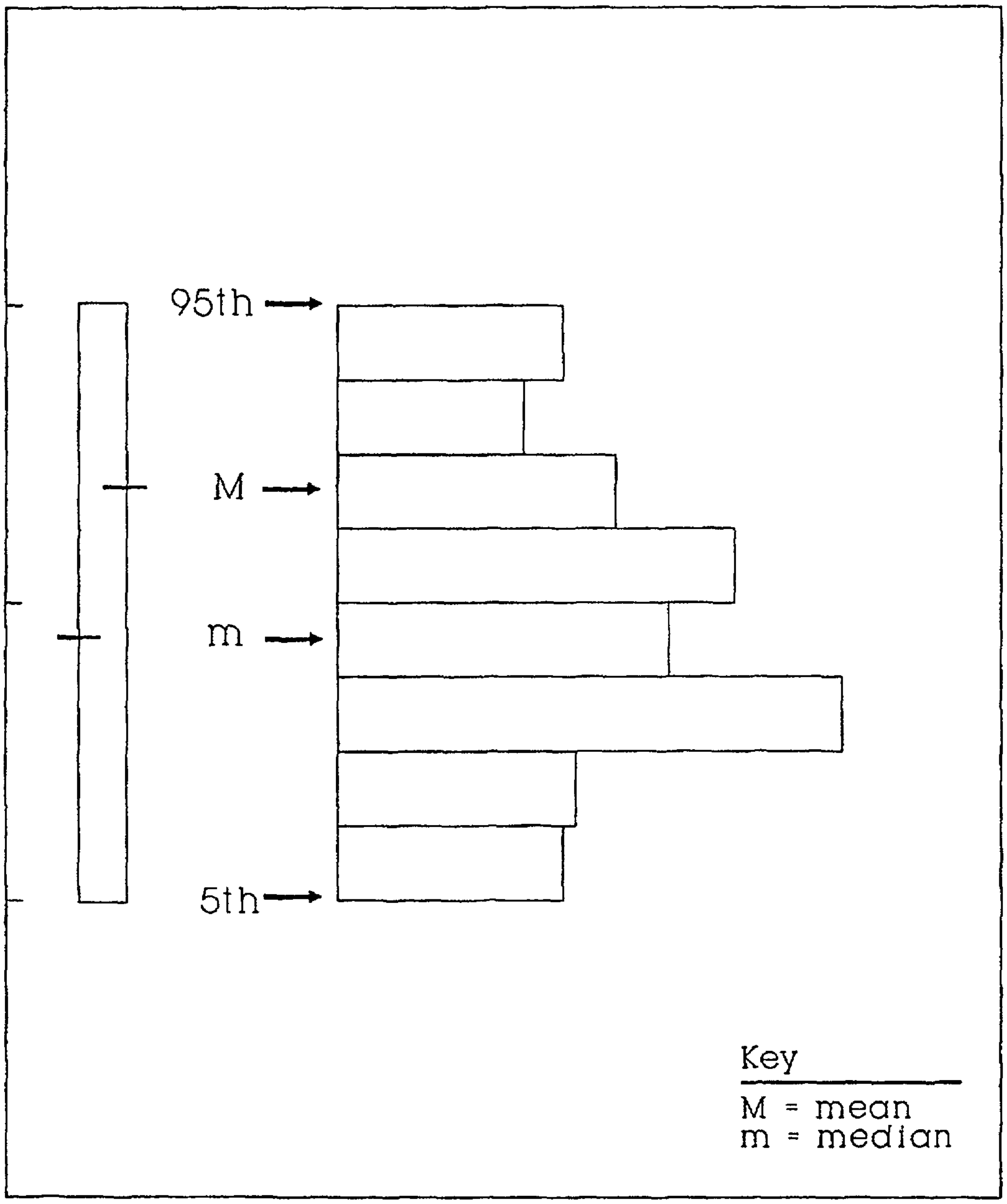


Figure A.6 Example display of core damage frequency distribution.

- The definitions and estimated frequencies of plant damage states

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

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

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

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

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

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

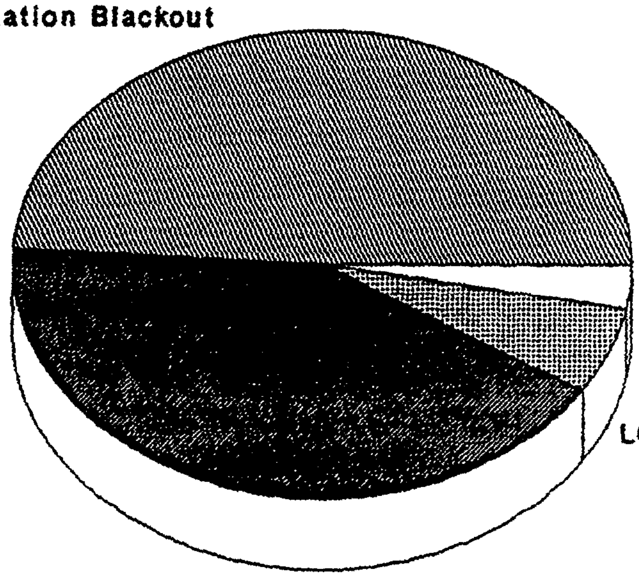
A.3 Accident Progression, Containment Loadings, and Structural Response Analysis**

A.3.1 Introduction

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

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

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



Total Mean Core Damage Frequency: 4.5E-6

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

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

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

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

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

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

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

Each of these steps is discussed below.

A.3.2 Development of Accident Progression Event Trees

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

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

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

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

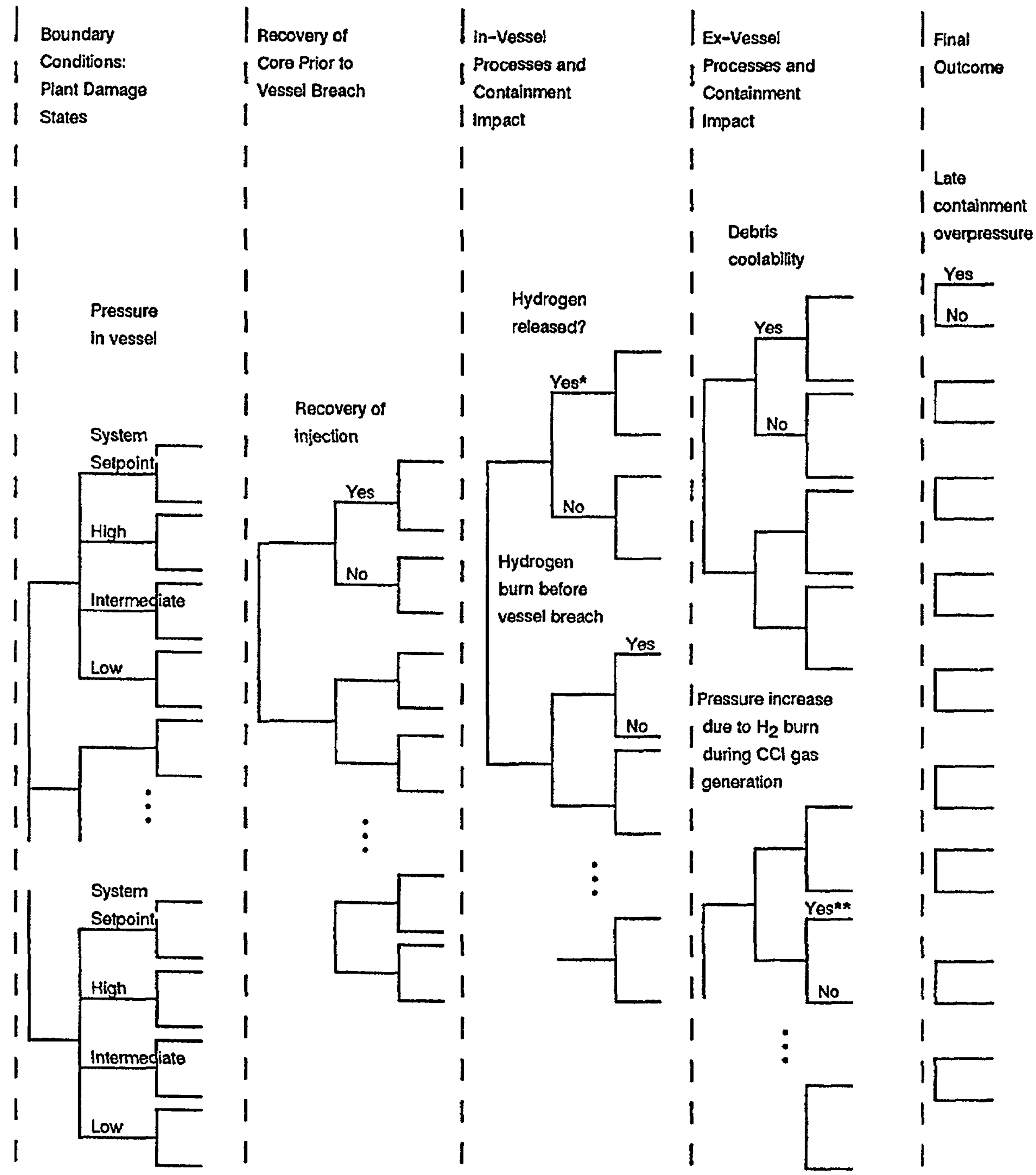
The final stage of the illustrated APET is related to the final status of containment building integrity. Long-term overpressurization, threats from combustion events, and similar questions were asked for this stage of the accident progression. For convenience, some questions that summarized the status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

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

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

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

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



*Amount of hydrogen released is sampled from continuous probability distribution

**Pressure increase is calculated from user function

Figure A.8 Schematic of accident progression event tree.

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

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

A.3.3 Structural Analyses

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

A.3.4 Probabilistic Quantification of APETs

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

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

A.3.5 Grouping of Event Tree Outcomes

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

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

A.3.6 Products of Accident Progression Analysis

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

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

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

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

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

  Measures of this distribution provided include:

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

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

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

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

A.4.1 Introduction

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

The principal steps in the source term analyses were:

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

Each of these steps will be discussed below.

A.4.2 Development of Parametric Models

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

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

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

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

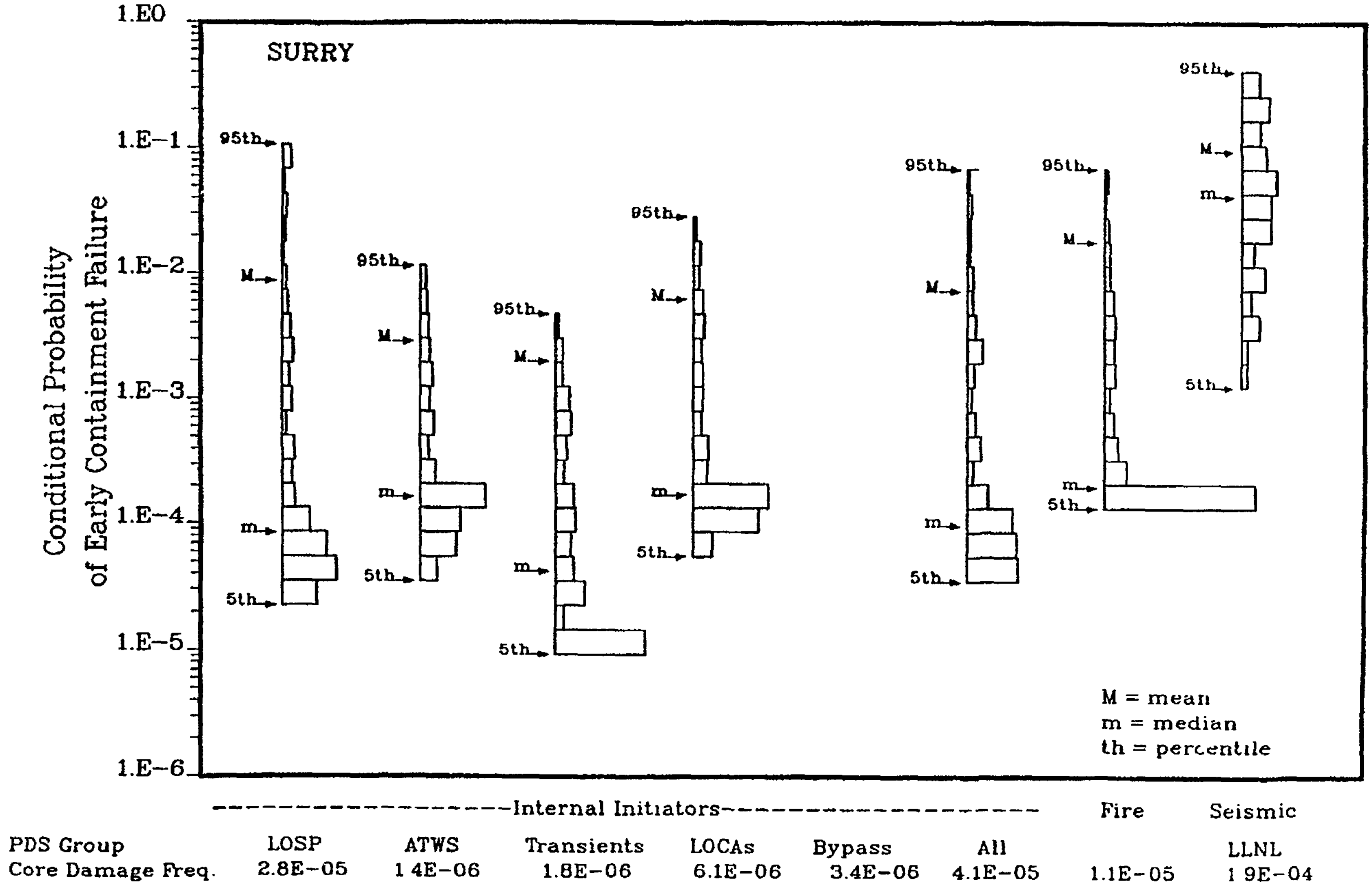


Figure A.9 Example display of early containment failure probability distribution.

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

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

SURRY

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

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

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

The basic parametric equations are of the form

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

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

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

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

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

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

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

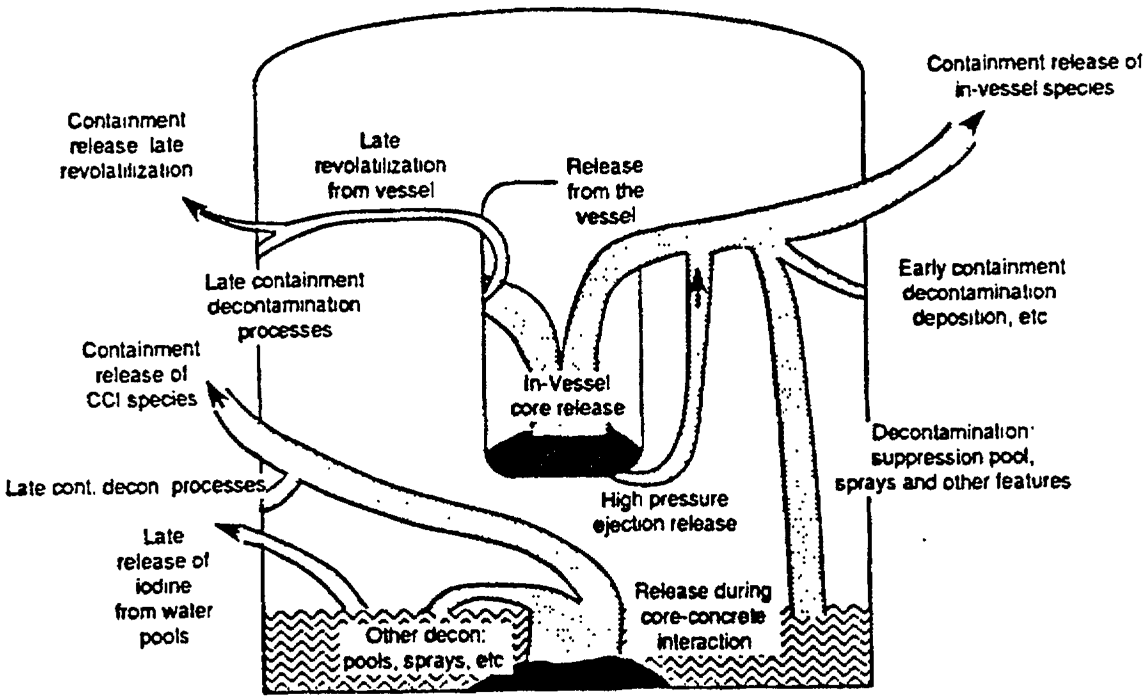


Figure A.11 Simplified schematic of source term (XSOR) algorithm.

A.4.3 Development of Values or Probability Distributions

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

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

A.4.4 Grouping of Radioactive Releases

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

A.4.5 Products of Source Term Analysis

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

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

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

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

A.5 Offsite Consequence Analysis

A.5.1 Introduction

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

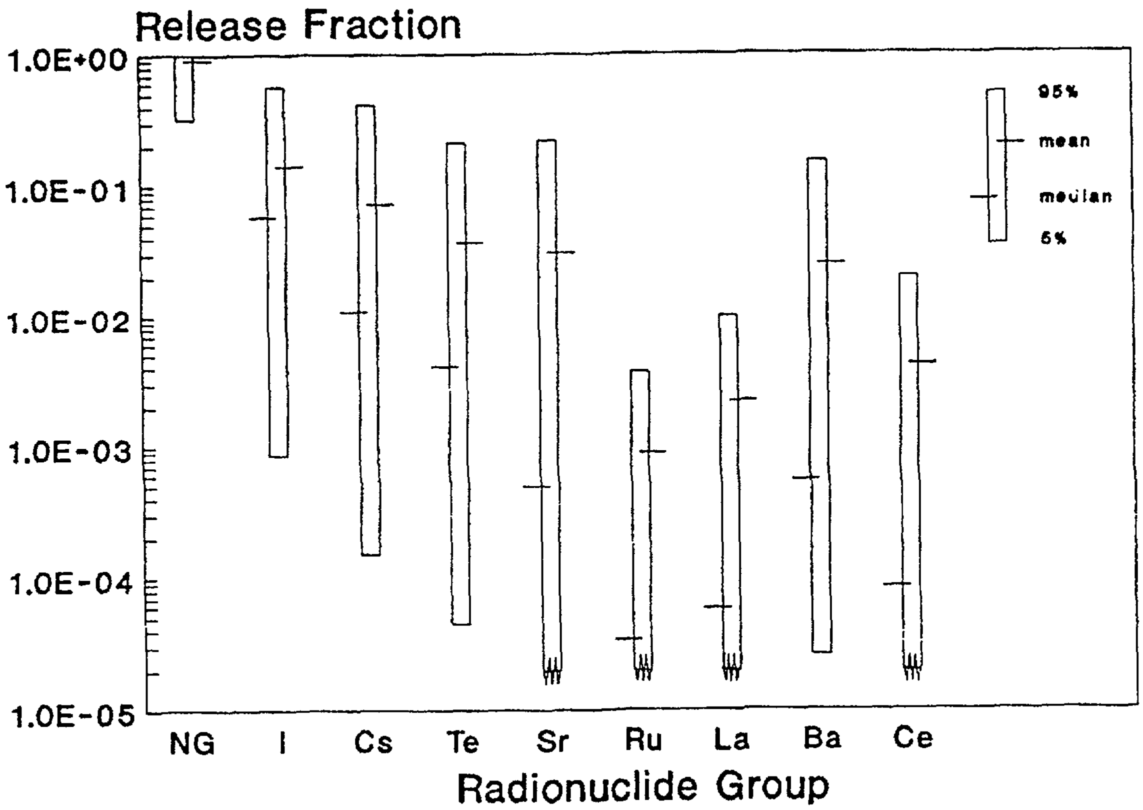


Figure A.12 Example display of radioactive release distributions for selected accident progression bin.

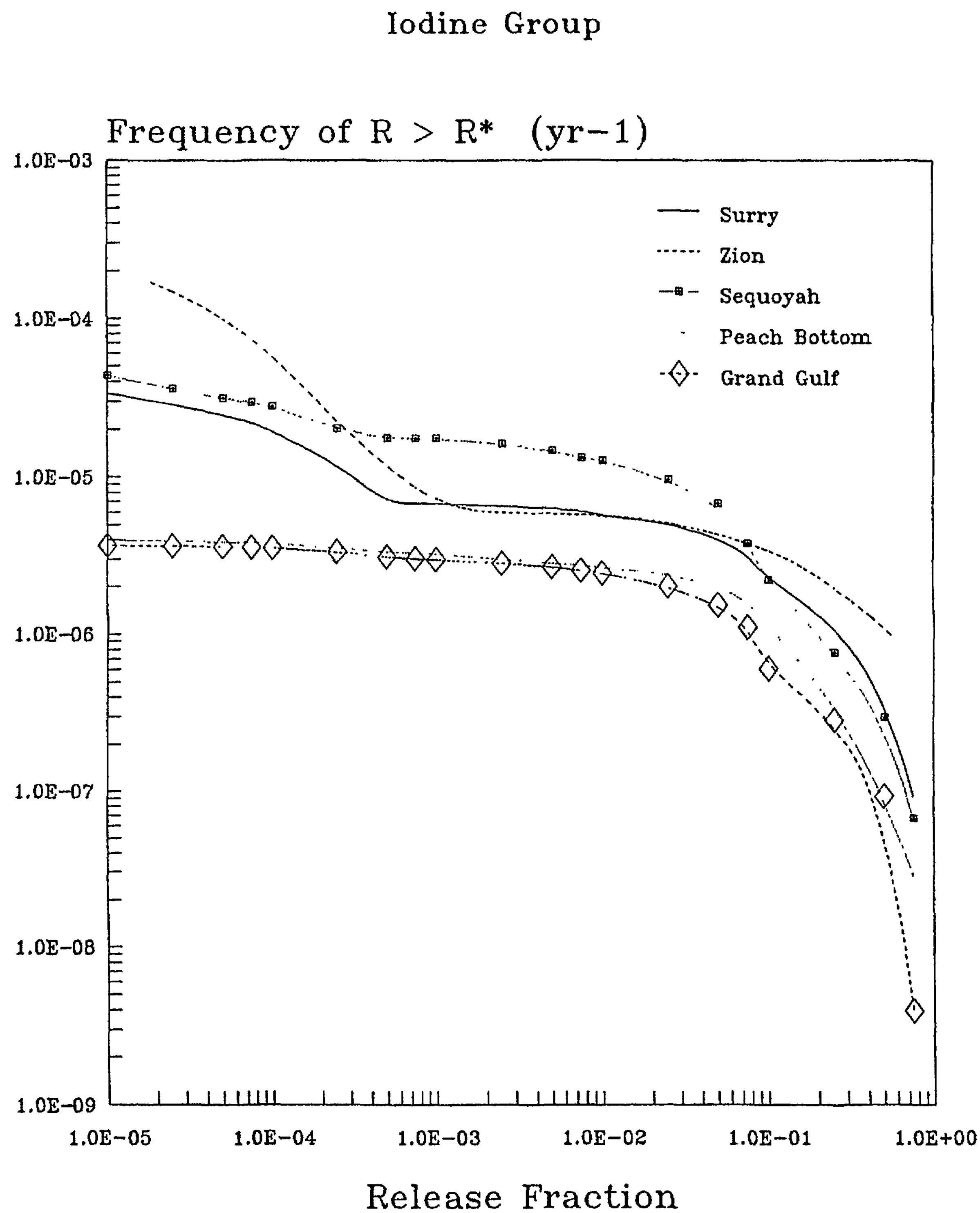


Figure A.13 Example display of source term complementary cumulative distribution function.

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

The principal steps in the offsite consequence analysis are:

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

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

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

A.5.2 Assessment of Pre-Accident Inventories

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

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

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

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

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

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

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

A.5.4 Calculation of Doses

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

A.5.5 Mitigation of Doses by Emergency Response Actions

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

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

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

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

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

A.5.6 Health Effects Modeling

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

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

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

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

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

A.5.7 Products of Offsite Consequence Analysis

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

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

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

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

A.6 Characterization and Combination of Uncertainties*

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

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

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

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

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

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

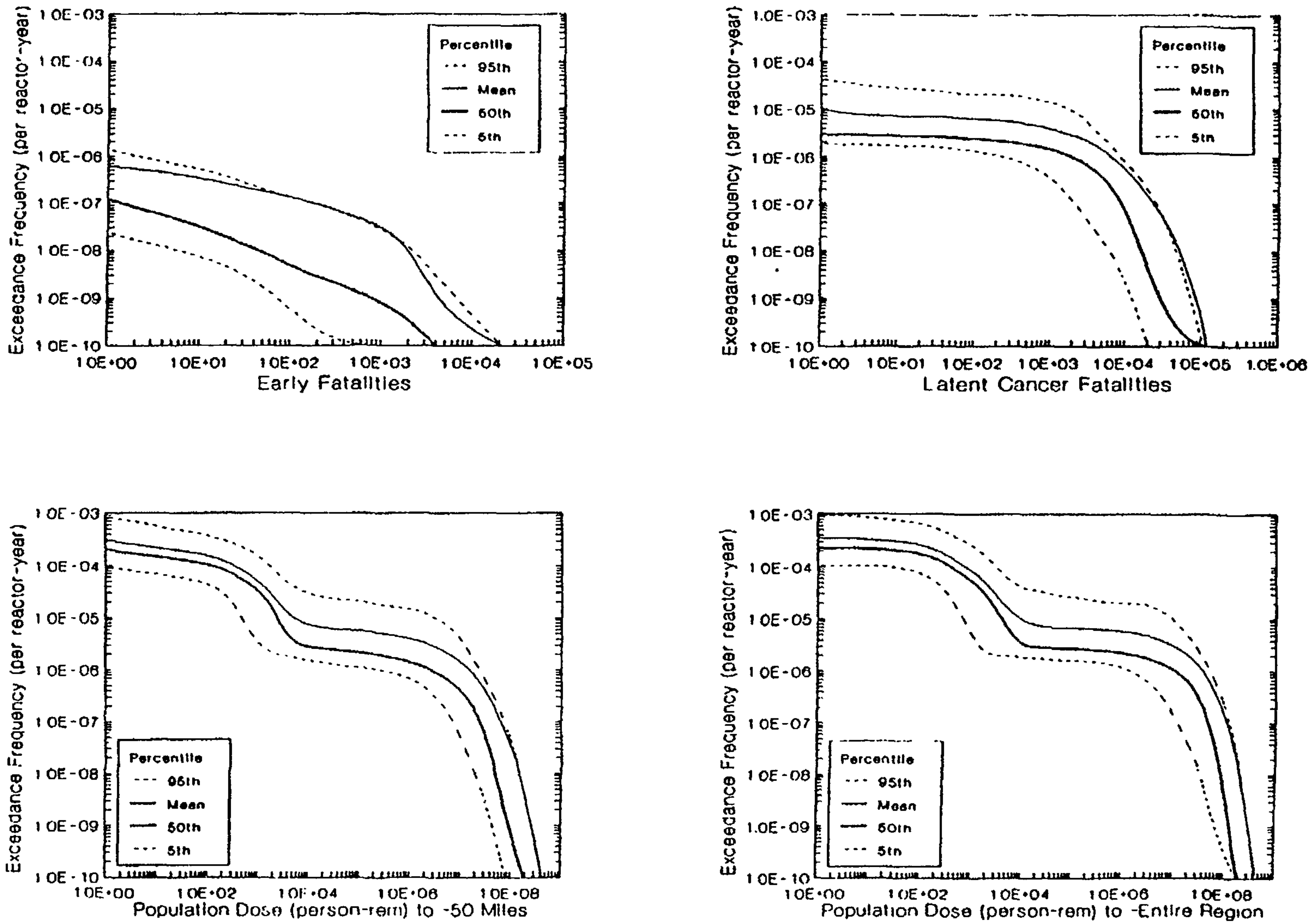


Figure A.14 Example display of offsite consequences complementary cumulative distribution function.

Table A.1 Issues considered by expert panels.

- **Accident Frequency Analysis Panel**

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

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

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

  Pipe rupture frequency in component cooling water system (Zion)

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

- **Reactor Coolant Pump Seal Performance Panel**

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

- **In-Vessel Accident Progression Panel**

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

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

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

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

- **Containment Loadings Panel**

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

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

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

- **Molten Core-Containment Interactions Panel**

  Drywell shell meltthrough (Peach Bottom)

  Pedestal erosion from core-concrete interaction (Grand Gulf)

- **Containment Structural Performance Panel**

  Static containment failure pressure and mode (PWRs and BWRs)

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

  Strength of reactor building (Peach Bottom)

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

  Pedestal strength during concrete erosion (Grand Gulf)

- **Source Term Expert Panel**

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

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

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

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

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

  Ice condenser decontamination factor (Sequoyah)

  Reactor building decontamination factor (Grand Gulf)

  Late sources of iodine (Grand Gulf)

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

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

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

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

where:

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

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

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

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

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

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

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

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

A.7 Elicitation of Experts*

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

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

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

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

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

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

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

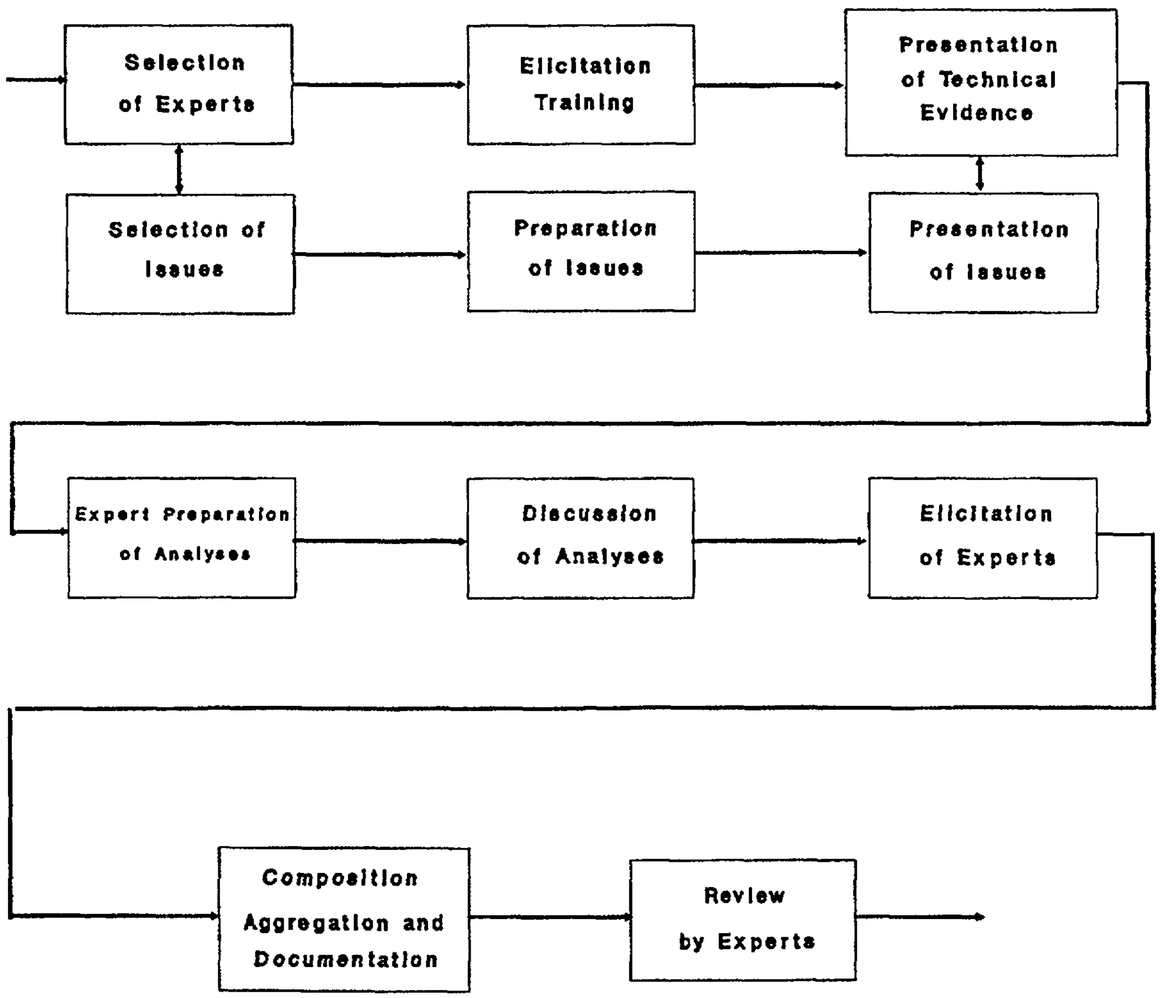


Figure A.15 Principal steps in expert elicitation process.

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

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

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

A.8 Calculation of Risk*

A.8.1 Methods for Calculation of Risk

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

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

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

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

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

LHS#1

f(PDS) P(PDS→APB) P(APB→STG) C(STG) RISK

$$\begin{bmatrix} f_1 \cdots\cdots f_{n_{PDS}} \end{bmatrix} \times \begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{APB}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{PDS}1} & & \cdots\cdots P_{n_{PDS}n_{APB}} \end{bmatrix} \times \begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{STG}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{APB}1} & & \cdots\cdots P_{n_{APB}n_{STG}} \end{bmatrix} \times \begin{bmatrix} C_{11} & C_{12} & \cdots\cdots C_{1n_C} \\ C_{21} & & \\ \vdots & & \\ C_{n_{STG}1} & & \cdots\cdots P_{n_{STG}n_C} \end{bmatrix} = \begin{bmatrix} RISK_1 \cdots\cdots RISK_{n_C} \end{bmatrix}$$

LHS#2

f(PDS) P(PDS→APB) P(APB→STG) C(STG) RISK

$$\begin{bmatrix} f_1 \cdots\cdots f_{n_{PDS}} \end{bmatrix} \times \begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{APB}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{PDS}1} & & \cdots\cdots P_{n_{PDS}n_{APB}} \end{bmatrix} \times \begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{STG}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{APB}1} & & \cdots\cdots P_{n_{APB}n_{STG}} \end{bmatrix} \times \begin{bmatrix} C_{11} & C_{12} & \cdots\cdots C_{1n_C} \\ C_{21} & & \\ \vdots & & \\ C_{n_{STG}1} & & \cdots\cdots P_{n_{STG}n_C} \end{bmatrix} = \begin{bmatrix} RISK_1 \cdots\cdots RISK_{n_C} \end{bmatrix}$$

$$\vdots$$

LHS#n_{LHS}

f(PDS) P(PDS→APB) P(APB→STG) C(STG) RISK

$$\begin{bmatrix} f_1 \cdots\cdots f_{n_{PDS}} \end{bmatrix} \times \begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{APB}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{PDS}1} & & \cdots\cdots P_{n_{PDS}n_{APB}} \end{bmatrix} \times \begin{bmatrix} P_{11} & P_{12} & \cdots\cdots P_{1n_{STG}} \\ P_{21} & & \\ \vdots & & \\ P_{n_{APB}1} & & \cdots\cdots P_{n_{APB}n_{STG}} \end{bmatrix} \times \begin{bmatrix} C_{11} & C_{12} & \cdots\cdots C_{1n_C} \\ C_{21} & & \\ \vdots & & \\ C_{n_{STG}1} & & \cdots\cdots P_{n_{STG}n_C} \end{bmatrix} = \begin{bmatrix} RISK_1 \cdots\cdots RISK_{n_C} \end{bmatrix}$$

Figure A.16 Matrix formulation of risk analysis calculation.

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

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

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

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

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

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

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

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

A.8.2 Products of Risk Calculation

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

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

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

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

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

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

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

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

A.9.1 Additional Explanation of Some Figures and Tables

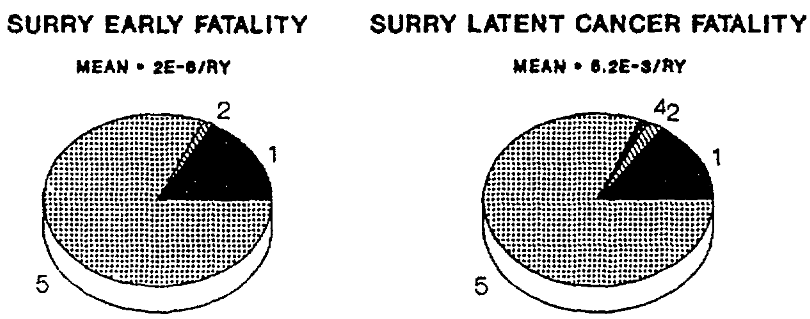
Most of the results presented in this report are generalized or summary results. They are similar to the intermediate results described in Section A.8.1. However, the groupings of postulated accidents that take place at the end of each constituent part of the risk calculation are more general in this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

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

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

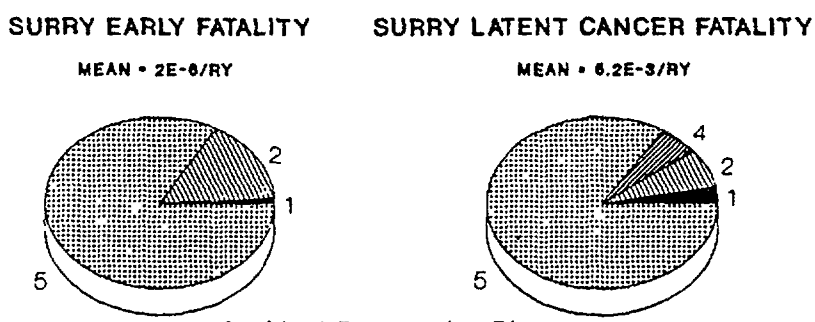
where $f(PDS_i)_n$ is the frequency of the generalized plant damage state i for Latin hypercube member n. Further, individual analysis results for the n_{LHS} sample elements can be ordered from the smallest to the largest and then used to estimate desired quantiles (i.e., 5th, median, and 95th), where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

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



Plant Damage States

1. SBO
2. ATWS
3. TRANSIENTS
4. LOCA
5. BYPASS



Accident Progression Bins

1. VB, Early CF, Alpha Mode
2. VB, Early CF, RCS Pressure >200 psia at VB
3. VB, Early CF, RCS Pressure <200 psia at VB
4. VB, BMT and Late Leak
5. Bypass
6. VB, No CF
7. No VB

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

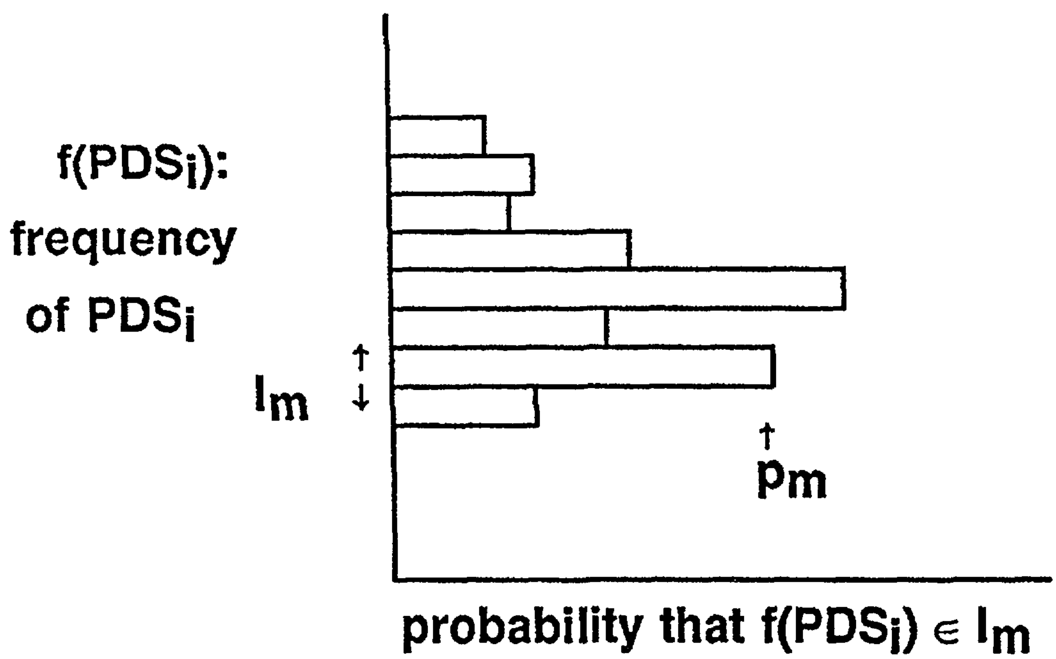


Figure A.18 Probability that $f(PDS_i)$ will fall in interval I_m.

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

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

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

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

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

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

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

A.9.2 Explanation of Some Terms

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

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

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

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

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

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

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

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

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

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

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

REFERENCES FOR APPENDIX A

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

A.2 E. D. Gorham-Bergeron et al., "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration, and Uncertainty Analyses," Sandia National Laboratories, NUREG/CR-4551, Vol. 1, Draft Revision 1, SAND86-1309, to be published.*

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

APPENDIX B

AN EXAMPLE RISK CALCULATION

CONTENTS

FIGURES

TABLES

B.1 Introduction

In this appendix, which is adapted from Reference B.1, an example calculation is followed through the entire analysis from the initiating event in the accident frequency analysis through to the offsite risk. This discussion has been prepared for the reader seeking detailed information on how the risk calculations were performed. It is assumed that the reader is familiar with nuclear power plants in general and with severe accident risk analysis in particular. Since the accident frequency analysis is generally more familiar to the PRA community, and the accident frequency analyses performed for NUREG-1150 have fewer novel features than the other analyses, the discussion of the accident frequency analysis in this appendix is abbreviated. Thus, even though the accident frequency analysis requires a level of effort comparable to that required for the other analyses, the discussion of the risk calculation from the identification of the initiating event through the definition of the plant damage state (PDS) does not reflect that fact.

The example selected for this discussion is a fast station blackout (SBO) accident for Surry. This accident, denoted TMLB'* in the Reactor Safety Study, is estimated to be one of the more likely accidents and is of historical interest. Surry was chosen because the accident progression event tree (APET) for Surry is simpler than the APETs for the other plants.

The PDS designation for the fast SBO accident is TRRR-RSR. (The PDS nomenclature is explained in Section B.2.3.) This PDS has the third highest mean core damage frequency (MCDF) at Surry. Several accident sequences comprise this PDS; the one chosen for this example is T1S-QS-L, which has the highest frequency of the sequences in TRRR-RSR. (This sequence is defined in detail in Section B.2.1.) PDS TRRR-RSR is the only PDS in PDS group 3, fast SBOs. The example will be followed through the APET to accident progression bin (APB) GFA-CAC-ABA-DA. (The APB nomenclature is explained in Section B.3.4.) For the observation chosen, this bin is the most likely to have both vessel breach (VB) and containment failure (CF). The computation of the source term for this bin will be followed through the source term analysis, and this source term will then be grouped with other similar source terms in the partitioning process.

Finally, offsite consequences will be determined for the subgroup to which the source term for GFA-CAC-ABA-DA was assigned, and the results of all the analyses will be combined to obtain the measures of risk.

To determine the uncertainty in risk, the accident frequency analysis, the accident progression analysis, and the source term analysis were performed many times, with different values for the important parameters each time. A sample of 200 observations was used for the Surry analysis. The Latin hypercube sampling method, a stratified Monte Carlo method, was used. In this example, one sample member or observation, Observation 4, will be followed all the way through the risk analysis. It was chosen because it was the median observation for early fatality risk (in the analysis in which 100 percent evacuation was assumed).

B.2 Accident Frequency Analysis

The accident frequency analysis determines the expected frequencies for the many different types of core damage accidents that can occur. This appendix is not intended to present methods, as those are summarized in Appendix A and presented in detail in Reference B.2. Nevertheless, many aspects of the methods will become apparent in this discussion. Section B.2.1 is an overview of the accident frequency analysis, and Section B.2.2 contains a description of the accident sequence. Section B.2.3 describes the quantification of the cut set, and Section B.2.4 discusses how the accident sequences are grouped into PDSs.

B.2.1 Overview of Accident Frequency Analysis

Development of the chronology and frequency of the accident sequences involves many tasks or constituent analyses. These include:

*TMLB' was defined in the Reactor Safety Study as a transient loss of offsite power (T) with failure of the power conversion system (M) and the auxiliary feedwater system (L), and failure of the emergency ac power system with no recovery of offsite ac power in 1 to 3 hours (B').

- Initiating event analysis, including determination of the system success criteria,
- Event tree analysis, including accident sequence delineation,
- Systems analysis, including fault tree construction,
- Dependent and subtle failure analysis,
- Human reliability analysis,
- Data base analysis, including development of the data base,
- Elicitation of expert judgment,
- Accident sequence quantification, including recovery actions,
- Grouping of the accident sequences into PDSs, and
- Uncertainty analysis.

These tasks are performed approximately in the order given above. The quantification and the assignment of the sequences to PDSs are performed several times in iterative fashion as the information available evolves and the requirements of the subsequent analyses change.

An accident sequence is a particular accident defined by the initiating event and failures of the systems required to respond to the initiator. Sequences are defined by specifying what systems fail to respond to the initiator. In the accident frequency analysis, models (event trees, fault trees) are constructed for all the important safety systems in the plant (usually at the pump and valve level of detail). Failure rates for equipment such as pumps and valves are developed from failure data specific to the plant being analyzed and from generic nuclear power plant data bases. The models and the failure rates are used by the computer program that calculates the thousands of possible failure combinations, denoted as cut sets, that lead to core damage.

Each cut set consists of the initiator and the specific hardware or operator failures that produce the system failures. The initiator and the failures are often referred to as "events." For example, a water injection system could fail because the pump failed to start or because the normally closed, motor-operated discharge valve failed to open. Cut sets that include the pump failure and cut sets that include the valve failure, but are otherwise identical, occur in the same accident sequence since the pump and valve failures have the same effect on a system level.

The accident sequence followed for this example is T1S-QS-L, which is the highest frequency sequence that contributes to PDS TRRR-RSR. This sequence is the most probable of several sequences that involve station blackout and early failure of the auxiliary feedwater system (AFWS). The mean frequency for TRRR-RSR is 4.8E-6/reactor year, and T1S-QS-L contributes about 75 percent of that. For Observation 4, the frequency of TRRR-RSR is 4.8E-7/reactor year, and the frequency of T1S-QS-L is 2.4E-7/reactor year. (It is purely coincidental that the frequency of TRRR-RSR for Observation 4 is one-tenth of the average frequency over all 200 observations.)

Sequence T1S-QS-L is comprised of 216 cut sets. The cut set with the highest frequency, consisting of nine events, is given in Table B.1. The cut set equation for T1S-QS-L is:

T1S-QS-L = (IE-T1) * (OEP-DGN-FS-DG01) * (/DGN-FTO) * (OEP-DGN-FS-DG03) *

(NRAC-1HR) * (REC-XHE-FO-DGEN) * (NOTQ) * (QS-SBO) * (AFW-XHE-FO-CST2)

+ ...... (215 other cut sets)

The frequency of each cut set varies from observation to observation because the probabilities of some of the events are sampled from distributions. For Observation 4, the frequency of the cut set in Table B.1 is

3.4E–8/reactor year. This cut set defines one group of specific failures that cause the accident, which will be followed through the entire analysis in this appendix. Each event listed in Table B.1 is discussed in some detail in Section B.2.3 below.

Table B.1 Most likely cut set in Surry sequence T1S-QS-L quantification for observation 4.

Event	Quantification	Description
IE-T1	0.0994	Initiating Event: LOSP
OEP-DGN-FS-DG01	0.0133	DG 1 fails to start
/DGN-FTO	0.966	Success of DG 2
OEP-DGN-FS-DG03	0.0133	DG 3 fails to start
NRAC-1HR	0.44	Failure to restore offsite electric power within 1 h
REC-XHE-FO-DGEN	0.90	Failure to restore a DG to operation within 1 h
NOTQ	0.973	RCS PORVs successfully reclose during SBO
QS-SBO	0.0675	Stuck-open SRV in the secondary system
AFW-XHE-FO-CST2	0.0762	Failure of operator to open the manual valve from the AFW pump suction to CST2
Entire cut set	3.4E–8	Frequency (per year) for Observation 4

Figure B.1 shows the event tree for T1S—station blackout at Unit 1. Three of the paths through this tree lead to core damage situations that are in PDS TRRR-RSR. Accident sequence 19, T1S-QS-L, is the most likely of these three. The logical expression for this sequence, according to the column headings or top events, is:

T1S-QS-L = T1S * NRAC-HALFHOUR * /Q * QS * L,

where /Q indicates not-Q, or success. System success states like /Q are sometimes omitted during quantification if the state results from a single event since the success value is very close to 1.0. T1 is a loss of offsite power (LOSP) initiator, and the "S" in T1S indicates that it is followed by failure of the emergency ac power system (EACPS). Failure of EACPS, although not shown explicitly in Figure B.1, is determined by a fault tree, and

T1S = T1 * Failure of EACPS,

where failure of EACPS is failure of diesel generator (DG) 1 and DG 3, or, failure of DG 1 and DG 2. (Failure of only DG 2 and DG 3 implies success of DG 1, which is not SBO for Unit 1. If DG 2 fails, it is assumed that DG 3 is assigned to Unit 2. Failure of all three DGs is included in a different sequence.) Note that T1 appears as IE-T1 in the cut set; the SBO is implied by events OEP-DGN-FS-DG01 and OEP-DGN-FS-DG03.

The cut set considered in this appendix has the failure of DG 1 and DG 3. A simplified depiction of the fault tree for DG 1 is shown in Figure B.2. The fault tree for DG 3 is similar. The heavier line in Figure B.2 indicates the failure, OEP-DGN-FS-DG01, in the cut set of interest. The other failures are included in other cut sets. (Figs. B.2 and B.3 are illustrative only and do not provide an accurate representation of the complete fault trees. The complete fault trees are given in Appendix B.2 to Ref. B.3.)

In Figure B.1, the cut set of interest is part of sequence 19, T1S-QS-L, which is shown by the heavier line. The first top event is the initiator, discussed above. The second top event concerns the recovery of offsite ac power within 30 minutes (NRAC-HALFHOUR). In the event of a loss of auxiliary feedwater, core uncovery will occur in approximately 60 minutes. A 30-minute time delay for reestablishing support systems (including canal water level) was assumed from the time that ac power was restored to the time that feed and bleed could be established. Thus, ac power must be recovered within 30 minutes in order to mitigate failure of auxiliary feedwater.

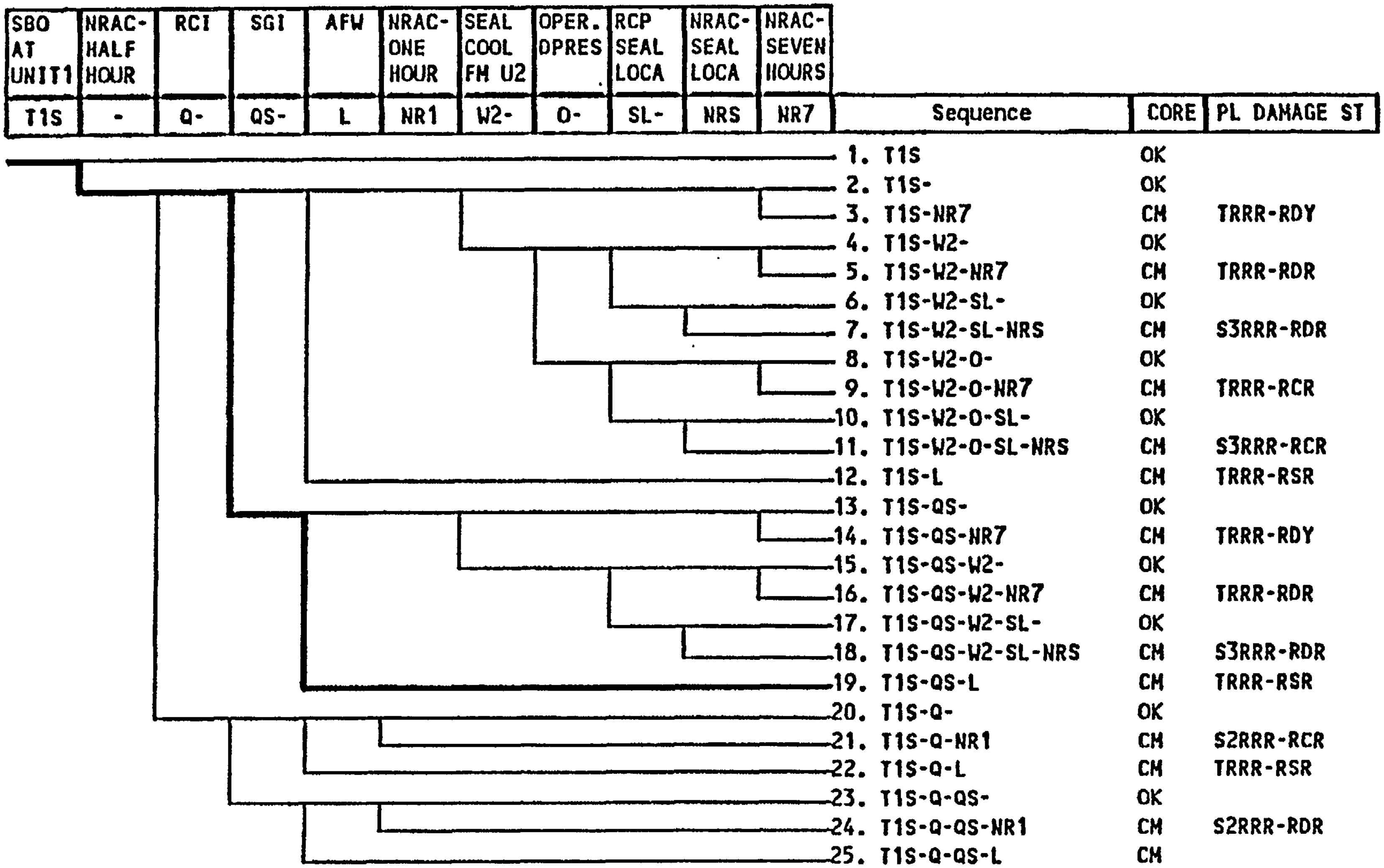


Figure B.1 Event tree for T1S-SBO at Surry Unit 1. (This figure is adapted from Section 4.4 of Ref. B.3. No PDS assignment is indicated for sequence 25 because the sequence frequency fell below the cutoff value.)

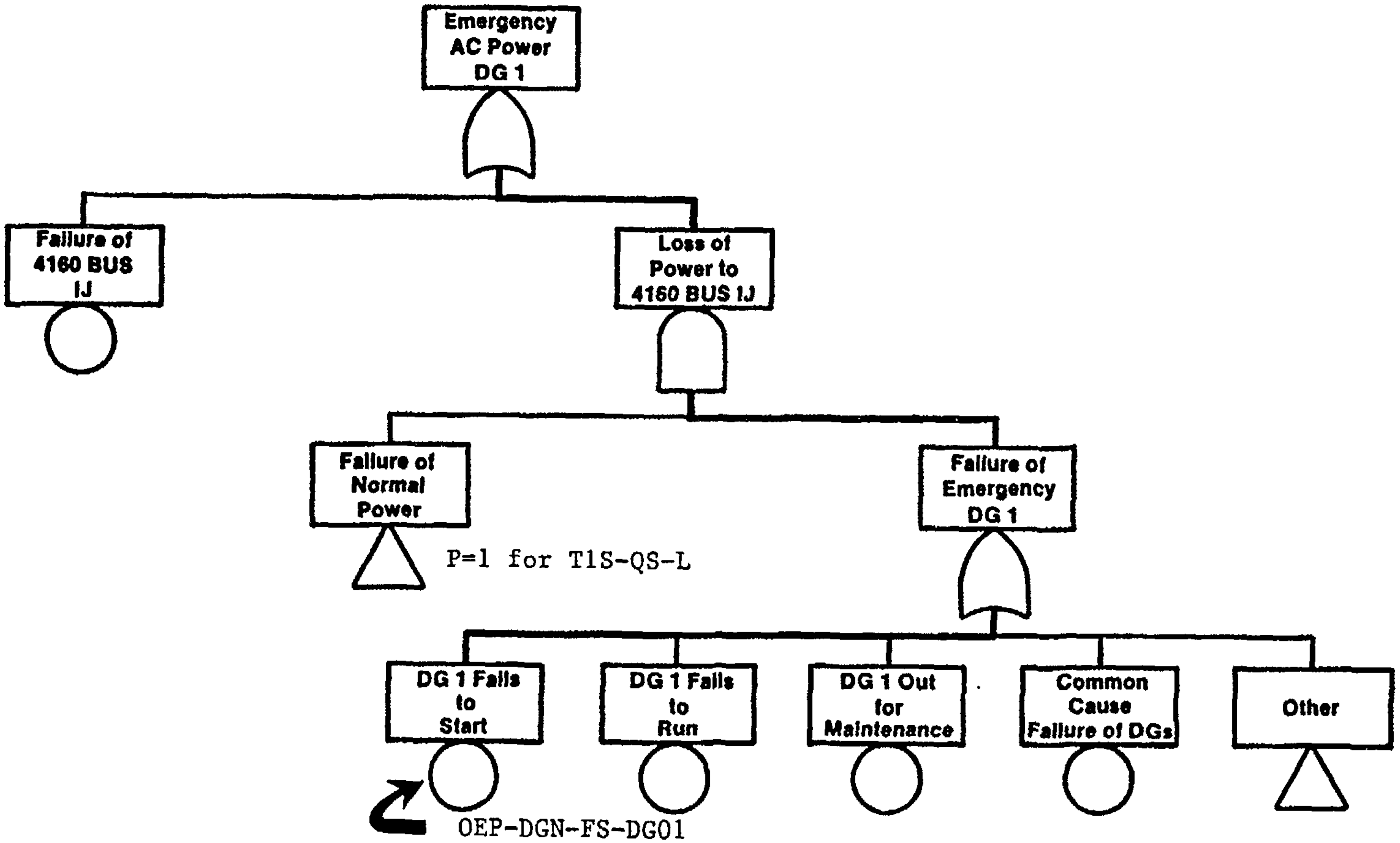


Figure B.2 Reduced fault tree for DG 1 at Surry Unit 1. (This figure is a greatly simplified version of the fault tree given in Appendix B.2 of Ref. B.3. P = 1 indicates that the failure probability is 1.0.)

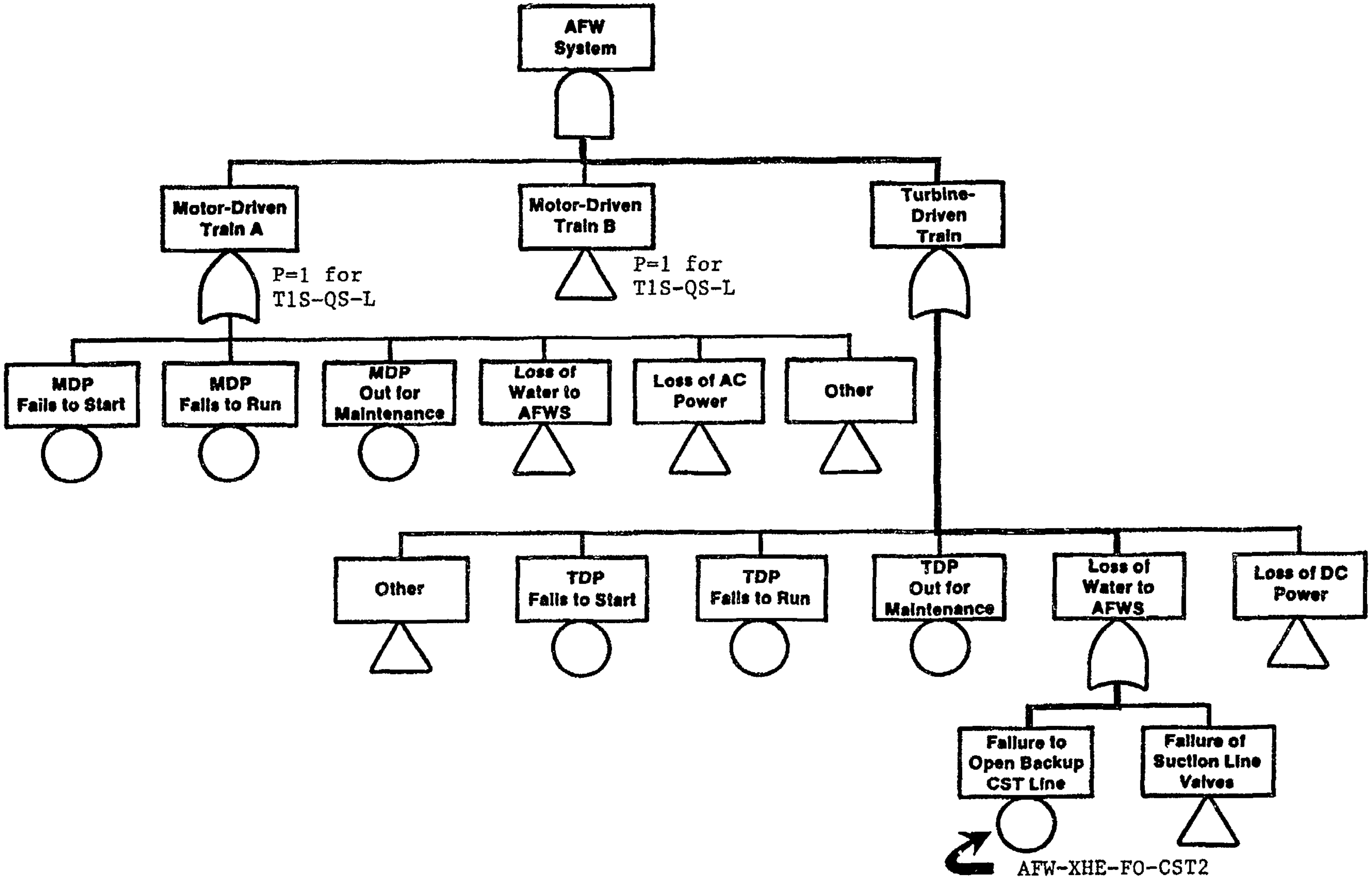


Figure B.3 Reduced fault tree for AFWS at Surry Unit 1. (This figure is a greatly simplified version of the fault tree given in Appendix B.2 of Ref. B.3. P = 1 indicates that the failure probability is 1.0.)

The third top event is RCI, failure of reactor coolant system integrity, Event Q. In the cut set being followed, the success branch is taken here, i.e., the PORVs cycle correctly and do not stick open. The fourth top event in Figure B.1 is SGI, failure of steam generator (secondary side) integrity. In this cut set, a relief valve on the main steam line sticks open, Event QS, so the failure branch is taken. Failure of QS causes rapid depressurization of the steam generator and rapid depletion of condensate water.

The fifth top event, L, is failure of the AFWS. In the cut set listed in Table B.1, the steam-turbine-driven AFWS runs until the condensate storage tank (CST) is depleted at about 60 minutes after the start of the accident. The AFWS fails at that time because the operators fail to switch the pump suction to the backup CST. This failure is event AFW-XHE-FO-CST2. Thus, the appropriate power recovery time for this cut set is NRAC–1HR, which replaces NRAC-HALFHOUR in the cut set, although the failure (lower) branch for NRAC-HALFHOUR is indicated on the event tree. None of the subsequent top events are applicable since the failures that have already occurred are sufficient to cause core damage, so there are no further branches on the path to sequence 19.

Figure B.3 is a simplified fault tree for the AFWS. The motor-driven AFW trains, of course, require ac electrical power and are not available for this accident. The heavier line in Figure B.3 indicates the failure that occurs in the cut set considered for this example. The other failures are included in other cut sets.

A general description of this accident sequence follows in the next section. More detail on the methods used in the accident frequency analysis may be found in Reference B.2. Details of the specific analysis for Surry may be found in Reference B.3.

B.2.2 Description of Accident Sequence

An LOSP initiator, IE-T1, starts this transient accident by tripping the reactor and the main steam turbine. The DG assigned to Unit 1, DG 1, fails to start, OEP-DGN-FS-DG01. DG 3 also fails to start, OEP-DGN-FS-DG03. (DG 2 is dedicated to Unit 2.) The event /DGN-FTO indicates that DG 2 is successfully powering Unit 2. The failure to start of DG 1 and DG 3 causes a complete failure of ac power at Unit 1. However, dc power is available from the Unit 1 batteries until they are depleted (in roughly 4 hours).

The pressure boundary of the reactor coolant system (RCS) is intact, so loss of water from the RCS is not an immediate problem. However, all the systems capable of injecting water into the RCS depend on pumps driven by ac electric motors. Thus, if decay heat cannot be removed from the RCS, the pressure and temperature of the water in the RCS will increase to the point where water will flow out through the PORVs, and there will be no way to replace this lost water.

Heat removal after shutdown is normally accomplished by the auxiliary feedwater system (AFWS). Surry's AFWS has three trains: two of these trains have pumps driven by ac electric motors, and these trains are unavailable due to the SBO. The only means of heat removal in a blackout situation is the steam-turbine-driven AFW train. In the accident defined by the cut set in Table B.1, the steam-turbine-driven AFW train is initially available as steam is being generated in the steam generators (SGs) to drive the steam turbine, and dc power is available for control purposes. The initiating LOSP causes the main steam isolation valves (MSIVs) to close, preventing the steam being generated in the SGs that is not needed by the AFWS turbine from flowing to the main condenser. The normal means of venting excess steam from the secondary system is through the atmospheric dump valves (ADVs), but in this sequence they are failed in the closed position because of the loss of 120 v ac power. Thus, pressure relief takes place through one or more of the secondary system safety-relief valves (SRVs).

In this accident sequence, at least one of the secondary system SRVs fails to reclose, which causes water to be lost at a significant rate from the secondary system. This is event QS in Figure B.1; it is denoted QS-SBO in the cut set. The AFWS initially draws from the 100,000-gallon condensate storage tank (CST). With an SRV stuck open, the AFWS will draw from the CST at 1,000 to 1,500 gpm to replace the water lost through the SRV, thus depleting the CST in 1.0 to 1.5 hours. A 300,000-gallon backup water supply (CST2) is available, but the AFWS cannot draw from this tank unless a manual valve is opened. In this cut set, the operators fail to open this valve, and the AFWS fails. This human error is event AFW-XHE-FO-CST2. There are two recovery actions in this cut set. One is the failure to restore offsite power within 1 hour (NRAC–1HR), and the other is the failure to recover a failed DG

(REC-XHE-FO-DGEN). In the path to sequence 19 shown in Figure B.1, the failure to recover offsite power is NRAC-HALFHOUR. In this particular cut set, the time to failure of the AFWS is longer than in the majority of cut sets in sequence T1S-QS-L, and this failure is replaced by NRAC–1HR.

With the failure of the turbine-driven AFW train, and no ac power to run the motor-driven AFW trains, the reactor coolant system (RCS) heats up until the pressure forces steam through the PORV(s). Water loss through the PORV(s) continues, with the PORV(s) cycling open and closed, until enough water has been lost to reduce the liquid water level below the top of active fuel (TAF). The PORVs do not stick open; this is event NOTQ. Without electric power, there is no way to replace the water lost from the RCS. The uncovering of the TAF (UTAF) marks the transition of the accident from the accident frequency analysis to the accident progression analysis. The onset of core degradation follows shortly after the UTAF.

B.2.3 Quantification of Cut Set

Table B.1 gives the specific cut set being considered in this example and shows the quantification of each event in the cut set for Observation 4. A discussion of how each quantification was derived follows.

IE-T1 is the initiating event: LOSP. The frequency of this initiating event was sampled from a distribution. The quantification for Observation 4 is 0.0994. This value is above the mean value of 0.077. The distribution for LOSP was derived from Surry station historical experience, using the methods in Reference B.4. This analysis uses Bayesian models for both the frequency of LOSP and the time to recovery of offsite power. Utility data from 63 LOSP incidents was analyzed to develop a composite offsite power model that combined the effects of failures of the grid, events at the plant (e.g., switchyard problems), and severe weather. The model can be adjusted to reflect specific switchyard design.

OEP-DGN-FS-DG01 is the failure of DG 1 to start. The probability of this event was sampled from a distribution. The quantification for Observation 4 is 0.0133. This value is slightly below the mean value of 0.022. The distribution for this event was derived from the Surry plant records of DG operation for 1980 to 1988. In this period, there were 484 attempts to start the DGs and 19 failures. Eight of these failures were ignored since they occurred during maintenance. A lognormal distribution with an error factor of three was used to model the uncertainty in this event. The error factor was based on a very narrow chi squared uncertainty interval.

/DGN-FTO indicates that DG 2 has started and is supplying power to Unit 2. Thus, DG 3, the "swing" DG at Surry, may be aligned to supply power to Unit 1. The Surry station consists of two units. Emergency power is supplied by three DGs; DG 1 can supply power only to Unit 1, DG 2 can supply power only to Unit 2, and DG 3 can be aligned to supply power to either unit. If DG 2 starts and runs initially, DG 3 is not required for Unit 2. The probability of this event was sampled from a distribution. The quantification for Observation 4 is 0.966, which is almost equal to the mean value (0.97) of this distribution. The distribution was developed from Surry plant data on DG operation at Unit 2 in a manner similar to that for the previous event.

OEP-DGN-FS-DG03 is the failure of DG 3 to start. The quantification for Observation 4 is 0.0133, the same as for OEP-DGN-FS-DG01 above. The same distribution was used for both DG 1 and DG 3, and the sampling was fully correlated.

NRAC–1HR is the failure to restore offsite power within 1 hour. Initially, the probability of this event was sampled from a distribution obtained using the offsite power recovery methods of Reference B.4. As the uncertainty in this event proved to be only a small contributor to the uncertainty in the core damage frequency in the uncertainty analysis performed for the accident frequency analysis alone, it was not sampled in the integrated analysis. For the integrated analysis, NRAC–1HR was set to the mean value of the distribution, 0.44, for every observation in the sample.

REC-XHE-FO-DGEN is the failure to restore a DG to operation within 1 hour. The probability of this event was sampled from the distribution for this operation that appears in the Accident Sequence Evaluation Program (ASEP) generic data base (Ref. B.2). The uncertainty in this event was not a significant contributor to the uncertainty in the core damage frequency. It was not sampled in the integrated analysis, and REC-XHE-FO-DGEN was set to the mean value of the distribution, 0.90, for every observation in the sample.

NOTQ indicates that the RCS PORV(s) successfully reclose during SBO. Event Q is the failure of the RCS PORV(s) to reclose in an SBO sequence, so NOTQ is success. The probability of this event was sampled from a distribution in the stand-alone version of the accident frequency analysis. Because the uncertainty in NOTQ was not a significant contributor to the uncertainty in the core damage frequency, NOTQ was set to 0.973, the complement of the mean value of the distribution for Event Q, for the integrated analysis. The distribution for Event Q was taken from the ASEP generic data base (Ref. B.2).

QS-SBO is the failure of a PORV or SRV in the secondary system to reclose after opening one or more times. For an SBO, the PORVs on the secondary side, also known as the atmospheric dump valves, are not operable, so it is the SRVs that open. The probability of this event was sampled from a distribution. The quantification for Observation 4 is 0.0675, which is considerably less than the mean value (0.27) of this distribution. The distribution for QS-SBO was determined from the ASEP generic data base (Ref. B.2). This analysis considered the number of times an SRV may be expected to open, and the rate at which the SRVs at Surry are expected to fail to reclose (Ref. B.3).

AFW-XHE-FO-CST2 is the failure of the operator to open the manual valves to the auxiliary condensate storage tank, CST2. This action is necessary to provide a supply of water for the AFWS after the primary condensate storage tank is depleted. The probability of this event was sampled from a distribution derived using a standard method for estimating human reliability. AFW-XHE-FO-CST2 is the failure to successfully complete a step-by-step operation following well-designed emergency operating procedures with a moderate level of stress. The method used is presented in Reference B.2, and detailed results may be found in Reference B.3. The quantification for Observation 4 is 0.0762, which is slightly above the mean value (0.065) of this distribution.

B.2.4 Accident Sequence and PDS

The cut set gives specific hardware faults and operator failures. In determining the general nature of the accident, however, many cut sets are essentially equivalent. These cut sets are grouped together in an accident sequence. For example, consider the cut set described above. In the description of the accident, it would have made little difference whether there was no ac power because DG 1 was out of service for maintenance (see Fig. B.2) or whether DG 1 failed to start as in the cut set in Table B.1. The fault is different, and the possibilities for recovery may be different, but the result is the same on a system level. Thus, both cut sets occur in accident sequence T1S-QS-L, along with many other cut sets that also result in the same combination of system failures. In the example, the important development for defining the accident is that DG 1 has failed. Exactly how it failed must be known to determine the probability of failure but is rarely important in determining how the accident progresses after UTAF.

The accident frequency analysis results in many significant accident sequences, typically dozens and perhaps a hundred or so. As the accident progression analysis is a complex and lengthy process, accident sequences that will progress in a similar fashion are grouped together into plant damage states (PDSs). That is, sequences with similar times to UTAF, similar plant conditions at UTAF, and that are expected to progress similarly after UTAF, are grouped together in a PDS. Figure B.1 shows the three sequences that are placed together in PDS TRRR-RSR. They are T1S-QS-L, T1S-L, and T1S-Q-L. (A fourth sequence, T1S-Q-QS-L, sequence 25, would have been placed in TRRR-RSR but was eliminated because of its low frequency.) T1S-QS-L is by far the most likely of these accident sequences and has been described above. Sequence T1S-L is similar to T1S-QS-L but has the AFWS failing at the very start of the accident because of failures in the steam-turbine-driven AFW train itself (such as fail to start, fail to run, etc.). In T1S-Q-L, which is much less probable than either T1S-QS-L or T1S-L, an RCS PORV sticks open, and there is no way to replace the water lost through this valve.

The process of assigning accident sequences to PDSs forms the interface between the accident frequency analysis and the accident progression analysis. The characteristics that define the PDSs are determined by the accident progression analysts based on the information needed in the APET. These characteristics are carefully reviewed with the staff that performs the accident progression analysis to ensure that all situations are included, that the definitions are clear, and that there are no ambiguous cases. Then, every cut set is examined to determine its appropriate PDS. This often requires an iteration through the event tree and fault tree analyses since assignment to the proper PDS may require information, for example, about the containment spray systems, that was not needed to determine the core damage frequency. Thus, it is possible that the cut sets that form a single accident sequence might be separated into two (or more) different PDSs, although this never occurs in the Surry analysis.

The seven letters that make up the Surry PDS indicator denote characteristics of the plant condition when the water level falls below the TAF and consideration of the accident passes from the accident frequency analysis to the accident progression analysis. For PDS TRRR-RSR, each character in the PDS designation is explained below. Recoverable means the system is not operating but can operate if ac power is recovered.

T – RCS is intact at the onset of core damage;

R – Emergency core cooling is recoverable;

R – Containment heat removal is recoverable;

R – ac power can be recovered from offsite sources;

R – The contents of the refueling water storage tank (RWST) have not been injected into the containment but can be injected if ac power is recovered;

S – The steam-turbine-driven AFWS failed at, or shortly after, the start of the accident; the electric-motor-driven AFWS is recoverable; and

R – Cooling for the reactor coolant pump (RCP) seals is recoverable.

A more complete description of the PDS nomenclature may be found in Reference B.1. The assignment of sequences to PDSs is discussed in Reference B.3. For internal initiators at Surry, 25 PDSs were above the cutoff frequency of 1.0E–7/reactor year for the accident progression analysis. They were placed in seven PDS groups based on the initiating events. The seven PDS groups for internal initiators at Surry, in order by decreasing mean core damage frequency, are:

1. Slow SBO;
2. Loss-of-coolant accidents (LOCAs);
3. Fast SBO;
4. Event V (interfacing-system LOCA);
5. Transients;
6. ATWS (failure to scram the reactor); and
7. Steam generator tube ruptures.

The example being followed here goes to the third PDS group, Fast SBO, which consists of only a single PDS, TRRR-RSR.

B.3 Accident Progression Analysis

The accident progression analysis considers the core degradation process and the response of the containment and other safety systems to the events that accompany core degradation. Of particular interest is whether the containment remains intact, since this determines the magnitude of the fission product release in many accidents. In the analyses conducted for NUREG–1150, the accident progression analysis is performed by use of a large event tree. While a simple event tree like that shown in Figure B.1 can be easily illustrated and evaluated with a hand calculator, the event trees used for the accident progression analysis are too large to be depicted in a figure and have so many paths through them that they can only be evaluated by a computer program.

B.3.1 Introduction

The APET for Surry consists of 71 questions. Many of these questions are not of particular interest for PDS TRRR-RSR; therefore, only about half the questions are listed in Table B.2 and shown in Figure B.4.

Table B.2 Selected questions in Surry APET.

	Question	Branch Taken or Parameter Defined	Source of Quantification	Meaning of Branch or Parameter
1.	RCS Integrity at UTAF?	Br.6	PDS Def.	RCS intact—water loss is through cycling PORVs
8.	Status of ac Power?	Br.2	PDS Def.	Will be available when offsite power recovered
10.	Heat Removal from SGs?	Br.2	PDS Def.	Will be available when offsite power recovered
12.	Cooling for RCP Seals?	Br.2	PDS Def.	Will be available when offsite power recovered
13.	Initial Cont. Condition?	Br.3	Acc.Freq.	Containment intact
15.	RCS Pressure at UTAF?	Br.1	Summary	RCS is at system setpoint pressure (2500 psia)
16.	PORVs Stick Open?	Br.2	Internal	PORVs do not stick open
17.	T-I RCP Seal Failure?	Br.1	Acc.Freq.	RCP seals fail
19.	T-I SGTR?	Br.2	Experts	No steam generator tube rupture
20.	T-I Hot Leg Failure?	Br.2	Experts	No hot leg or surge line failure
21.	AC Power Early?	Br.2	Distrb.	Offsite ac power is not recovered before VB
23.	RCS Pressure at VB?	Br.3	Internal	The RCS is at intermediate pressure (200 to 600 psia)
28.	Cont. Pressure before VB?	Par.1	Summary	The containment is at 26 psia just before VB
29.	Time of Accm. Discharge?	Br.2	Summary	The accumulators discharge during core melt
30.	Fr. Zr Oxidized In-Ves.?	Par.2	Experts	0.866 of the Zr is oxidized in-vessel
31.	Amt. Zr Oxidized In-Ves.?	Br.1	Summary	A high fraction of Zr is oxidized in-vessel
32.	Water in Cavity at VB?	Br.2	Summary	The reactor cavity is dry at VB
33.	Fr. Core Released at VB?	Par.3	Experts	0.544 of the core is released at VB
34.	Amt. Core Released at VB?	Br.1	Summary	A high fraction of the core is released at VB
35.	Alpha-Mode Failure?	Br.2	Experts	There is no alpha-mode failure
36.	Type of Vessel Breach?	Br.1	Experts	High-pressure melt ejection occurs at VB
38.	Size of Hole in Vessel?	Br.1	Internal	The hole in the vessel is large
39.	Pressure Rise at VB?	Par.4	Experts	The pressure rise at VB is 56.8 psig
41.	Ex-Vessel Steam Explosion?	Br.2	Internal	There is no ex-vessel steam explosion at VB

Table B.2 (Continued)

Question	Branch Taken or Parameter Defined	Source of Quantification	Meaning of Branch or Parameter
42. Cont. Failure Pressure?	Par.7	Experts	The containment failure pressure is 148.4 psig
	Par.8		The LHS number for failure mode is 0.808
43. Containment Failure?	Br.4	Calc.	The containment does not fail at VB
45. AC Power Late?	Br.1	Distrb.	Offsite ac power is recovered during early CCI
46. Late Sprays?	Br.1	Summary	Containment sprays are recovered during early CCI
49. How much H_2 Burns at VB?	Par.8	Internal	0.30 of the hydrogen burns at VB
50. Late Ignition?	Br.1	Experts	Ignition occurs during early CCI
	Par.9	Internal	95 percent of the hydrogen burns if ignition occurs
	Par.10	Internal	The pressure rise scale factor is 1.12
51. Late Burn? Pressure Rise?	Br.1	Calc.	Hydrogen combustion occurs during early CCI
	Par.11	Calc.	The load pressure is 100.2 psia
52. Containment Failure?	Br.4	Calc.	The containment does not fail during early CCI
53. Amount of Core in CCI?	Br.2	Internal	A medium amount of the core is involved in CCI
54. Is Debris Bed Coolable?	Br.1	Internal	The debris bed is coolable if water is available
55. Does Prompt CCI Occur?	Br.1	Summary	Prompt CCI occurs
62. Very Late Ignition?	Br.2	Experts	Ignition does not occur during or after late CCI
68. Basemat Meltthrough?	Br.1	Internal	The basemat eventually melts through
71. Final Cont. Condition?	Br.3	Summary	The only containment failure is basemat meltthrough

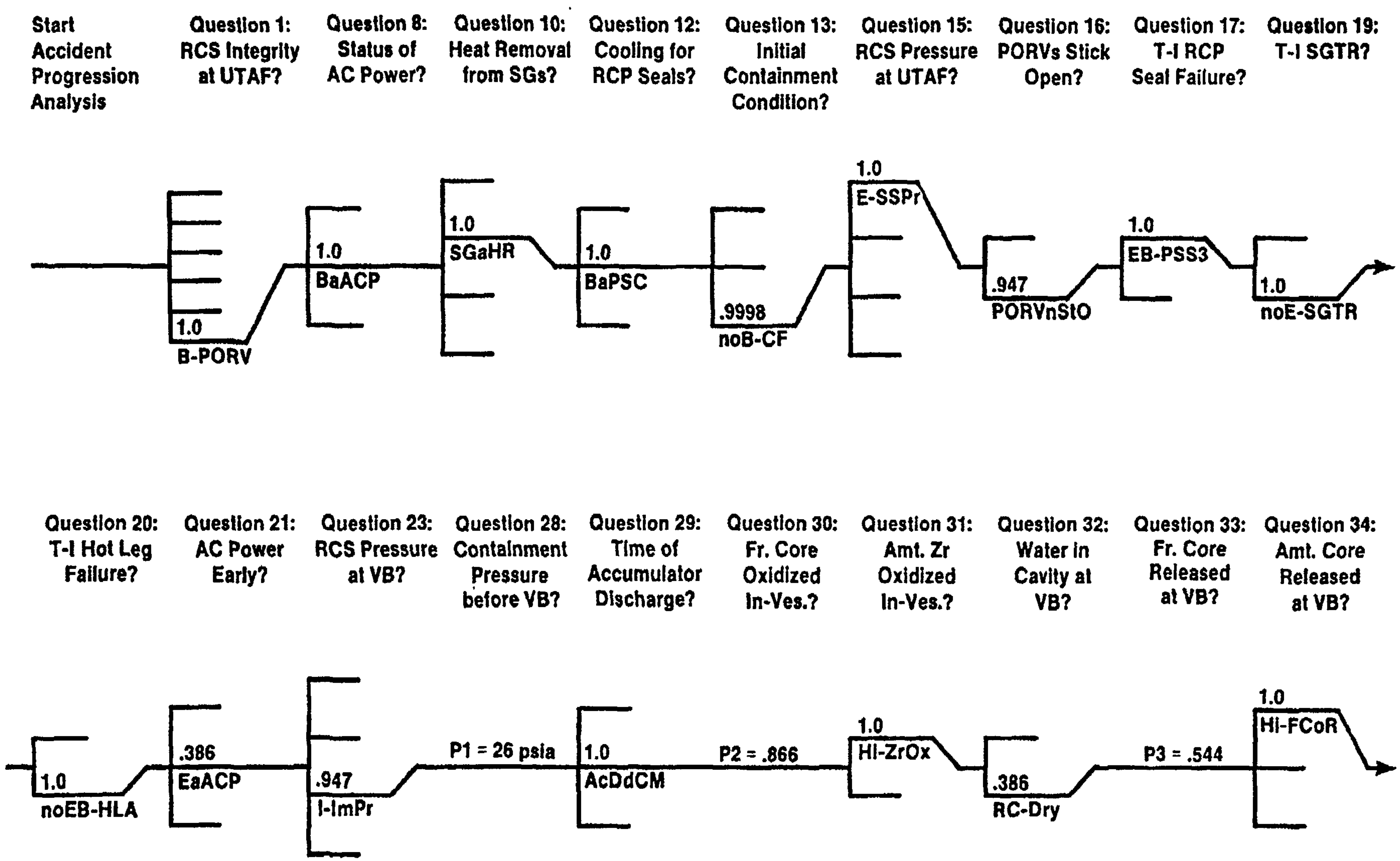


Figure B.4, Sheet 1 Simplified diagram of first part of Surry accident progression event tree. (The complete tree is too large to be depicted graphically. The complete tree is listed and discussed in Appendix A to Ref. B.1.)

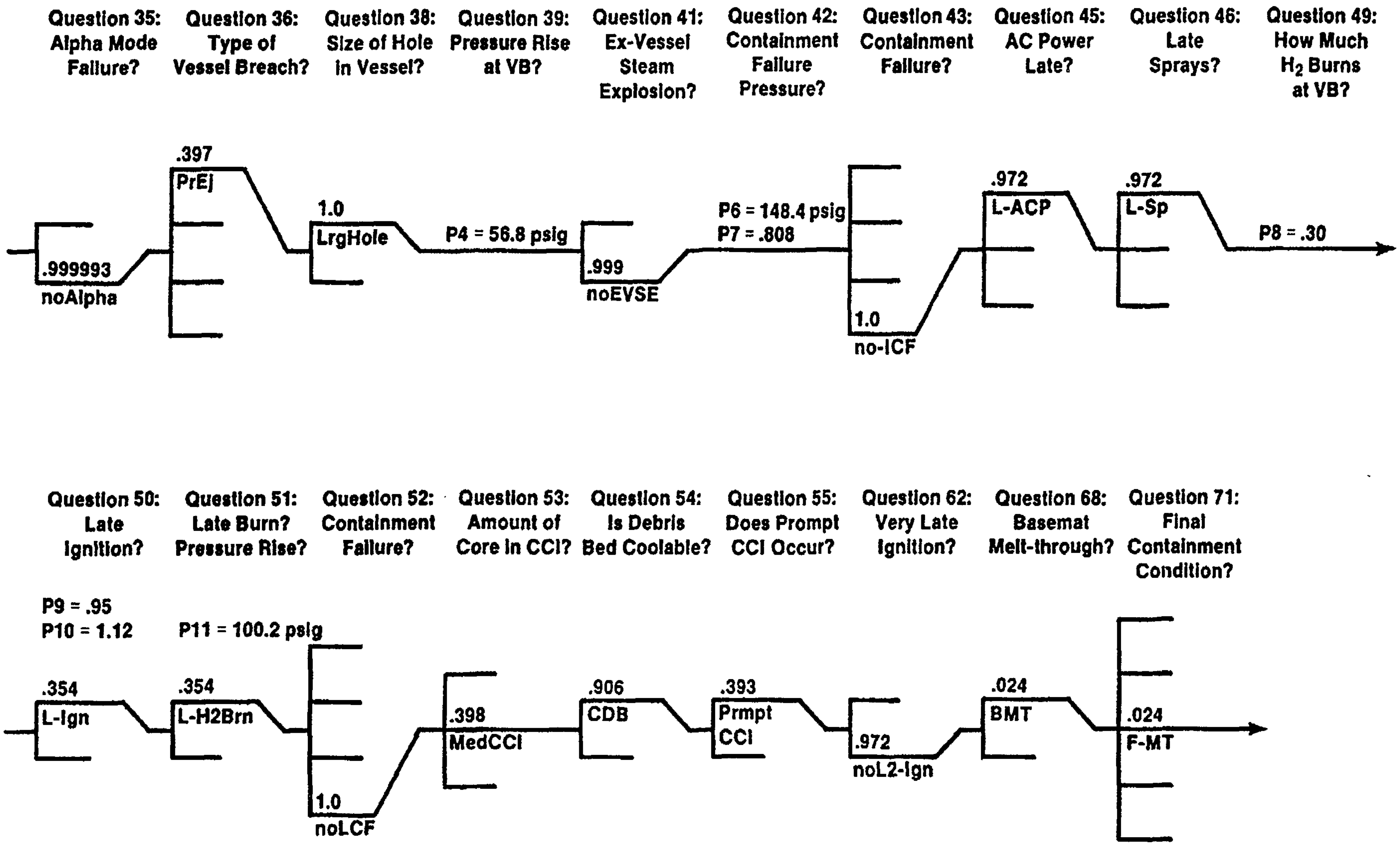


Figure B.4, Sheet 2 Simplified diagram of second part of Surry accident progression event tree.

A full listing of the questions in the Surry APET and detailed discussions of them may be found in Appendix A to Reference B.1. A discussion of how the event trees are defined and evaluated may be found in the methodology discussion in Reference B.5. Many of the branching ratios and parameter values used were determined by expert panels. More detail on this subject may be found in Part I and Part VIII of Reference B.6. EVNTRE, the computer code used to evaluate the APET, is documented in Reference B.7.

Figure B.4 shows the 38 questions displayed and discussed for this example. Only the path chosen for this example is followed from beginning to end in this figure. That is, at each question, only the branch chosen for this example continues on to the next question. In the complete evaluation of the APET for Observation 4 for PDS group 3, many of the branches shown as ending in Figure B.4 do terminate because they have zero probability.

However, many other branches shown as ending in Figure B.4 have nonzero probability and do propagate to the end of the tree. They are undeveloped in Figure B.4 because of space limitations.

In Figure B.4, which is best read in conjunction with Table B.2, the probability of the branch taken is shown above the line. It is the probability of that branch for the entire question and may have contributions from paths other than the one followed for this example. That is, all paths through the APET pass through every question. The probability of a particular branch in Figure B.4 reflects all paths, not just the one being followed in this example, and thus may be different from the probability for this path. Below the line in Figure B.4 is the branch mnemonic abbreviation. This is a succinct way of referring to each branch in the tree, and it is useful to have this information when relating this abbreviated Surry APET to the complete APET listed in Appendix A to Reference B.1.

The complete APET contains case structure, which is not shown in Figure B.4. By defining different cases for a question, different branch probabilities may be defined that depend on the branches taken at previous questions. For example, the branch taken at Question 15, RCS Pressure at UTAF, depends upon the RCS Integrity at UTAF, Question 1. This dependency is implemented by defining a number of cases. Case 2 is the system setpoint pressure (2500 psia) case for Question 15. One of the applicability conditions for Case 2 is that there be no break in the RCS at UTAF, i.e., that Branch 6 was taken at Question 1. For Case 2, the probability for the first branch, system setpoint pressure, is 1.0. Only the total branch probability for the path of interest can be shown in Figure B.4. There is no way to show branching probabilities as functions of the case structure for each question in a compact plot of the APET such as this.

As discussed above, for Observation 4, the accident frequency analysis determined that PDS TRRR-RSR had a frequency of 4.8E–7/reactor year. As PDS group 3 consists solely of TRRR-RSR, the frequency of group 3 is also 4.8E–7/reactor year for Observation 4. The APET is evaluated without regard to this frequency, and the result is a conditional probability for each path given the occurrence of PDS group 3. There are too many paths through the APET for us to be able to keep and treat each path individually. Therefore, paths that are similar as far as the release of fission products and risk are placed together in accident progression bins (APBs or just "bins") as explained in Section B.3.4. For the bin that results from the path followed in this example, denoted GFA-CAC-ABA-DA, the conditional probability is 0.017. The absolute frequency of this bin from PDS group 3 is the product of these two values, or 8.1E–9/reactor year.

Table B.2 lists the 38 questions shown in Figure B.4. These are the most important questions for following TRRR-RSR through the APET. The question is often given in abbreviated form to avoid using two lines. The "Branch Taken or Parameter Defined" column gives the branch taken at that question for the path being followed through the APET. If a parameter is defined in the question, the parameter number is given. The "Source of Quantification" column gives the source of the branch probability or the distribution for the parameter value for this question. PDS Def. means that the branch taken is determined by the definition of the PDS. Acc. Freq. means that the split between the branches at this question was determined in the accident frequency analysis. "Summary" indicates that the branch taken at this question is determined solely by the branches taken at previous questions. "Internal" means that the split between the branches, or the parameter value, was determined by the NUREG–1150 team of analysts, usually with assistance from other experts in various national laboratories. "Distrb." means that the probability of offsite power recovery was determined from distributions of power recovery as a function of time prepared for each reactor site. "Experts" indicates that the sampling is from a distribution determined by one of the expert panels that considered the most important issues for risk.

A discussion of each question follows in Section B.3.2. An expanded discussion of a few questions that were quantified by panels of experts follows in Section B.3.3. Finally, the binning of the results of the evaluation of the APET is discussed in Section B.3.4.

B.3.2 Discussion of APET Questions

Question 1. RCS Integrity at UTAF?

This question defines the state of the RCS at the start of the accident progression analysis. UTAF indicates the uncovering of the TAF, which is the nominal starting point for this analysis. The first character in the PDS definition, "T", indicates that TRRR-RSR has no failures of the RCS pressure boundary. Branch 6 is chosen; the water loss is through the cycling PORVs.

Question 8. Status of ac Power?

Branch 2 is chosen as indicated by the fourth character in the PDS definition. This is the "available" state, and it indicates that ac power will be available throughout the plant if offsite power is recovered after UTAF. The accident frequency analysis concluded that recovery of power from the diesel generators was of negligible probability. Recovery of offsite power in time to prevent core damage was considered by the accident frequency analysis. Recovery of offsite power after the ostensible onset of core damage but before vessel failure is more likely than not for TRRR-RSR. Recovery of power would allow the high-pressure injection system (HPIS) and the containment sprays to operate as these are also in the available state at UTAF. (The questions concerning emergency core cooling system (ECCS) and spray states are not listed in the interest of brevity.)

Question 10. Heat Removal from SGs?

As determined by the sixth letter of the PDS indicator, Branch 2 is chosen. This branch indicates that the steam-turbine-driven AFWS is failed, but the electric-motor-driven AFWS is available to operate when power is restored.

Question 12. Cooling for RCP Seals?

The last character of the PDS definition indicates that the accident frequency analysis concluded that there would be no cooling water flow to the RCP seals unless ac power was recovered. Thus, Branch 2 is taken.

Question 13. Initial Containment Condition?

The Surry containment is maintained below atmospheric pressure, at about 10 psia, during operation. The accident frequency analysis concluded that the probability of a pre-existing leak is negligible and that the probability of an isolation failure at the start of the accident was 0.0002. The more likely branch, no containment failure (Branch 3), is followed in this example.

Question 15. RCS Pressure at UTAF?

This question summarizes the information in the previous questions to determine the RCS pressure at the onset of core damage. As there is no break in the pressure boundary and no heat removal by the AFWS, the only water loss mechanism is the cycling PORVs: the RCS must be at the setpoint pressure of the PORVs, about 2500 psia. This pressure range is indicated by Branch 1.

Question 16. PORVs Stick Open?

After the core degradation process has proceeded for some time, the PORVs will be passing hydrogen and superheated steam and will be operating at temperatures well in excess of those for which they were designed. Based on the rate at which PORVs fail to reclose at normal operating conditions, the number of cycles expected, and allowing for degraded performance at high temperatures, failure of the PORVs was estimated to be of indeterminate probability. As there was no information available on PORV performance at temperatures considerably above the design temperature, a uniform probability distribution from 0.0 to 1.0 was used for this question. That is, the probability that the PORVs will stick open is equally likely to be anywhere between 0.0 and 1.0. In Observation 4, the value for PORV failure is 0.0528. This example follows the more likely branch, Branch 2, and the PORVs reclose.

Question 17. Temperature-Induced (T-I) RCP Seal Failure?

In normal operation, the seals around the shafts of the reactor coolant pumps (RCPs) are kept from overheating by a flow of relatively cool water. If this cooling flow is not available, the seal material may become too hot and fail. Failure of the RCP seals is important in both the accident frequency analysis and the accident progression analysis. In the accident frequency analysis, whether the seals fail, and when they fail, determines the time to UTAF and the RCS pressure at UTAF. In the accident progression analysis, if the seals have not failed before UTAF or whether the seals fail after UTAF may determine the RCS pressure when the vessel fails. The containment loads at VB are strongly dependent on the RCS pressure at that time.

As part of the accident frequency analysis, an expert panel was convened specifically to consider the failure of RCP seals. One of their conclusions was that the seals must be deprived of cooling for some time before failure is likely. In TRRR-RSR, UTAF occurs fast enough that the probability of RCP seal failure calculated in the accident frequency analysis was negligible. That is, by the time the seals have been without cooling long enough to have a significant chance of failure, the water level has dropped below the TAF and the consideration of the accident has passed to the accident progression analysis. In the accident sequence chosen for this example, then, seal failure only occurs in the accident progression analysis.

In the accident frequency analysis, the question of RCP seal failure is sampled zero-one; that is, in some observations a seal-failure branch has a probability of 1.0, and in other observations the no-seal-failure branch has a probability of 1.0. The accident progression analysis samples RCP seal failure the same way for consistency. For the entire sample, the probability of seal failure for this case where the RCS is at setpoint pressure (2500 psia) is 0.71. That is, of the 200 observations, 142 have seal failure and 58 have no seal failure. In Observation 4, the seals fail, so Branch 1 is taken. More discussion on the matter of RCP seal failure may be found in Section B.3.3 and in Reference B.8.

Question 19. Temperature-Induced (T-I) Steam Generator Tube Rupture (SGTR)?

After some period of core melt, the gases leaving the core region are expected to be quite hot. If these gases heat the steam generator (SG) tubes sufficiently, failure of the tubes may be possible. The expert panel that considered this issue concluded that T-I SGTR was possible but very unlikely if the RCS was at PORV setpoint pressure, and not possible if the system was at less than setpoint pressure (Ref. B.6). The failure of the RCP seals has reduced the RCS pressure below the setpoint of the PORVs, so, for Observation 4, there is no possibility of T-I SGTR, and Branch 2 is taken.

Question 20. Temperature-Induced Hot Leg Failure?

The very hot gases leaving the core region during melt may also heat the hot leg or the surge line to temperatures where failure is possible. The experts considered this failure much more likely than T-I SGTR, but only if the RCS was at, or near, the PORV setpoint pressure (Ref. B.6). The failure of the RCP seals has reduced the RCS pressure considerably below the setpoint of the PORVs, so, for Observation 4, there is no possibility of T-I hot leg or surge line failure. Branch 2 is taken.

Question 21. AC Power Early?

This question determines whether offsite power is recovered in time to restore coolant injection to the core before vessel failure. Distributions giving the probability of offsite power recovery as a function of time for the Surry plant are sampled to obtain the values used in this question (Ref. B.4). The times marking the beginning and the end of the time period considered were determined by considering the rate at which this accident progresses and the nature of the plant. For PDS TRRR-RSR, case 2 of this question is applicable; the time period is 0.5 to 2.0 hours after the start of the accident (LOSP). The average value for power recovery in this period for this case is 0.565. The value in Observation 4 is slightly above average at 0.614. If power is recovered during this period, it is likely that vessel breach will not occur. Because an example that proceeds to vessel breach is desirable, the less likely branch is chosen at this question. Branch 2 indicates that offsite power is not available in the plant during this period but may still be recovered in the future.

Question 23. RCS Pressure at VB?

This question determines the pressure in the RCS, including the vessel, just before the vessel fails. For the cases with large breaks in the RCS or with no breaks in the RCS, this pressure is well known. For cases with small (S2) or very small (S3) breaks, the pressure at VB depends upon the time between core slump and VB and the rate at which the pressure decays away following the steam spike at core slump. The RCP seal failure may be of large S3 or small S2 size although all are classed as S3 breaks in this analysis. Taking the range of break sizes and the likely delay between core slump and vessel breach into account, it was estimated that it was equally likely that the RCS pressure at VB would be in the High range, the Intermediate range, or the Low range (Ref. B.6). This question is sampled zero-one. In Observation 4, the Intermediate range is selected. Therefore, all of the accident, except the 5.3 percent with the PORVs stuck open, goes to Branch 3.

Question 28. Containment Pressure before VB?

The total pressure in the containment just after vessel breach consists of the baseline pressure before breach plus the pressure rise associated with the events at VB. (The pressure rise at VB is considered in Questions 39 and 40.) The containment pressure before VB is a function of spray operation and the magnitude of the blowdown from the RCS. The path followed in this example has no sprays and no large break. The results of detailed mechanistic simulation codes indicate that the containment atmospheric pressure will be around 26 psia in this case. Parameter 1 is set to 26 in this question. As the RCS pressure was above the accumulator setpoint when the core uncovered, and is below the setpoint (due to the RCP seal failure) at VB, the accumulators must have discharged during the core melt. Branch 2 is chosen.

Question 30. Fraction of Zr Oxidized In-Vessel?

The fraction of the Zr oxidized in the vessel before VB determines the rate of the core degradation process and temperatures of the gases leaving the core region. The amount of unoxidized Zr in the core debris leaving the vessel is also important in determining the nature of the core-concrete interaction (CCI). The expert panel provided distributions for this parameter for cases that depended upon the RCS pressure and the time of accumulator discharge (Ref. B.6). The path followed here has setpoint pressure in the RCS at the start of core melt and accumulator discharge during core melt. Observation 4 contains the value 0.866 for parameter 2 for this case. The median value for this distribution is 0.45; the value in Observation 4 is the 91st percentile value. As the fraction of Zr oxidized in the vessel is related to the temperature of the gas leaving the core by a known physical mechanism, the value for this parameter is as rank correlated with the probability of T-I hot leg failure as possible.

Question 31. Amount of Zr Oxidized In-Vessel?

The expert panel that considered containment loads at vessel breach gave distributions for two discrete levels of in-vessel Zr oxidation. Therefore, the oxidation fractions obtained from a continuous distribution in the previous question must be sorted into two ranges or classes. This is accomplished by Question 31; the fraction 0.40 divides the fraction of Zr oxidized in-vessel into High and Low ranges. The value of parameter 2 selected from the experts' distribution in the previous question, 0.866, falls in the High range; Branch 1 is taken.

Question 32. Water in Reactor Cavity at VB?

At Surry, the cavity is not connected to the containment sumps at a low level. The only way to get an appreciable amount of water in the cavity before VB is for the sprays to operate. As there is no electric power to operate the spray pumps in this blackout accident, the cavity is dry at VB in the path followed in this example. This is indicated by Branch 2.

Question 33. Fraction of Core Released from Vessel at Breach?

The expert panel provided a distribution for the amount of the core ejected promptly when the vessel fails (Ref. B.6). This is the fraction of the core that can be redistributed in the containment by the subsequent gas blowdown in a direct containment heating event. Observation 4 contains the value 0.544 for parameter 3. This is the 92nd percentile value. The median value is 0.27.

Question 34. Amount of Core Released from Vessel at Breach?

This question sorts the parameter values obtained from the experts' distribution in the previous question into three classes. The fraction 0.40 divides the High range from the Medium range for the fraction of core released at VB. The value of parameter 3 selected from the experts' distribution in the previous question falls in the High range; Branch 1 is chosen.

Question 35. Alpha-Mode Failure?

An alpha-mode failure is a steam explosion (fuel-coolant interaction) in the vessel that fails the vessel in such a way that a missile fails the containment pressure boundary as well. The distribution for this failure mode was constructed from the individual distributions contained in the Steam Explosion Review Group report (Ref. B.9) modified and updated as explained in Reference B.6. The alpha-mode failure probability in Observation 4 is 0.00011. This is considerably less than the mean value. It is so low that alpha-mode failures are truncated within the tree and do not appear in the results. The path selected for this example follows the more probable branch, Branch 2.

Question 36. Type of Vessel Breach?

This question determines the way in which the vessel fails. The possible failure modes are pressurized ejection, gravity pour, or gross bottom head failure. A panel of experts considered the relative likelihood of these possible failure modes (Ref. B.6). Their aggregate conclusion is sampled zero-one. The mode selected in Observation 4 is pressurized ejection (also denoted high-pressure melt ejection). For the whole sample, this failure mode is selected 60 percent of the time for the case where the vessel is at a high or intermediate pressure. Branch 1 indicates pressurized ejection upon vessel breach.

Question 38. Size of Hole in Vessel?

The experts who considered the loading of the containment at vessel breach gave pressure rise distributions that depend upon the size of the hole in the vessel. Hole size was also to have been determined by the experts, but no usable results were obtained. The hole size question was considered by a national laboratory expert in this field (Ref. B.6). He concluded that a small hole (nominal size = 0.1 m^2) was much more likely than a large hole (nominal size = 2.0 m^2). This question is sampled zero-one. Only 10 percent of the time is the large hole branch, Branch 1, selected as it was in Observation 4.

Question 39. Pressure Rise at VB?

The magnitude of the pressure rise in containment that accompanies vessel breach was determined by a panel of experts (Ref. B.6). In defining their distributions, the experts took into account all the pressure rise mechanisms, including vessel blowdown, steam generation, hydrogen burns, ex-vessel steam explosions, and direct containment heating. The pressure rise at vessel breach is treated in two questions, 39 and 40, in the Surry APET because the experts considering this issue defined so many cases. The large hole cases are considered in Question 39. The applicable case for the path being followed in this example is case 11: large hole, high fraction of the core ejected at breach, RCS at intermediate pressure, and dry cavity. For Observation 4, the 34th percentile value, 56.8 psig, was selected for this case. Parameter 4 is set to this value. This issue is discussed further in Section B.3.3.

Question 41. Ex-Vessel Steam Explosion?

This question determines whether a significant steam explosion occurs when the hot core debris falls into water in the reactor cavity upon vessel breach. In the path for this example, the cavity is dry, so there is no steam explosion, which is indicated by Branch 2.

Question 42. Containment Failure Pressure?

Two sampled variables are determined in this question. The first is the failure pressure of the containment. It is sampled from a distribution provided by structural experts who considered the Surry containment specifically. The other value is a random number between 0.0 and 1.0 that is used to determine the mode of failure if the containment fails. The value for the failure pressure in Observation 4 is 148.4 psig. This is the 93rd percentile value. The mean and the median failure pressures are around 126 psig. The random number selected for determining the mode of failure is 0.808 for Observation 4. Thus, in this question, parameter 6 is assigned a value of 148.4 psig and parameter 7 is assigned a value of 0.808. This issue is discussed further in Section B.3.3 and in Reference B.6.

Question 43. Containment Failure and Type of Failure?

This question determines if the containment fails shortly after vessel breach, and, if it fails, the mode of failure. This calculation is done in a FORTRAN "user function," which is evaluated at this question in the APET. Failure is determined by comparing the load pressure with the failure pressure (Refs. B.5 and B.6). In the user function, the failure pressure is converted to absolute pressure (163.1 psia) and the load pressure is calculated by summing the baseline containment pressure (parameter 1, see Question 28), 26 psia, and the pressure rise at VB (parameter 4, see Question 39), 56.8 psi. The load pressure, 82.8 psia, is less than the failure pressure so there is no containment failure at vessel breach in Observation 4. No containment failure is indicated by Branch 4.

Question 45. AC Power Late?

This question determines whether offsite power is recovered after vessel breach and during the initial period of CCI. The same basic distributions sampled in Question 21 are sampled again to obtain the probability of power recovery in this period. The average value for power recovery in this period for this case is 0.888. The value in Observation 4 is slightly above average at 0.927. This is the probability that power is recovered in this period if it was not recovered in the previous period, and it applies only to the fraction, 0.386, that did not have power recovered in the previous period. The most likely branch, Branch 1, is taken here; the path being followed in this example thus has power recovery at this point.

Question 46. Late Sprays?

As the sprays were available to operate at the start of the accident (Question 6, not discussed in the interest of brevity), they operate now that power has been restored throughout the plant. Branch 1 is selected for the path of interest.

Question 49. How Much Hydrogen Burns at Vessel Breach?

The restoration of power means that the sprays will begin to operate in the containment and that ignition sources will probably be present. The sprays will condense most of the steam in the containment and may convert the atmosphere from one that was inert because of the high steam concentration to one that is flammable. To determine the hydrogen concentration in the containment atmosphere during this period, the fraction of the available hydrogen burned at VB must be known. For the path of interest, pressurized ejection at VB with no sprays operating (the sprays were recovered after VB), there is a good chance that all or most of the the containment would have been effectively inert at VB because of the steam concentration. It was estimated internally that, on the average, 30 percent of the hydrogen produced in-vessel would burn at VB. Thus, parameter 8 is set equal to 0.30.

Question 50. Late Ignition?

This question determines the likelihood of ignition and sets the values of two parameters. The experts who considered ignition concluded that, if electric power were available, ignition was almost ensured in a matter of seconds or minutes, given that the atmosphere was flammable. In the path of interest, due to power recovery and the de-inerting of the containment, ignition is essentially ensured. Parameter 9 is the conversion ratio for hydrogen combustion, i.e., the fraction of the hydrogen that burns if there is ignition. The Surry containment is fairly open, and steam condensation due to the spray action is expected to make it well mixed at this time. The conversion factor is estimated to be 0.95, and parameter 9 is set to this value. Parameter 10 is the scale factor applied to the adiabatic pressure rise. A distribution was obtained for this value internally. The value for Observation 4 is 1.12, the 91st percentile value, and parameter 10 is set to this value. (Values of the scale factor greater than 1.0 account for the possibility that local flame acceleration will result in pressures greater than those calculated for deflagrations using the adiabatic assumptions. Global detonations were not considered at Surry.)

Question 51. Late Burn? Pressure Rise?

In this question, a FORTRAN "user function" is evaluated to determine if the containment atmosphere is flammable and, if it is, the total pressure that results from the ensuing deflagration. The amount of hydrogen in the containment is computed from the fraction of the Zr oxidized before vessel failure (parameter 2, see Question 30) and the fraction of the existing hydrogen that burned at vessel failure (parameter 8, see Question 49). This assumes that the ignition takes place before CCI or early in the CCI, i.e., before any appreciable amount of hydrogen has been generated by the CCI. The fraction of the hydrogen available that is consumed in the deflagration is given by the conversion ratio, parameter 9, read in the previous question. The baseline pressure is determined from the masses of the different gas species in the containment assuming a 50 percent steam mole concentration. The pressure rise calculated with the adiabatic assumptions is multiplied by the scale factor (parameter 10, Question 50) to obtain the final load pressure. For Observation 4 and the path of interest, 253 kg-moles of hydrogen burned resulting in an adiabatic pressure rise of 64.7 psia. The scaled pressure rise is 72.6 psia, and the total load pressure is 100.2 psia. Parameter 11 is set to this value.

Question 52. Containment Failure and Type of Failure?

This question determines if the containment fails several hours after vessel breach. If CCI occurs, failure at this time would be during the initial portion of CCI. This is designated the "Late" period. If the containment fails, the mode of failure is determined. This calculation is done in a FORTRAN "user function" as in Question 43. Failure is determined by comparing the load pressure with the failure pressure (parameter 6, see Question 42). The failure pressure is 163.1 psia. The load pressure is 100.2 psia, so there is no late containment failure for Observation 4. This is indicated by Branch 4.

Question 53. Amount of Core in CCI?

This question determines the amount of core available for CCI, should it take place. The path being followed has pressurized ejection at VB and a large fraction of the core ejected from the vessel. Pressurized ejection means that a substantial portion of the core material was widely distributed throughout the containment. For this case, it was estimated that between 30 and 70 percent of the core would be available to participate in CCI. This is the Medium range for CCI, indicated by Branch 2.

Question 54. Is Debris Bed Coolable?

This question determines if the core debris in the reactor cavity will be coolable, assuming that water is available. The path being followed has pressurized ejection at VB, so a substantial portion of the core material was widely distributed throughout the containment, and this portion of the core debris is likely to be coolable. It was internally estimated that, for this case, the probability of the debris in the cavity being in a coolable configuration is 80 percent (Ref. B.6). Note that for the debris to actually be cooled, in addition to the debris being in a coolable configuration, water must be present in the cavity at vessel breach and must be continuously replenished thereafter. This question only determines whether the debris configuration is coolable. The most likely branch, Branch 1, is followed for the example path, indicating that the debris bed configuration is potentially coolable. In the path being followed, the reactor cavity is dry at vessel breach, so whether the debris bed is coolable is a moot point.

Question 55. Does Prompt CCI Occur?

The reactor cavity is dry at vessel breach since the sprays did not operate before VB, so CCI begins promptly. While the sprays are recovered in the period following VB, they may not start to operate until some time after vessel breach. It was internally concluded that if the cavity was dry at VB, the debris would heat up and form a noncoolable configuration, and that, even if water was provided at some later time, the debris would remain noncoolable. Thus, prompt CCI occurs, and Branch 1 is chosen.

Question 62. Very Late Ignition?

Ignition leading to a significant hydrogen burn does not occur during the late portion of CCI, or after CCI, in the path being followed through the Surry APET for this example. Ignition occurred in the previous period and ac power has been available since that time. As an ignition source has been present since the late burn, any hydrogen that accumulates after the burn will burn off whenever a flammable concentration is reached. Burns at the lower flammable concentration limit will not threaten the Surry containment. Therefore, Branch 2, no ignition, is taken at this question.

Question 68. Basemat Meltthrough?

The path of interest has a medium amount of the core involved in CCI and the sprays start after VB and operate continuously thereafter. As the basemat at Surry is 10 feet thick, eventual penetration of the basemat by the CCI was internally judged to be only 5 percent probable for this case (Ref. B.6). Branch 1 is followed at this question. Although this branch indicates basemat meltthrough and is less probable than the other branch, it is taken because the source term and risk analyses are not of much interest if there is no failure of the containment.

Question 71. Final Containment Condition?

This is the final question in the Surry APET; it summarizes the condition of the containment a day or more after the start of the accident. Only the most severe failure is considered, that is, if the containment failed at vessel breach, a later basemat meltthrough would be ignored. In the path followed through the APET, there were no aboveground failures, so Branch 3 is selected, indicating basemat meltthrough.

B.3.3 Quantification of APET Questions by Expert Judgment

This section contains detailed quantification of three questions in the APET that were considered by the expert panels. The first is Question 17: probability of RCP seal failure. The second is Question 39: pressure rise in the containment at VB. The last is Question 42: containment failure pressure.

Temperature-Induced RCP Seal Failure

Question 17 determines whether there is a temperature-induced failure of the RCP seals. This failure mechanism is considered in the accident frequency analysis as well as in the accident progression analysis as it is important to both. The panel of experts that considered RCP seal failure was convened as part of the accident frequency analysis, and the results of that panel were used here as well. These experts concluded that the seal degradation depended primarily on the amount of time the seals had spent at elevated temperatures. For fast SBO accidents such as TRRR-RSR, the seal failure would not occur before UTAF. (It could, however, occur after UTAF.) Thus, for the accident sequence and PDS considered in this example, RCP seal failure is primarily of interest in the accident progression analysis.

The RCP seal is designed to allow a small amount of leakage (3 gpm) of primary coolant water during normal operation. The purpose of the leakage is to cool the shaft of the pump. This leak rate is well within the capacity of the normal makeup system. During an SBO with loss of the AFWS, there is no heat removal from the RCS and no cooling flow to the RCP seals. As the temperature and pressure of the reactor coolant system rise, the ability of the RCP seals to control leakage at acceptable levels determines whether the integrity of the RCS will be maintained. Significant leakage of RCS water through the seals will hasten the uncovering of the core and reduce the time available for restoration of ac power and core cooling.

The RCP seal is a complex multistage labyrinth seal that uses elastomer o-rings and free-floating seal plates. The integrity of the o-rings and the stability of the plates depend on the pressure in the RCS and the temperature of the water passing through the seal. Should the RCP seals fail, the size of the leak and the time of failure are functions of the combination of o-ring and seal plate failures in the seal assembly.

In the operating history of Westinghouse reactors, there has never been a seal failure caused by loss of seal cooling. However, there have been six incidents where seal cooling has been lost in U. S. Westinghouse reactors. In each case, the loss of seal cooling lasted less than 1 hour, which is the minimum time considered necessary to degrade the seal o-rings. While instability of the seal plates could occur at any time after the loss of seal cooling, this phenomenon has not been observed in any of the incidents to date. The o-ring material has been tested by both the Idaho National Engineering Laboratory (INEL) (Refs. B.10 and B.11) and the French national electrical utility, EDF. These tests showed that the o-ring material can be degraded when subjected to off-normal temperatures and pressures.

Both Westinghouse (Ref. B.12) and Atomic Energy of Canada Ltd. (AECL) (the latter under contract to the NRC) have performed extensive analyses of the performance of the RCP seal assemblies under off-normal conditions. Neither these tests or analyses, nor the incidents to date, have provided sufficient data for a quantitative probability model of RCP seal leak rate as a function of time after loss of cooling on which all parties can agree. Furthermore, the analyses by Westinghouse and AECL are proprietary. For these reasons, the resolution of this issue was delegated to a separate panel of three experts who were familiar with the problem and who had access to this proprietary information.

The three experts on this panel were:

Michael Hitchler, Westinghouse,
Jerry Jackson, U.S. Nuclear Regulatory Commission, and
David Rhodes, Atomic Energy of Canada Limited.

Which expert provided each distribution is not identified. The experts are described below as A, B, and C, which is not necessarily the order given above. They were asked to determine the probability of failure of the Westinghouse RCP shaft seals and corresponding leak rates under SBO conditions. More detail on this issue may be found in Reference B.8.

With the approval of the panel, the issue of RCP seal failures was decomposed into two questions:

1. What is the likelihood of the various combinations of o-ring and seal plate failures in a single RCP, and what is the resulting leak rate for each combination of failures?

2. What correlation, if any, exists between pumps for each combination of similar o-ring and seal plate failures?

The first question simplified the issue by focusing attention on the specific leak paths that might develop in a single pump. The second question expanded the scope of the panel's analysis to develop total leak rates for all of the RCPs. (Surry has three pumps.)

In resolving the first question, the experts agreed to develop a single event tree that would represent the set of all possible failure combinations and their corresponding leak rates for a single pump. A consensus was reached on the expected leak rate assigned to each set of failures. Each expert assigned his own probabilities to the various events of the event tree to arrive at his own estimate of single pump leak rate probabilities. To resolve the second question, each expert gave his judgments regarding the correlation of failures of event tree events between pumps. Then the experts' correlation elicitations were used to extend each expert's single pump model to obtain leak rates and their probabilities for all three pumps.

The single pump event tree is shown in Figure B.5. The probabilities on the tree are those for Expert A. It should be noted that some of the event probabilities on the tree are shown as functions of time. The experts concluded that degradation of o-ring elastomer material is dependent on the length of time that the o-rings are exposed to uncooled RCS water. The extension of Expert A's elicitation to a three-pump model is shown in Figures B.6 and B.7. Figure B.7 is the continuation of Figure B.6; it shows the various failure combinations for the first stage seal plates of the three pumps, based on Expert A's elicitation on the correlation of first stage seal failures. The five outcomes on Figure B.6 are passed on to Figure B.7, where the first stage o-ring and second stage component failures are shown. The result is 16 possible outcomes on Figure B.7, each with a time-dependent probability. Similar trees were developed for Experts B and C, but each expert's tree was unique because of differences in their elicitations.

To illustrate the method used, the path to outcome 5 in Figure B.7 will be followed. Expert A concluded that this was the most likely outcome; the path starts on Figure B.6, where the upper branch taken at top event B1 indicates success; that is, the first stage seal ring of pump 1 does not fail. The first stage seal rings also do not fail for pumps 2 and 3, so transfer path 1 is reached on Figure B.6. Transfer path 1 is the top entry path on Figure B.7, and Expert A concluded that this was the most likely transfer path (probability = 0.951). In the path to outcome 5 in Figure B.7, the first stage o-ring fails, so the lower branch is taken at

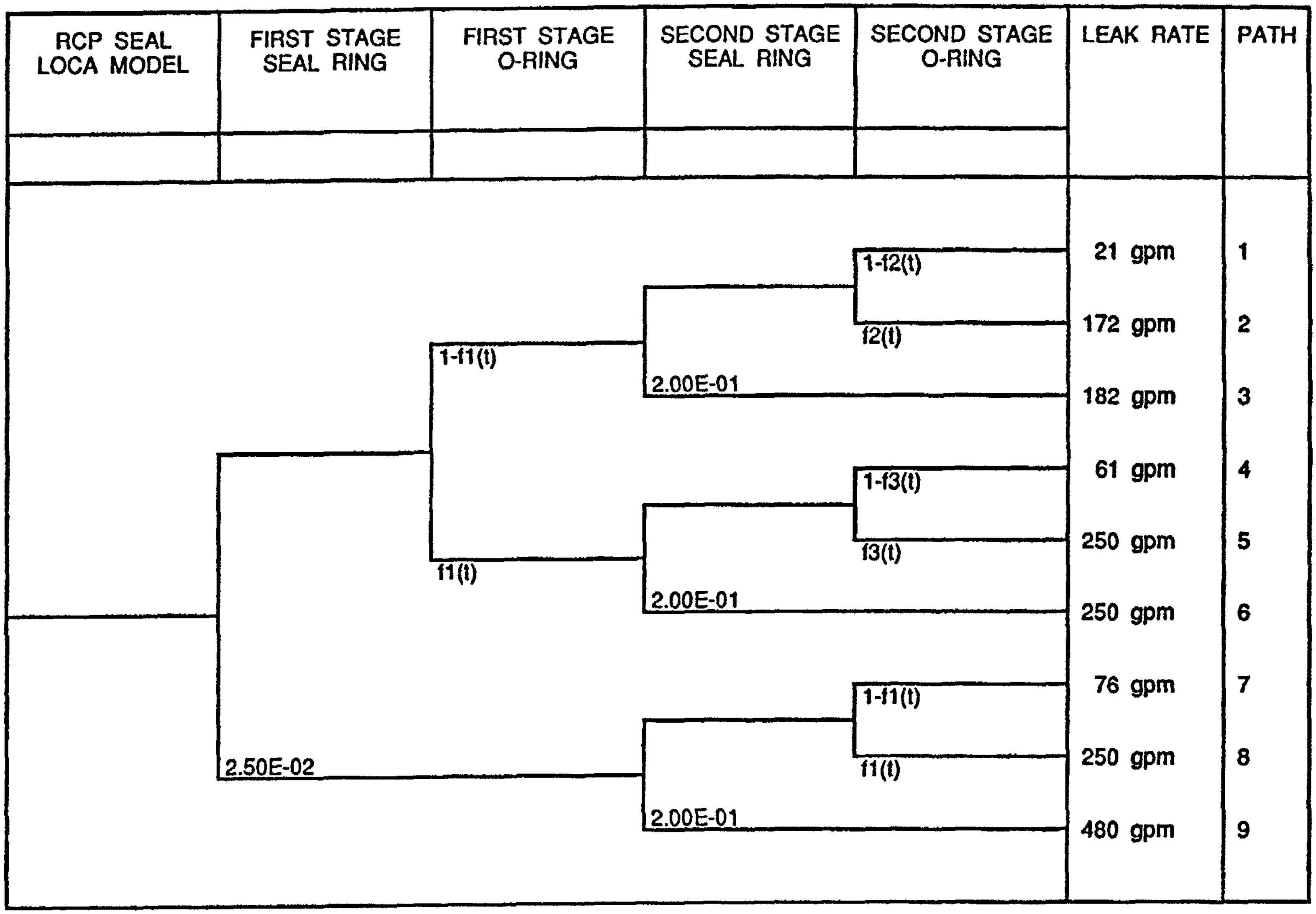


Figure B.5 Event tree used by all three experts in determining the probabilities of different leak rates for a single reactor coolant pump. The branch fractions shown are for Expert A. (This figure is adapted from Section C.4 of Ref. B.8.)

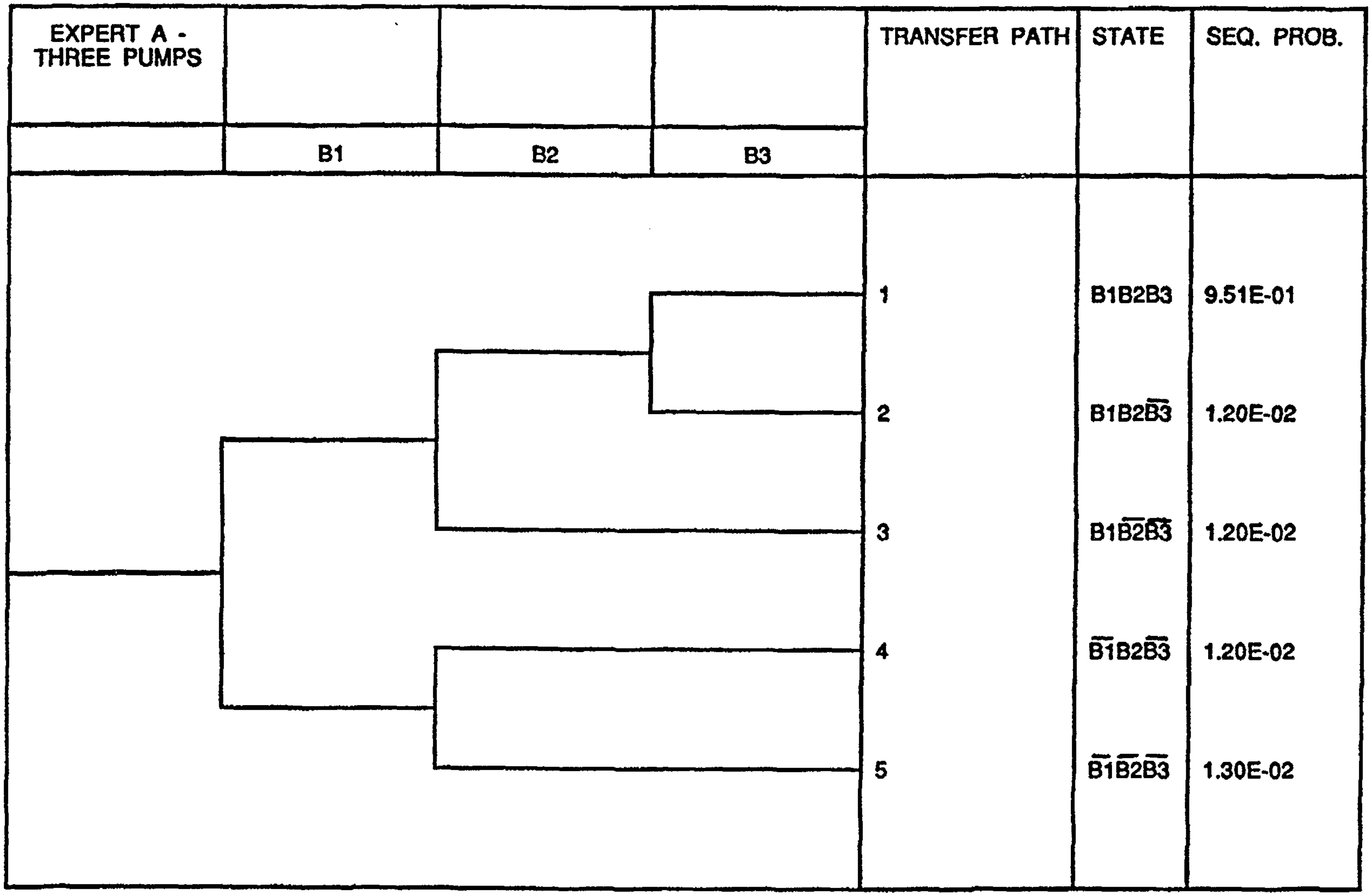


Figure B.6 First part of the event tree used by Expert A in determining the probabilities of different leak rates for all three reactor coolant pumps. The transfer paths indicate the entry point on the second part of this event tree. The top events concern failure of the first stage seal rings; B1 for pump 1, B2 for pump 2, and B3 for pump 3. In the State column, Bi indicates no failure of the seal rings, and $\overline{Bi}$ indicates failure of the seal rings, for pump i. (This figure is adapted from Section C.4 of Ref. B.8.)

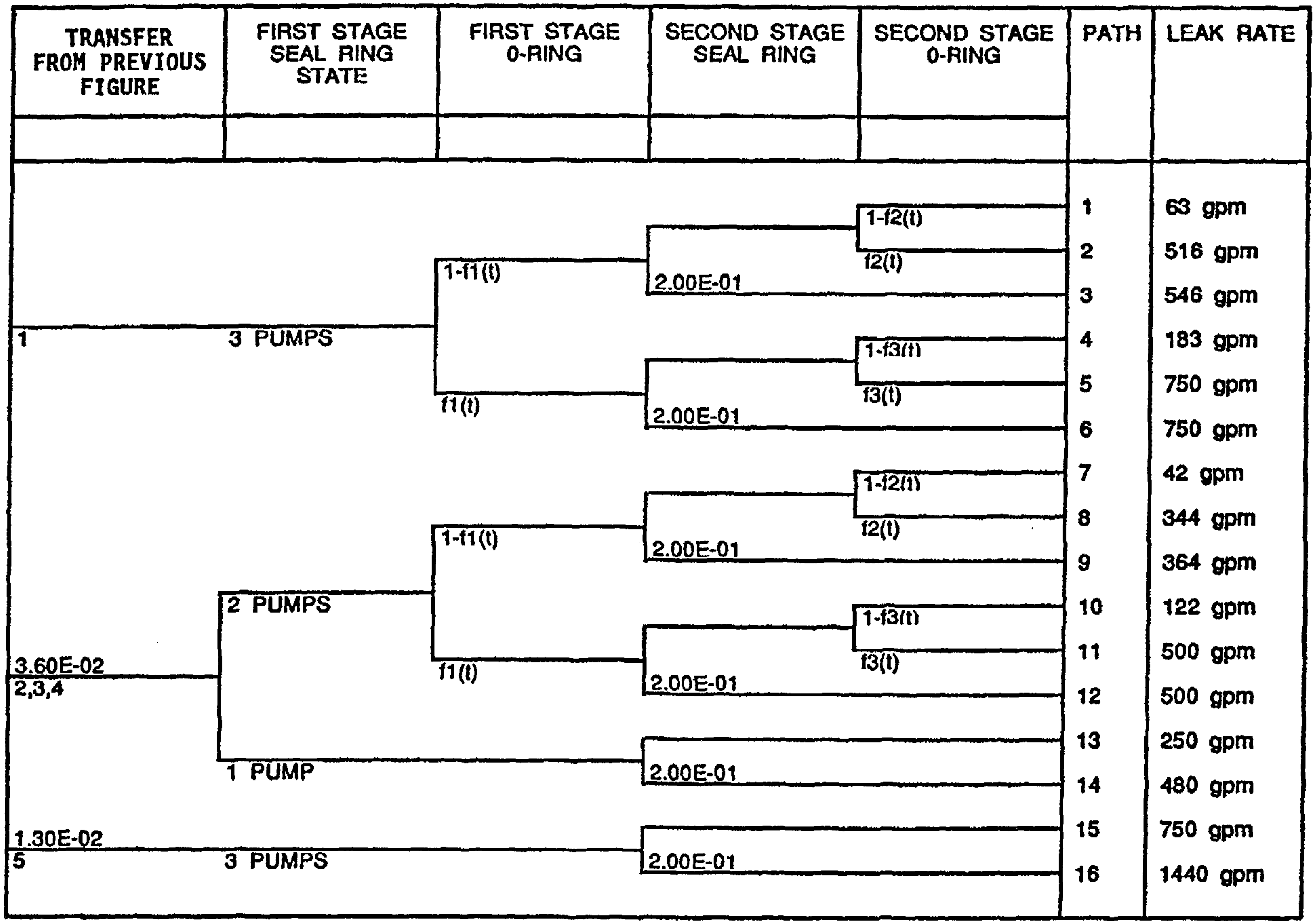


Figure B.7 Second part of the event tree used by Expert A in determining the probabilities of different leak rates for all three reactor coolant pumps. The transfer paths indicate the exit point from the first part of this event tree. (This figure is adapted from Section C.4 of Ref. B.8.)

top event "First Stage O-Ring." For the probability of this branch, Expert A developed a time-dependent model, denoted f1(t) on Figure B.7. Expert A was of the opinion that if the o-ring failed in the seal for one pump, they would fail on the other pumps as well, so the lower path at top event "First Stage O-Ring" represents the failure of the o-rings in all three pumps. At the next top event, the second stage seal rings do not fail, so the upper branch is taken. Expert A assigned a probability of 0.80 to this branch. At the final top event, the second stage o-ring fails in all three pumps. Expert A represented the probability of this failure by another function of time, denoted f3(t). Outcome 5 on Figure B.7 is a 250-gpm leak in all three pumps, for 750 gpm total. The probability of this outcome is a function of time, rising from zero at 1 hour after the loss of core cooling to 0.76 at 2.5 hours.

Experts A and C had fairly similar models for the single pump fault tree. Both treated failure of the first stage o-rings as a step function of time. Experts A and C concluded that failure would be virtually certain by 1.5 hours and 2.0 hours, respectively. Both reasoned that the first stage seal plates would be very reliable, but that integrity of the seals would be compromised by high probability failures of the first and second stage o-rings and second stage seal plates. Experts A and C judged that the likelihood of a second stage failure was somewhat dependent on the status of the first stage as first stage failure could compromise the ability of the second stage to succeed. Expert B's model was considerably more optimistic than those of Experts A and C. He also concluded that the probability of o-ring failure would be a function of time, but with a maximum value of 0.15 for the first stage and 0.50 for the second stage. His probability for seal plate failure was similar to those of Experts A and C, but he did not think that the second stage was dependent on the status of the first stage.

The most significant difference between Expert B and Experts A and C is the failure of.the o-rings as a function of time. Expert B thought that the o-rings would degrade slowly, and, by 4 hours after loss of cooling to the RCS, the RCS would have been depressurized by the operators. He believed that the o-rings would not fail in the depressurized environment. Experts A and C were of the opinion that the degradation of the o-rings would be so rapid that the question of depressurization within 4 hours was moot.

With respect to the correlation of o-ring and seal plate faults between pumps, Expert C's elicitation was the most simplistic. He concluded that similar components would behave similarly in different pumps. Thus, his three-pump leak rate model was exactly the same as his single-pump model, except that the leak rates of the single-pump model are multiplied by three. Experts A and B had significantly more complex elicitations for correlation of faults between pumps. Both had similar models for the correlation of first stage seal plate failures.

They both judged that the first stage seal plates could fail independently of each other, but they agreed to a simplifying assumption that, should similar components in any two pumps fail, the third pump would experience the same failure. Thus, Expert B's model for first stage seal plate failures is the same as that of Expert A in Figure B.6. The probabilities for several of the five outcomes for the first stage seal plate failure tend to be somewhat lower for Expert B than for Expert A. However, both models show the first outcome (all three first stage seals succeed) to be the most dominant outcome by far.

For the second stage, Experts A and B both concluded that the second stage o-rings would all fail in the same manner. But Expert A concluded that the second stage seal plates would all fail in the same manner, while Expert B judged that the second stage seal plates would fail independently.

The final RCP seal LOCA leak rates were calculated by averaging the leak rate probabilities of the three experts for various time intervals. Each expert's leak rate probabilities were given equal weight with respect to the others. The results are shown in Table B.3. (The o-rings in the RCP seals can be made from two types of material. The new material is much more resistant to degradation at high temperatures. The experts considered both types of material. All the pressurized water reactors (PWRs) considered for NUREG-1150 had o-rings made of the old material when these analyses were performed. Table B.3 shows only the results for seals with o-rings composed of the old, less heat-resistant material.)

The entries in the table give the probability of having the total leak rate shown at the times listed. Values in parentheses denote the probabilities that apply if the RCS is not depressurized.

Table B.3 Aggregate results for RCP seal failure with existing o-ring material.

Leak Rate (gpm)	1.5 (h)	2.5 (h)	3.5 (h)	4.5 (h)	5.5 (h)
63	0.31	0.29	0.27	0.27(0.26)	0.27(0.24)
183 to 224	0.15	0.04	0.05	0.05(0.06)	0.05(0.08)
372	0.008	0.005	0.005	0.004	0.003
516 to 546	0.0004	0.0003	0.0003	0.0003	0.0003
602 to 614	0.001	0.0	0.0	0.0	0.0
750	0.53	0.66	0.66	0.66	0.66
1440	0.004	0.004	0.004	0.004	0.004

The time dependence shown in Table B.3 could not be incorporated directly into the accident frequency analysis. Instead, eight RCP seal states were defined, and Table B.3 was used to derive probabilities for these states. Some of the less likely leak rates were combined with similar leak rates. The result for the Surry accident frequency analysis was:

Seal State	Probability	Total Leak Rate and the Time Seals Fail
1.	0.29	Design leakage (no failure)
2.	0.014	183 gpm at 90 minutes
3.	0.53	750 gpm at 90 minutes
4.	0.0043	1440 gpm at 90 minutes
5.	0.016	183 gpm at 150 minutes
6.	0.13	467 gpm at 150 minutes
7.	0.0040	561 gpm at 150 minutes
8.	0.016	183 gpm at 210 minutes

In the accident frequency analysis, each of these eight RCP seal states was considered separately as the different flow rates and different times of failure led to UTAF at different times. This level of detail could not be accommodated in the accident progression analysis. The APET considered only two RCP seal states: failed and not failed. Based on the results of the expert panel given above, the failed state has a probability of 71 percent. The failed state was designated as an S3 break (less than 2-in. diameter) even though the most likely flow rate, 750 gpm total, is in the lower end of the range of flows of the S2 breaks (0.5-in. to 2-in. diameter). This assignment, initiated in the accident frequency analysis, keeps the RCP seal failures separate from the stuck-open PORV cases, since the latter were all classified as S2 breaks and avoids having to split the RCP seal failures between the two break sizes.

As mentioned in the discussion of Question 17, the accident frequency analysis sampled this issue in the zero-one manner, i.e., there were eight states for the RCP seals: seven failure states and one design leakage (no failure) state. In each observation, one of these states was assigned a probability of 1.0 and the other seven were assigned a probability of 0.0. The relative frequency of each state in the entire sample corresponded to the aggregate distribution of the experts, e.g., 29 percent of the observations had the design leakage state with a probability of unity. The accident progression analysis samples RCP seal failure the same way for consistency except that there are only two states. The sample for the accident progression analysis consists of 200 observations, so 142 observations had the failure state selected and 58 had the no-failure state selected.

Pressure Rise at Vessel Breach

Questions 39 and 40 determine the pressure rise at VB in the Surry APET. Two questions are required because of the number of cases to be considered. Vessel failure usually causes the pressure to rise in the containment, sometimes dramatically. A number of mechanisms may contribute to this pressure rise: vessel blowdown, steam generation by the expelled debris, hydrogen combustion, ex-vessel steam explosions, and direct containment heating (DCH). The expert panel convened to consider the containment loads at VB concluded that the contributions of each of these mechanisms were generally not separable. Thus, the distributions for pressure rise provided by the experts include the contributions from all the pressure rise mechanisms. RCS blowdown and DCH cause significant loads to the containment only if the RCS pressure is 10 to 20 atmospheres or more above that of the containment at vessel breach.

After some discussion with the panel, the following case structure for Surry was adopted:

Case	RCS Pressure (psia)	Cavity Water	Sprays Operating
1	2000 to 2500	Full	Yes
1a	2000 to 2500	Half	Yes
1b	2000 to 2500	Dry	No
1c	2000 to 2500	Full	No
3	500 to 1000	Full	Yes
3a	500 to 1000	Half	Yes
3b	500 to 1000	Dry	No
4	15 to 200	Half	Yes

The panel defined eight subcases by considering the following variations (nominal values in parentheses):

Zr Oxidation—High (60 percent) and Low (25 percent),

Melt Fraction Ejected—High (75 percent) and Low (33 percent), and

Initial Hole Size—Large (2 m^2) and Small (0.1 m^2).

As there were eight cases, eight subcases for each meant that each expert provided 64 distributions for pressure rise at VB for Surry. Four members of the containment loadings expert panel considered the pressure rise at Surry. They were:

Kenneth Bergeron, Sandia National Laboratories,
Theodore Ginsberg, Brookhaven National Laboratory,
James Metcalf, Stone and Webster, and
Alfred Torri, Pickard, Lowe, and Garrick.

Expert A approached the problem by using the available CONTAIN (Refs. B.13 and B.14), MAAP (Ref. B.15), and Surtsey results (Refs. B.16 through B.19) to assess pressure rise distributions for three base cases. His base cases were chosen to represent the most severe pressure rises for the three different RCS pressure levels analyzed and were 1b, 3b, and 4. The low Zr oxidation, large hole, and large fraction ejected subcase was used for each base case.

For the middle portions of his base case distributions, Expert A placed the most reliance on the CONTAIN results as reported in NUREG/CR-4896 (Ref. B.20) and some subsequent calculations (Refs. B.21 through B.23). He obtained the extreme values from energy balance calculations. Using a PC spreadsheet program, he then adjusted these base cases for the effects of hole size, the amount of core ejected, and the fraction of Zr oxidized in-vessel to get values for the other 61 subcases.

Expert B also based his "best estimates" on CONTAIN calculations and on scaled experiments. The case for the 500 to 1,000 psia pressure range was taken as a base, and the cumulative distribution function (CDF) for that case was modified to obtain the CDFs for other cases. Expert B concluded that the presence of water in the cavity could either enhance or reduce the pressure, so the median for the wet cavity cases was kept the same as for the dry cavity cases, but the distribution was stretched at both ends. On the low side, an overabundance of water might reduce pressure by two bars. On the high side, calculations indicate the possibility of increasing pressure by one bar. Expert B took his high extreme values from a one cell adiabatic equilibrium code he had written to analyze Zion and Surry. While calculating the low side of the distribution, he considered phenomena that might reduce pressure, such as larger drop diameter or faster trapping.

Dependence on the extent of Zr oxidation, the VB area (hole size), and the fraction of melt ejected was also considered by Expert B. CONTAIN calculations (Refs. B.20 through B.23) have indicated that there is little dependence on previous Zr oxidation, probably because of oxidation starvation in the cavity. The effect of greater hole area is to give higher pressure rises across the entire distribution because the gas would exit with higher velocity. The effect of fraction of core ejected was handled by scaling the base case ratio of final to initial pressure.

Expert C used HMC calculations (Ref. B.24), CONTAIN calculations (Refs. B.20 through B.23), and MAAP calculations (Ref. B.15). He tabulated the cases described in the issue description and applied the code results that appeared to be the most applicable to each case. He was forced to modify the code results in many instances to account for differences between the initial conditions in the code calculations and the case under consideration. Expert C used the HMC calculations for the several cases in which there was water in the cavity and considered the highest pressures calculated by HMC to be the upper bounds of his distributions. The pressure rise without direct heating formed his lower bound.

Expert C relied on CONTAIN and MAAP results in cases in which the cavity was dry. CONTAIN calculations with unconditional hydrogen burn and default burn were averaged and used for the upper part of his distributions, while the MAAP results were used for the lower part of his distributions. Although he believed the CONTAIN calculations to be consistently above the median, he considered the results quite credible. From CONTAIN sensitivity calculations, Expert C was able to estimate the effects of changes in initial conditions, and using these estimates he obtained distributions for the subcases for which no HMC, CONTAIN, or MAAP results were directly applicable.

Expert D used CONTAIN results (Refs. B.20 through B.23) as the basis for his analysis because CONTAIN is currently the only code that has a DCH model. For his base case, he took the high-pressure case with a large fraction of melt ejected (75 percent) and a small initial hole. No further definition of the base case was necessary because Expert D was of the opinion that the effects of co-dispersed water should not be included and that the fraction of the Zr oxidized in-vessel was not particularly important. (CONTAIN runs in which the in-vessel oxidation was varied showed small differences in pressure rise.) To obtain his distribution for this base case, he started from results of the 18-node Surry model with unconditional hydrogen burn as defined in References B.21 and B.22. Expert D adjusted these results to account for alternate particle sizes, an alternate trapping model, and the effect of the thin steel in the containment on peak pressure.

For the small hole cases, Expert D adjusted the CONTAIN pressures upward somewhat since there is the possibility that more than one penetration may fail at or about the same time. For the cases with the sprays operating, he reduced the pressures about 1.5 to 2.5 bars below the pressures in the equivalent cases without the sprays operating. Expert D concluded that changing the particle size assumed in CONTAIN could only decrease the pressure rise. If the particle size assumed in the CONTAIN calculations (1.0 mm) is increased, the pressure rise will decrease because the material in the center of the particle will not have reacted before the particle is quenched. If the particle size is decreased from that assumed in CONTAIN, there is a negligible effect since all the metal in the particle is already reacting. CONTAIN assumes that the core debris distributed throughout the containment during the blowdown phase of the DCH process is homogeneous. Expert D expects the entrained material to be richer in oxides than a homogeneous mixture, which would decrease the pressure rise somewhat. He also pointed out that, when DCH occurs, only a very small portion of the hydrogen pre-existing in the containment or produced during the high-pressure melt ejection (HPME) can be expected to remain unburned after the event is over.

Results for all 64 subcases may be found in Reference B.6. Statistical tests on the 64 subcases showed that many of them could be combined, that the differentiation made on the fraction of Zr oxidized in-vessel could be dropped, and that all the subcases for the low-pressure case (case 4) could be consolidated. The result of the statistical analysis was that there were 13 distinct cases for Surry.

However, the dividing point between high and low fraction ejected used by the expert panel on containment loads, 50 percent ejected, was very near the high end of the aggregate distribution given for fraction ejected by the in-vessel experts. As defined by the loads panel, the high-fraction-ejected subcase has the fraction ejected greater than 50 percent with a 75 percent nominal value and the low-fraction-ejected subcase has the fraction ejected less than 50 percent with a 33 percent nominal value. The aggregate distribution from the in-vessel panel for core fraction ejected has a maximum value of 60 percent, and the probability that the fraction ejected will exceed 50 percent is only about 11 percent.

Not wishing to place 89 percent of the samples in the low-fraction-ejected subcase of the loads panel, and as the "high-low" division was more coarse than necessary, the core-fraction-ejected distribution of the in-vessel panel was divided into three ranges: 0 to 20 percent, 20 to 40 percent, and 40 to 60 percent. The pressure rise distributions from the loads panel were then adjusted to provide pressure rises for these three ranges.

For the 0 to 20 percent ejected range, the average of the low-fraction-ejected results and the case 4 results (RCS pressure < 200 psia) were used. As the low-fraction-ejected case had 33 percent (nominally) ejected, and case 4 had, in effect, no core ejected at high pressure, this appeared to be appropriate. For the 20 to 40 percent ejected range, the low-fraction-ejected results from the loads panel were be used directly since the nominal value used by the loads experts was 33 percent ejected. For the 40 to 60 percent ejected range, the loads low-fraction-ejected distributions and high-fraction-ejected distributions were averaged. The average of the nominal fractions ejected is 54 percent, which is reasonably close to the center of this range. This treatment of the distributions was discussed with, and approved by, a member of the containment loadings expert panel.

This expands the number of cases for Surry from 13 to 19. Plots of the aggregate distributions for these 19 cases are contained in Reference B.6.

For the example being followed through Observation 4, the path through the APET went to case 11 of Question 39. This case has intermediate pressure in the RCS at VB, dry cavity, large hole, and high (40 to 60 percent) core fraction ejected at breach. This is case 3b of the loads panel. The statistical analysis found no significant differences between the expert's results for cases 1, 1a, and 3b. As explained above, the 40 to 60 percent ejected distribution is the mean of the loads panel low-fraction-ejected aggregate distribution and high-fraction-ejected aggregate distribution. Figure B.8 shows the distributions of the four experts and the aggregate for case 3b, large hole, for both core fractions ejected. Also included in Figure B.8 is a plot showing the two aggregates for case 1/1a/3b, large hole, and the aggregate for case 4, as received from the loads panel, and the three aggregate distributions derived therefrom for the three ranges of the in-vessel panel distribution for core fraction ejected. The distribution for 40 to 60 percent ejected was used in the sampling process to obtain the value of 56.8 psig, the 34th percentile value, used for Question 39, case 11, in Observation 4.

Containment Failure Pressure

The value for the containment failure pressure is determined in Question 42. The Surry containment is a cylinder with a hemispherical dome roof. Both the cylinder and the dome are constructed of reinforced concrete. The foundation is a reinforced concrete slab. The containment is lined with welded 0.25-inch plate steel. The containment is maintained below ambient atmospheric pressure, at about 10 psia, during operation. The design pressure is 45 psig. The free volume is about 1,850,000 cm^3. A section through this containment is shown in Figure B.9.

A panel of structural experts was convened to determine the loads that would cause containment failure at Surry and the other plants. As the probability of a global detonation in the Surry containment was considered to be quite small, only static loads were treated for Surry. Such loads would result from the pressure rise that accompanies VB or a deflagration. Typical pressure rise times would be on the order of a few seconds, which is longer than the containment response time.

Four members of the structural expert panel considered failure pressure and failure mode for the Surry containment. They were:

Joseph Rashid, ANATECH Research Corp.,
Richard Toland, United Engineers and Constructors,

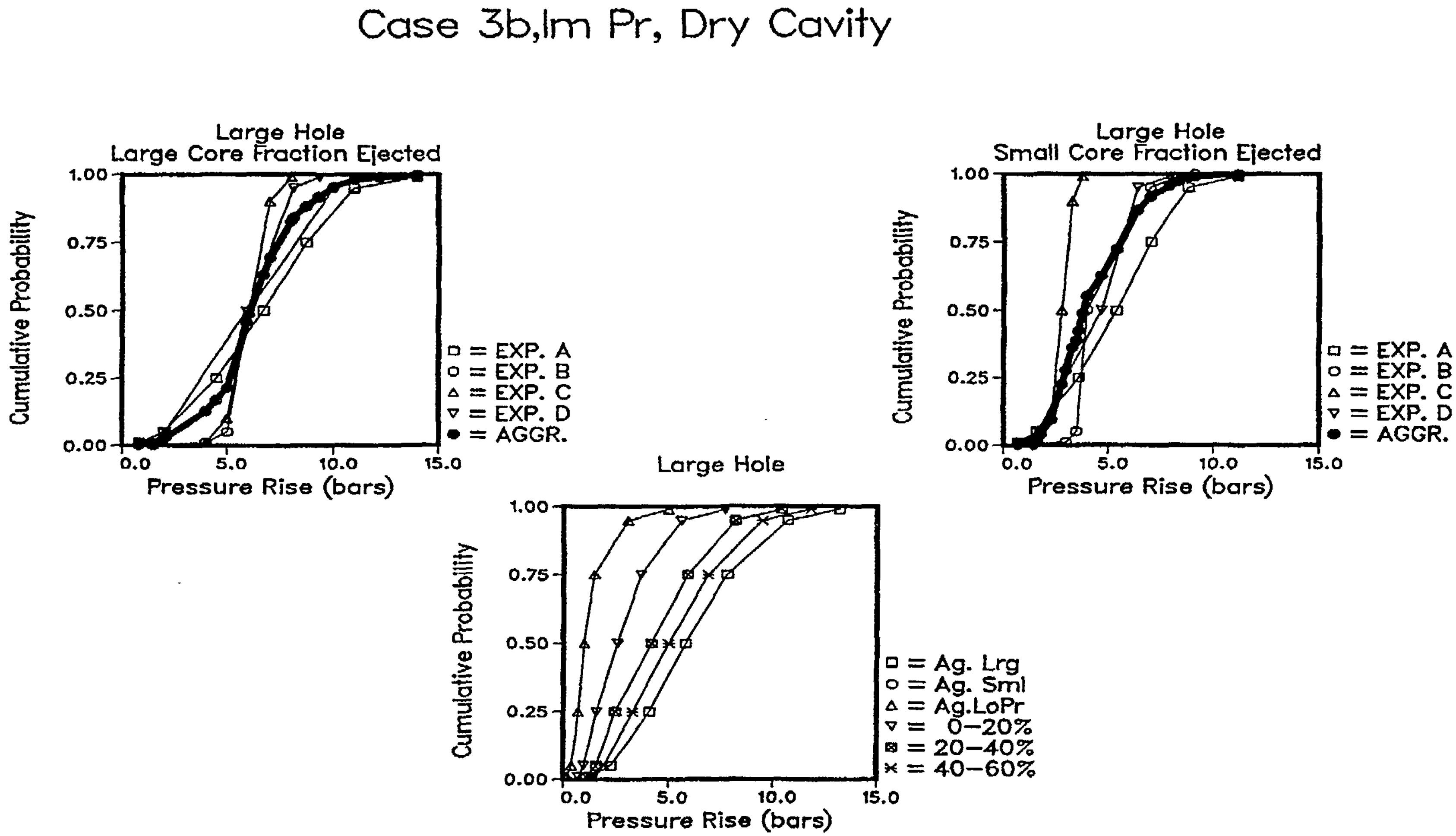


Figure B.8 Results of expert elicitation for pressure rise at vessel breach for Surry. [The pressure rise is shown for the case where the RCS is at intermediate pressure (200–600 psia), the cavity is dry, and the sprays are not operating when the vessel fails by the formation of a large hole. In the top two plots, the first four distributions are those given by the experts and the fifth plot is the aggregate distribution. The lower plot shows the aggregate distributions provided by the experts and the distributions actually used in the APET. The first curve is the large-core-fraction-ejected aggregate from the upper left plot, and the second curve is the small-fraction-ejected aggregate from the upper right plot. The third curve is the aggregate for low pressure (less than 200 psia) in the RCS at VB. The last three curves are the actual distributions used in the APET evaluation. The distribution for 20–40 percent ejected is coincident with the second curve (small fraction ejected).]

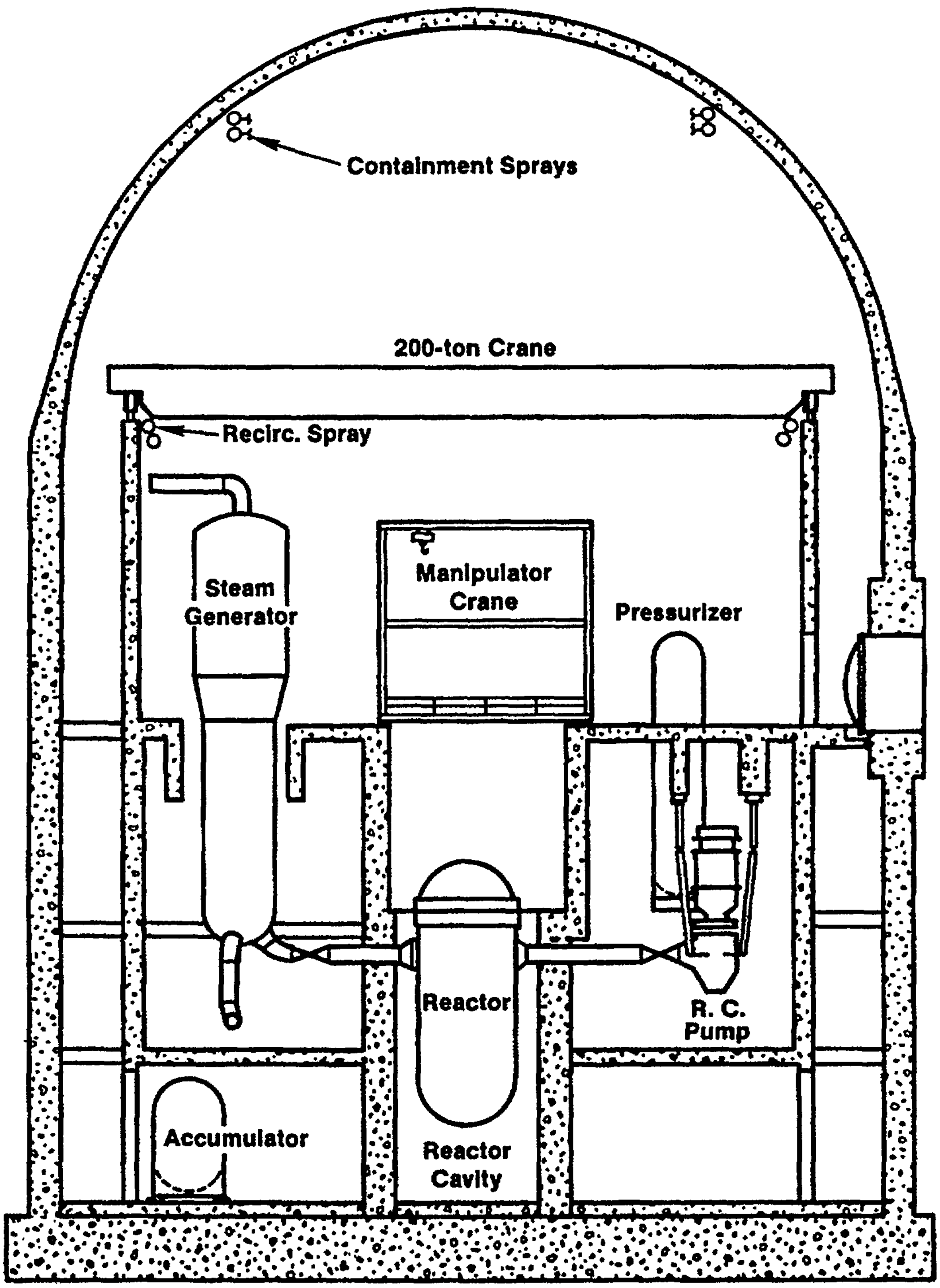


Figure B.9 Simplified schematic of Surry containment.

Adolph Walser, Sargent and Lundy, and
J. Randall Weatherby, Sandia National Laboratories.

They did not differentiate on the basis of failure location since any failure location except shear at the basemat-cylinder junction would result in a direct path to the outside. The reinforcing and concrete details in this junction area were such that three of the four experts ruled out failure in this location. (The fourth expert did not specify failure location explicitly.)

The experts treating Surry did not perform any extensive new calculations. They reviewed the previous detailed calculations and the drawings of the containment, including reinforcing details, penetrations, and hatches and airlocks. Their experience allowed them to judge how comprehensive the previous analyses had been and, when there were conflicting results, which result was more likely to be correct.

Expert A based his conclusions on previous analyses of the Indian Point containment (Ref. B.25), the Surry containment (Ref. B.26), and the drawings of the Surry containment structure. He considered four failure modes: hoop failure in the cylinder, hoop failure in the dome, shear failure at the cylinder-basemat junction, and penetration failure. Meridional failure in the dome will be similar to the hoop failure and was not considered explicitly.

On the basis of the detailed drawings and some brief calculations, Expert A concluded that the cylinder-basemat junction was a very strong region and ruled out failure at this location. He looked briefly at the equipment hatch, personnel airlock, pipe penetrations, and electrical penetrations and concluded that they were sufficiently similar to those at Zion that failures at these locations were of relatively low probability. At low and medium stress levels, with the liner taken into account, Expert A concluded that the dome is stronger than the cylinder. However, the way the rebar was placed at the top of the dome led Expert A to question whether the dome would be stronger than the cylinder at high stress levels.

For the cylinder, the hoop stress can adequately be calculated by hand. In this manner, Expert A concluded that general yield of the rebar would occur at 119 psig, which agrees with the Stone & Webster analysis (Ref. B.26). This is the lowest pressure for which Expert A would expect to find any chance of failure; at this pressure the cylinder wall has moved out 2 inches. Expert A then calculated that 2 percent hoop strain corresponded to 150 psig, including the effects of strain hardening of the rebar. At this level of strain, he concluded that liner tear is certain at discontinuities such as those around penetrations and stiffener plates. Further, concrete cracking at 2 percent general strain will have removed much of the liner support. At 2 percent strain, the cylinder wall has moved out 16 inches.

In summary, Expert A concluded that the containment would fail between 120 and 150 psig and that the probability density of failure was uniform in that range. His median value was 135 psig.

Expert B based his analysis on the Stone & Webster study of the Surry containment (Ref. B.26), studies of other plants such as Indian Point 2 and 3 (Ref. B.25), Seabrook (Ref. B.27), and the test of the 1/6-scale model at Sandia (Ref. B.28). Expert B's hoop membrane stress analysis showed that there would be general yielding of the shell and rebar at 120 psig, and that rebar that just met the minimum requirements would fail at 144 psig. If all the rebar were of average strength, the rebar would fail at 166 psig.

Based on the reference analyses and this information, Expert B placed his median failure pressure at 120 psig and his upper bound at 165 psig. He placed his lower bound at 70 psig. This took into account the possibility of faulty rebar joints or liner tears due to stress concentrations around openings.

Expert C based his conclusions on an analysis of the mid-section of the cylindrical portion of the containment. His study of the drawings and the results of other analyses led him to conclude that this was the weakest portion of the containment. His conclusions about the leak failure mode and liner tear are largely based on the 1/6-scale model test at Sandia (Ref. B.28). Once a liner tear has developed, it is difficult to see how it could be kept from expanding with a continued increase in pressure.

Expert C concluded that failure was most likely in the 135 to 147 psig range, and he placed 70 percent of his probability there. He placed 10 percent of his probability below 135 psig to allow for his uncertainty about the actual rebar properties.

Expert D's analysis led him to conclude that a leak was certain to develop by 130 psig. At this pressure the rebar has yielded considerably and reached a strain of about 1 percent. He would expect leaks to develop because of dislocation at discontinuities (Ref. B.29). There is no possibility of a leak developing at pressures below 75 psig. This value was obtained by hoop membrane stress analysis assuming that the liner is at its yield stress of 35,000 psi. If the liner and the hoop reinforcement are both at their respective yield stress, which is 55,000 psi for the reinforcement and 35,000 psi for the liner, the pressure would be 110 psig. Expert D took 110 psig to be his median value for leaks. He noted that the specified minimum yield strength is 55,000 psi for the reinforcement and 35,000 psi for the liner.

Expert D took the lower threshold for rupture to be 140 psig, which was determined by a local effects analysis of the discontinuity at the basemat-cylinder junction (Ref. B.30). He expected that a crack would open at this junction for a substantial portion of the circumference. Although the crack might be very small, it would be long enough to depressurize the containment in less than 2 hours. He concluded that rupture was certain when the main reinforcement reaches its specified minimum ultimate strength. For the Surry containment, Expert D considered catastrophic rupture to be impossible.

Figure B.10 shows the distributions of the four experts and the aggregate distribution for total cumulative failure probability. Experts A and C concluded that there is little or no chance of failure by 120 psig, while Expert D concluded that failure is almost certain by 120 psig. The aggregate distribution for the failure pressure of the Surry containment was formed by weighting equally the individual distributions of the four structural experts who considered this issue.

From the information provided by the experts, aggregate distributions were also obtained for the mode of containment failure. Because the containment did not fail in this example, the question of the mode of failure is not discussed here. The results of the experts' elicitations on the mode of failure may be found in Reference B.6, and the method used to determine the mode of failure in the APET is discussed in References B.1 and B.5.

For use in Question 42, a value for the containment failure pressure is obtained from the aggregate distribution by a random sampling process. The value for the failure pressure in Observation 4 is 148.4 psig. This is the 93rd percentile value. The mean and the median failure pressures are around 127 psig.

B.3.4 Binning Results of APET

There are so many paths through the APET that they cannot all be considered individually in the source term analysis. The results of evaluating the APET are therefore condensed into accident progression bins (APBs) or just bins. The computer code, EVNTRE, that evaluates the APET places the paths through the tree in the bins as it evaluates them. At Surry, each bin is defined by 11 characteristics of the path taken through the event tree. (For the summary discussions contained in Volume 1 of NUREG-1150, these detailed bin definitions were collapsed into a smaller set.) The bin definition provides sufficient information for the algorithm used in the source term calculation. The binning method provides the link between the accident progression analysis and the source term analysis, which calculates the fission product release.

The computer input file that contains the binning instructions is referred to as the "binner." It is listed and discussed in detail in Reference B.1. A discussion of the binning process may be found in the methodology discussion in Reference B.5.

In computer files, the bin is represented as an unbroken string of 11 letters. For presentation here, hyphens have been inserted every three characters to make the bin more readable. A given letter in a given position has a definite meaning. For example, the first characteristic primarily concerns the time of containment failure. If the first character in the bin designator is a "C", containment failure before VB is indicated.

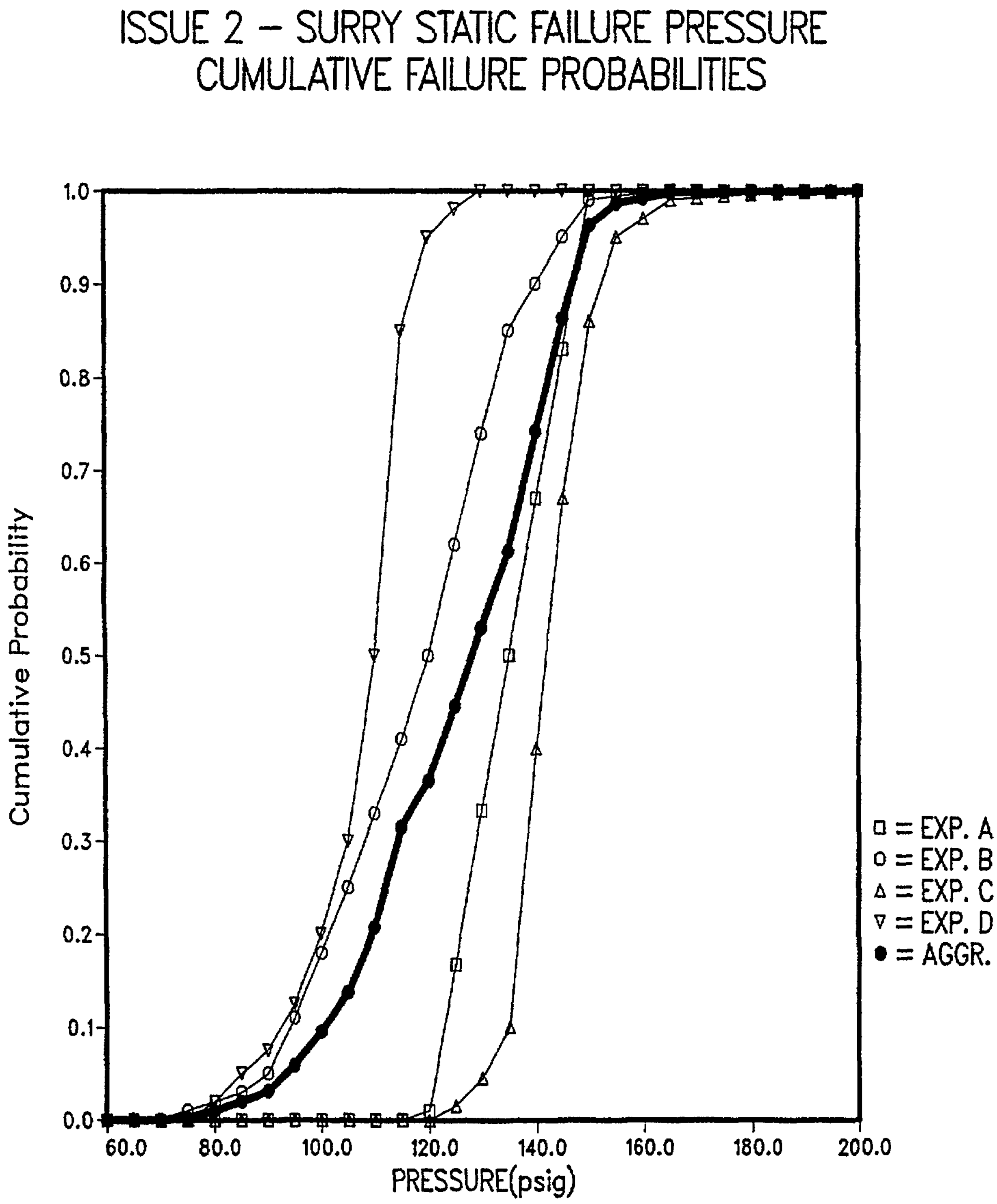


Figure B.10 Results of expert elicitation for static failure pressure of Surry containment. (The first four curves are the distributions of the four experts, and the fifth curve is the aggregate distribution.)

For PDS group 3, Observation 4 produced 22 bins. These resulted from all the paths that remained above the cutoff probability (1.0E–7). For example, the alpha-mode probability was so low in Observation 4 that all the alpha-mode paths were truncated and there are no bins with alpha-mode failures of the containment. The most probable bin (0.55) in Observation 4 is HDC-CFC-DBD-FA, which has no VB and no containment failure. It results from offsite ac power recovery before the core degradation process had gone too far.

Bin GFA-CAC-ABA-DA results from the path followed through the tree in this example for Observation 4. It is the most likely (0.017) bin for Observation 4, which has both VB and containment failure. Basemat meltthrough occurred a day or more after the start of the accident. Containment failure in this time period is indicated by the character "G" in the first position. The other ten characteristics are defined in a similar manner. For bin GFA-CAC-ABA-DA, each character in the bin designation has the following meaning:

G - Containment failure in the final period;

F - Sprays only in the Late and Very Late periods;

A - Prompt CCI, dry cavity;

C - Intermediate pressure in the RCS at VB;

A - High-pressure melt ejection (HPME) occurred at VB;

C - No steam generator tube rupture;

A - A large fraction of the core was available for CCI;

B - A high fraction of the Zr was oxidized in-vessel;

A - High amount of core in HPME;

D - Basemat meltthrough; and

A - One effective hole in the RCS after VB.

The binning follows directly from the path through the APET with one exception. At Question 53, the amount of core in CCI was determined to be medium (Branch 2). The binning above shows that the fraction of the core involved in the CCI is large. The reason for this is that the computer code that performs the source term analysis, SURSOR, subtracts the amount of the core involved in HPME from the total passed to it. To avoid subtracting this amount twice, whenever HPME occurs, the amount of the core involved in CCI is set to Large in the binner.

It is common to keep more information in the binner than that actually used in the source term code. The reason is so that the results of the accident progression analysis can be examined in more detail. By reducing the amount of information passed on to the source term analysis in a "rebinning" step, the amount of source term calculation time can be reduced. Thus, the APBs from an evaluation of the APET by EVNTRE are processed or rebinned by a small computer program, PSTEVNT (Ref. B.31) before the source term analysis.

SURSOR does not distinguish between the various after-VB containment failure times. So PSTEVNT combines the "Very Late" and "Final" containment failure times. The result is that the indicator for failure in the Final period is changed from a "G" in the 1st character to an "F". Bin characteristics 2 through 9 and characteristic 11 are unchanged by the processing with PSTEVNT. The other change is in the 10th character. SURSOR treats BMT in the same manner as it treats a leak in the final period, so Leak, "C", and BMT,"D", are combined and appear as "C". SURSOR also determines whether a bypass of the containment has occurred directly from character 1 ("A" or "B" for Event V) and from character 6 ("C" for no SGTR), so Bypass ("E") and NoCF ("F") are combined as "D" in the rebinner.

(At one time BMT was considered separately from final leaks in SURSOR; the releases of inert gases and organic iodine from BMT were lower than those from a late leak. It turned out to be very difficult to determine, with any certainty at all, just how much lower than the final leak releases the BMT releases should be. As the BMT releases were not expected to be substantial contributors to risk, in the interest of simplicity, the BMT releases were conservatively assumed to be equivalent to the final leak releases.)

Thus the "rebinned" bin equivalent to GFA-CAC-ABA-DA is FFA-CAC-ABA-CA. The meaning of each rebinned character is:

F - Containment failure in the Very Late or Final period;

F - Sprays only in the Late and Very Late periods;

A - Prompt CCI, dry cavity;

C - Intermediate pressure in the RCS at VB;

A - HPME occurred at VB;

C - No steam generator tube rupture;

A - A large fraction of the core was available for CCI;

B - A high fraction of the Zr was oxidized in-vessel;

A - High amount of core in HPME;

C - Leak or basemat meltthrough; and

A - One effective hole in the RCS after VB.

As mentioned in the introduction to this section, for Observation 4 the conditional probability of bin FFA-CAC-ABA-CA is 0.017 (given that PDS group 3 has occurred) and the absolute frequency is 8.1E–9/reactor year. For Observation 4, PDS group 3 is not the only group to produce this bin when the APET is evaluated. Group 1, slow SBO, also produces this bin. For Observation 4, the frequency of PDS group 1 is 9.3E–6/reactor year, and the conditional probability of APB FFA-CAC-ABA-CA is 2.6E–3, so the absolute frequency is 2.4E–8/reactor year. In the source term calculation, there is no point in calculating a source term twice for FFA-CAC-ABA-CA for Observation 4. Therefore, the bins resulting from the seven PDS groups for internal initiators are combined to produce a master bin list for each observation. In producing the master bin list, FFA-CAC-ABA-CA from group 3 is combined with FFA-CAC-ABA-CA from group 1; the total frequency for FFA-CAC-ABA-CA is 3.2E–8/reactor year for Observation 4.

B.4 Source Term Analysis

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

The source term analysis is performed by a relatively small computer code: SURSOR. The aim of this code is not to calculate the behavior of the fission products from their chemical and physical properties and the flow and temperature conditions in the reactor and the containment. Instead, the purpose is to represent the results of the more detailed codes that do consider these quantities. The release fractions are calculated in SURSOR using a limited number of factors. Many of these factors were considered by a panel of experts. Collectively, they provided distributions for these factors, and the value used in any particular observation is determined by a sampling process. The sampling process used is Latin hypercube sampling (LHS) (Ref. B.32); it is a stratified Monte Carlo method and is more efficient than straightforward Monte Carlo sampling.

The 60 radionuclides (also referred to as isotopes or fission products) considered in the consequence calculation are not dealt with individually in the source term calculation. Some different elements behave similarly enough both chemically and physically in the release path that they can be considered together. The 60 isotopes are placed in nine radionuclide classes as shown in Table B.4. It is these nine classes that are treated individually in the source term analysis. A more complete discussion of the source term analysis, and of SURSOR in particular, may be found in Reference B.33. The methods on which SURSOR is based are presented in Reference B.5, and the source term issues considered by the expert panels are described more fully in Part IV of Reference B.6.

The example being followed has led to accident progression bin FFA-CAC-ABA-CA for Observation 4. The total absolute frequency for this APB is 3.2E-8/reactor year, which comes from PDS group 1 and PDS group 3. The path followed to this point came through PDS group 3, Fast SBO.

Table B.4 Isotopes in each radionuclide release class.

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

B.4.1 Equation for Release Fraction for Iodine

In this example of a complete calculation, only the computation of the release fraction for iodine will be presented in detail. The releases of the other fission products are calculated in an analogous fashion. The total release is calculated in two parts as if the containment failed before, at, or a few tens of minutes after vessel breach. The early release occurs before, at, or within a few tens of minutes of vessel breach. The late release occurs more than a few tens of minutes, typically several hours, after vessel breach. In general, the early release is due to fission products that escape from the fuel while the core is still in the RCS, i.e., before vessel breach (VB), and is often referred to as the RCS release. The late release is largely due to fission products that escape from the fuel during the CCI, i.e., after VB, and is referred to as the CCI release. For situations where the containment fails many hours after VB, the "early" release equation is still used, but the release is better termed the RCS release, and after both releases are calculated in SURSOR, both releases are combined into the late release and the early release is set to zero. The "late" release includes not only fission products released from the core during CCI, but also material released from the fuel before VB that deposits in the RCS or the containment and then is revolatilized after VB.

The early or RCS iodine release is calculated from the following equation:

$$ST = [FCOR * FVES * FCONV / DFE] + DST.$$

And the late or CCI iodine release is calculated from:

$$STL = [(1 - FCOR) * FPART * FCCI * FCONC / DFL] + FLATE + LATEI.$$

In these equations, some terms that pertain only to steam generator tube ruptures (SGTRs) have been omitted since bin FFA-CAC-ABA-CA has no SGTR. The meaning of the terms is as follows:

ST = fraction of the core iodine in the RCS release to the environment;

FCOR = fraction of the iodine in the core released to the vessel before VB;

FVES = fraction of the iodine released to the vessel that is subsequently released to the containment;

FCONV = fraction of the iodine in the containment from the RCS release that is released from the containment in the absence of any mitigating effects;

DFE = decontamination factor for RCS releases (sprays, etc.);

DST = fraction of core iodine released to the environment due to direct containment heating at vessel breach;

STL = fraction of the core iodine in the late release to the environment;

FPART = fraction of the core that participates in the CCI;

FCCI = fraction of the iodine from CCI released to the containment;

FCONC = fraction of the iodine in the containment from the CCI release that is released from the containment in the absence of any mitigating effects;

DFL = decontamination factor for late releases (sprays, etc.);

FLATE = fraction of core iodine remaining in the RCS that is revolatilized and released late in the accident; and

LATEI = fraction of core iodine remaining in the containment that is converted to volatile forms and released late in the accident.

Like ST and STL, DST, FLATE, and LATEI are expressed as fractions of the initial core inventory. DST, FLATE, and LATEI are not independent of the other factors in the equations given above. Complete expressions for these three terms and an expanded discussion of them may be found in Reference B.18.

Some of these factors are determined directly by sampling from distributions provided by the expert panels. Others are derived from such values, and still others were determined internally. In Section B.4.2, each factor in the equation above will be discussed briefly, and the source of the value used for each factor will be given. In Section B.4.3, three of the factors are discussed in more detail.

For Observation 4, the following values were used in the equation for the RCS iodine release for bin FFA-CAC-ABA-CA:

FCOR = 0.98
FCONV = 1.0E−6
DST = 0.0
FVES = 0.86
DFE = 34.0
resulting in ST = 2.5E−8.

ST is a very small fraction of the original core inventory of iodine because the containment failure takes place many hours after VB and there is a long time for natural and engineered removal process to operate.

For Observation 4, the following values were used in the equation for the late or CCI iodine release for bin FFA-CAC-ABA-CA:

1 - FCOR = 0.02
FCCI = 1.0

DFL = 82.2
LATEI = 0.0044
FPART = 0.57
FCONC = 1.2E-4
FLATE = 7.2E-9
resulting in STL = 0.0044

Containment failure occurs a long time after most of the radionuclides have been released from the fuel during CCI, so there is a long period in which the aerosol removal processes operate. Thus, the CCI release of iodine in nonvolatile form is very small, and the total late iodine release is almost all due to the late formation of volatile iodine in the containment.

B.4.2 Discussion of Source Term Factors

As most of the parameters in the source term equations were determined by sampling from distributions provided by a panel of experts, Part VI of Reference B.6 is not cited for each of the parameters. The parameters not determined by expert panels are discussed in References B.1, B.6, and B.33.

The values for many of the parameters defined above are obtained from distributions when SURSOR is evaluated, most of which were provided by experts. They determined distributions for the nine radionuclide release classes defined in Table B.4. Only the distributions for iodine (class 2) are discussed here, but distributions exist for the other eight classes as well (Ref. B.6). These distributions are not necessarily discrete. While the experts provided separate distributions for all nine classes for FCOR, for other factors, for example, they stated that classes 5 through 9 should be considered together as an aerosol class. Note that the distributions for the nine radionuclide classes are assumed to be completely correlated. That is, a single LHS number is obtained for each factor in the source term equation, and it applies to the distributions for all nine radionuclide classes. For example, in Observation 4 the LHS number for FCOR is 0.828. That means the 82.8th percentile value is chosen from the iodine distribution, the cesium distribution, the tellurium distribution, etc., for FCOR.

FCOR is the fraction of the fission products released from the core to the vessel before vessel failure. The value used in each observation is obtained directly from the experts' aggregate distribution. There are separate distributions for each fission product group (inert gases, iodine, cesium, etc.) for high and low Zr oxidation in-vessel. Each distribution takes the form of a curve that relates the values of FCOR to a cumulative probability. A value of FCOR is obtained in the following manner: the LHS program (Ref. B.32) selects a number between zero and 1.0 that is the cumulative probability. Using this value, the value of FCOR is obtained from the experts' aggregate cumulative probability distribution. The LHS number in Observation 4 for FCOR is 0.828, and the corresponding FCOR value for iodine is 0.98. For Observation 4, then, almost all the iodine is released from the core to the vessel before breach. FCOR is discussed in more detail in Section B.4.3.

FVES is the fraction of the fission products released to the vessel that is subsequently released to the containment before or at vessel failure. As for FCOR, the value used in each observation is obtained directly from the experts' aggregate distribution, and there are separate distributions for each fission product group. The LHS number in Observation 4 for FVES is 0.931. The corresponding value in the experts' aggregate distribution for FVES for iodine is 0.86. So, in this example, most of the iodine in the vessel before breach is released from the vessel to the containment.

FCONV is the fraction of the fission products in the containment from the RCS release that is released from the containment in the absence of mitigating factors such as sprays. The expert panel provided distributions for FCONV for four cases, each of which applies to all species except the noble gases. These cases apply to containment failure at or before VB, or within a few hours of VB. (There is a fifth distribution that applies to Event V.) None of these distributions is used in the path followed for this example since containment failure happens a day or more after the start of the accident. Because of the long time period for the engineered and natural removal processes to reduce the concentration of the fission products in the containment atmosphere, the fraction of the fission products released before or at VB remaining airborne at the time of containment failure is very small. This fraction was estimated internally to be 1.0E−6, and FCONV is set to that value for final period releases. (The particular value of 1.0E−6 is not important; any very small value would be satisfactory.) This value is used whether the release is due to aboveground failure or basemat meltthrough. FCONV is discussed in more detail in Section B.4.3.

DFE is the decontamination factor for early releases. For APB FFA-CAC-ABA-CA, the containment sprays are the only mechanisms that contribute to DFE. The expert panel concluded that the distributions used for the spray decontamination factors (DFs) were less important to determining offsite risk and the uncertainty in risk than whether the sprays were operating and other factors, so the spray DF distributions were determined internally. There are two spray distributions that apply to the fission products released from the RCS before or at VB: the first applies when the containment fails before or at VB and the RCS is at high pressure at VB; and the second applies when the containment fails after VB or when the containment fails at VB but the RCS is at low pressure. Each distribution applies to all species except the noble gases. The LHS number for the spray distribution for Observation 4 was 0.928. Using the distribution for late containment failure, a spray DF value of 3.4 is obtained. For failures of the containment in the final period, the value from the distribution is multiplied by 10 to account for the very long period that the sprays have to wash particulate material out of the containment atmosphere. Thus, DFE is increased from 3.4 to 34.

DST is the fission product release (in fraction of the original core inventory) from the fine core debris particles that are rapidly spread throughout the containment in a direct containment heating (DCH) event at VB. The experts provided distributions for the fractions of the fission products that are released from the portion of the core involved in DCH for VB at high pressure (1,000 to 2,500 psia) and for VB at intermediate pressure (200 to 1,000 psia). There are separate distributions for each fission product group (inert gases, iodine, cesium, etc.). However, neither the high-pressure nor the low-pressure set of distributions was used in calculating the source term for FFA-CAC-ABA-CA because the containment failure occurs so long after VB. It was internally estimated that the amount of fission products from DCH remaining in the atmosphere many hours after VB would be negligible, so DST is set to zero for this APB.

FPART is the fraction of the core that participates in the CCI. Bin FFA-CAC-ABA-CA has a "large" fraction, nominally 40 percent, of the core participating in HPME. As 5 percent of the core is estimated to remain in the vessel indefinitely, the fraction participating in DCH is 0.95 * 0.40 = 0.38. The fraction of the core available to participate in CCI is thus 0.95 - 0.38 = 0.57.

FCCI is the fraction of the fission products present in the core material at the start of CCI that is released to the containment during CCI. The experts provided distributions for four cases that depended upon the fraction of the Zr oxidized in-vessel and the presence or absence of water over the core debris during CCI. There are separate distributions for each fission product group. For the path being followed in this example, bin FFA-CAC-ABA-CA indicates that a large fraction of the Zr was oxidized in-vessel before VB and that the cavity was dry at the start of CCI. However, for iodine, the case is immaterial since all the iodine remaining in the core debris is released during CCI for any case and for every point on the distribution. Thus, FCCI is 1.0.

FCONC is the fraction of the fission products released to the containment from the CCI that is released from the containment. The expert panel provided distributions for FCONC for five cases. There are separate distributions for each fission product group (inert gases, iodine, cesium, etc.). None of these cases applies directly to the situation for APB FFA-CAC-ABA-CA since this bin has containment failure in the final period (after 24 hours). Since containment failure occurs many hours after most of the fission products have been released from CCI, only a very small fraction of these fission products will still be in the containment atmosphere at the time of containment failure. This fraction was estimated internally to be on the order of 1.0E−4. The exact value is determined by using the FCONC distribution for case 3, rupture before the onset of CCI. The ratio of the LHS value from the distribution to the median value times 1.0E−4 is the value of FCONC used for final period containment failure. In Observation 4, the LHS number for determining FCONC is 0.777. The iodine value of FCONC for this point on the FCONC, case 3, for the CDF is 0.78, and for the median value of the distribution is 0.63. Thus, FCONC is set to 0.78/0.63 * 1.0E−4 = 1.2E−4. This value is used whether the release is due to aboveground failure or basemat meltthrough.

DFL is the decontamination factor for late releases. At Surry, DFL can be due to either the containment sprays or a pool of water over the core debris during CCI. Since the CCI began in a dry cavity, only the spray DF applies for bin FFA-CAC-ABA-CA. The procedure used to obtain the spray DF for the CCI release for final period CF is similar to that used to obtain the value for DFE (discussed above). There is only one distribution for the spray DF for the CCI release, and it applies to all species except the noble gases. The same LHS number (0.928) is used for all the spray distributions, giving a CCI spray DF value of 8.2. As for DFE, because the containment fails in the final period, the value from the distribution is multiplied by 10 to account for the very long time the sprays have to wash particulate material out of the containment atmosphere. Thus, DFL is 82.

FLATE accounts for the release of iodine from the RCS late in the accident. Like DST, it is a fraction of the original core inventory. Iodine that had been deposited in the RCS before VB may revert to a volatile form after the vessel fails and make its way to the environment. This term considers only revolatilization from the RCS; revolatilization from the containment is considered in the next term. The experts provided distributions for the fraction of the radionuclides remaining in the RCS that are revolatilized. The amount remaining in the RCS is a function of FCOR, FVES, and other terms and is calculated in SURSOR (Ref. B.33). The experts concluded that whether there was effective natural circulation through the vessel was important in determining the amount of revolatilization. Thus, there are two cases: one large hole in the RCS and two large holes in the RCS.

The experts provided separate distributions only for iodine, cesium, and tellurium. (Revolatilization is not possible for the inert gases as they would deposit, and it is negligible for radionuclide classes 5 through 9.) For accident progression bin FFA-CAC-ABA-CA, the last character indicates that there is only one effective hole in the RCS: the hole formed in the bottom head when the vessel failed. The other failure of the RCS pressure boundary is the RCP seals, and the path through them is considered too tortuous to allow effective natural circulation to develop. The LHS number for late revolatilization of Observation 4 is 0.412, and the corresponding value for iodine from the experts' distribution for iodine is 0.033. This number is applied to the fraction of the iodine remaining in the RCS, which is small, and then the FCONC value for tellurium is applied to that value to determine how much of the iodine that is revolatilized from the RCS escapes from the containment. (The Te value for FCONC is considered to be generally appropriate for revolatilized material since it, like Te, is slowly released over a long time period.) The resulting value for revolatilized iodine that escapes from the containment is very low, 7.2E–9.

LATEI accounts for iodine in the containment that may assume a volatile form and may be released late in the accident. The volatile forms are typically organic iodides such as methyl iodide, but are not limited to organic forms. The primary source of this iodine is the water in the reactor cavity and the containment sumps (which are separate at Surry). This term is added to the late release only for radionuclide class 2, iodine. The experts provided only one distribution. The LHS number for late revolatilization in Observation 4 is 0.055, and the corresponding value for iodine from the experts' distribution is 0.0051. This number is applied to the fraction of the iodine remaining in the containment. Based on the values of FCOR, FVES, FCCI, and other factors, the fraction of the original core iodine still in the containment and available to assume a volatile form was determined to be 85 percent for Observation 4. Applying the release fraction obtained from the experts' distribution to this gives a late revolatilization iodine release fraction of 0.0044. LATEI is discussed in more detail, and the expression used to calculate it is given in Section B.4.3.

While the total iodine release is small compared to a case where the containment fails at VB or is bypassed from the start (such as Event V), the iodine release is very large compared to the other radionuclide classes except inert gases (see Section B.4.4). This relatively large release fraction for iodine is entirely due to the LATEI term. Even though the release point for basemat meltthrough is underground, no allowance is made for attenuation or decontamination of the late iodine release represented by the LATEI term. For the example considered, the very slow passage of the gases through wet soil with a low driving pressure would undoubtedly result in some reduction in the late iodine release. This reduction could be quite large. Although giving no credit for removal in the wet soil is conservative, it is unimportant for the sample as a whole. Other observations and other modes of containment failure dominate risk. For the mode of containment failure in this example, basemat meltthrough, however, the release of late organic iodine is a major contributor to risk, and the risk from this release may be overestimated by the neglect of iodine removal in the wet soil. Even with this conservative estimate of the late iodine release, the total iodine release and the risk therefrom are very small compared to the releases and risks from accidents and pathways in which the containment fails at or before vessel breach, or where the containment is bypassed.

It could be argued that, since BMT is so much more likely than early CF, overstating the BMT release results in a significant overestimate of the population dose and latent cancer fatality estimates. However, bypass accidents (V or SGTR) are twice as likely at Surry as nonbypass accidents that lead to BMT. As the iodine releases from the bypass accidents are more than an order of magnitude higher than BMT iodine releases, this argument is not valid.

B.4.3 Quantification of Source Term Factors by Experts

In this section, the quantification of three factors in the source term equation that were considered by the expert panel is presented in more detail. The eight issues considered by the source term expert panel are:

1. FCOR and FVES
2. Ice Condenser DF (not applicable to Surry)
3. FLATE
4. FCCI
5. FCONV and FCONC
6. LATEI (not used for PWRs)
7. Reactor Building DF (not applicable to PWRs)
8. DCH Releases (DST)

Three of these issues are not applicable to Surry. Of the eight factors in the iodine equation for Surry, only three are discussed here. More extensive documentation of all the issues may be found in Part IV of Reference B.6. The source term factors chosen for discussion here are FCOR, FCONV, and LATEI. The consideration for FVES is similar to that for FCOR, only there are more cases. The consideration for FCONC is similar to that for FCONV, except that the experts provided a distribution for each fission product group for FCONC and they did not for FCONV. Of the remaining factors, LATEI made the largest contribution to the iodine example considered above.

B.4.3.1 FCOR

FCOR is the fraction of the fission products released from the core to the vessel before vessel failure. Four members of the source term expert panel provided distributions for FCOR:

Peter Bieniarz, Risk Management Associates,
Robert Henry, Fauske and Associates, Inc.,
Thomas Kress, Oak Ridge National Laboratory, and
Dana Powers, Sandia National Laboratories.

Two of these four experts concluded that there were no significant differences between PWRs and BWRs as far as FCOR was concerned, and each provided one distribution that applied to both types of reactors. The other two experts provided separate PWR and BWR distributions and further subdivided this issue by providing different distributions for high Zr oxidation in-vessel and low Zr oxidation in-vessel.

Expert A based his analysis for FCOR upon the experimental work done on the release of fission products from fuel (Refs. B.34 and B.35). He concluded that the results for cesium could be well represented by an equation similar to the diffusion equation and that the constants in the solution could be determined from the data. He obtained release rates for the other fission products by considering their "relative volatilities." The results of applying this method of calculating release rates appeared to him to agree reasonably well with experiments. Expert A then wrote a simple computer program to vary the temperature rise with time over a range of reasonable scenarios and keep track of the amount of each fission product released. Expert A provided FCOR distributions for both high and low Zr oxidation in-vessel for both types of reactors.

Expert B based his conclusions for FCOR on a large number of MAAP (Ref. B.15) calculations for various accident scenarios. He also relied on Reference B.36 and the evidence from TMI-2 (Refs. B.37 and B.38). The MAAP results served as the basis for his conclusions, but he included uncertainty for phenomena not modeled in MAAP and phenomena that MAAP currently does not treat in sufficient detail. For example, Expert B thought that MAAP sometimes overestimated the releases of certain nuclide groups because the process of core collapse imposes physical limitations on other processes that MAAP does not consider adequately at this time. He also concluded that neither the reactor type nor the amount of Zr oxidation in core had a significant effect on FCOR, so he provided a single distribution for FCOR.

Expert C reasoned that even if the dependency of the fission product release rates on temperature were much better known, the release rates, and thus FCOR, could not be much better predicted because the variations of the temperatures in the core by time and location are so crudely known at this time, especially after the onset of relocation. The extent of metal oxidation is also a significant uncertainty. Relocation not only changes the surface to volume ratio, but it alters the hydrogen-steam ratio, which in turn affects the diffusion and transport rates of the fission products. Thus, the current models, which largely depend upon Arrhenius-type equations, have definite limitations. For example, the STCP (Ref. B.39) tends to overpredict FCOR because it poorly treats the formation of eutectics and the gradual relocation of the core. Expert C provided separate FCOR distributions for high and low Zr oxidation in-vessel for both types of reactors.

Expert D did not consider the amount of Zr oxidation in-vessel or the type of reactor to be important for FCOR; he provided one distribution for FCOR. Expert D was of the opinion that all the noble or inert gases (Xe and Kr) would escape from the fuel. For tellurium, he concluded that the data were so ambiguous and conflicting that he could not support any particular distribution. He thus specified that a uniform distribution between zero and one be used. For the other seven radionuclide groups, he provided nonuniform distributions. His conclusions were based on a set of experimental work that he has performed or was performed by others (Refs. B.40 and B.41). He made use of several small computer programs to manipulate the experimental results to obtain release fractions for different pressures and temperatures.

The aggregate distributions for the two PWR cases are shown in Figures B.11 and B.12, as are the distributions of each of the four experts who considered this issue. The estimated fraction released depends strongly on the volatility of the fission products, as might be expected. The differences between I and Cs are not great. The differences between the less volatile fission products Ba, Sr, Ru, La, and Ce are small. Note that the differences between the high-Zr-oxidation case and the low-Zr-oxidation case are small compared to the differences between the experts. Furthermore, the differences between the radionuclide classes are often less than the differences between the experts for a given class. This is indicative of the uncertainty in the source term area.

B.4.3.2 FCONV

This issue concerns the fraction of radionuclides released to the containment atmosphere from the vessel before it fails or at failure that is subsequently released to the environment if the containment fails. FCONV may be defined by the equation:

$$\text{FCONV} = \text{mVout}/\text{mVin}$$

where:

mVin = mass (kg) of a radionuclide (or radionuclide class) released from the vessel to the containment atmosphere at or before vessel breach (VB); and

mVout = mass (kg) of a radionuclide (or radionuclide group) released from the vessel to the containment atmosphere at or before VB that is subsequently released from containment.

That is, FCONV is the fraction of mVin that is released to the environment when the containment fails.

FCOR – Case PWR–1 – High Zr Oxidation

Release Fraction (abscissa) by

Cumulative Probability (ordinate)

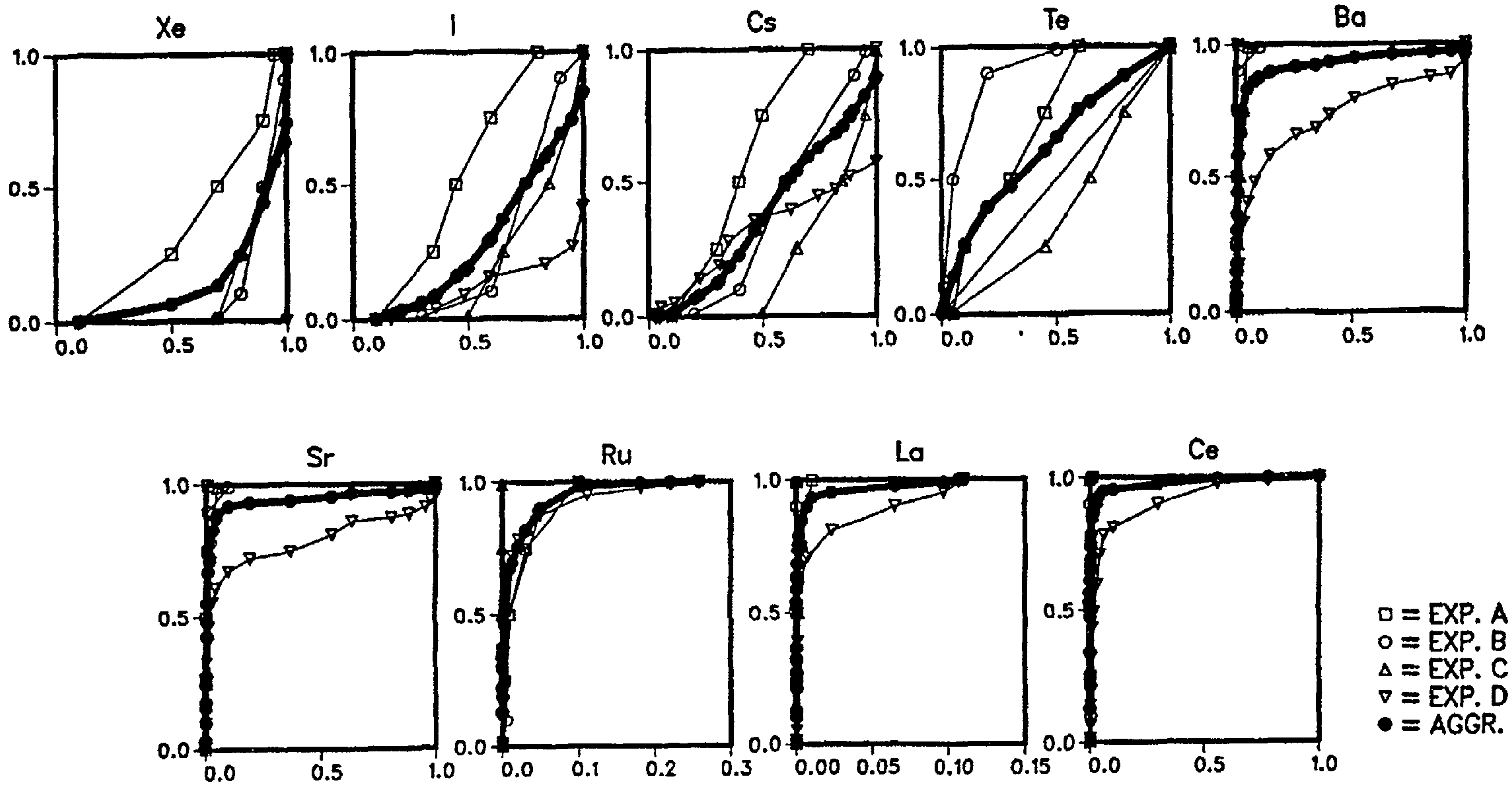


Figure B.11 Results of expert elicitation for FCOR, fraction of the fission products released from core to vessel for the nine radionuclide groups. (This figure shows the results when a high fraction of the core Zr is oxidized in-vessel. In each plot, the first four curves are the distributions of the experts, and the fifth curve is the aggregate distribution.)

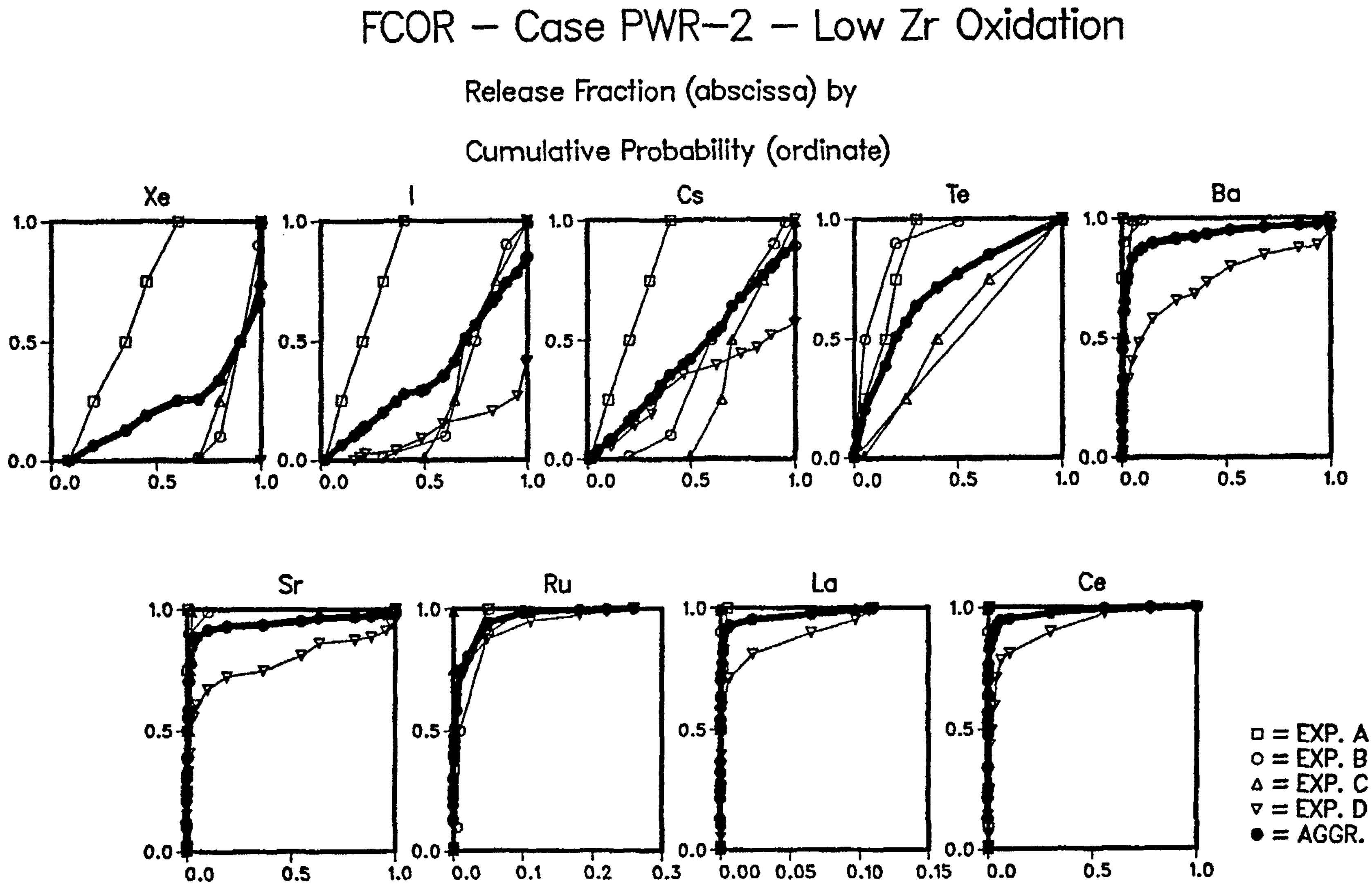


Figure B.12 Results of expert elicitation for FCOR, fraction of fission products released from core to vessel for the nine radionuclide groups. (This figure shows the results when a low fraction of the core Zr is oxidized in-vessel. In each plot, the first four curves are the distributions of the experts, and the fifth curve is the aggregate distribution.)

Three cases were defined for FCONV for the large, dry PWR containments:

1. Early containment failure, leak
2. Early containment failure, rupture
3. Late containment failure, rupture

Early containment failure means at or before vessel breach, and "late" means at least 3.5 hours and nominally 6 hours after VB. A "leak" is a failure of the containment that results in leakage significantly larger than design leakage but is small enough so that the containment does not depressurize in less than 2 hours. The nominal leak is a hole with an area of 0.1 ft^2. A "rupture" is a containment failure sufficient to depressurize the containment in less than 2 hours; the nominal hole size is 7 ft^2. The releases from late leaks were deemed to be low enough that the value of FCONV for late leaks could be derived from these three cases without significantly affecting risk.

FCONV is defined to be the release fraction from containment excluding the effects of engineered safety features such as containment sprays. One expert, however, concluded that one of the principal removal mechanisms, aerosol agglomeration, depended upon the humidity of the atmosphere for case 1. While the humidity may depend on whether the sprays are operating, aerosol removal from the atmosphere by the sprays is considered separately.

Five members of the source term panel considered FCONV:

Andrzej Drozd, Stone and Webster,
James Gieseke, Battelle Columbus Division,
Thomas Kress, Oak Ridge National Laboratory,
Y. H. (Ben) Liu, University of Minnesota, and
David Williams, Sandia National Laboratories.

They all concluded that the inert gases would be completely released and that all the other radionuclides would behave as aerosols. Thus, their distributions for FCONV apply to fission product classes 2 through 9.

Expert A obtained his estimates of the event timing and, thus, residence times from References B.42 and B.43. Expert A concluded that the most important factor in determining FCONV was the residence time of the aerosols in the containment; the longer the time between the formation of the aerosols and the failure of the containment, the smaller the release. Because of the opposing effects of the dynamic shape factor on coagulation and settling, the uncertainty in the dynamic shape factor has little effect on the fraction released. Expert A did not distinguish between the volatile fission products and the refractory groups because he concluded that a significant fraction of the volatiles is released from the fuel prior to VB and deposit on the surfaces of the reactor coolant system. The refractory fission products are released from the fuel at a slower rate and a significant fraction is released after VB and has a direct pathway to the containment. Thus both the volatile and nonvolatile species have similar release rates during the times of interest. He also stated that the aerosol concentration in the containment dropped dramatically in 1 to 2 hours and did not change much after that. The atmospheric humidity has little effect; high humidity makes particles more compact. The compact particles settle out faster but do not agglomerate as fast.

Expert B used NAUA (Ref. B.44) calculations done in conjunction with STCP calculations (Refs. B.42, B.43, and B.45) as a basis for his results, obtaining values directly from NAUA computer output as well as from published reports. For practical considerations, only Xe, I, Cs, and Te were considered for FCONV, and these were deemed to be applicable to all the fission product groups. Other sources consulted by Expert B are the Brookhaven National Laboratory uncertainty study (Ref. B.46), an Electric Power Research Institute (EPRI) calculation for Peach Bottom (Ref. B.47), the recent CONTAIN calculations (Refs. B.20 through B.23), the MELCOR analysis of Peach Bottom (Ref. B.48), and other MELCOR calculations (Refs. B.49 and B.50).

Expert B intended his distributions to include uncertainties from:

1. Surface area (deposition area or compartment height),
2. Natural circulation,
3. Hygroscopic nature of aerosols (primarily I and Cs groups),
4. Particle shape factors (not a big effect),
5. H_2 burn, and
6. Residence time.

Expert C examined the available code calculations relevant to aerosol and fission product behavior in, and release from, the containment. In most cases, these calculations were performed with the STCP (Ref. B.39) or CONTAIN (Ref. B.14) codes. He developed base distributions for FCONV from the code results and then modified them for the effects of factors not considered by the codes. The scale factors applied to the base distributions took into account factors such as aerosol agglomeration, aerosol source strength, timing, shape factors, and containment volume.

Finally, Expert C modified the resulting distributions if they were greatly different from his intuitive expectations.

Expert D considered calculations performed in the GREST exercise (Ref. B.51), by Sandia with MELCOR (Ref. B.48), and by the ANS (Ref. B.52). He concluded that the factors affecting the value of FCONV include:

1. Aerosol characteristics (shape factors, distribution, density);
2. Residence time (size and time of containment failure);
3. Whether the containment was open or divided into many compartments;
4. The effective height of the containment;
5. Thermodynamic state of the atmosphere (superheated or condensing); and
6. Hygroscopic nature of the aerosols.

Expert D noted that the ANS parametric study showed a decrease in aerosol concentration by a factor of 10 in 2 hours and that both the ANS parametric study and KfK DEMONA experiments (Ref. B.53) showed that the existence of many compartments in the containment reduced the release by about a factor of 1.6. Expert D pointed out that the LACE experiments (Ref. B.54) show that, if the hygroscopic effect is present, it can be dominant. A hydrogen burn, by decreasing the residence time and reducing the condensation in the atmosphere, can increase the release fraction FCONV by a factor of 2.

Expert E used an EPRI/FAI aerosol behavior algorithm (Ref. B.55) to perform an independent uncertainty analysis for this issue. He directly varied the aerosol source rates and the aerosol form factors (gamma and chi). To study the impact of the timing and mode of containment failure, he varied the containment failure time and leak rates, assuming choked flowthrough holes from 0.1 ft^2 to 7 ft^2. He also considered pre-existing leakage, steam condensation onto walls, and the impact of pool flashing in his calculations.

Expert E assumed that the aerosols were released directly from the reactor vessel and obtained his aerosol form factors from the QUASAR (Ref. B.46) and QUEST (Ref. B.56) studies. He concluded that the timing and mode of containment failure is the major source of uncertainty. Because it affects agglomeration, the level of turbulence in containment is also an important uncertainty.

The distributions of the five experts who considered this issue are shown in Figure B.13. Case 1 is divided into wet and dry subcases because one of the experts concluded that the release fraction depended on the

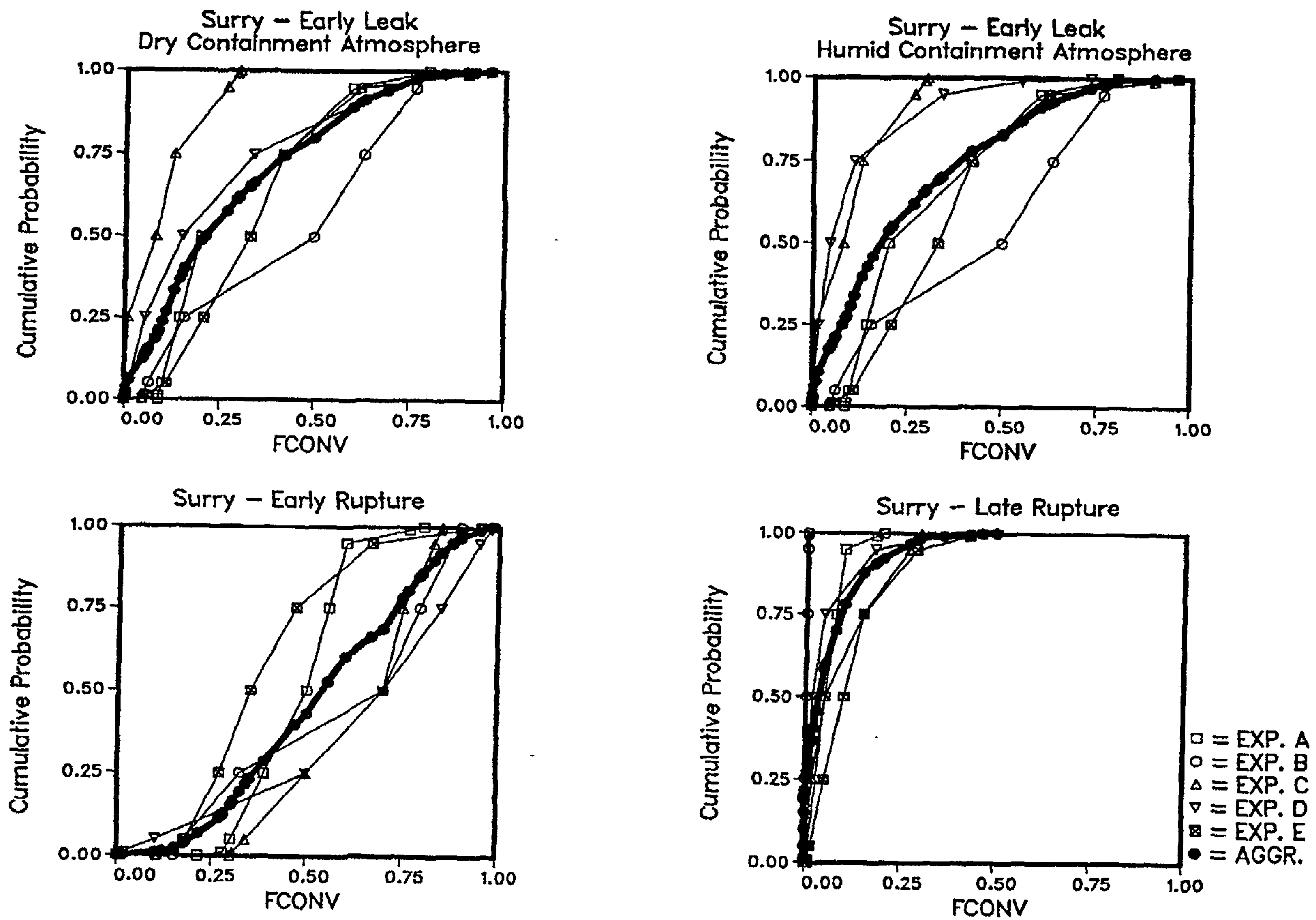


Figure B.13 Results of expert elicitation for FCONV, fraction of fission products in containment from RCS release that is released to environment. (In each plot, the first five curves are the distributions of the experts, and the sixth curve is the aggregate distribution.)

humidity for this case. The differences between the wet and dry atmosphere subcases are small compared with the differences between the experts, so this distinction was dropped. The aggregate distributions are also shown on Figure B.13. The differences between the experts are large compared to the differences between cases 1 and 2. As explained in the discussion of FCONV in Section B.4.2, none of these four distributions was used for bin FFA-CAC-ABA-CA since the containment failure time was so late.

B.4.3.3 LATEI

The question of interest for this issue is how much of the iodine in the containment late in the accident assumes a volatile form (typically organic) and is released to the environment. This volatile iodine is assumed to be unaffected by all removal mechanisms (pool scrubbing, sprays, deposition, etc.). The release fraction determined in this issue applies to almost all the iodine released from the fuel and retained in the containment. The bulk of this iodine is expected to be in aqueous solution, so the issue was specifically framed as release from water pools.

The late release of volatile iodine was deemed to be much more important for BWRs than for PWRs because the BWR design often results in most of the iodine released during core degradation being transported to and retained in the suppression pool. Therefore, the panel of experts was asked only about BWRs directly. They were asked to consider the release of volatile iodine from a BWR suppression pool following containment failure and from water in the pedestal region beneath the reactor pressure vessel (RPV) during CCI.

For the release of volatile iodine from the suppression pool after the containment has failed, two cases were defined: (1) the pool remains subcooled, and (2) the pool is at the saturation temperature. In case 1, considerable surface evaporation is expected but no bulk boiling. In case 2, substantial flashing of the pool would accompany containment failure.

For the release of volatile iodine from water that overlies the core debris in the RPV pedestal, there are also two cases: (1) the drywell is flooded at the time of VB and the entire CCI takes place beneath a pool at least a few feet deep; and (2) the RPV pedestal area contains some water at the time of VB but most of this water is boiled away during CCI.

The results of the expert elicitation on this issue are contained in detail in Part IV of Reference B.6. They are not summarized here because the source term calculation for Surry did not use the results of any of the BWR cases. The PWR situation is somewhat different since the bulk of the iodine is expected to be contained in solution in the sump water. The sump water does not play the same role in heat removal that the suppression pool does in the BWR, and the sump water at Surry is separate from the water in the reactor cavity. Thus, none of the BWR cases is directly applicable although the subcooled suppression pool case is the most applicable. Instead of using this BWR case, the distribution obtained specifically for PWRs in the first draft of NUREG-1150 (Ref. B.57) was used. This is discussed further in Part VI of Reference B.6.

The equation used to calculate the late release of iodine in volatile form is:

LATEI = XLATE * [{FCOR * FVES + (1 - FCOR) * FPART * FCCI} - ST - STL + FLATE]

where XLATE is the fraction of the iodine in the containment late in the accident that assumes a volatile form and is released to the environment. The other terms have been defined above. The term in brackets [ ] is the fraction of the initial core inventory that is in the containment at late times. FCOR * FVES is the RCS release to the containment, and ST is the RCS release from the containment. Similarly, (1 - FCOR) * FPART * FCCI is the CCI release to the containment, and STL is the CCI release from the containment. The FLATE iodine is not considered amenable to this release mechanism because its residence time in the containment is short.

Figure B.14 displays the four aggregate distributions obtained for late volatile iodine release fraction, XLATE, for the BWR cases described above and the distribution for XLATE used for Surry. The range of release fractions used for Surry is the same as for the most applicable BWR case—subcooled suppression pool. The details of the distribution used for Surry are not particularly important as the risk is

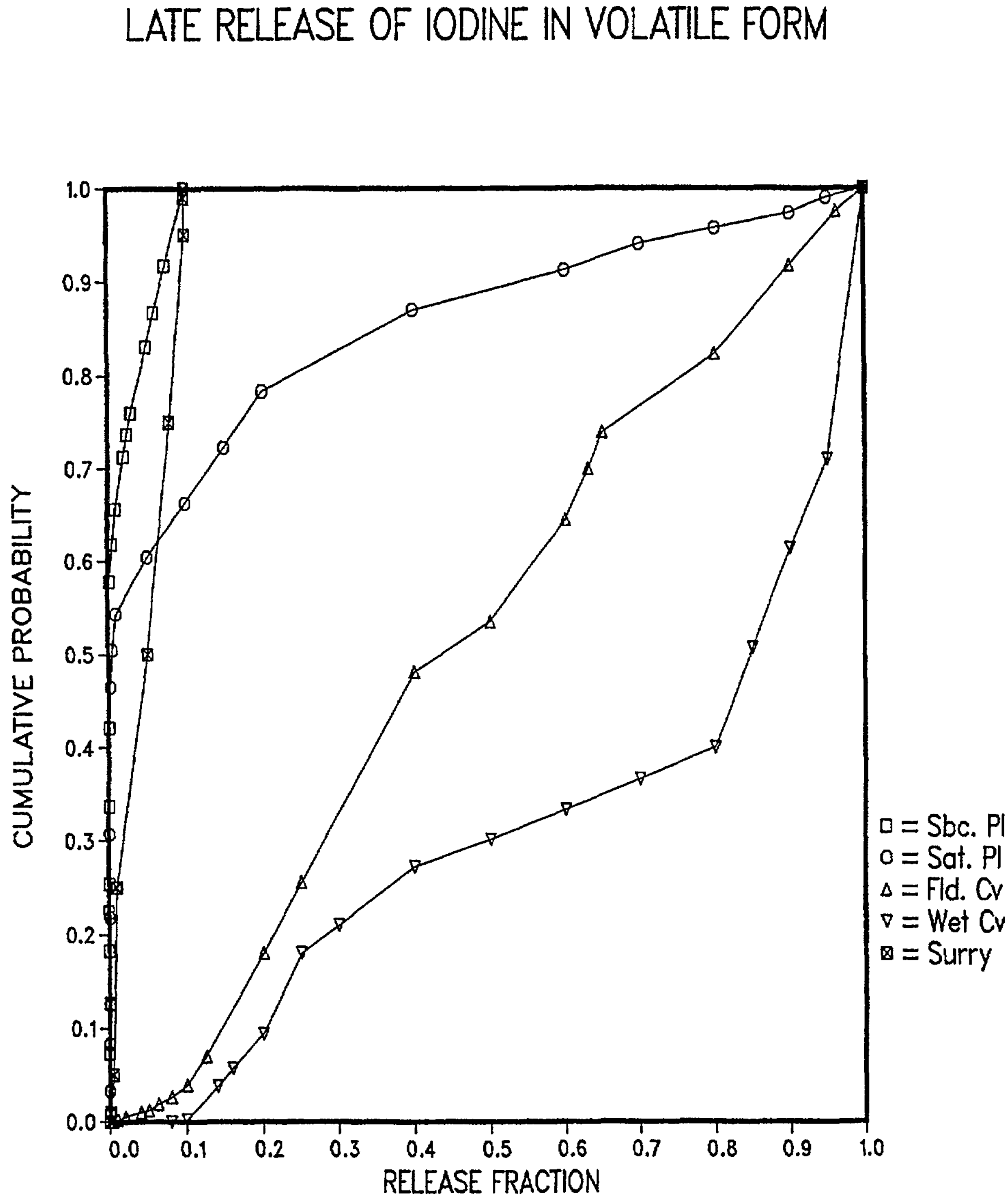


Figure B.14 Distributions for late release of iodine from containment in volatile form. (The first four curves are the aggregate distributions for BWRs: release from a subcooled pool, release from a saturated pool, release from a flooded cavity, and release from a wet cavity. The fifth curve is the distribution used for Surry.)

dominated by accident scenarios in which the path from the reactor vessel to the atmosphere is quite direct (Event V and SGTRs). As this volatile release of iodine occurs late in the accident, its contribution to early fatality risk is negligible. In the accidents that contribute the most to the latent cancer fatality risk, there is little iodine remaining in the containment at late times to be released by this mechanism.

B.4.4 Releases for All Fission Products

'. release that commences a day or more after the onset of core damage or 10 hours or more after VB would be expected to have very small releases, and such is observed to be the case here. The iodine release is dominated by volatile (mostly organic) species that form late in the accident. When the release is so late in the accident, there is only one release, and the distinction between an RCS or early release and a CCI or late release is not kept. The RCS release is put in the late release with the CCI release, and the early release is set to zero. Thus, the complete early and late release fractions for bin FFA-CAC-ABA-CA are:

Fission Products	Early Release	Late Release	Total Release
Xe, Kr	0.0	1.0	1.0
I	0.0	4.4E-3	4.4E-3
Cs, Rb	0.0	8.6E-8	8.6E-8
Te, Sc, Sb	0.0	2.3E-7	2.3E-7
Ba	0.0	2.8E-7	2.8E-7
Sr	0.0	1.2E-9	1.2E-9
Ru, etc.	0.0	3.0E-8	3.0E-8
La, etc.	0.0	3.1E-8	3.1E-8
Ce, Np, Pu	0.0	2.0E-7	2.0E-7

SURSOR also provides the times and energies associated with the early and late releases, the release elevation, and the time that a general emergency is declared. For bin FFA-CAC-ABA-CA, the times for the early release are irrelevant. The other parameters are:

TW = time warning is given = 6.1 h

T2 = start of late release = 36 h

DT2 = duration of late release = 6 h

E2 = energy release rate = 3600 W

ELEV = height of release = 10 m

If BMT releases had been calculated separately, the release height would have been zero. Since BMT and leak releases are treated together, a height more appropriate to a leak above ground is used.

B.5 Partitioning of Source Terms

The accident progression analysis and the source term analysis, each performed once for the 200 observations that constitute the sample, produced 18,591 source terms. This is far too many to be able to perform a consequence analysis for each, so a reduction step is performed before the consequence analysis. This step is called partitioning. Partitioning is performed for all the observations in the sample together.

B.5.1 Introduction

Partitioning is a grouping of the source terms based on the radiological potential of each source term to cause adverse effects on humans. The factors used in partitioning are those most important for the magnitude of the risk: the release fractions for the early release, the release fractions for the late release, and the time between start of the evacuation in the surrounding region and the start of the first release. It is difficult to take each of the nine radionuclide groups into account separately when grouping the source terms, so "effect weights" are determined for each release. These provide a means of considering all the fission products as if they were just one. The partitioning process consists of grouping the source terms together based on these effect weights. Each source term group is then further subdivided based on evacuation timing relative to the start of the release. For releases that have the possibility of causing early fatalities as well as latent cancer fatalities, the grouping is two-dimensional and is based on the source term's potential to cause both types of fatalities. In this analysis for Surry, there were 6,820 source terms with nonzero early fatality weights (EFWs) and nonzero chronic fatality weights (CFWs). For releases that have the possibility of causing only latent cancer fatalities, the grouping is one-dimensional; 11,771 source terms were of this nature.

The partitioning process is carried out by a computer code—PARTITION (Ref. B.58). The process is an interactive one, with the user choosing the number of cells or divisions to be used and selecting those grids that have so few points that they may be combined with a neighboring cell.

B.5.2 Effects Weights

The early fatality weight (EFW) of a source term is a measure of the radiological potential of a source term to cause early fatalities in the absence of any mitigating effects except relocation. The chronic fatality weight (CFW) of a source term is, similarly, a measure of the radiological potential of a source term to cause latent cancer fatalities.

The calculation of the EFW has two parts. First, the releases of the 59 radionuclides other than I-131 are converted into equivalent amounts of I-131. Then the total release, expressed in an equivalent amount of I-131, is used to estimate the number of resultant early fatalities.

The isotope conversion factor used to determine the equivalent amount of I-131 for each isotope is based on the bone marrow dose from the three pathways that give an acute dose: cloudshine, groundshine, and inhalation.

The cloudshine or immersion dose results from being surrounded by air containing radioactive molecules. The inhalation dose comes from breathing the contaminated air. Radioactive molecules are absorbed into the body from the air in the lungs. The groundshine dose comes from standing or walking on ground on which radioactive particles have been deposited. The cloudshine and groundshine doses are immediate, as the body receives beta or gamma radiation from radionuclides that decay outside the body. The inhalation dose is received some time later when the radionuclides absorbed from the air decay inside the body.

The conversion factor is computed from an equation that is presented and derived in Reference B.5. It depends upon the dose factor for each pathway, the shielding factor for each pathway, the breathing rate, and the deposition velocity. For each pathway and organ of the body, the dose factor is a constant that depends upon the type of radiation emitted by the isotope, the energy of that radiation, and, for the inhalation dose factor, the selective transport of the isotope to, and absorption in, the specific organ. For cloudshine, the dose factor relates the dose rate to the concentration in the air. For groundshine, the dose factor relates the dose rate to the concentration on the ground. For inhalation, the dose factor relates the dose rate to the amount of the isotope inhaled. The shielding factor accounts for the fact that some of the time the people will be indoors and will be partially shielded by the building. The deposition velocity measures the rate at which solid particles are deposited from the plume.

Table B.5 lists the dose factors used in calculating the isotope conversion factors and the isotope conversion factor itself for 11 representative isotopes. Complete information about the calculation of effects weights, with the values of the parameters used for all 60 isotopes, may be found in Reference B.5. The groundshine dose factor is the factor for an exposure of 8 hours; radioactive decay during this time is accounted for in computing this factor. The applicable concentration is the concentration at the beginning of the period. The inhalation dose factor used is the acute inhalation factor. The cloudshine and groundshine dose factors for Sr-90 are zero because it produces no gamma radiation. The groundshine dose factor for Kr-88 is zero because it is not deposited.

Table B.5 Partitioning parameters and results.

Isotope	Dose Factors: Cloud-shine (1E-15 Sv-m³/Bq-s)	Dose Factors: Ground-shine (8 h) (10^{-12} Sv-m³/Bq)	Dose Factors: Inhal-ation (Acute) (10^{-9} Sv/Bq)	Half-Life	Reactor Inventory (10E+18 Bq)	EF Isotope Conv. Factor	($ELCF_k$ + $CLCF_k$) /R_k (LCF/ 10E+15 Bq)
Co-60	99.6	50.3	0.40	5.3 yr	0.025	6.6	59.2
Kr-88	116.0	0.0	0.0004	2.8 h	2.9	2.0	0.0003
Sr-90	0.0	0.0	1.7	28 yr	0.19	4.3	118.
Zr-95	29.2	0.17	0.28	66 d	5.5	2.5	2.1
Ru-106	8.1	4.6	0.087	369 d	1.0	0.71	5.2
Te-132	7.6	35.3	0.25	3.2 d	4.7	3.4	0.13
I-131	14.5	8.7	0.035	8.0 d	3.2	1.0	0.29
Cs-137	22.2	12.6	0.56	30 yr	0.24	2.8	114.0
Ba-140	7.1	7.3	0.47	13 d	6.2	1.9	0.53
La-140	94.8	44.2	0.14	40 h	6.4	5.4	0.083
Pu-239	0.0017	0.0014	2.4	24000 yr	0.0008	6.0	1565.0

Each isotope is converted into an equivalent amount of I-131 by the equation:

$$EQN_k = CF_k \; I_k \; STN_k \; \exp[\,-\lambda_k\,(TN + DTN/2)],$$

where:

N = 1 for the early release and N = 2 for the late release,

EQN_k = the equivalent amount of I-131 for isotope k for release N,

CF_k = the isotope conversion factor for isotope k,

I_k = the inventory of isotope k at the start of the accident,

STN_k = the release fraction of isotope k for release N,

λ_k = the decay constant for isotope k,

TN = the time of the start of release N, and

DTN = the duration of release N.

Release fractions are determined in the source term calculation for nine radionuclide groups; each isotope is assigned to one of these groups. The total I-131 equivalent release is then found from:

$$EQ = \Sigma_k \, EQ1_k + \Sigma_k \, EQ2_k.$$

The relationship between the size of the equivalent release and the number of early fatalities is nonlinear because of the threshold effect and is complicated by the variability of the weather and the uneven distribution of the population around the site. The last factors are treated by using the weather-averaged, mean, or expected value for early fatalities as calculated by MACCS (Ref. B.59), using the actual site weather and demographic data. The effects of release magnitude were determined by making complete MACCS runs for I-131 releases of different sizes. The result is shown in Figure B.15. The MACCS

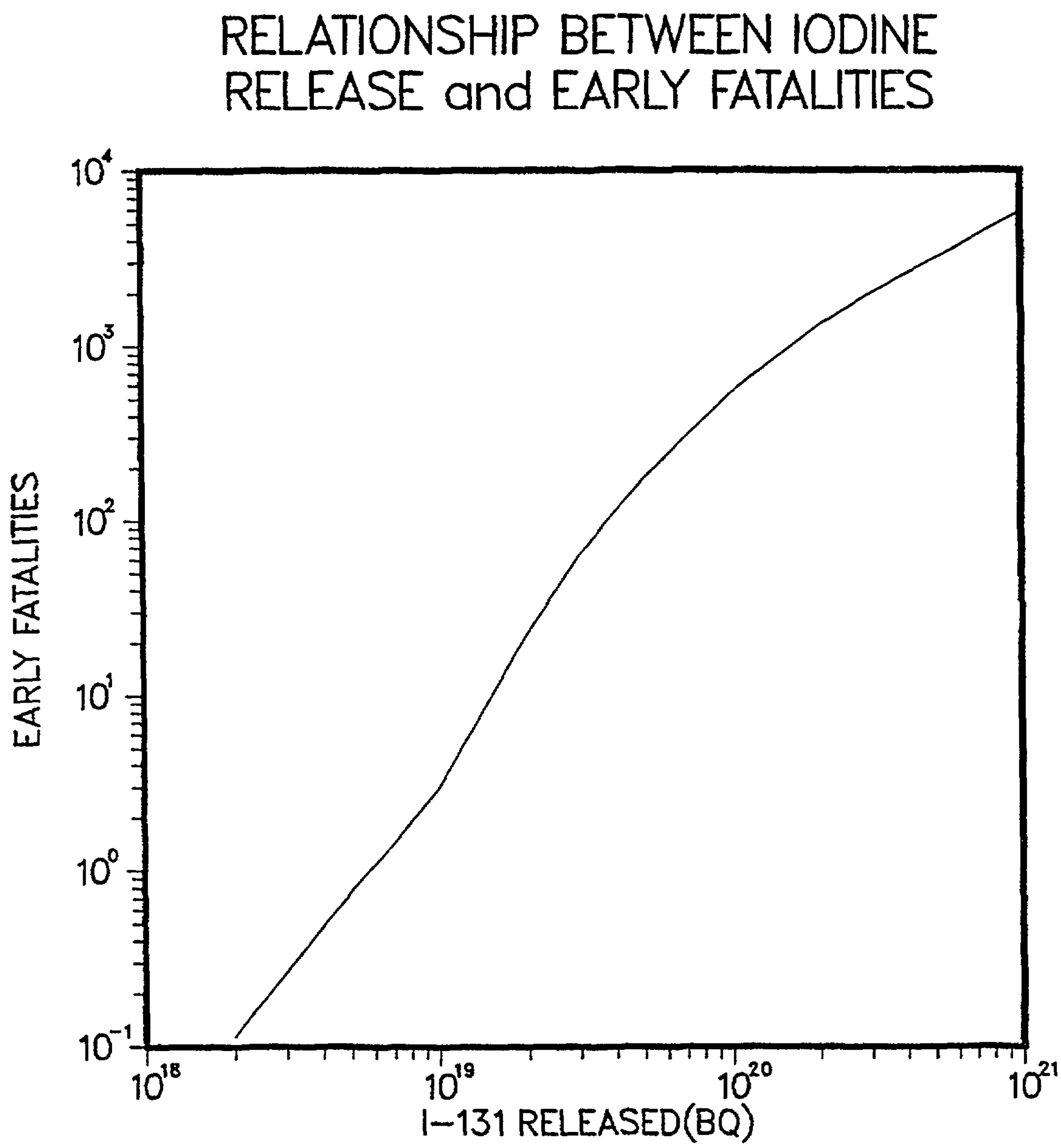


Figure B.15 Relationship between I-131 release and mean early fatalities used in determining early effects weights for partitioning.

calculations assumed an instantaneous release at ground level, with no evacuation or sheltering. No immediate mitigative action was used because the purpose of the EFW is to measure the radiological potential for early fatalities. The relocation criteria normally used for MACCS were not changed, however.

In summary, the procedure in calculating the EFW for a release is to compute the I–131 equivalent for each isotope for both the early and late release and then to total these values to obtain a total I–131 equivalent release. The curve relating the total equivalent release to mean early fatalities, Figure B.15, is then used to find the number of early fatalities, which is the early fatality weight, EFW. Releases with EQs less than 2E+18 Bq are assigned an early fatality weight of zero.

The method used to determine the chronic fatality weight is different. Because of the nearly linear relationship between the amount of an isotope released and the number of latent cancer fatalities, there is no need to convert the amount released for each isotope to an equivalent amount of a reference radionuclide. Instead, MACCS was used to determine the relationship between the amount released and the mean number of latent cancer fatalities for each site, using the weather and population data appropriate for that site. For each isotope, a MACCS run was made in which only that isotope was released. The entire inventory of the isotope in the reactor was released for the inert gases and 10 percent of the entire inventory for the other radioisotopes. The latent cancer fatalities due to the dose obtained in the first 7 days and those due to the chronic dose (received after the first 7 days) were obtained from the MACCS output. Since these fatalities are approximately linear with the amount released, the ratio of their total number to the amount released gives a reliable measure of such fatalities per unit release in the absence of any mitigating actions. This method is not applicable to early fatalities because of the threshold effect.

The chronic fatality weight for each isotope is given by the equation

$$CFW_k = I_k\,(ST1_k + ST2_k)(ELCF_k + CLCF_k)/R_k$$

and the total chronic fatality weight is then

$$CFW = \Sigma_k\, CFW_k,$$

where the summation is over all isotopes. I_k and STN_k have been defined above, and

CFW_k = the chronic fatality weight, in latent cancer fatalities, for a release of an amount $I_k\,(ST1_k + ST2_k)$ of isotope k;

$ELCF_k$ = the number of latent cancer fatalities due to early exposure from a release of an amount R_k of isotope k;

$CLCF_k$ = the number of latent cancer fatalities due to late exposure from a release of an amount R_k of isotope k; and

R_k = the amount of isotope k released in the MACCS calculation used to determine $ELCF_k$ and $CLCF_k$.

The early exposure is that obtained in the first 7 days after the accident and the late exposure is the exposure obtained after the first 7 days. The distinction is made because slightly different methods of these two periods are used in MACCS. Table B.5 lists values of the ratio $(ELCF_k + CLCF_k)/R_k$ for 11 representative isotopes. A total of 60 radionuclides are used in the consequence calculation. A complete listing of the conversion factors and CFWs for each may be found in Reference B.5.

Table B.5 also lists the half-life of the isotopes, and the reactor inventory at the time the accident starts. The half-life, the inventory, the EF isotope conversion factor, and the specific CFW, together with a rough idea of the release fractions for each radionuclide group, can be used to assess roughly the importance of the isotope to early and late offsite health effects. The 11 isotopes listed are all fairly important for at least one of the two consequence measures as this was one of the criteria for their selection. However, Kr-88 is clearly of little importance for chronic fatalities because the half-life and the specific CFW are low. Pu-239, on the other hand, can be seen to be of less importance than other isotopes for early fatalities because the inventory is very low and the isotope conversion factor is about the same size as that for more common fission products. The release fractions for Pu-239 also tend to be low.

The early and chronic fatality weights described here are not used in calculating consequences; they are only used in partitioning the source terms into groups of similar source terms. An average source term for each subgroup is used to calculate the offsite consequences with MACCS.

The process described in this section is applied to the source term for bin FFA-CAC-ABA-CA in Observation 4; the result is EFW = 0.0 and CFW = 2.7. The EFW is zero for this bin because the release is so low. Evacuation is not taken into account in determining the EFW, so the fact that the evacuation is complete before the FFA-CAC-ABA-CA release starts has no effect on the EFW although it may be very important in computing the actual consequences.

B.5.3 Partitioning Process and Results

The partitioning process divides the "space" defined by the logarithm of the effect weights into a number of cells. For the source terms for which both EFW and CFW are nonzero, this produces a rectangular grid on a two-dimensional plot. For the source terms for which the EFW is zero, the "space" to be divided is one-dimensional; that is, partitioning involves defining cells based on CFW alone. In the Surry analysis, there are 11,771 source terms with EFW = 0.0. The source term for bin FFA-CAC-ABA-CA is one of these.

For the 11,771 source terms with EFW = 0.0, the range of $\log_{10}$(CFW) is from −4.2 to 3.7. This range was divided into six groups or cells. The results of the initial division was:

Group		1		2		3		4		5		6	
$\log_{10}$(CFW)	−4.2		−2.9		−1.6		−0.2		+1.1		+2.4		+3.7
Count		349		1553		2892		1599		2438		2940	
% Weighted Freq.		7.9		33.2		49.3		2.5		4.9		2.1	

The second line gives the values of $\log_{10}$(CFW) that divide the range into six cells or groups. The third line gives the number of source terms in each group, and the last line gives the percentage of the weighted frequency in that group. (The weighted frequency of a bin is the absolute frequency of the bin, or the PDS frequency multiplied by the bin probability for the observation that applies.) The sixth group has slightly more source terms in it than the third group, but the frequencies of the source terms in the sixth group are very low. Thus the percentage of the frequency in group 3 is 25 times as high as that in group 6. The source term for bin FFA-CAC-ABA-CA has CFW = 2.7, and $\log_{10}(2.7)$ = 0.43, so the source term for bin FFA-CAC-ABA-CA goes into group 4. Group 4 includes all the source terms that have EFW = 0.0 and for which $\log_{10}$(CFW) lies between −0.2 and +1.1. Source terms are placed in this group if they meet these criteria no matter what PDS or APB they represent. Although the number of source terms in group 4 is fairly high at 1,599, the frequency of these source terms on the whole is fairly low, and the fraction of the frequency in group 4 is only 2.5 percent.

It is not worth making separate calculations for groups that have a small fraction of the weighted frequencies. Based on the information given in the table immediately above, it was decided that three CFW groups would be sufficient. Groups 1, 4, and 6 were eliminated, and the source terms in those groups were pooled with the neighboring groups. Groups 4 and 6 were pooled with groups 3 and 5 because fractions of the weighted frequencies in groups 4 and 6 were so low. The frequency of group 1 is higher than that of group 5, but the consequences of group 1 are very low, so the absorption of the source terms of group 1 into group 2 has a negligible effect on the risk.

The final partitioning is:

Group		1		2		3		4		5		6	
$\log_{10}$(CFW)	−4.2		−2.9		−1.6		−0.2		+1.1		+2.4		+3.7
Count		0		1902		3223		0		6646		0	
% Weighted Freq.		0		41.1		49.8		0		9.1		0	

The $\log_{10}$(CFW) for bin FFA-CAC-ABA-CA is just above the center value for group 4, so it was placed in group 5. The average values of the release fractions and the release characteristics for group 5 at the end of partitioning include the source terms originally in groups 4 and 6 that have been absorbed into group 5, not just the source terms originally in group 5.

Group 5 in the partitioning process for source terms with zero EF weight becomes source term 17 when the partitioning for source terms with nonzero EF weights are considered and empty groups are eliminated. The mean frequency of this group is 3.4E−6/reactor year, and the conditional probability of this group, with respect to the source terms with EFW = 0.0, is 0.084.

Each source term group is subdivided into three source term subgroups (STGs) on the basis of evacuation timing:

STG1 (early evacuation): Evacuation starts at least 30 minutes before the release begins.

STG2 (synchronous evacuation): Evacuation starts between 30 minutes before and 1 hour after the release begins.

STG3 (late evacuation): Evacuation starts 1 or more hours after the release begins.

For source term 17, STG1 has 94 percent of the source terms, and STG2 has 6 percent of the source terms. There are no source terms in STG3. As would be expected from the very late release time for the source term for bin FFA-CAC-ABA-CA for Observation 4, this source term goes to STG1.

Consequence calculations are not done for source term group 17 as a whole; they are done for the subgroups since the timing differs markedly between the subgroups. The mean source terms for each subgroup form the basis for each consequence calculation. The properties of subgroup 1 of source term 17 are given in Table B.6. This information is used by the computer code MACCS (Ref. B.59) to calculate the offsite consequences. Although the source term for bin FFA-CAC-ABA-CA of Observation 4 had a zero early release, some of the other source terms placed in STG1 of source term 17 in the partitioning process had nonzero early releases, thus the mean early release fractions for STG1 are nonzero. The representation of the source term for FFA-CAC-ABA-CA by a source term with a nonzero early release does introduce a slight distortion. However, the early release is small, and it occurs at 13.3 hours, which is well after the time at which the warning to evacuate is given, 6.9 hours. The small early release and the fact that the evacuation is completed before the release commences mean that there are no early fatalities from the early release for STG1. As the latent cancer fatalities and population dose do not depend on the release timing to any significant degree, the error introduced by the nonzero early release is negligible.

The results of partitioning are contained in two files. One file contains all the source terms that MACCS will use to calculate consequences. STG1 of source term 17 is the 49th source term in this file and is referred to as SUR−49 for the consequence calculation. The second output file indicates the STG in which each bin of each observation was placed in the partitioning process. For bin FFA-CAC-ABA-CA of Observation 4, this was STG1 of source term 17, now known as SUR−49.

Table B.6 Properties of source term 17, subgroup 1.

Property	Minimum Value	Maximum Value	Frequency Weighted Mean
Release Height	10	10	10
Warning Time	2.2E+4	3.6E+4	2.5E+4
Start Early Release	4.7E+4	5.1E+4	4.8E+4
Duration Early Release	0.0	3.6E+3	3.3E+2
Energy Early Release	0.0	7.0E+8	9.2E+5
ERF Xe, Kr	0.0	1.0E 0	1.4E-1
ERF I	0.0	1.5E-1	7.3E-3
ERF Cs, Rb	0.0	1.1E-1	5.4E-3
ERF Te, Sc, Sb	0.0	2.9E-2	1.2E-3
ERF Ba	0.0	1.4E-2	1.2E-4
ERF Sr	0.0	2.4E-3	2.3E-5
ERF Ru, etc.	0.0	1.1E-3	6.6E-6
ERF La, etc.	0.0	5.2E-3	2.8E-5
ERF Ce, Np, Pu	0.0	1.4E-2	1.4E-4
Start Late Release	4.7E+4	1.3E+5	1.1E+5
Duration Late Release	1.0E+1	2.2E+4	1.2E+4
Energy Late Release	0.0	7.0E+8	9.2E+5
LRF Xe, Kr	0.0	1.0E 0	8.1E-1
LRF I	5.0E-6	1.3E-1	4.0E-2
LRF Cs, Rb	0.0	5.0E-2	3.9E-4
LRF Te, Sc, Sb	3.4E-11	9.6E-2	2.7E-4
LRF Ba	6.3E-14	1.7E-2	4.9E-5
LRF Sr	1.0E-18	1.4E-3	2.7E-6
LRF Ru, etc.	5.2E-18	1.6E-3	4.2E-6
LRF La, etc.	5.2E-18	1.7E-3	6.5E-6
LRF Ce, Np, Pu	1.6E-13	1.4E-2	4.2E-5

Note: ERF means early release fraction, LRF means late release fraction, the release height is in meters, the energy of the release is in watts, and all times are in seconds.

B.6 Consequence Calculation

The computer code MACCS (Ref. B.59) is used to determine the consequences of a release of fission products from the damaged reactor. Consequences are the offsite results of the accident expressed in societal terms; for example, number of early fatalities or the risk of latent cancer fatality to the population within 10 miles of the plant. A separate MACCS calculation is performed for the mean source term associated with each STG.

B.6.1 Description of Consequence Calculation

The consequence calculation is an extensive calculation. The inventory of fission products in the reactor at the time of the accident and the release fractions for each radionuclide class are used to calculate the amount released for each of the 60 isotopes considered by MACCS. Then, for a large number of weather situations, the transport and dispersion of these radionuclides in the air downstream from the plant is calculated. The amounts deposited on the ground are computed for each distance downwind. Doses are computed for a hypothetical human at each distance from immersion in the contaminated air, from breathing the contaminated air, from the exposure due to radioactive material deposited on the ground, and from drinking water and eating food contaminated by deposited radioactive particles. The first three of these dose pathways result in immediate doses that can cause health effects within a few hours or days of the release. These are called acute effects. In addition to the three pathways that cause acute effects, long-term exposure from contaminated ground and ingestion also contributes to latent cancer fatalities and other delayed effects. These are known as chronic effects. Doses to an individual are converted to population doses using population data. Doses and health effects are calculated for nine organs of the human body.

The consequence calculation requires a large amount of supporting data. Files of weather data and demographic data specific to the plant region are required. Information is needed on land use (crops grown or dairy use) and land value in the surrounding area. Dose factors are used to relate the dose to each organ to the concentration of each of the isotopes as explained in Section B.5.2.

To assess the effects of different weather on the consequences, the complete transport, deposition, and dose calculation is repeated over 1,000 times for each source term. For each of 16 wind directions, the consequence calculation is performed for about 130 different weather situations. Weather data from the specific plant being modeled are used. The wind direction determines the population over which the plume from the accident passes. The atmospheric stability is also important as it determines the amount of dispersion in the plume downwind from the plant. Deposition is much more rapid when it is raining than when it is not. Each weather sequence contains information about how the wind direction, stability, and precipitation changes from hour to hour. The consequence calculation also computes the effects of the evacuation of the population from the immediate area around the plant. More information on the consequence calculation may be found in Reference B.59.

B.6.2 Results of Consequence Calculation

As discussed in Section B.5.3, the source term for bin FFA-CAC-ABA-CA for Observation 4 was included in group SUR-49 in the partitioning process. The eight consequence measures used in the risk computation, with the mean or expected values computed for partition group SUR-49, are:

Early Fatalities	0.0
Early Injuries	4.2E-6
Latent Cancer Fatalities	1.1E+2
Population Dose—50 miles	2.7E+5 person-rem
Population Dose—region	6.9E+5 person-rem
Economic Cost (dollars)	1.8E+8
Individual Early Fatality Risk—1 mile	0.0
Individual Latent Cancer Fatality Risk—10 miles	7.6E-5

The distribution for latent cancer fatalities is shown in Figure B.16. Note that these consequences assume that the release SUR-49 occurred.

The population doses are the effective dose equivalent whole body doses. The individual early fatality risk is the risk to an average individual within 1 mile of the site boundary. It is calculated using the actual population distribution within 1 mile of the site boundary. For SUR-49, there are no early fatalities and the early fatality risk within 1 mile is zero as well. This is due to the very small magnitude of the early release. As 0.5 percent of the population is assumed not to evacuate when told to do so, neither of these measures would be zero if the release were large enough to cause early fatalities. The individual latent cancer fatality risk is the risk to an average individual within 10 miles of the plant, excluding the food pathways. The low total release fractions for SUR-49 result in a low value for this risk measure also.

After the MACCS calculations are completed, the results for each weather trial (wind direction and weather sequence) for each STG are used in constructing complementary cumulative distribution functions, as discussed below in Section B.7.3. Averages over the weather trials determine the expected consequence values for each STG. These are extracted and assembled into a matrix for use in determining distributions of expected values of risk.

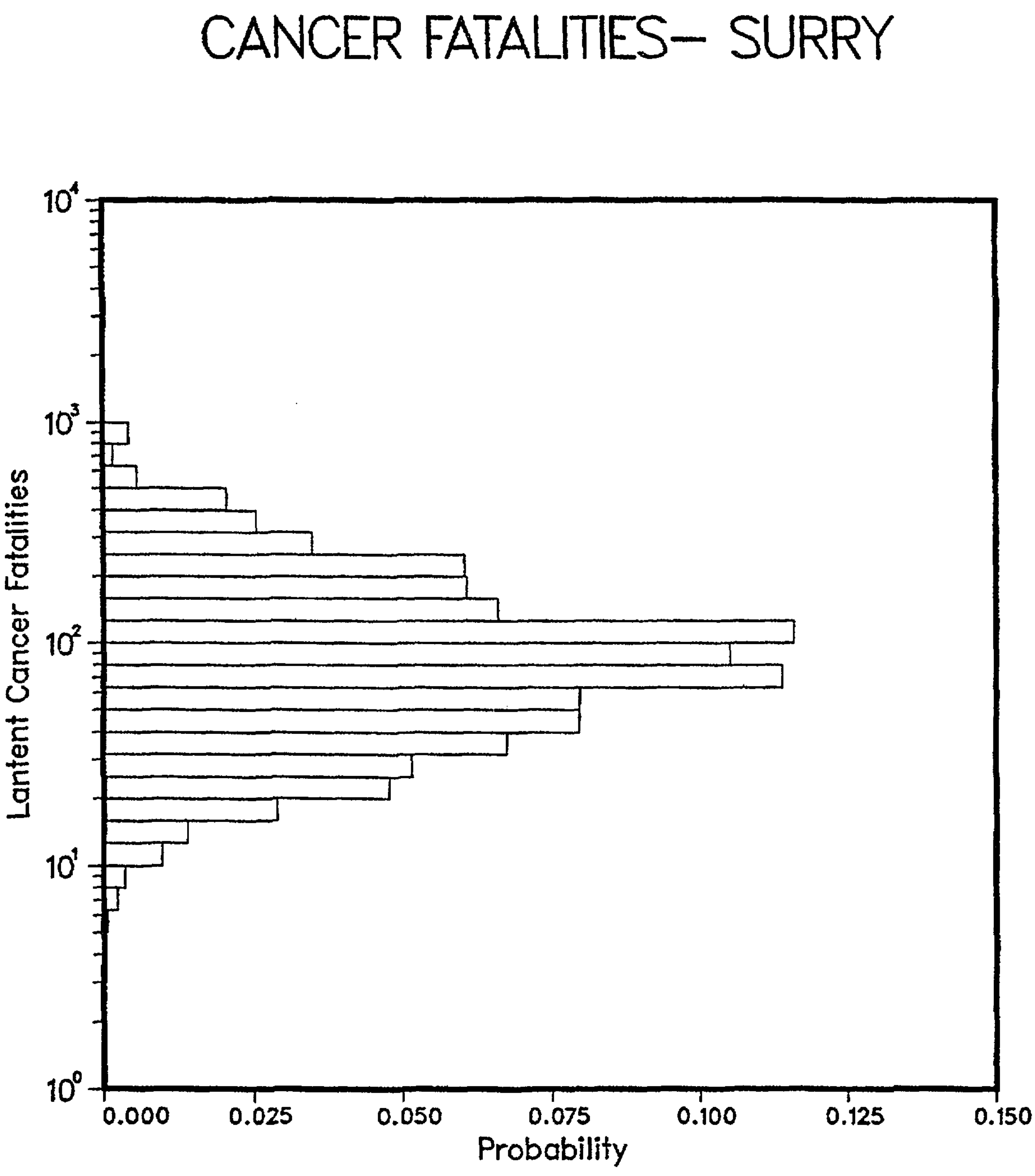


Figure B.16 Distribution of latent cancer fatalities computed for STG SUR–49.

B.7 Computation of Risk

The risk measures that form the final result of a PRA are formed by merging the results of all the preceding analyses. Section B.7.1 introduces the concepts and terms. Mean risk is discussed in Section B.7.2, and displays that show the variability in risk due to the weather and other variables are discussed in Section B.7.3.

B.7.1 Introduction

The final step in a complete risk calculation is putting together the results of all the steps described above to produce a measure of risk. The accident progression analysis and the source term analysis were carried out on conditional bases. That is, the accident progression analysis was performed for each PDS group for each observation without regard for the frequency of that PDS group. Similarly, for each observation in the sample, the source terms for each bin were calculated without taking the bin frequency into account. The partitioning process, however, used the absolute frequency of each source term in determining which groups of source terms could be combined into neighboring groups and in determining the mean values used to represent each STG. To accomplish this, for each observation, the absolute frequency of each PDS group was obtained from the accident frequency analysis, and the conditional probability of the bins was obtained from the accident progression analysis. The consequence analysis was performed for each source term subgroup without regard for its absolute or relative frequency.

Risk is generally displayed in three forms: mean risk, histograms of risk, and complementary cumulative distribution functions (CCDFs). For each sample member and for each consequence measure, the results are averaged over the weather conditions to obtain an expected value. If, for one consequence measure, the average of the 200 expected values is taken, the result is actually the mean value of the expected risk, or the mean value of the weather-averaged mean risk. It is usually referred to as just the mean risk. The mean values of expected risk do provide single numbers (for each consequence measure) that characterize the risk. However, they provide no information about the variability or uncertainty in the risk. Histograms of expected risk display the variation in weather-averaged expected risk among the members of the sample. Mean risk is discussed in Section B.7.2 below.

To display the variability in risk due to the weather, CCDFs are used. While a separate CCDF is formed for each observation, not all 200 are usually plotted at once. Instead, statistical measures of the sample of CCDFs are plotted. CCDFs are presented and explained in Section B.7.3.

B.7.2 Calculation and Display of Mean Risk

As described in Section B.6, a separate consequence calculation is performed with MACCS for each source term subgroup, and the result was a large number of consequences. They are represented by C_{lvwk}, which is the value of consequence measure l for wind direction v, weather sequence w, and source term subgroup k. When a frequency-weighted average is formed over the wind direction and the weather sequences, a set of mean or expected consequence measures, C_{lk}, is formed. As there are 52 source term subgroups, 52 expected values are computed for each of the eight consequence measures.

The expected risk for measure l and observation n can be expressed in terms of the results of the individual analyses by:

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

where:

Risk_{ln} = expected risk for consequence measure l for observation n (consequences/reactor year);

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

$P_n\, (\mathrm{IE}_h \rightarrow \mathrm{PDS}_i)$ = probability that initiating event h leads to PDS_i for observation n;

$P_n\ (PDS_i \rightarrow APB_j)$ = probability that PDS_i leads to APB_j for observation n;

$P_n\ (APB_j \rightarrow STG_k)$ = probability that APB_j was partitioned into source term subgroup k for observation n; and

C_{lk} = expected value of consequence measure l for source term subgroup k.

All the terms in the equation change from observation to observation except C_{lk}. The distribution for initiating event frequency is part of the data used in the calculations performed in the accident frequency analysis. The summation over all the different initiating events is performed in the accident frequency analysis, and the output of the accident frequency analysis is actually

$$f_n(PDS_i) = \Sigma_h f_n(IE_h)\ P_n(IE_h \rightarrow PDS_i),$$

the frequency for each PDS group for each observation.

$P_n\ (PDS_i \rightarrow APB_j)$ is the result of the accident progression analysis and the evaluation of the accident progression event tree. It consists of a list of APBs for each PDS group, with a probability for each, for each observation. The APB probability is conditional on the occurrence of the PDS group.

$P_n\ (APB_j \rightarrow STG_k)$ represents the combined result from the source term calculation and the partitioning process. For each observation, a source term is computed for each APB in that observation. The source terms for all 200 observations in the sample are placed in source term subgroups in the partitioning process. The outcome of the source term analysis and partitioning is the STG to which each APB in each observation is assigned and a mean source term for each STG. $P_n\ (APB_j \rightarrow STG_k)$ is an element of the matrix that contains this information; the matrix element for given values of j and k is 1.0 if APB_j was placed in STG_k and 0.0 otherwise.

$Risk_{ln}$ is the expected risk for each observation. What is reported as the mean risk for measure l is really the expected or mean value of expected risk:

$$Risk_l = \Sigma_n Risk_{ln}/n_{LHS}$$

where n_{LHS} is the number of observations in the LHS (200 for Surry). That is, the expected or mean value of risk is the average of 200 values of weather-averaged expected risk.

The distribution of the 200 values for the mean latent cancer fatality (LCF) risk are displayed in the form of a histogram. Such a histogram for LCFs is shown in Figure B.17 for 99.5 percent evacuation. The quantiles are obtained by ordering the 200 observations by the value of the independent variable. The 5th and 95th percentiles, the mean, and the median are shown. Not shown are the extreme values (Observation 123, 7.8E–5 LCFs/reactor year and Observation 76, 7.5E–2 LCFs/reactor year) since they are not indicative of the sample as a whole. The mean for Observation 4 is 5.0E–3 LCFs/reactor year, which is near the 75th percentile and is well above the median.

The equation for risk given above provides the means for determining the risk due to certain subsets or groups of events and the contribution of these subsets to mean risk. For example, say the contributions of the APBs were desired. The risk due to APB_k for observation n is:

$$Risk_{lnj} = \Sigma_h \Sigma_i \Sigma_k\ f_n(IE_h)\ P_n(IE_h \rightarrow PDS_i)\ P_n(PDS_i \rightarrow APB_j)\ P_n(APB_j \rightarrow STG_k)\ C_{lk}.$$

For each observation, the fractional contribution of APB_j is:

$$Risk_{lnj}/Risk_{ln}.$$

The risk and the fractional contributions for combinations of APBs can be determined in an analogous manner.

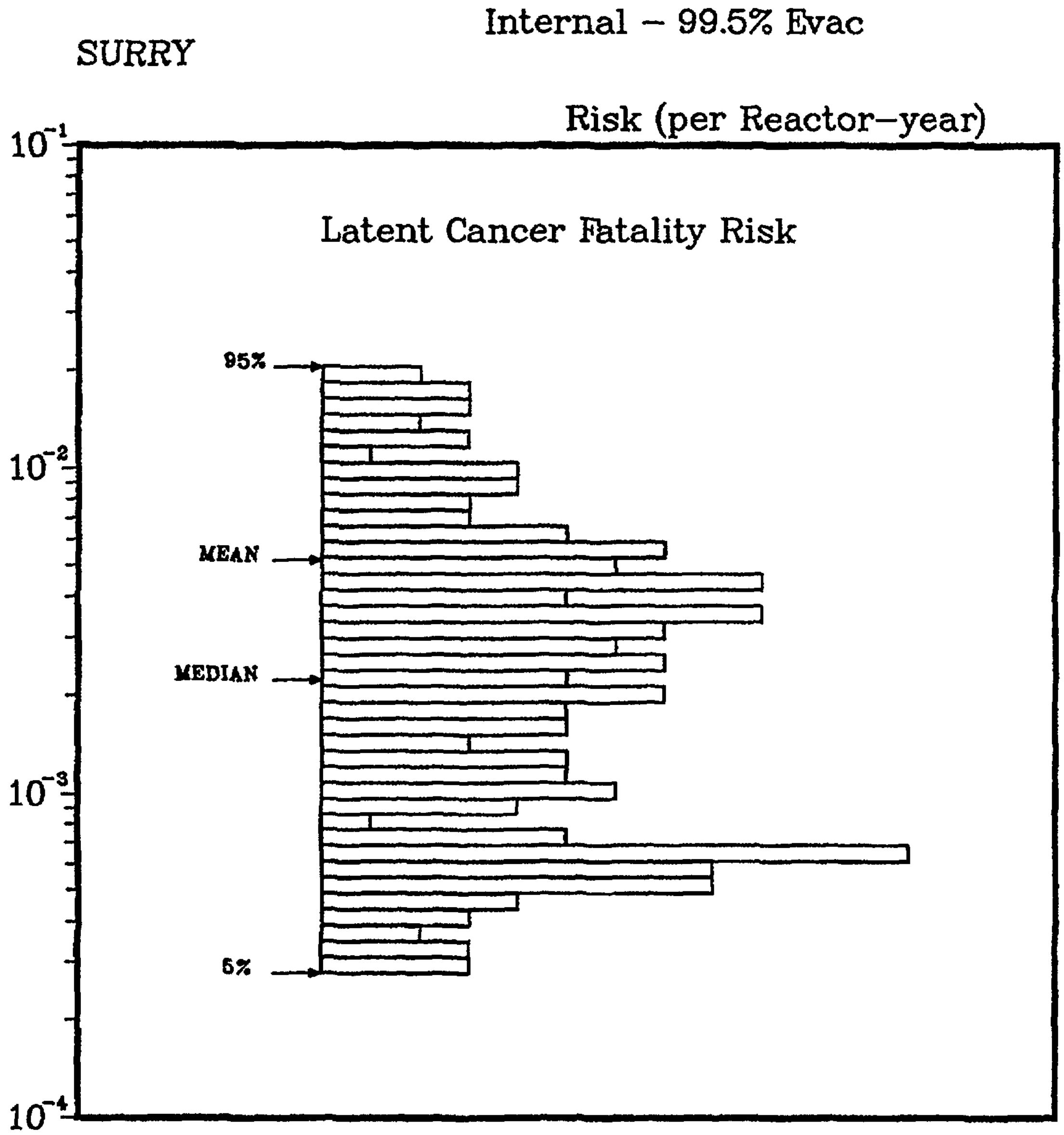


Figure B.17 Distribution of expected (weather-averaged) latent cancer fatality risk for Surry.

For the sample as a whole, the mean risk from APB_k is:

$$(\Sigma_n Risk_{lnj})/n_{LHS}.$$

Fractional contributions to mean risk for the entire sample can be determined in two ways. The expression

$$[\Sigma_n (Risk_{lnj})/Risk_{ln}]/n_{LHS}$$

is termed the mean fractional contribution to risk for APB_j. The expression

$$[(\Sigma_n\ Risk_{lnj})/n_{LHS}]/Risk_l$$

is termed the fractional contribution to mean risk for APB_j. Figure 3.14 of the main report shows the fractional contribution to mean risk for groups of APBs. Because the distributions for risk often have very long tails and the mean is then determined by the few observations with the highest risk, the fractional contribution to mean risk and the mean fractional contribution to risk can be quite different. There is no consensus that one method of calculating the contribution to mean risk is preferable to the other.

The summations in the expression for $Risk_{ln}$, and the computations necessary to obtain the statistical measures, are performed by a computer program named RISQUE. A description of this code may be found in an appendix to Reference B.5.

B.7.3 Calculation and Display of CCDFs

As already indicated, the output from MACCS is a large number of consequences represented by C_{lvwk}, the magnitude of consequence measure l for wind direction v, weather sequence w, and source term subgroup k. When the results for each weather trial (combination of wind direction and weather sequence) are kept separate and the probability of each weather trial is taken into account, a complementary cumulative distribution function (CCDF) may be obtained for each STG. Figure B.18 displays the latent cancer fatality CCDFs for STG SUR-49 and for all 52 STGs. The CCDF for a single STG displays the effects of the variability of the weather and relates the magnitude of the consequence to the probability that it will be exceeded. (The cumulative distribution function (CDF) displays the probability that a certain value will not be exceeded. The CCDF displays the probability that a certain value will be exceeded. CCDFs are usually shown for historical reasons.) The CCDFs in Figure B.18 have conditional probability on the ordinate: each curve displays the latent cancer fatality results conditional on the occurrence of a release. For example, the top plot in the figure displays the results of the consequence analysis conditional on the SUR-49 release.

To understand how the CCDF for a single STG is formed, let the subscript u represent a combination of wind direction and weather sequence; then C_{lvwk} may be written C_{luk}. Each combination of wind direction and weather sequence is denoted a weather trial. C_{luk} is now the value of consequence measure l for weather trial u and source term subgroup k. The CCDF for consequence measure l and STG_k is formed from a set of results of the form:

$$(C_{luk}, P_u) \qquad u = 1, 2, ..., nWT,$$

where:

P_u = probability that weather trial u will occur, and

nWT = number of weather trials (about 2,500).

The set (C_{luk}, P_u) is ordered on the consequence measure (i.e., $C_{luk} < C_{l,\ u+l,\ k}$). The results of the consequence evaluation include both C_{luk} and P_u.

Since it is the complementary cumulative probability that is plotted in Figure B.18, this figure results from plotting the set $(C_{luk}, 1 - CP_u)$ where CP_u is the cumulative probability defined by:

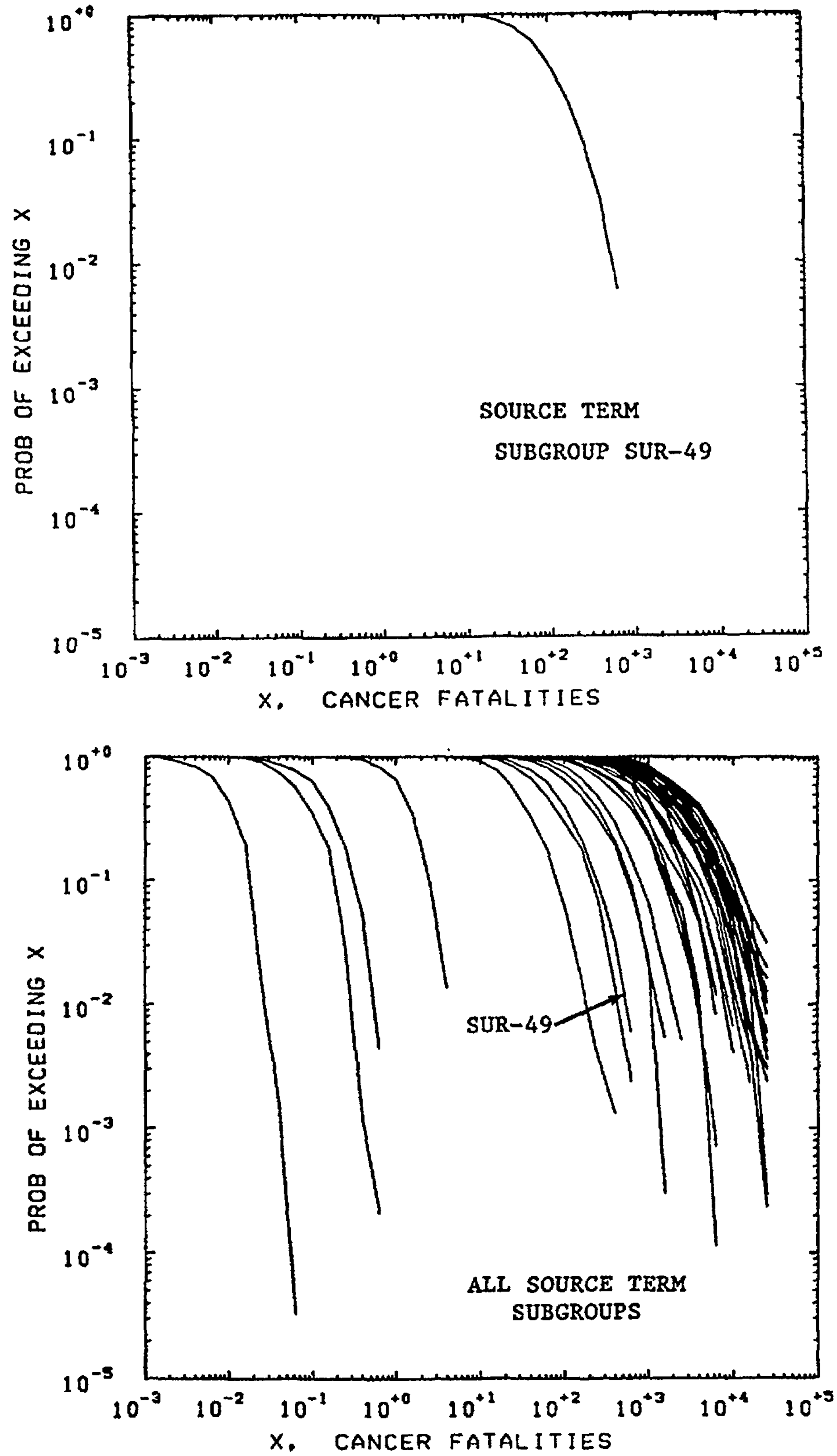


Figure B.18 CCDFs for latent cancer fatalities for STG SUR-49 and for all 52 STGs.

$$CP_u = \sum_{q=1}^{u} P_q.$$

Figure B.18 shows the probability of exceeding a given consequence. For instance, the top plot shows that the conditional probability of exceeding 100 LCFs is about 0.10, given the occurrence of SUR-49.

The next step is to form the CCDF for each of the n_{LHS} (200) observations that comprise the sample. This is done by considering a large number of values on the abscissa in the lower plot in Figure B.18. Let X represent a value of a consequence measure; in this figure X would be a number of latent cancer fatalities. For each STG_k, the ordinate (i.e., Y) of the point (X,Y) on the curve defined by $(C_{luk}, 1 - CP_u)$ gives the probability of exceeding X consequences. Let this probability be represented by $P_{xc_k}(X)$. The next factor needed is f_{nk}, the frequency of STG_k for observation n. The equation for f_{nk} is analogous to the equation for $Risk_{ln}$:

$$f_{nk} = \Sigma_i \Sigma_j \, f_n(PDS_i) \, P_n(PDS_i \rightarrow APB_j) \, P_n(APB_j \rightarrow STG_k),$$

where all the terms have already been defined in Section B.7.2.

So for one STG_k, the contribution to the CCDF for observation n is the set of points $(X, P_{xc_k}(X) f_{nk})$, where the product $P_{xc_k}(X) f_{nk}$ represents the frequency for observation n that STG_k will result and that X will be exceeded. When all the STG_k are considered, it may be seen that the set of points that defines the CCDF for observation n becomes

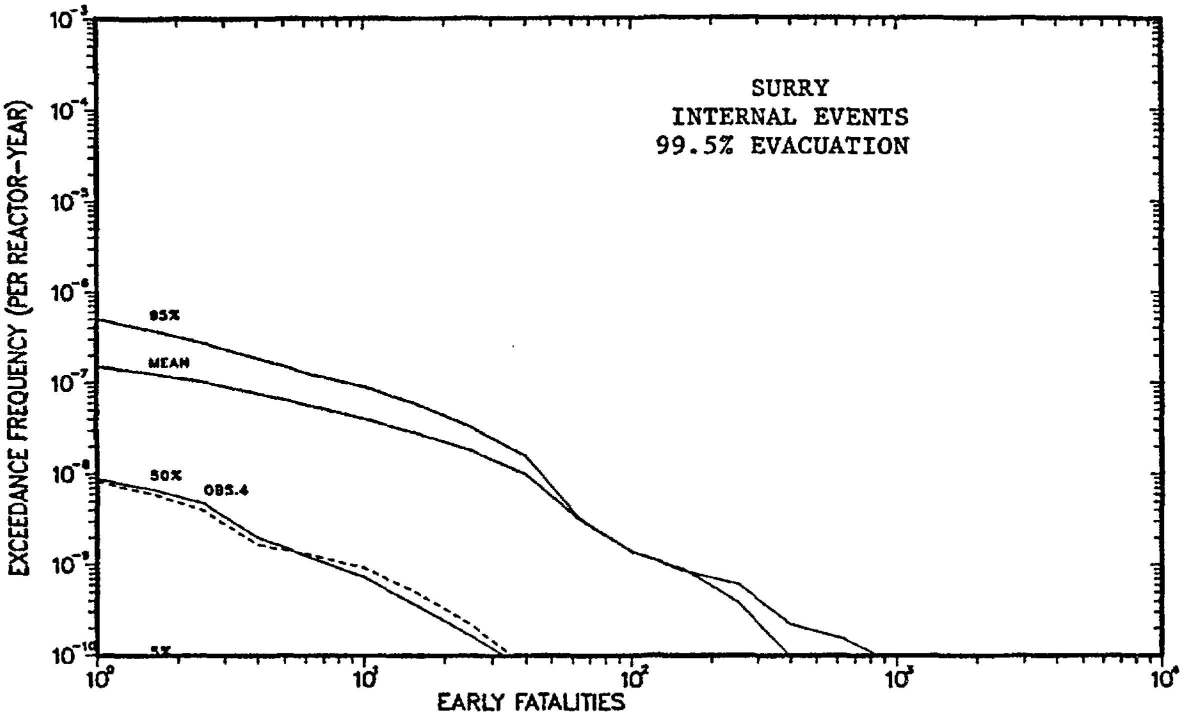
$$(X, \Sigma_k [P_{xc_k}(X) f_{nk}]),$$

where the summation is over all the source term subgroups. As there are 200 observations in the sample, 200 such CCDFs are calculated. The dashed lines on the plots in Figure B.19 are the CCDFs for Observation 4 for early fatalities (EFs) and LCFs.

While the CCDFs for all 200 observations for consequence measure l could be placed on one plot, the result is cluttered and hard to read. Instead, four statistical measures of the set of 200 curves are plotted. The four measures of the sample are the 5th percentile, the 95th percentile, the median, and the mean. They are shown on Figure B.19. These four curves do not represent specific observations. Instead, each point on these curves is a statistical measure of the sample at that value of the consequence. For example, consider the mean curve for latent cancer fatalities in Figure B.19. For 100 LCFs, there are 200 values of the exceedance frequency, one for each observation in the sample. As each observation is as likely as the next, these 200 values are summed and divided by 200 to determine the mean for 100 LCFs. This process is repeated for 105 LCFs, 110 LCFs, and so on. In this way, the points that comprise the mean curve are obtained. The result does not coincide with any one of the 200 observations. The 5th and 95th percentiles are obtained in an analogous manner. Since the 200 values for a given LCF value are ordered, the 5th percentile value is the value of the 10th observation.

The CCDF for each observation displays the effects of the variability of the weather, through the term P_u, as well as some variability due to the sampling process, through the term f_{nk}. The shape of the CCDF for a single observation depends upon the relative frequency of each STG in that observation, which in turn depends on all the sampled factors in the accident frequency analysis, the accident progression analysis, and the source term analysis. As the CCDFs for each STG are the same for all the observations in the sample, the differences among the CCDFs for the 200 observations are due to the different frequencies for each STG in each observation.

The CCDF can be read in a number of ways. Consider the median curve for LCF in Figure B.19; it shows that the median frequency of exceeding one to ten LCFs is between 3E-6/reactor year and 4E-6/reactor year; the median frequency of exceeding 100 LCFs is about 2E-6/reactor year; the median frequency of exceeding 1,000 LCFs is 5E-7/reactor year; and the median frequency of exceeding 10,000 LCFs is 5E-9/reactor year. The same interpretation is valid for the other measures of the sample of 200 CCDFs. Another way to read a CCDF is to consider one particular value of the consequence. Take 1,000 LCFs for



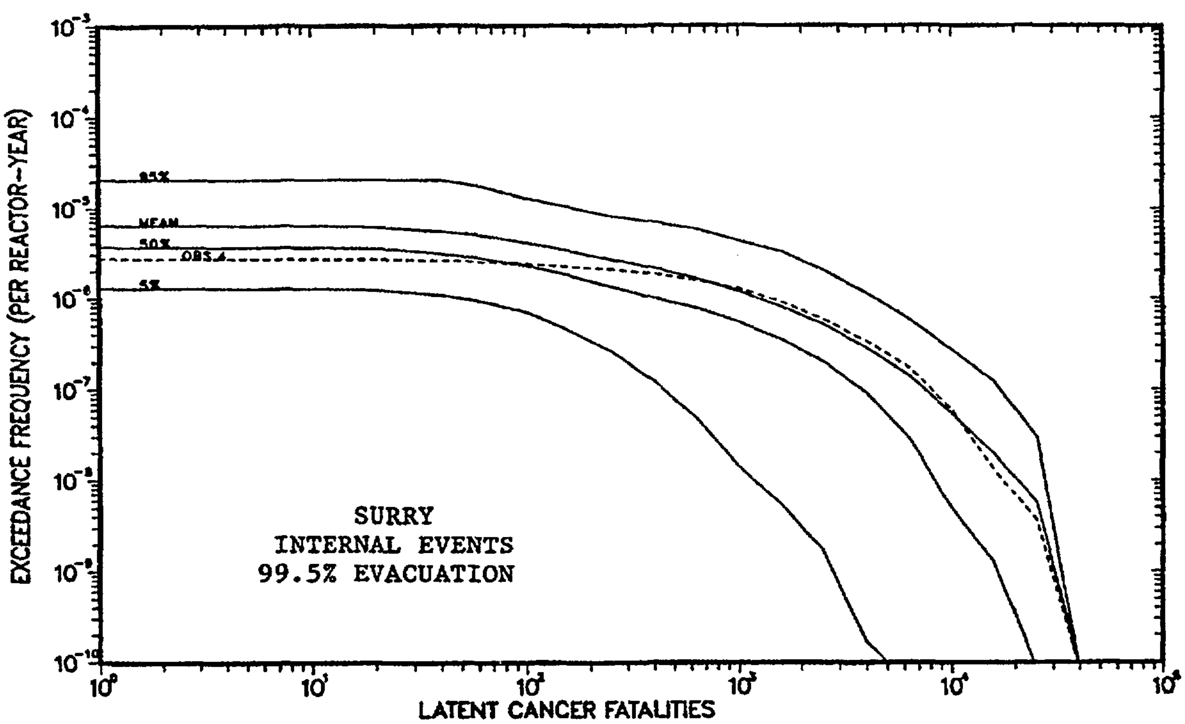


Figure B.19 Computed curves showing four statistical measures of 200 CCDFs for Surry for early fatalities and latent cancer fatalities. (The CCDF for Observation 4 is also shown.)

example: in 95 percent of the observations, the frequency that 1,000 LCFs would be exceeded is less than 4E–6/reactor year; while in half of the observations, the frequency that 1,000 LCFs would be exceeded is less than 5E–7/reactor year.

Because of the wide range of the consequences, the CCDFs are plotted with a logarithmic scale on both axes. The distributions typically have long "tails" in both directions. For distributions like this, the mean value often reflects just a few high observations. Such is the case for the early fatalities in Figure B.19. Above about 150 EFs, the mean value of the sample exceeds the 95th percentile. This means that the mean in this portion of the plot is determined by fewer than 10 observations out of the 200 in the sample.

B.8 Summary

In this section, a specific accident at Surry has been followed all the way through a complete analysis, from the initiating event in the accident frequency analysis through to the offsite risk values computed by the consequence analysis. The example selected was a fast SBO accident; the sequence is denoted T1S1-QS-L, and the PDS is TRRR-RSR. To determine the uncertainty in risk, a sample of 200 observations was used for the Surry analysis. Observation 4 was followed in this example, and specific numbers for the important events and parameters for this observation were given above.

This accident is initiated by the LOSP. The emergency DG dedicated to Unit 1 (DG1) fails to start, and the swing DG (DG3) also fails to start, so all ac power is lost, which is designated a station blackout. The only means of heat removal from the reactor core in a station blackout situation is the steam-turbine-driven (STD) AFW train, which operates as designed initially. However, a secondary system safety relief valve fails to close, and the water lost out this valve fails the AFWS in about an hour. Offsite power is not restored by this time, and the plant staff fails to get one of the two DGs capable of supplying power to Unit 1 to start. With the failure of the STD-AFW train, and no ac power to run the motor-driven AFW trains, there is no heat removal from the reactor. The reactor coolant system (RCS) heats up until the pressure forces open the PORVs. Water (steam) loss through the PORVs continues, with the PORVs cycling open and closed, until enough water has been lost to reduce the liquid water level below the top of active fuel. Without electric power, there is no way to replace the water lost from the RCS. After the water level has dropped some distance below the top of active fuel, the fuel above the water level heats up sufficiently to degrade.

The accident frequency analysis determined the frequency of the initiating events and the probabilities of the other failures that led to this state. The frequency of the initiating event for Observation 4 is 0.099/reactor year, and the frequency of the cut set defined in Table B.1 is 3.4E–8/reactor year. This cut set was placed with many others in sequence T1S1-QS-L, which has a frequency, for Observation 4, of 2.4E–7/reactor year. Sequence T1S1-QS-L was grouped with other fast station blackout sequences in plant damage state TRRR-RSR, which has a frequency of 4.8E–7/reactor year for Observation 4. As TRRR-RSR is the only PDS in PDS group 3, this is also the group frequency.

The core melt process and the response of the containment were treated in the accident progression analysis. Although the most likely outcome for TRRR-RSR was termination of the accident without failure of either the reactor vessel or the containment due to the recovery of offsite power, the path followed through the APET took the branch in which power was not restored in time to avert vessel breach. The RCS was at the PORV setpoint pressure at the start of core degradation since there were no breaks in the system. However, the path through the event tree took the branch in which the RCP seals failed, and the vessel failed with the RCS at intermediate pressure. While high-pressure melt ejection and direct containment heating accompanied the failure of the vessel, the pressure rise in the containment was insufficient to fail the containment. However, the core material attacked the concrete floor of the reactor cavity, and the containment failed by basemat meltthrough many hours after the start of the accident. This path through the event tree is designated accident progression bin FFA-CAC-ABA-CA. For the observation chosen, this bin is the most likely bin that has both vessel breach (VB) and containment failure (CF).

For Observation 4, the evaluation of the APET for PDS group 3 resulted in a conditional probability of 0.017 for bin FFA-CAC-ABA-CA. Bin FFA-CAC-ABA-CA was also generated in evaluating the APET for PDS group 1. The absolute frequency of this bin for both PDS groups, for Observation 4, is 3.2E–8/reactor year. Only a small portion of this frequency came from the cut set listed in Table B.1.

In the source term analysis, the release fractions, the timing of the release, and the energy and height of the release were determined for bin FFA-CAC-ABA-CA. These values define the source term. In the partitioning process, the source term for bin FFA-CAC-ABA-CA of Observation 4 was grouped with other similar releases into source term group SUR-49. In the consequence analysis, the offsite consequences of the mean source term for SUR-49 were computed. SUR-49 has such small release fractions that it caused no early fatalities. The total release caused (in this calculation) 113 latent cancer fatalities when averaged over all weather conditions.

Finally, the results of the four constituent analyses were combined to determine risk. For the consequences averaged with respect to the weather variability, when all PDS groups and APBs are considered, the contribution of Observation 4 was slightly below the median for early fatalities and near the mean value for latent cancer fatalities. Although bin FFA-CAC-ABA-CA was the most likely bin in Observation 4 that had both vessel breach and containment failure, it was a relatively improbable bin in the entire sample, and its contribution to the risk measures was very small. The position of the CCDFs for Observation 4 is shown in Figure B.19.

REFERENCES FOR APPENDIX B

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

B.2 D. M. Ericson, Jr., (Ed.) et al., "Analysis of Core Damage Frequency: Internal Events Methodology," Sandia National Laboratories, NUREG/CR-4550, Vol.1, Rev. 1, SAND86-2084, January 1990.

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

B.4 R. L. Iman and S. C. Hora, "Modeling Time to Recovery and Initiating Event Frequency for Loss of Offsite Power Incidents at Nuclear Power Plants," Sandia National Laboratories, NUREG/CR-5032, SAND87-2428, January 1988.

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

B.6 F. T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

B.7 J. M. Griesmeyer and L. N. Smith, "A Reference Manual for the Event Progression Analysis Code (EVNTRE)," Sandia National Laboratories, NUREG/CR-5174, SAND88-1607, September 1989.

B.8 T. A. Wheeler et al., "Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation," Sandia National Laboratories, NUREG/CR-4550, Vol. 2, SAND86-2084, April 1989.

B.9 Steam Explosion Review Group (SERG), U.S. Nuclear Regulatory Commission (USNRC), "A Review of the Current Understanding of the Potential for Containment Failure from In-Vessel Steam Explosions," NUREG-1116, June 1985.

B.10 D. B. Rhodes et al., "Reactor Coolant Pump Shaft Seal Stability During Station Blackout," Chalk River Nuclear Laboratories, NUREG/CR-4821, AECL-9342, May 1987.

B.11 C. A. Kittmer et al., "Reactor Coolant Pump Shaft Seal Behavior During Station Blackout," Idaho National Engineering Laboratory, NUREG/CR-4077, EGG-2365, April 1985.

B.12 C. H. Campen and W. D. Tauche, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," Westinghouse Electric Corporation, Nuclear Energy Systems, WCAP-10542, 1986.

B.13 K. D. Bergeron et al., "User's Manual for CONTAIN 1.0, A Computer Code for Severe Reactor Accident Containment Analysis," Sandia National Laboratories, NUREG/CR-4085, SAND84-1204, July 1985.

B.14 K. D. Bergeron et al., "User's Manual for CONTAIN 1.1, A Computer Code for Severe Reactor Accident Containment Analysis," Sandia National Laboratories, NUREG/CR-5026, SAND87-2309, to be published.*

B.15 Fauske and Associates, Inc., "MAAP, Modular Accident Analysis Program User's Manual," Vols. I and II, IDCOR Technical Report 16.2-3, February 1987.

B.16 M. Pilch and W. W. Tarbell, "Preliminary Calculations of Direct Heating of a Containment Atmosphere by Airborne Core Debris," Sandia National Laboratories, NUREG/CR-4455, SAND85-2439, July 1986.

*Available at the USNRC Public Document Room, 2120 L Street NW., Washington, DC.

B.17 W. W. Tarbell et al., "Results from the DCH-1 Experiment," Sandia National Laboratories, NUREG/CR-4871, SAND86-2483, June 1987.

B.18 M. Pilch and W. W. Tarbell, "High Pressure Injection of Melt from a Reactor Pressure Vessel—The Discharge Phase," Sandia National Laboratories, NUREG/CR-4383, SAND85-0012, September 1985.

B.19 W. W. Tarbell et al., "Pressurized Melt Ejection into Scaled Reactor Cavities," Sandia National Laboratories, NUREG/CR-4512, SAND86-0153, October 1986.

B.20 D. C. Williams et al., "Containment Loads Due to Direct Containment Heating and Associated Hydrogen Behavior: Analysis and Calculations with the CONTAIN Code," Sandia National Laboratories, NUREG/CR-4896, SAND87-0633, May 1987.

B.21 D. C. Williams and D. L. Y. Louie, "Containment Analysis of Direct Containment Heating Events in the Surry Plant," *Proceeding of the American Nuclear Society/European Nuclear Society International Meeting*, Thermal-Hydraulics Division (Washington, DC), October 31–November 4, 1988.

B.22 D. C. Williams, Sandia National Laboratories, letter to the NUREG-1150 Loads and Source Term Panels, "CONTAIN Calculations for Some DCH Scenarios in Sequoyah," dated April 6, 1988.

B.23 D. C. Williams et al., "Integrated Phenomenological Analysis of Containment Response to Severe Core Damage Accidents," *Progress in Nuclear Energy*, 19, 69, 1987.

B.24 L. Pong et al., "Surry S2D Severe Accident Containment Loads Calculations Using HMC," University of Wisconsin (Madison), UWRSR-34, March 1986.

B.25 United Engineers & Constructors, "Indian Point Units 2 and 3, Containment Capability Analysis," report prepared in support of the Indian Point Probabilistic Safety Study, March 1980.

B.26 W. J. Pananos and C. F. Reeves, "Containment Integrity at Surry Nuclear Power Station," Stone & Webster Engineering Corp., TP84-13, 1984.

B.27 United Engineers & Constructors, "Containment Ultimate Capacity of Seabrook Station Units 1 & 2 for Internal Pressure Loads," February 1983.

B.28 D. S. Horschel, Sandia National Laboratories, letter to J. F. Costello, USNRC, on the testing of a reinforced concrete containment, dated August 2, 1987.

B.29 Y. R. Rashid et al., "State-of-the-Art Review of Concrete Containment Response to Severe Overpressurization," *Transactions of the 9th SMIRT Conference* (Lausanne, Switzerland), August 17-21, 1987.

B.30 R. A. Dameron et al., "Analytical Correlation and Post-Test Analysis of the Sandia 1:6-Scale Reinforced Concrete Containment Test," Fourth Workshop on Containment Integrity (Arlington, VA), June 14-17, 1988.

B.31 S. J. Higgins, "A User's Manual for the Postprocessing Program PSTEVNT," Sandia National Laboratories, NUREG/CR-5380, SAND88-2988, November 1989.

B.32 R. L. Iman and M. J. Shortencarier, "A FORTRAN 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use with Computer Models," Sandia National Laboratories, NUREG/CR-3624, SAND83-2365, June 1984.

B.33 H. N. Jow et al., "XSOR Codes User's Manual," Sandia National Laboratories, NUREG/CR-5360, SAND89-0943, to be published.*

B.34 R. A. Lorenz, J. L. Collins, and A. P. Malinauskus, "Fission Product Source Term for the Light Water Reactor Loss-of-Coolant Accident," *Nuclear Technology*, 46, 404 (1979).

*Available at the USNRC Public Document Room, 2120 L Street NW., Washington, DC.

B.35 R. A. Lorenz, D. O. Hobson, and G. W. Parker, "Fuel Rod Failure Under Loss-of-Coolant Conditions in TREAT," *Nuclear Technology*, 11, 502 (1971).

B.36 USNRC, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," NUREG–0772, June 1981.

B.37 E. L. Tolman et al., "TMI–2 Accident Scenario Update," Idaho National Engineering Laboratory, EGG-TMI-7489, December 1986.

B.38 D. W. Akers and R. K. McCardell, "Fission Product Relocation and Behavior in the TMI–2 Reactor Vessel," *Trans. Am. Nuclear Soc.*, Vol. 57, pp. 418-19, Int. Conf. on Nuclear Fission: Fifty Years of Progress in Energy Security, and Topical Meeting on TMI–2 Accident: Materials Behavior and Plant Recovery Technology, TANSAO 57, October 30–November 4, 1988.

B.39 J. A. Gieseke et al., "Source Term Code Package: A User's Guide," Battelle Columbus Division, NUREG/CR–4587, BMI–2138, July 1986.

B.40 M. D. Allen et al., "ACRR Source Term Experiment: ST–1," Sandia National Laboratories, NUREG/CR–5345, SAND89–0308, to be published.*

B.41 R. A. Sallach, "Vapor Pressure of Liquid CsOH," Sandia National Laboratories, *Reactor Safety Research Semiannual Report, January-June*, 1987, NUREG/CR–5039, Vol. 1., SAND87–2411, January 1988.

B.42 J. A. Gieseke et al., "Radionuclide Release under Specific LWR Accident Conditions," Battelle Columbus Laboratory, Vols. I-VI, BMI-2104, July 1983–July 1984.

B.43 R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios," Battelle Columbus Division, NUREG/CR–4624, Vols. 1-5, BMI–2139, July 1986.

B.44 H. Bunz et al., "NAUA Mod4," Kerforschungzentrum Karlsruhe, KfK–3554, West Germany, 1983.

B.45 R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios: Supplemental Calculations," Battelle Columbus Division, NUREG/CR–4624, Vol. 6, BMI–2139, August 1990.

B.46 M. Khatib-Rahbar et al., "On the Uncertainties in Core Melt Progression, Fission Product Release, and Pressurization Loads for a BWR with Mark I Containment," Brookhaven National Laboratory, Draft BNL Technical Report A–3286, 1988.

B.47 E. Fuller, Electric Power Research Institute, letter to D. C. Williams, Sandia National Laboratories, on the decontamination factor of the Peach Bottom reactor building and fission product release from corium during core-concrete interaction, dated March 14, 1988.

B.48 S. E. Dingman et al., "Analysis of Peach Bottom Station Blackout with MELCOR," *Proceedings of the 14th Water Reactor Safety Information Meeting* (Gaithersburg, MD), NUREG/CP–0082, SAND86–2129C, February 1987.

B.49 R. M. Summers et al., "MELCOR In-Vessel Modeling," *Proceedings of the 15th Water Reactor Safety Information Meeting* (Gaithersburg, MD), NUREG/CP-0091, February 1988.

B.50 S. E. Dingman et al., "MELCOR Analyses for Accident Progression Issues," Sandia National Laboratories, NUREG/CR–5331, SAND89–0072, to be published.*

B.51 H. A. Morewitz et al., "Results of the GREST Code Comparison Exercise," Committee on the Safety of Nuclear Installations, Organisation for Economic Co-operation and Development, Nuclear Energy Agency, Paris, France, CSNI Report 116, 1986.

*Available at the USNRC Public Document Room, 2120 L Street NW., Washington, DC.

B.52 American Nuclear Society, "Report of the Special Committee on Source Terms," September 1984.

B.53 W. Schock, "General Valuation of the Results of the DEMONA Program," presentation at the DEMONA Final Colloquium, Kernforschungszentrum Karlsruhe, West Germany, June 1987.

B.54 P. Pasanen et al., "Behavior of Hygroscopic Aerosols: Test Results on Sedimentation in Tubes," Electric Power Research Institute, LACE TR-030, 1987.

B.55 "Evaluation of Empirical Aerosol Correlations," performed by Rockwell International Corp., published by Electric Power Research Institute, EPRI NP-4927, 1985.

B.56 R. J. Lipinski et al., "Uncertainty in Radionuclide Release Under Specific LWR Accident Conditions," Sandia National Laboratories, SAND84-0410, Vols. 1-4, Draft, February-December 1985.

B.57 USNRC, "Reactor Risk Reference Document," NUREG-1150, Vols. 1-3, Draft for Comment, February 1987.

B.58 R. L. Iman et al., "PARTITION: A Program for Defining the Source Term/Consequence Analysis Interface in the NUREG-1150 Probabilistic Risk Assessments," Sandia National Laboratories, NUREG/CR-5253, SAND88-2940, May 1990.

B.59 D. I. Chanin, H. N. Jow, J. A. Rollstin et al., "MELCOR Accident Consequence Code System (MACCS)," Sandia National Laboratories, NUREG/CR-4691, Vols. 1-3, SAND86-1562, February 1990.

APPENDIX C

ISSUES IMPORTANT TO QUANTIFICATION OF RISK

CONTENTS

FIGURES

Page

TABLES

C.1 Introduction

It is well known that the methods of probabilistic risk assessment (PRA) have many kinds of uncertainties associated with them. To a varying extent, these uncertainties contribute to the imprecision in the estimates of risk to public health and safety from nuclear reactor accidents. The NRC contractor work underlying this report (Refs. C.1.1 through C.1.14) addresses many of these uncertainties and quantifies their impact on selected measures of risk. The method to incorporate uncertainties in the quantification of reactor risk involves identifying uncertainty "issues." In this context, an issue is a physical parameter, process, or event that cannot be characterized precisely but is potentially important to the frequency of core damage or to severe accident progression. Examples are operator error rates, hydrogen generation during core meltdown, and direct containment heating.

The total number of issues that was considered in the present analyses is quite large. A complete description of all issues is available in the contractor reports. To give the reader a feel for what information is available, three tables and the description of those tables from the Surry analysis (Ref. C.1.10) have been included in this section.

This appendix summarizes the way in which a few important issues were treated in the five PRAs addressed in this report. In this context, "important" refers to subjective judgments made by the NRC staff and its contractors (based on results of the detailed analyses) that a particular issue has substantial influence on the quantification of risk. The objective of these descriptions is twofold:

1. To provide an answer to the following question: What aspects of the knowledge base supporting probabilistic risk assessment for nuclear power reactors are the principal contributors to our inability to precisely calculate risk?

2. To describe how areas known from previous work to have substantial uncertainty were addressed in the analyses described in NUREG–1150.

It should be noted that issues contributing to the uncertainty in risk are not necessarily significant contributors to a particular estimate of reactor risk. For example, issues that are threshold in nature (e.g., those governing the outcome of an event that either occurs or does not occur) can be important contributors at one end of the spectrum of risk estimates, yet be an insignificant contributor to risk estimates at the opposite end of the spectrum. Such issues may not even be major contributors to the mean value of risk. It is important to identify these issues—particularly those that contribute to estimates of risk near the high end of the spectrum. Improvements in the precision with which reactor risk analysis can be performed may be achieved by focusing future research on topics that are major contributors to the uncertainty in risk. Confidence that a selected measure of reactor risk is below some value can be improved by focusing research on topics that contribute to estimates of risk near the upper end of the spectrum.

Issues important to risk uncertainty are described in the following sections, which are organized in a similar fashion. First, an issue is defined in the context of its application within the risk analyses in this study. Since most issues are relatively high-level representations of uncertainty (i.e., they represent a composite of several interrelated sources of uncertainty), the specific source(s) of uncertainty included within each issue are delineated as part of the definition. The process of characterizing the contributing factors to the uncertainty associated with an issue is termed "issue decomposition." An example of issue decomposition is provided in Figure C.1.1, which considers the hypothetical issue of containment bypass. Underlying this hypothetical issue are a variety of more basic events and processes. Each of these may have an associated uncertainty. Quantification of the uncertainty associated with the main issue, therefore, involves the aggregation of uncertainties of several interrelated items. This process can become quite complicated and is not addressed in detail in this appendix. A summary of each issue's quantification and the technical basis that supports this quantification is provided. For greater detail regarding issue decomposition and quantification of individual contributors to uncertainty, the reader is referred to References C.1.1 through C.1.7 for issues related to estimating core damage frequency and References C.1.8 through C.1.14 for issues related to accident progression and consequences. Finally, the manner in which an issue was incorporated in the PRA(s) is described. Results of statistical analyses and other indicators of an issue's importance to risk uncertainty are presented.

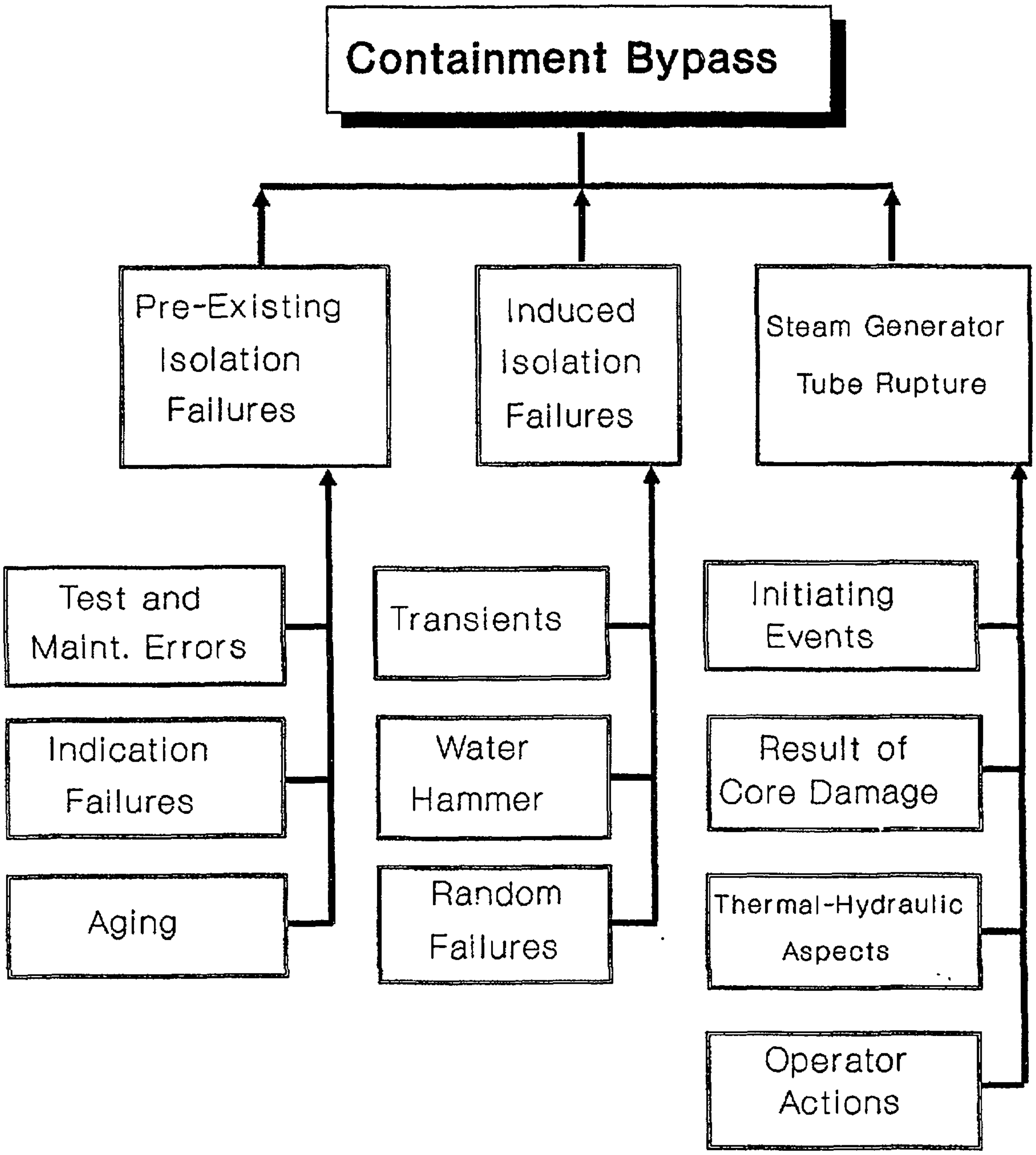


Figure C.1.1 Example of NUREG–1150 "issue decomposition."

C.1.1 Description of Table C.1.1; Variables Sampled in Accident Frequency Analysis

In the accident frequency analysis for internal events, a large number of variables were sampled. (A list of these variables may be found in Ref. C.1.3.) Only those variables found to be important to the uncertainty in the accident frequencies were selected for sampling in the integrated risk analysis. These variables are listed and defined in Table C.1.1. For the regression analysis, identifiers of eight characters or less were required, and these are listed in the first column. Where these differ from the identifiers used in the fault trees, these identifiers are listed in the description in brackets. Generally, the eight-character identifiers have been selected to be as informative as possible to those not familiar with the conventions used in systems analysis. For example, while Event K is commonly used to indicate the failure of the reactor protection system (RPS) to insert enough control rods to make the reactor subcritical, the identifier AU-SCRAM was chosen since it was felt that "auto scram" conveys more meaning to most readers than "K."

The second column in Table C.1.1 gives the range of the distribution for the variable, and the third column indicates the type of distribution used and its mean value. The entry "Experts" for the distribution indicates that the distribution came from the accident frequency analysis expert panel. The fourth and fifth columns in Table C.1.1 show whether the variable is correlated with any other variable, and the last column describes the variable. More complete descriptions and discussion of these variables may be found in the Surry accident frequency analysis report (Ref. C.1.3). This report also gives the source or the derivation of the distributions for all these variables.

Only two accident frequency variables were correlated in the integrated analysis. As indicated in Table C.1.1, DG-FRUN1 and DG-FRUN6 were correlated with each other since they represent failures to run for different times for the same equipment. The failures to run for the steam-turbine-driven auxiliary feedwater (AFW) pump (ATP-FR6 and ATP-FR24) should have been correlated for the same reason, but this correlation was omitted because of an oversight. Neither of the AFW pump failure-to-run variables was important in determining the uncertainty in risk, so the effect of omitting the correlation between them is not significant.

Table C.1.1 Variables sampled in accident frequency analysis for internal initiators.

Variable	Range	Distribution	Correlation	Correlation With	Description
V-TRAIN	1.8E-13 1.5E-5	Experts Mean=5.5E-7	None		Initiating event: frequency (1/yr) of check valve failure in one of the LPIS trains.
IE-LOSP	2.6E-5 0.28	LOSP Data Mean=0.077	None		Initiating event: frequency (1/yr) of LOSP. [IE-T1]
IE-A	5.0E-5 0.0032	Lognormal Mean=5E-4	None		Initiating event: frequency (1/yr) of a large (dia. > 6 in.) break in the RCS (LOCA).
IE-S1	1.0E-4 0.0063	Lognormal Mean=0.001	None		Initiating event: frequency (1/yr) of an intermediate size (6 in. > dia. > 2 in.) LOCA.
IE-S2	1.0E-4 0.0063	Lognormal Mean=0.001	None		Initiating event: frequency (1/yr) of a small break (2 in. > dia. > 0.5 in.) in the RCS.
IE-S3	0.0013 0.082	Lognormal Mean=0.013	None		Initiating event: frequency (1/yr) of a very small (0.5 in. > dia.) break in the RCS (LOCA).

Table C.1.1 (continued)

Variable	Range	Distribution	Correlation	Correlation With	Description
IE-T-ALL	0.67 41.6	Lognormal Mean=6.6	None		Initiating event: frequency (1/yr) of all transients that require scram (Surry data). [IE-T]
IE-T-HIP	0.60 37.2	Lognormal Mean=5.9	None		Initiating event: frequency (1/yr) of all transients from high (>25%) power that require scram (Surry data). [IE-TN]
IE-LMFWS	0.096 5.9	Lognormal Mean=0.94	None		Initiating event: frequency (1/yr) of transients due to loss of the main feedwater system (Surry data). [IE-T2]
IE-SGTR	0.001 0.063	Lognormal Mean=0.01	None		Initiating event: frequency (1/yr) of SGTRs (PWR data). [IE-T7]
IE-DCBUS	2.5E-5 0.14	Lognormal Mean=0.005	None		Initiating event: frequency (1/yr) for loss of a dc power bus. [IE-T5]
DG-FRUN1	9.9E-6 0.057	Lognormal Mean=0.002	Rank 1	DG-FRUN6	Probability that the diesel generator fails to run for 1 h, given that it starts. [DGN-FR-1HR]
DG-FRUN6	6.0E-5 0.34	Lognormal Mean=0.012	Rank 1	DG-FRUN1	Probability that the diesel generator fails to run for 6 h, given that it starts. [DGN-FR-6HR]
DG-FSTRT	0.0022 0.14	Lognormal Mean=0.022	None		Probability that the diesel generator fails to start, given a demand to start. [DGN-FS]
UNFV-MOD	1.8E-4 0.27	Lognormal Mean=0.014	None		Fraction of the time that the reactor operates with an unfavorable moderator temperature coefficient. [Z]
AU-SCRAM	1.8E-6 7.6E-4	Lognormal Mean=6E-5	None		Probability of failure of the RPS to automatically insert sufficient control rods to terminate the reaction. [K]
MN-SCRAM	0.017 1.0	Max. Entropy Mean=0.17	None		Probability of failure to effect manual scram due to operator error and hardware faults. [R]
AUTO-ACT	4.8E-5 0.020	Lognormal Mean=0.0016	None		Probability of failure of one train of an automatic actuation system (generic). [ACT-FA]

Table C.1.1 (continued)

Variable	Range	Distribution	Correlation	Correlation With	Description
CCF–RWST	1.5E–6 0.0085	Lognormal Mean=3E–4	None		Probability of common-cause failure of the recirculation mode transfer system due to miscalibration of the water level sensors in the RWST (human error). [RMT–CCF–FA–MSCAL]
BETA2MOV	0.0089 0.55	Lognormal Mean=0.088	None		Beta factor for common-cause failure of two motor-operated valves (generic). [BETA–2MOV]
BETA–AFW	0.0057 0.35	Lognormal Mean=0.056	None		Beta factor for common-cause failure of the AFWS motor-driven pumps (generic).
BETA–LPI	0.015 0.94	Lognormal Mean=0.15	None		Beta factor for common-cause failure of the LPIS pumps (generic).
AFW–STMB	2.0E–8 0.0070	Lognormal Mean=1.0E–4	None		Probability of common-cause failure of all AFWS due to steam binding (backleakage through check valves from MFWS). [CCF–LK–STMBD]
MDP–FSTR	1.5E–5 0.085	Lognormal Mean=0.003	None		Probability of failure to start (per demand) for motor-driven pumps for which specific plant data were not available (generic). [MDP–FS]
AFWMP–FS	6.4E–4 0.040	Lognormal Mean=0.0063	None		Probability of failure to start (per demand) for AFW motor-driven pumps (from Surry data). [AFW–MDP–FS–FW3B]
AFWTP–FS	5.5E–5 0.31	Lognormal Mean=0.011	None		Probability of failure to start (per demand) for AFW steam-turbine-driven pump (from Surry data). [TDP–FS]
ATP–FR6	1.5E–4 0.85	Lognormal Mean=0.030	None		Probability of failure to run for 6 h for the AFW steam-turbine-driven pump (generic). [TDP–FR–6HR]
ATP–FR24	0.01 1.0	Max. Entropy Mean=0.12	None		Probability of failure to run for 24 h for the AFW steam-turbine-driven pump (generic). [TDP–FR–24HR]
PORV–BLK	0.0041 0.25	Lognormal Mean=0.040	None		Probability of failure to open (per demand) for the PORV block valves (MOVs). [PPS–MOV–FT]

Table C.1.1 (continued)

Variable	Range	Distribution	Correlation	Correlation With	Description
LPRS–MOV	2.6E–5 0.15	Lognormal Mean=0.0052	None		Probability of failure (per demand) for the suction MOVs in the LPRS due to hardware failures or plugging. [LPR–MOV–FT]
MOV–FT	1.5E–5 0.085	Lognormal Mean=0.003	None		Probability of failure to transfer (per demand) for motor-operated valves (generic).
MNV–PG1	4.1E–6 2.5E–4	Lognormal Mean=3.6E–5	None		Probability of failure due to plugging for manual valves that are flow-tested every month (generic). [XVM–PG–1MO]
MOV–PG3	1.0E–5 6.3E–4	Lognormal Mean=1.0E–4	None		Probability of failure due to MOVs that are flow-tested every 3 months (generic). [MOV–PG–3MO]
MOV–PG12	4.5E–5 0.0028	Lognormal Mean=4.4E–4	None		Probability of failure due to plugging for MOVs that are flow-tested every 12 months (generic). [MOV–PG–12MO]
AFW–OCC	1.5E–5 9.5E–4	Lognormal Mean=1.5E–4	None		Probability of common-cause failure of AFWS due to an inadvertently open crossconnect to Unit 2 (flow diversion). [AFW–PSF–FC]
PORV–REC	1.5E–4 0.85	Lognormal Mean=0.030	None		Probability of failure of the pressurizer PORVs to reclose after opening (generic). [SOV–OO]
SSRVO–SB	0.030 1.0	Max. Entropy Mean=0.27	None		Probability of failure of an SG SRV to reclose within 1 h during SBO (faulted steam generator). [QS–SBO]
SSRVO–U2	0.016 1.0	Max. Entropy Mean=0.16	None		Probability of failure of a secondary system SRV at Unit 2 to reclose within 1 h during SBO at both units. [QS–UNIT2]
SOV–FT	1.0E–4 0.0063	Lognormal Mean=0.001	None		Probability of failure to transfer (per demand) for solenoid-operated valves (generic).
CKV–FT	1.0E–5 6.3E–4	Lognormal Mean=1E–4	None		Probability of failure to open (per demand) for check valves (generic).

Table C.1.1 (continued)

Variable	Range	Distribution	Correlation	Correlation With	Description
HE-FDBLD	0.0071 0.71	Max. Entropy Mean=0.071	None		Probability of failure of the operator to initiate feed and bleed (human error—open PORVs, and start charging pump and align suction and discharge valves). [HPI-XHE-FO-FDBLD]
HE-PORVS	0.0044 0.44	Max. Entropy Mean=0.044	None		Probability of failure of the operator to initiate feed and bleed (human error; diagnose situation and open PORVs). [PPS-XHE-FO-PORVS]
HE-CST2	0.0065 0.65	Max. Entropy Mean=0.065	None		Probability of failure of the operator to align the AFWS suction to the backup CST during an SBO with a faulted SG. [AFW-XHE-FO-CST2]
HE-UNIT2	0.0036 0.36	Max. Entropy Mean=0.036	None		Probability of failure of the operator to provide AFW from Unit 2 via the crossconnect. [XHE-FO-UNIT2]
HE-SKILL	1.3E-5 0.077	Lognormal Mean=0.0026	None		Probability of human error for skill-based human errors (rudimentary actions performed from memory). [XHE-FO-SKILLBASE]
RCP-SL-F		Experts	None		Probability of RCP seal failure before the onset of core damage.

C.1.2 Description of Table C.1.2; Questions in Surry APET

In addition to the number and name of the question, Table C.1.2 indicates if the question is sampled, and how the question is evaluated or quantified. In the sampling column, an entry of DS indicates that the sampling is from a distribution provided by one of the expert panels, or from the electric power recovery distribution. The item sampled may be either the branching ratios or the parameter defined at that question. For questions that are sampled and that were quantified internally, the entry ZO in the sampling column indicates that the question was sampled zero-one, and the entry SF means the questions were sampled with split fractions. The difference may be illustrated by a simple example. Consider a question that has two branches and a uniform distribution from zero to 1.0 for the probability for the first branch. If the sampling is zero-one, in half of the observations the probability for the first branch will be 1.0, and in the other half of the observations it will be zero. If the sampling is split fraction, the probability for the first branch for each observation is a random fractional value between zero and 1.0. The average over all the fractions in the sample is 0.50. The implications of ZO or SF sampling are discussed in Reference C.1.8.

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

The abbreviations in the quantification column of Table C.1.2 are given below, with the number of questions that have that type of quantification.

Type of Quantification	Number of Questions	Comments
PDS	11	Determined by the PDS
AcFrqAn	1	Determined in the accident frequency analysis
Other	4	See Notes 1 through 4
Internal	17	Quantified internally in this analysis
Summary	17	The branch taken at this question follows directly from the branches taken at previous questions
ROSP	3	The probability of the recovery of offsite power is determined by distributions derived from the electric power recovery data for this plant
UFUN-Str.	3	Calculated in the User Function, using distributions from the structural response expert panel
UFUN-Int.	2	Calculated in the User Function, using an adiabatic pressure rise calculation determined internally
In-Vessel	5	Distributions from the in-vessel accident progression expert panel
Loads	2	Distributions from the containment loadings expert panel
Struct.	1	Distribution from the containment structural performance expert panel
N.A.	5	Fan cooler questions not applicable to Surry

In some cases, a question may have more than one function so the entry under Quantification in Table C.1.2 can be only indicative. For example, Questions 43, 52, and 64 are listed as being quantified by the user function, based on distributions generated by the containment structural performance expert panel. The actual situation is that a portion of the user function is evaluated which determines whether the containment fails using the load pressure and the failure pressure. The load pressure is determined in Questions 39 and 40 based on aggregate distributions from the containment loadings expert panel. The containment failure pressure is determined in Question 42 from the aggregate distributions from the containment loadings expert panel. If the failure pressure is lower than the load pressure, then the containment fails and the mode of failure is determined using the random number defined in Question 42 and a table of conditional failure mode probabilities contained in the user function. This table was also generated by the containment structural performance expert panel. The sampling is indicated to be zero-one because one of the four branches of these questions always has a probability of 1.0, and the other three always have a probability of zero.

Table C.1.2 Questions in Surry APET.

Question Number	Question	Sampling	Quantification
1.	Size & location of RCS break when the core uncovers?	SF	PDS
2.	Has the reaction been brought under control?	SF	PDS
3.	For SGTR, are the secondary system SRVs stuck open?	SF	PDS
4.	Status of ECCS?	SF	PDS
5.	RCS depressurization by the operators?	SF	PDS
6.	Status of sprays?	SF	PDS
7.	Status of fan coolers?		N.A.
8.	Status of ac power?		PDS
9.	RWST injected into containment?	SF	PDS
10.	Heat removal from the steam generators?	SF	PDS
11.	Did the operators depressurize the secondary before the core uncovers?	SF	PDS
12.	Cooling for RCP seals?	SF	PDS
13.	Initial containment condition?		AcFrqAn
14.	Event V—break location under water?	SF	Note 1
15.	RCS pressure at the start of core degradation?		Summary
16.	Do the PORVs stick open?	SF	Note 2
17.	Temperature-induced RCP seal failure?	ZO	Note 3
18.	Is the RCS depressurized before breach by opening the pressurizer PORVs?		Internal
19.	Temperature-induced SGTR?	DS	In–Vessel
20.	Temperature-induced hot leg or surge line break?	DS	In–Vessel
21.	Is ac power available early?	SF	ROSP
22.	Rate of blowdown to containment?		Summary
23.	Vessel pressure just before vessel breach?	ZO	Internal
24.	Is core damage arrested? No vessel breach?	SF	Internal
25.	Early sprays?		Summary
26.	Early fan coolers?	N.A.	
27.	Early containment heat removal?		Summary
28.	Baseline containment pressure before VB?		Internal
29.	Time of accumulator discharge?		Summary
30.	Fraction of zirconium oxidized in-vessel during core degradation?	P	In–Vessel
31.	Amount of zirconium oxidized in-vessel during core degradation?		Summary
32.	Amount of water in the reactor cavity at vessel breach?		Summary

Table C.1.2 (continued)

Question Number	Question	Sampling	Quantification
33.	Fraction of core released from the vessel at breach?	P	In-Vessel
34.	Amount of core released from the vessel at breach?		Summary
35.	Does an alpha event fail both vessel & containment?	SF	Note 4
36.	Type of vessel breach?	ZO	In-Vessel
37.	Does the vessel become a "Rocket" and fail the cont.?		Internal
38.	Size of hole in vessel (after ablation)?	ZO	Internal
39.	Total pressure rise at vessel breach? Large hole cases	P	Loads
40.	Total pressure rise at vessel breach? Small hole cases	P	Loads
41.	Does a significant ex-vessel steam explosion occur?		Internal
42.	Containment failure pressure?	P	Struct.
43.	Containment failure and type of failure?	ZO	UFUN-Str.
44.	Sprays after vessel breach?		Internal
45.	Is ac power available late?	SF	ROSP
46.	Late sprays?		Summary
47.	Late fan coolers?		N.A.
48.	Late containment heat removal?		Summary
49.	How much hydrogen burns at vessel breach?	SF	Internal
50.	Does late ignition occur?		Internal
51.	Resulting pressure in containment?		UFUN-Int.
52.	Containment failure and type of failure?	ZO	UFUN-Str.
53.	Amount of core available for CCI?		Summary
54.	Is the debris bed in a coolable configuration?		Internal
55.	Does prompt CCI occur?		Summary
56.	Is ac power available very late?	SF	ROSP
57.	Very late sprays?		Summary
58.	Very late fan coolers?		N.A.
59.	Very late containment heat removal?		Summary
60.	Does delayed CCI occur?		Summary
61.	How much hydrogen is produced during CCI?		Internal
62.	Does very late ignition occur?	P	Internal
63.	Resulting pressure in containment?		UFUN-Int.
64.	Containment failure and type of failure?	ZO	UFUN-Str.
65.	Sprays after very late CF?		Internal
66.	Fan coolers after very late CF?		N.A.
67.	Containment heat removal after very late CF?		Summary
68.	Eventual basemat meltthrough (BMT)?		Internal

Table C.1.2 (continued)

Question Number	Question	Sampling	Quantification
69.	*Eventual overpressure failure of containment?*		*Internal*
70.	BMT before overpressure failure?		Internal
71.	Final containment condition?		Summary

Note 1. Whether the location of the break in the low-pressure piping would be under water in Event V at the time the core was uncovered was determined by a special panel that considered only this problem for the draft version of this analysis. As there was no new information available, there was no reason to change the conclusions reached by this group.

Note 2. There is little or no data on the failure rate of PORVs when passing gases at temperatures considerably in excess of their design temperature. The quantification was arrived at by discussions between the accident frequency analyst and the plant analyst.

Note 3. In the accident frequency analysis, a special panel was convened to consider the issue of the failure of RCP seals. The quantification of this question is not as detailed as that done in the accident frequency analysis but relies on the information produced by this panel.

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

Key to Abbreviations in Table C.1.2

AcFrqAn The quantification was performed as part of the accident frequency analysis.

DS The branch probabilities are taken from a distribution; depending on the distribution, the sampling may be SF or ZO.

Internal The quantification was performed at Sandia National Laboratories by the plant analyst with the assistance of other members of the laboratory staff.

In-Vessel This question was quantified by sampling from an aggregate distribution provided by the expert panel on in-vessel accident progression.

Loads This question was quantified by sampling from an aggregate distribution provided by the expert panel on containment loadings.

N.A. Not Applicable.

P A parameter is determined by sampling from a distribution, in most cases an aggregate distribution from an expert panel.

PDS The quantification follows directly from the definition of the plant damage state.

ROSP This question was quantified by sampling from a distribution derived from the offsite power recovery data for the plant.

SF Split fraction sampling - the branch probabilities are real numbers between zero and one.

Struct. This question was quantified by sampling from an aggregate distribution provided by the containment structural performance expert panel.

Summary The quantification for this question follows directly from the branches taken at preceding questions or the values of parameters defined in preceding questions.

UFUN-Str. This question is quantified by the execution of a part of the User Function, using distributions from the containment structural performance expert panel.

UFUN-Int. This question is quantified by the execution of a part of the User Function, using an adiabatic calculation for the pressure rise due to hydrogen combustion.

ZO Zero-One sampling–the branch probabilities are either 0.0 or 1.0.

C.1.3 Description of Table C.1.3; Variables Sampled in Source Term Analysis

The variables that were sampled in the source term analysis are listed and summarized in Table C.1.3. Those variables quantified by the source term expert panel are marked with an asterisk in Table C.1.3.

Table C.1.3 Variables sampled in source term analysis.

Variable	Description
FCOR*	Fraction of each fission product group released from the core to the vessel before or at vessel breach. There are two cases: high and low zirconium oxidation.
FVES*	Fraction of each fission product group released from the vessel to the containment before or at vessel breach. There are four cases: RCS at system setpoint pressure, RCS at high or intermediate pressure, RCS at low pressure, and Event V.
VDF	Decontamination factor for pool scrubbing for Event V when the break location is underwater at the time of the release. There is one distribution, which applies to all radionuclide classes except inert gases.
FCONV*	Fraction of each fission product group in the containment from the RCS release that is released from the containment in the absence of mitigating factors such as sprays. There is one distribution for each case, which applies to all radionuclide classes except inert gases. There are five cases: containment leak at or before vessel breach with sprays operating, containment leak at or before vessel breach with sprays not operating, containment rupture at or before vessel breach, very late containment rupture, and Event V. Note that FCONV does not account for fission product removal by the sprays. The case differentiation on spray operation is to account for differences in containment atmosphere temperature and humidity between the two cases.
FCCI*	Fraction of each fission product group in the core material at the start of CCIs that is released to the containment. There are four cases: low zirconium oxidation in the core and no overlaying water, high zirconium oxidation in the core and no overlaying water, low zirconium oxidation in the core with overlaying water, and high zirconium oxidation in the core with overlaying water.
FCONC*	Fraction of each fission product group in the containment from the CCI release that is released from the containment in the absence of mitigating factors such as sprays. The five cases are the same as those for FCONV, but there are separate distributions for each radionuclide class.
SPRDF	Decontamination factor for sprays. Internal elicitation was used to develop a distribution for this variable, which was used for all fission product groups except the noble gases. There is one distribution for each case, which applies to all radionuclide classes except inert gases. There are three cases: RCS release at high pressure and CF at VB, RCS releases not covered by the first case, and CCI releases.

Table C.1.3 (continued)

Variable	Description
LATEI*	Fraction of the iodine deposited in the containment that is revolatilized and released to the environment late in the accident. This variable applies only to iodine.
FLATE*	Fraction of the deposited amount of each fission product group in the RCS that revolatilized after VB and released to the containment. There are two cases: one large hole in the RCS and two large holes in the RCS.
DST*	Fraction of each fission product group in the core material that becomes aerosol particles in a direct containment heating event at VB that is released to the containment. There are two cases: VB at high pressure (1000 to 2500 psia) and VB at intermediate pressure (200 to 1000 psia).
FISGFOSG	Fraction of each fission product group released from the reactor vessel to the steam generator, and from the steam generator to the environment, in an SGTR accident. There are two separate distributions, FISG and FOSG, each of which has two cases: SGTRs in which the secondary SRVs reclose and SGTRs in which the secondary SRVs stick open.
POOL-DF	Decontamination factor for a pool of water overlaying the core debris during CCI. There are two cases: a completely full (depth about 14 ft) cavity and a partially full cavity (accumulator water only, depth about 4 ft).

*Quantified by source term expert panel.

REFERENCES FOR SECTION C.1

C.1.1 D.M. Ericson, Jr., (Ed.) et al., "Analysis of Core Damage Frequency: Internal Events Methodology," Sandia National Laboratories, NUREG/CR-4550, Vol. 1, Revision 1, SAND86-2084, January 1990.

C.1.2 T.A. Wheeler et al., "Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation," Sandia National Laboratories, NUREG/CR-4550, Vol. 2, SAND86-2084, April 1989.

C.1.3 R.C. Bertucio and J.A. Julius, "Analysis of Core Damage Frequency: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Revision 1, SAND86-2084, April 1990.

C.1.4 A.M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

C.1.5 R.C. Bertucio and S.R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

C.1.6 M.T. Drouin et al., "Analysis of Core Damage Frequency: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 6, Revision 1, SAND86-2084, September 1989.

C.1.7 M.B. Sattison and K.W. Hall, "Analysis of Core Damage Frequency: Zion Unit 1," Idaho National Engineering Laboratory, NUREG/CR-4550, Vol. 7, Revision 1, EGG-2554, May 1990.

C.1.8 E.D. Gorham-Bergeron et al., "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration, and Uncertainty Analyses," Sandia National Laboratories, NUREG/CR-4551, Vol. 1, Draft Revision 1, SAND86-1309, to be published.*

C.1.9 F.T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

C.1.10 R.J. Breeding et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, October 1990.

C.1.11 A.C. Payne, Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft Revision 1, SAND86-1309, to be published.*

C.1.12 J.J. Gregory et al., "Evaluation of Severe Accident Risks: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, December 1990.

C.1.13 T.D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 6, Draft Revision 1, SAND86-1309, to be published.*

C.1.14 C.K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

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

C.2 Common-Cause and Dependent Failures

Since the completion of the Reactor Safety Study over 10 years ago, more than 25 PRAs have been completed. These PRAs and other reliability analyses performed on nuclear power plant systems indicate that among the major contributors to the estimated total frequency of core damage are events that involve dependent failures.

Dependent failures are failures that defeat the redundancy or diversity in engineered plant safety systems. In the absence of dependent failures, separate trains of a redundant system, or diverse methods of providing the same safety function, may be regarded as independent. However, actual operating experience indicates that not all components of redundant systems are free from dependent failures; simultaneous failure of similar independent components occur as a result of a common cause. Such failures occur infrequently, and their interdependence can be very subtle. As a result, dependent failure data are too sparse to accurately estimate common-cause failure rates for many types of components. Also, dependent failure mechanisms are often plant specific in nature, further limiting the availability of directly usable data.

System analysts generally try to include explicit dependencies in the basic plant logic model (i.e., in the structure of the fault and event trees). Functional dependencies arising from the reliance of frontline systems on support systems, such as emergency coolant injection on service water or on electrical power, are examples of types of dependent failures that are usually modeled as an integral part of the fault and event tree structure. Interaction among various components within systems, such as common maintenance or test schedules, common control or instrumentation circuitry, and co-location within plant buildings (common operating environments), are often included as basic events on system fault trees. Even though the fault and event tree models include the major dependencies that have been identified, in some cases it is not possible to identify the specific mechanisms of common-cause failure from the available data bases (e.g., Licensee Event Reports—LERs). In other cases, where there can be many different types of common-cause failure, each with low probability, it is not practical to model each type separately. A relatively simple method is often used to account for the collective contribution of these residual common-cause failures to system or component failure rates. The method correlates the common-cause failure rate of multiple similar components to the failure rate of a single component of the same type. This method, known as the modified beta factor method (Ref. C.2.1), was applied in the system analyses for this study. Quantification of the beta factors (the common-cause failure rate correlation parameters) for important components is based on limited data and was treated as an uncertainty issue.

C.2.1 Issue Definition

The Fleming report is used as a basis for beta factors for common-cause faults involving the failure of two out of two components (Ref. C.2.2). Quantification of higher order common-cause events, such as three of three or four of four, is based on an additional set of multipliers (e.g., those developed by Atwood (Ref. C.2.3)). These multipliers are applied to the beta factors to calculate failure rates for higher order common-cause faults. For the elicitation, the higher order multipliers are not treated specifically in this issue; but because the quantifiable estimates of higher order common-cause failures are functions of Fleming-based data factors, the treatment of the beta factors in this issue will also affect the higher order factors. The uncertainties associated with this issue center on the appropriateness, robustness, and interpretation of the Fleming data. A beta factor represents that fraction of component faults that could also result in faults for similar components in the same service. It is also the conditional probability of a component failure, given that a similar component has failed. Such failures are concurrent, or approximately so, and are not due to any other component fault. Mathematically, Fleming's data are manipulated to derive beta factors defined by

$$\beta = A / (A+B),$$

where

$$A = N_{ac} + W_c N_{pc} + 1$$

$$B = N_{ai} + W_i N_{pi} + 1$$

N_{ac} = number of actual component failures due to common cause,

N_{pc} = number of potential* component failures due to common cause,

N_{ai} = number of actual independent failures,

N_{pi} = number of potential independent failures, and

W_i, W_c = weighting factors for considering potential failures as actual failures.

Not all common-cause beta factors used in the present analyses are based on the Fleming report because either a more component-specific analysis existed elsewhere or the Fleming report did not analyze data for certain components. The beta factors *not* based on Fleming's work, are:

- BWR safety relief valves (fail to reclose)
- Batteries
- Air-operated valves

The BWR SRV failure to reclose common-cause event was modeled with data from Reference C.2.2, using a nonparametric model instead of the beta factor model. BWRs have from eight to ten SRVs, so it was necessary to model the failure of various combinations of these valves. This is an exception to the assumption that was used for most other components, where the common-cause failure of k redundant components was modeled for only one failure combination, all k components. These SRV failures include all multiple SRV failures to reclose. The probabilities of these outcomes include the contribution of combinations of independent failures as well as common-cause failures. Reference C.2.4 discusses in detail the approach that was used to derive the common failure rate used in these situations for BWR SRVs failing to reclose.

The dc power study (Ref. C.2.5) was the source for the beta factor for a common-cause failure of two redundant batteries. That study suggests a worst case beta factor of 0.4 for a two-battery configuration in the minimum standard dc power system reported. Higher order beta factors were calculated with a formula based on the assumption that the conditional probability of the k_{th} (k>2) battery failing, given that "k−1" have failed, is the average of 1.0 and the beta for "k−1" of k batteries failing:

$$\beta_k = \prod_{i=2}^{k} \left[(2^{i-2} - 1.0 + B_2)/(2^{i-2})\right]$$

Air-operated valve (AOV) failures were not specifically addressed in the various references on common-cause failures. A screening value of 0.1 was chosen as a beta factor for two or even more AOVs failing from a common cause. This was the result of an expert judgment elicitation performed among the project staff (Ref. C.2.6).

C.2.2 Technical Bases for Issue Quantification

The results of the common-cause beta factor statistical analysis of the Fleming data are shown on Table C.2.1 for pumps and Table C.2.2 for valves. Using Fleming's model, 5^{th}, 50^{th}, and 95^{th} binomial confidence intervals were calculated to measure the uncertainty in the data. The Fleming model weights the potential failures by a factor of 0.1 (i.e., 10 percent of potential failures evolve into actual failures). The importance of considering potential failures in quantifying common-cause beta factors was examined in a sensitivity study that examined two extreme cases. The first (denoted β_a) case assumed all potential failures become actual component failures (W_c, W_i = 1.0); the second (denoted β_d) case assumed no contribution from potential failures (W_c, W_i = 0.0). The impact of these assumptions on the median value of the common-cause beta factor for each of several components is indicated in Tables C.2.1 and C.2.2.

*Potential failures involve components that are capable of performing their functions but exhibit a degraded performance or an incipient condition which, if not corrected, could lead to failure.

Table C.2.1 Beta factor analysis for pumps—based on Fleming data.

Pumps		Binomial Confidence Intervals 5	50	95	Data* N_{ac}	N_{pc}	N_{ai}	N_{pi}
LPCI/LPCS/RHR	β	0.10	0.15	0.25	7	4	40	27
	β_a		0.16					
	β_d		0.13					
PWR Safety Injection	β	0.15	0.21	0.26	15	4	59	18
	β_a		0.20					
	β_d		0.21					
PWR Aux. Feedwater	β	0.036	0.056	0.093	9	6	107	11
	β_a		0.079					
	β_d		0.053					
PWR Containment Spray	β	0.047	0.11	0.25	2	2	25	7
	β_a		0.14					
	β_d		0.11					
Service Water/	β	0.012	0.026	0.065	2	10	111	4
Component	β_a		0.075					
Cooling Water	β_d		0.026					

β = Beta factor
β_a = Beta factor calculated by weighting all potential failures (N_{pc}, N_{pi}) at 1.0.
β_d = Beta factor calculated by weighting all potential failures from the model.
* See text for definition of terms.

Table C.2.2 Beta factor analysis for valves—based on Fleming data.

Valves		Binomial Confidence Intervals 5	50	95	Data* N_{ac}	N_{pc}	N_{ai}	N_{pi}
Motor-Operated Valves	β	0.08	0.09	0.11	72	43	778	64
	β_a		0.12					
	β_d		0.08					
Safety Relief Valves	β	0.022	0.07	0.30	0	0	11	19
(PWR)	β_a		0.03					
	β_d		0.08					
Relief Valves	β	0.16	0.22	0.28	27	23	107	29
	β_a		0.27					
	β_d							

β = Beta factor
β_a = Beta factor calculated by weighting all potential failures (N_{pc}, N_{pi}) at 1.0.
β_d = Beta factor calculated by weighting all potential failures from the model.
* See text for definition of terms.

The β_a and β_d values are not always respectively higher and lower than the base case values. This is because the assumption to disregard or fully credit potential events also affects the denominator of the Fleming model, which includes terms for potential common-cause and independent failures.

Expert judgment was elicited from two experienced system analysts regarding the uncertainty in beta factor estimates due to potential misclassification of available data:

Alan Kolaczkowski—Science Applications International Corp., and
Arthur Payne, Jr.—Sandia National Laboratories.

Their consensus is that this uncertainty is adequately accounted for in current models. Their rationale is as follows:

1. **Inclusion of Potential Failures in Data Base**

   The β_a and β_d factors on Tables C.2.1 and C.2.2 indicate that the inclusion of potential common-cause and independent failures in the data base does not represent a significant source of model uncertainty. The most significant impact of assuming that all potential events in the data are actual failures is an increase by a factor of 2.9 (service water system pump). There is almost no impact of deleting all potential failures from the data base.

2. **Classification of Independent Failures**

   The beta factor model is highly sensitive to the number of independent failures. This number dominates the denominator of the beta factor equation. A factor of *n* increase in the number of independent failures would result roughly in a factor of *n* decrease in the beta factor. A factor of *n* decrease in independent failures would have the inverse effect. It seems highly unlikely that the data classification could be so erroneous that enough independent failures could have been miscategorized to create significant error in the parameter estimates.

3. **Classification of Common-Cause Failures**

   A sensitivity analysis was performed to examine the impact of reclassifying common-cause data. In this analysis, Fleming's common-cause data were assumed to have been miscategorized by a factor of two (i.e., the observed failures were assumed to be common cause in nature twice as often or, alternatively, half as often as categorized by Fleming). The resulting range of beta factor values for these cases fell well within the uncertainty ranges of the current models (Ref. C.2.6). As a result, the experts whose judgments were elicited on this issue believe it unreasonable that the data could have been misinterpreted to such an extent that current models inadequately represent this uncertainty.

Because it is unlikely that significant misclassifications of events have occurred, the experts believe that the distributions for common-cause beta factors are peaked near the median and fall off rapidly from the median. Given the lack of information and historical insensitivities of the accident sequence analysis results to the actual distribution selected, the experts believe that the lognormal distributions indicated on Table C.2.3 adequately characterize the data and modeling uncertainties for this issue.

C.2.3 Treatment in PRA and Results

The beta factors described above were used in the quantification of the system fault trees for each plant. An indication of the importance of individual common-cause and dependent failures in the fault tree analysis for these plants is the decrement by which the total core damage frequency would be reduced if these failures were not to occur. This decrement (known as the risk-reduction measure) is shown in Table C.2.4 for selected common-cause events in the Surry and Peach Bottom analyses. Note that several of the common-cause events shown in Table C.2.4 were not quantified using Fleming's data (e.g., diesel generator failures). For these events, plant-specific information was used when available. A complete listing of the risk-reduction measures is provided in References C.2.7 through C.2.11.

The collective contribution of common-cause failures to the mean total core damage frequency was investigated by performing a sensitivity study in which all beta factors were assigned a single (point

Table C.2.3 Beta factor models from EPRI NP-3967.

Pumps	Mean	Error Factor	Valves	Mean	Error Factor
Low-Pressure Coolant Injection	0.15	3	Motor Operated	0.088	3
Low-Pressure Core Spray	0.15	3	Safety Relief (PWR)	0.07	3
Residual Heat Removal	0.15	3	Relief (BWR)	0.22	3
High-Pressure Safety	0.21	3			
PWR Aux. Feedwater (Motor-driven)	0.056	3			
PWR Containment Spray	0.11	3			
Service Water/Component Cooling Water	0.026	3			

All distributions are assumed to be lognormal.

Table C.2.4 Risk-reduction measures for selected common-cause events in Surry and Peach Bottom analysis.

Common-Cause Event	Mean Event Probability	Risk-Reduction Measure*
Surry (mean total core damage frequency = 4.01E-5)		
BETA-2MOV (failure of 2 motor-operated valves)	8.80E-2	2.72E-6
BETA-3DG (failure of 3 diesel generators)	1.80E-2	2.66E-6
BETA-2DG (failure of 2 diesel generators)	3.80E-2	2.25E-6
BETA-LPI (failure of multiple motor-driven pumps, low-pressure injection)	1.50E-1	6.75E-7
Peach Bottom (mean total core damage frequency = 4.50E-6)		
BETA-5BAT (failure of 5 station batteries)	2.50E-3	1.97E-7
BETA-3AOVS (failure of 3 air-operated valves)	5.50E-2	9.75E-8
BETA-4DGNS (failure of 4 diesel generators)	1.30E-2	3.52E-8
BETA-2SIPUMPS (failure of 2 safety injection pumps)	2.10E-1	1.81E-8

*Decrement by which the total core damage frequency would be reduced if this event were not to occur.

estimate) value of zero, and the core damage frequency distribution was recalculated. The results of this analysis are summarized in Table C.2.5, which shows the extent to which the mean total core damage frequency for Surry, Sequoyah, Peach Bottom, and Grand Gulf is reduced when common-cause failures are eliminated.

Table C.2.5 Results of sensitivity study in which common-cause failures were eliminated from fault trees.

Plant	Base Case Analysis	Sensitivity Study No Common-Cause Failures	Percent Reduction
Surry	4.01E-5	3.08E-5	23
Sequoyah	5.72E-5	4.57E-5	20
Peach Bottom	4.50E-6	4.07E-6	10
Grand Gulf	4.05E-6	3.10E-6	26

REFERENCES FOR SECTION C.2

C.2.1 Pickard, Lowe and Garrick, Inc., "Procedures for Treating Common Cause Failures in Safety and Reliability Studies, Procedural Framework and Examples," NUREG/CR-4780, Vol. 1, EPRI NP-5613, January 1988.

C.2.2 K.N. Fleming et al., "Classification and Analysis of Reactor Operation Experience Involving Dependent Failures," Pickard, Lowe and Garrick, Inc., EPRI NP-3967, June 1985.

C.2.3 C.L. Atwood, "Common Cause Fault Rates for Pumps," EG&G Idaho, Inc., NUREG/CR-2098, EGG-EA-5289, February 1983.

C.2.4 D. M. Ericson, Jr., (Ed.) et al., "Analysis of Core Damage Frequency: Internal Events Methodology," Sandia National Laboratories, NUREG/CR-4550, Vol. 1, Revision 1, SAND86-2084, January 1990.

C.2.5 P. W. Baranowsky et al., "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants," NUREG-0666, April 1981.

C.2.6 T.A. Wheeler et al., "Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation," Sandia National Laboratories, NUREG/CR-4550, Vol. 2, Part 2 of 2—"Project Staff," SAND86-2084, April 1989.

C.2.7 R.C. Bertucio and J.A. Julius, "Analysis of Core Damage Frequency: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Revision 1, SAND86-2084, April 1990.

C.2.8 A.M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

C.2.9 R.C. Bertucio and S.R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

C.2.10 M.T. Drouin et al., "Analysis of Core Damage Frequency: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 6, Revision 1, SAND86-2084, September 1989.

C.2.11 M.B. Sattison and K.W. Hall, "Analysis of Core Damage Frequency: Zion Unit 1," Idaho National Engineering Laboratory, NUREG/CR-4550, Vol. 7, Revision 1, EGG-2554, May 1990.

C.3 Human Reliability Analysis

Human performance has been found to be a dominant factor in major safety-related incidents at nuclear power plants, both in the United States and elsewhere. Examples include events such as those at Three Mile Island Unit 2, Davis-Besse, and Oyster Creek in the United States and at Chernobyl in the U.S.S.R. In each of these, a complex interaction between humans and hardware led to a significant hazardous event and, in two cases, to offsite releases. Deficiencies in human performance occurred both before the initiation of the event, in areas such as maintenance, training, and planning, and in response to the event.

In the evaluation of human performance, different types of human errors have been identified. The first is generally an error where an intended action is not carried out, usually because of lapses in memory or lack of attention. Examples of these types of errors involve an operator missing a step in a procedure or accidentally selecting a wrong switch. The second type of human error is generally an action performed in accordance with a plan that is inadequate for the situation. The plan may be inadequate because there is an error in diagnosing the type of event or because the type of event has not been considered in preparing the plan and is not part of the operator's experience and training. The third type of human error is a deliberate deviation from practices thought necessary to maintain safety. These kinds of errors can be either routine (as in taking shortcuts) or exceptional (as in the case of Chernobyl).

Techniques have been developed for modeling some, but not all, of these types of human errors in PRAs. In particular, the first two kinds of error described above are analyzed in the present analysis. Other types of errors have not been addressed in this analysis, principally because no methods have been developed to provide quantitative estimates of error rates for them. Those errors considered in this study, and the methods for modeling them, are discussed below.

C.3.1 Issue Definition

Human reliability was not analyzed as a separate issue in these analyses; that is, the influence of alternative methods, models, or data were not evaluated. Rather, uncertainty distributions of the individual failure probabilities were estimated using standard human reliability methods. These failure probabilities were incorporated in the accident sequence quantifications.

In most cases, human errors were modeled as failures of people to take actions specified in procedures, including maintenance procedures, operating procedures, and emergency operating procedures. In a few cases, innovative actions were identified as ways to arrest sequences prior to the onset of core damage; failure probabilities for such actions were estimated. There were no evaluations of the consequences of mistakes, as in "if the operators mistook scenario A for scenario B, then they would ...."

The kinds of human actions represented in the analyses included human errors before the onset of an accident and errors and recovery actions following the start of an accident. The pre-accident errors are mostly failures by test and maintenance personnel to restore components to operation following maintenance (hence, rendering a system unavailable) or miscalibration of multiple sensors, such as containment pressure or reactor level sensors (hence, failing automatic initiation signals at the correct setpoint). Other pre-accident errors are failures of operators to perform tests correctly, such as failing to restore the standby liquid control system after testing, resulting in its being unavailable in the event of a demand.

Post-accident failures include failures to initiate or control emergency core cooling systems (ECCS), control rods, the standby liquid control system, etc., or their critical support systems, following their failure to start or run automatically during an accident. Examples include recovering the operability of failed diesel generators and arranging crossconnections of service water systems between units following single-unit failures. In addition, there are post-accident actions that must be performed manually during certain accidents to prevent core damage; these are not simply starting systems that failed to start automatically. Examples include the implementation of feed and bleed or the changeover from high-pressure injection to recirculation cooling following depletion of the refueling water storage tank (RWST) during loss-of-coolant accidents at some PWRs (e.g., Sequoyah).

C.3.2 Technical Bases for Issue Quantification

Quantitative estimates were made for the likelihoods of human errors using documented human reliability models. Failures in test and maintenance actions were quantified using a simplified version of the technique for human error rate prediction (THERP) (Ref. C.3.1). The original THERP method (Ref. C.3.2) was developed and applied in the Reactor Safety Study to model human errors that can be analyzed using a task analysis (a step-by-step decomposition of an activity into simple items, such as "read meter," "turn switch," etc.). The THERP documentation provided a data base and rules of application for this approach to human reliability analysis. In the simplified approach developed for this study, bounding values were initially assigned for overall tasks, such as "restore pump," without performing a task analysis. A nominal failure probability of 3E-2 was assigned for all pre-accident failures, with adjustments made for factors such as people performing independent checks and the use of written verification sheets. Each of these factors reduced the nominal value by a factor of 10. Hence the existence of three factors would reduce the failure probability from 3E-2 to 3E-5.

The post-accident actions were categorized as to whether misdiagnosis of the plant state was considered credible. Misdiagnosis was not considered credible for manual scrams following failure of the automatic scram system, where operators are well trained and written procedures exist. Misdiagnosis was included for operators failing to recognize the condition of the plant and responding with an inappropriate strategy. These misdiagnosis errors were quantified, using a time-reliability correlation described in Reference C.3.1. A time-reliability correlation provides an estimate of failure probability based on a time available for operators to take action following the onset of an event; this is a commonly used type of technique for this type of error.

These human reliability techniques provide single best estimate values with associated ranges of uncertainties. Table C.3.1 shows representative errors and associated uncertainty ranges used in the Grand Gulf accident sequence analysis.

Table C.3.1 Representative ranges of human error uncertainties (taken from Grand Gulf analysis).

Human Error	Error Rate*	Uncertainty Range*
Common-mode miscalibration of instrument	2.5E-5	10
Failure during isolation and repair of pump	3.0E-5	16
Operator failure to initiate level control (ATWS)	1.0E-3	5

*Error rates and uncertainty ranges are expressed as the median and error factors of the distributions used in the sequence quantification.

C.3.3 Treatment in PRA and Results

In this analysis, as in most published PRAs, human errors are most commonly represented in the system fault trees (much like component failures within systems), in the event trees (representing procedural actions), and in the recovery analysis of accident sequence cut sets. Therefore, many human errors are scattered throughout the system analysis models.

A small number of operator actions are represented in event trees. These are where a single action has a direct effect on the progression of an accident, as in the case of manual depressurization of a PWR to achieve "feed and bleed." Similarly, manual reactor trips are represented in ATWS-related event trees.

It is not possible to state what range of uncertainty in the core damage frequencies and other risk measures results only from uncertainties in human reliability. However, analyses were performed to evaluate the sensitivity of core damage frequencies to human reliability values. These sensitivity studies were conducted by setting the human error probabilities for post-accident actions to zero and comparing the resulting core damage frequencies to those for the base case analyses. Requantifying the core damage frequency with these human error probabilities equal to zero led to reductions in the range of 3.5 to 6.6—a significant potential reduction. These are summarized in Table C.3.2. The highest factor, 6.6 for Grand

Gulf, resulted largely from eliminating all cut sets involving diesel failure, because zero human error implied perfect recovery of the failed diesel. The core damage frequency for Grand Gulf is dominated by station blackout, and, hence, eliminating diesel failure results in a significant reduction in core damage frequency. It must be remembered that the calculated reduction in core damage frequency includes unrealistic equipment behavior as well as unrealistic human performance as a result of simplifying assumptions in the analysis. Specifically, "perfect" human repairs would be effective for only a subset of possible equipment failure modes. Recovery of diesel operability following major equipment damage within a short time is meaningless, regardless of the quality of the human performance. (To be more realistic, the analysis would require separating recoverable failure modes from unrecoverable failure modes and associating perfect recovery only with recoverable failure modes.)

In a different sense, these calculated reductions in core damage frequency are an upper bound. During analyses such as this, recovery actions (human actions to terminate sequences prior to core damage) are identified only for sequences important to the total core damage frequency; this is done to simplify the analysis and focus the analysts' efforts on the important factors in the analysis. Setting the human error probability to zero eliminates the initially dominant sequences that already include recovery actions, but no reconsideration of recovery was evaluated for the remaining sequences that did not include recovery. It is likely that recovery actions could be postulated in some of these sequences. Adding recovery actions to these newly dominant sequences would yield further reductions in core damage frequencies.

Table C.3.2 Core damage frequencies with and without human errors.

	Core Damage Frequency		
Plant	**Base Case**	**No Errors**	**Factor of Reduction**
Grand Gulf	4.1E-6	6.2E-7	6.6
Peach Bottom	4.5E-6	9.5E-7	4.8
Sequoyah	5.7E-5	2.5E-5	3.5
Surry	4.0E-5	1.1E-5	3.8

REFERENCES FOR SECTION C.3

C.3.1 A.D. Swain III, "Accident Sequence Evaluation Program—Human Reliability Analysis Procedure," Sandia National Laboratories, NUREG/CR-4772, SAND86-1996, February 1987 (with errata).

C.3.2 A.D. Swain III and H.E. Guttmann, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," Sandia National Laboratories, NUREG/CR-1278, Revision 1, SAND80-0200, October 1983.

C.4 Hydrogen Combustion Prior to Reactor Vessel Breach

LWR fuel assemblies and core structures contain substantial quantities of metallic materials that oxidize when heated to sufficiently high temperatures (i.e., those calculated to accompany core meltdown accidents). Oxidation of these metals, principally Zircaloy and stainless steel, can liberate sufficient quantities of hydrogen to generate substantial containment loads if released to the containment building and allowed to accumulate and subsequently burn. It is estimated (Ref. C.4.1) that approximately 270–370 kilograms of hydrogen were generated and released to the containment during the accident at Three Mile Island Unit 2 (TMI–2). Combustion of this hydrogen during the TMI–2 accident resulted in a containment pressure spike of approximately 28 psig (peak pressure). Since the design pressure of the TMI–2 containment is approximately 60 psig, this pressure rise did not pose a serious threat to containment integrity. However, the pressure spike observed during the accident at TMI–2 provided much of the motivation for subsequent changes in NRC regulations regarding containment hydrogen control (10 CFR 50.44). These changes involved hardware backfits for plants with pressure-suppression containments—BWR Mark I, II, and III and PWR ice condensers.

Hydrogen combustion is the dominant contributor to early containment failure for both Grand Gulf (BWR Mark III) and Sequoyah (PWR ice condenser). The importance of hydrogen combustion for these plants stems from the fact that both plants have relatively small free volumes, have low containment failure pressures, and incorporate pressure-suppression systems, which condense steam released from the vessel and thereby allow flammable mixtures of hydrogen and air to form. Accidents that result in early containment failure allow radionuclides that are released during the core damage process to escape to the environment. Because these accidents result in an early release, the time available for many of the emergency response actions to mitigate the accident is reduced. Because of its importance to risk, a large amount of effort was devoted to modeling and representing the uncertainty in these events.

Hydrogen can be generated during three phases of the accident: during the core damage process, at vessel breach, and late in the accident from the interaction of the core debris with concrete. The first two sources of hydrogen are generally more important to risk because they can lead to early containment failure.

During the core degradation process, the oxidation of metal with steam in the reactor produces hydrogen, which is subsequently released to the containment. During transient-type accidents in a BWR, hydrogen is released through the safety relief valve (SRV) tailpipes into the suppression pool. Similarly, during transients in a PWR, the hydrogen is released through the power-operated relief valves (PORVs) into the containment. Hydrogen can also be released directly to the containment during a loss-of-coolant accident (LOCA). If this hydrogen is allowed to accumulate, combustible mixtures can form. Although both Grand Gulf and Sequoyah have hydrogen ignition systems that are designed to burn the hydrogen at low concentrations, during station blackouts (i.e., accidents in which both onsite and offsite ac power are lost), these systems are not available. Subsequent ignition of this hydrogen by either random ignition sources or by the recovery of ac power can result in loads that can threaten the integrity of the containment. Depending on the concentration of the hydrogen at the time of ignition, both deflagrations and detonations are possible.

In the analyses performed for this study, hydrogen combustion phenomena were decomposed into smaller parts or issues. The various issues that were considered included in-vessel hydrogen production, ignition probability, detonation probability, peak pressure rise from a deflagration, dynamic load from a detonation, structural capacity of the containment to quasistatic pressurization events, and structural capacity of the containment to dynamic loads. Each of these issues in itself is very complicated and involves a substantial amount of uncertainty. Because of the large amount of uncertainty associated with these issues and because of their importance to early containment failure and risk, many of these issues were presented to panels of experts. The amount of hydrogen produced in the reactor during the core degradation process was addressed by the in-vessel accident progression expert panel. Hydrogen combustion phenomena before vessel breach and containment loads at vessel breach were addressed by the containment loadings expert panel. The structural response to these loads was provided by the containment structural performance expert panel.

The interfaces and the transfer of information between the various related issues are managed in the accident progression event tree (APET). A brief description of the relationship between these issues is provided below. The amount of hydrogen that is produced in-vessel is obtained from distributions provided by the expert panel. This amount of hydrogen depends on the type of accident that is being analyzed and the conditions in the reactor vessel during the core damage process. A certain fraction of hydrogen produced during core damage is released to the containment. The distribution of the hydrogen in the various compartments of the containment depends on the accident. Based on the amounts of hydrogen, steam, and air that are present in the containment, the flammability of the mixture is determined. If the mixture is flammable, the likelihood that it is ignited by a random source is determined. The ignition probability in the absence of igniters was provided by the containment loadings expert panel and is a function of the concentration of steam and hydrogen in the containment. If the hydrogen concentration is high enough, it is possible for a deflagration-to-detonation transition to occur. If the hydrogen burns as a deflagration, a quasistatic pressure load results, whereas a detonation would impose a dynamic load on the containment.

The peak pressure that results from a deflagration depends on hydrogen concentration, burn completeness, effectiveness of the heat transfer to surrounding structures, and the degree to which gases are vented to other compartments within the containment. The magnitude of the dynamic load that results from a detonation is a function of hydrogen concentration, steam concentration, and the geometry of the containment compartment within which the burn occurs. Given that a hydrogen combustion event occurs, it must be determined if the load is sufficient to fail the containment. The load obtained from the combustion event is compared to structural capacity of the containment. If the load is greater than the containment failure pressure, the containment fails. Based on the pressure at which the containment fails and the magnitude of the load, the mode of failure is determined (i.e., leak or rupture). The APET accesses Fortran subroutines, which are used to track the amount of hydrogen and oxygen in the containment during the course of the accident, to determine the hydrogen concentration and the flammability of the atmosphere in the containment during various time periods, and to determine the pressure rise due to hydrogen burns.

Combustion events can also occur at the time of vessel breach or late in the accident. At the time of vessel breach, hydrogen is produced by the rapid oxidation of metal that accompanies energetic events such as ex-vessel steam explosions and direct containment heating. The ejection of hot core debris from the vessel also provides numerous ignition sources. Because of the availability of both hydrogen and ignition sources, the likelihood of hydrogen combustion is high at the time of vessel breach. However, because the pressure rise at vessel breach results from many phenomena (e.g., direct containment heating (DCH), ex-vessel steam explosions, and hydrogen combustion) and hydrogen combustion is only one component, a single distribution was used to characterize the uncertainty in the pressure rise from these events. Thus, the pressure rise at vessel breach is addressed as a separate issue. Late in the accident, the interaction of the core debris with concrete produces both hydrogen and carbon monoxide, both of which are combustible. The approach used to determine the likelihood of containment failure from late hydrogen and carbon monoxide combustion is similar to the approach used before vessel breach.

As an example, a detailed description of the hydrogen issue, hydrogen combustion prior to reactor vessel breach, is presented below.

C.4.1 Issue Definition

Because of significant differences in containment configuration and other design features, this issue was posed in a slightly different way for each plant. However, in each case, the issue was posed to answer the following two fundamental questions:

1. What distributions characterize the uncertainty in the probability that hydrogen combustion will occur in the containment building prior to vessel breach?

2. Given that combustion occurs, what distributions characterize the uncertainty in the attendant peak static pressure and the maximum impulse loading (to the drywell wall for Grand Gulf and to the ice condenser walls for Sequoyah)?

The answers to these questions may depend on the accident scenario postulated. Therefore, a case structure was established to distinguish the initial conditions associated with accident sequences found to be important contributors to a plant's estimated core damage frequency. The case structure also provides a convenient tool for applying the generated probability distributions to the Grand Gulf and Sequoyah PRAs (as described in Section C.4.3).

Containment loads due to hydrogen combustion in Grand Gulf represent a significant challenge to containment integrity only during station blackout accident sequences, during which the igniters are inoperable. The case structure for Grand Gulf combustion loads, therefore, considers three variations of the station blackout accident sequence. These are summarized below:

Case	Steam Partial Pressure (psia)	Steam Partial Pressure (kPa)	Air Partial Pressure (psia)	Air Partial Pressure (kPa)	ac Power Recovered?	Containment Sprays Operate After Power Recovery?
1	7	5	17	115	No	No
2	7	50	17	115	Yes	Yes*
3	20	135	18	120	Yes	Yes*

Case 1 represents station blackout scenarios during which ac power is not recovered and a hydrogen burn is ignited spontaneously from a random ignition source. Cases 2 and 3 represent station blackout scenarios during which ac power is recovered prior to vessel breach; they differ from each other only in containment initial conditions.

Variations of station blackout are also the only accident scenarios in Sequoyah for which containment loads from hydrogen combustion represent a significant challenge to containment integrity. As in Grand Gulf, the ignition system is inoperable during these sequences; however, numerous "random" ignition sources are available in the Sequoyah containment. For example, sparks can be generated from movement of the intermediate deck doors in the ice condenser. The case structure for Sequoyah involves four variations of station blackout accident sequences, each of which yields different thermodynamic initial conditions. These may be summarized as follows:

Case 1: Station blackout; cycling power-operated relief valve (PORV) with the reactor pressure vessel at 2000–2500 psia.

Case 2: Station blackout; loss of pump seal cooling induces failure(s) of one or more reactor coolant pump (RCP) seals, resulting in a relatively low leak rate.

Case 3: Station blackout; loss of pump seal cooling induces failure(s) of one or more RCP seals, resulting in a relatively large leak rate.

Case 4: Station blackout; high temperatures in the reactor vessel upper plenum induce a creep rupture failure of hot leg piping of sufficient size to rapidly depressurize the reactor vessel.

For readers familiar with Reactor Safety Study (Ref. C.4.2) nomenclature for labeling accident sequences, Case 1 is the classic TMLB' accident scenario. Case 2 represents accident scenarios similar to S_3B, Case 3 represents scenarios similar to S_2B, and Case 4 represents scenarios similar to AB.

The following discussion of this issue only addresses the potential range of containment loads that may result from the combustion of hydrogen prior to reactor vessel breach for the Grand Gulf and Sequoyah plants. Containment response analyses for the other three plants addressed in the present work did consider hydrogen combustion in the evaluation of containment loads; however, the importance of early hydrogen combustion to the uncertainty in reactor risk for these plants is minor in comparison to that observed in the Grand Gulf and Sequoyah analyses. Additionally, questions regarding safety concerns that may result from hydrogen combustion in the containment, such as equipment survivability, initiation of building fires, etc., are not part of this issue.

*Sprays are assumed to initiate approximately 90 seconds after ignition.

C.4.2 Technical Bases for Issue Quantification

The principal bases for quantifying the probability of hydrogen combustion during station blackout accident scenarios and the accompanying containment loads are calculations performed with several computer codes. These include early calculations performed with the MARCH2 code (Ref. C.4.3), more recent MARCH3 analyses (Refs. C.4.4 and C.4.5), HECTR calculations of hydrogen burns in an ice condenser containment (Refs. C.4.6 and C.4.7), parametric analyses of Grand Gulf containment response to hydrogen burns using MELCOR (Ref. C.4.8), and calculations performed in support of the IDCOR program (Ref. C.4.9). These calculations are supplemented by a large body of information, available in the open literature, which discusses the strengths and weaknesses of the models employed by these codes, the experimental evidence on which many of the models are based, and the sensitivity of calculated containment loads to uncertain modeling parameters. The number of reports describing this information is too large to list here; however, a reasonably complete list is presented in References C.4.1 and C.4.10.

It is not uncommon for a relatively wide range of estimates of containment loads due to hydrogen combustion to be generated if several different analysts are asked to provide an estimate. Even for well-defined initial and boundary conditions (e.g., rate of release of a hydrogen/steam mixture to containment), the selection of different analytical tools and judgments regarding appropriate modeling parameters (e.g., flame speed, burn completeness) inevitably results in significant differences in containment loads. Although the sensitivity of estimated containment loads to many important modeling parameters has been quantified in a fairly comprehensive fashion, an analyst's judgment is still required to select the combination of parameters most appropriate to the particular problem being evaluated.

To characterize the distribution of containment loads that reflects the current state of knowledge and uncertainties in hydrogen combustion modeling, this issue was presented to a panel of experienced severe accident analysts. For each plant, three analysts were asked to provide a distribution representing the probability that various mixtures of combustible gas, air, and diluents would ignite (in the absence of an operating igniter system) and a distribution for the attendant peak static pressure and/or maximum impulse. Participating in the Grand Gulf evaluations were:

James Metcalf—Stone & Webster Engineering Corp.,
Louis Baker—Argonne National Laboratory, and
Martin Sherman—Sandia National Laboratories.

Participating in the Sequoyah evaluations were:

Patricia Worthington—U.S. Nuclear Regulatory Commission,
James Metcalf—Stone & Webster Engineering Corp., and
Martin Sherman—Sandia National Laboratories.

The following discussion summarizes the collective judgment of the panels and selected individual assessments where significant differences in judgment were observed.

Quantification of Ignition Frequency

As indicated in the issue definition, there is some uncertainty regarding how and with what frequency a combustible mixture would ignite in the Grand Gulf or Sequoyah containment if the igniter system is inoperative. There is some evidence that the propensity for combustible mixtures to spontaneously ignite (because, for example, of the discharge of accumulated static charges) increases with increasing molar concentrations of combustible gas. The likelihood of spontaneous ignition of a given combustible gas concentration also increases with time.

An additional source of ignition during loss-of-ac-power accident scenarios in Sequoyah is the generation of sparks from movement of the intermediate and deck doors of the ice condenser. These doors, illustrated in Figure C.4.1, are normally closed to isolate the ice compartment from warm regions of the containment. Under accident conditions, when steam or other reactor coolant system effluents enter the bottom of the ice condenser, the doors open upward against their own weight. During severe accidents, gas flow rates through the ice condenser are not likely to be sufficiently high to hold all the doors open, and several doors may cycle open and shut as flow fluctuates. A comparable ignition source was not identified in the Grand Gulf containment building.

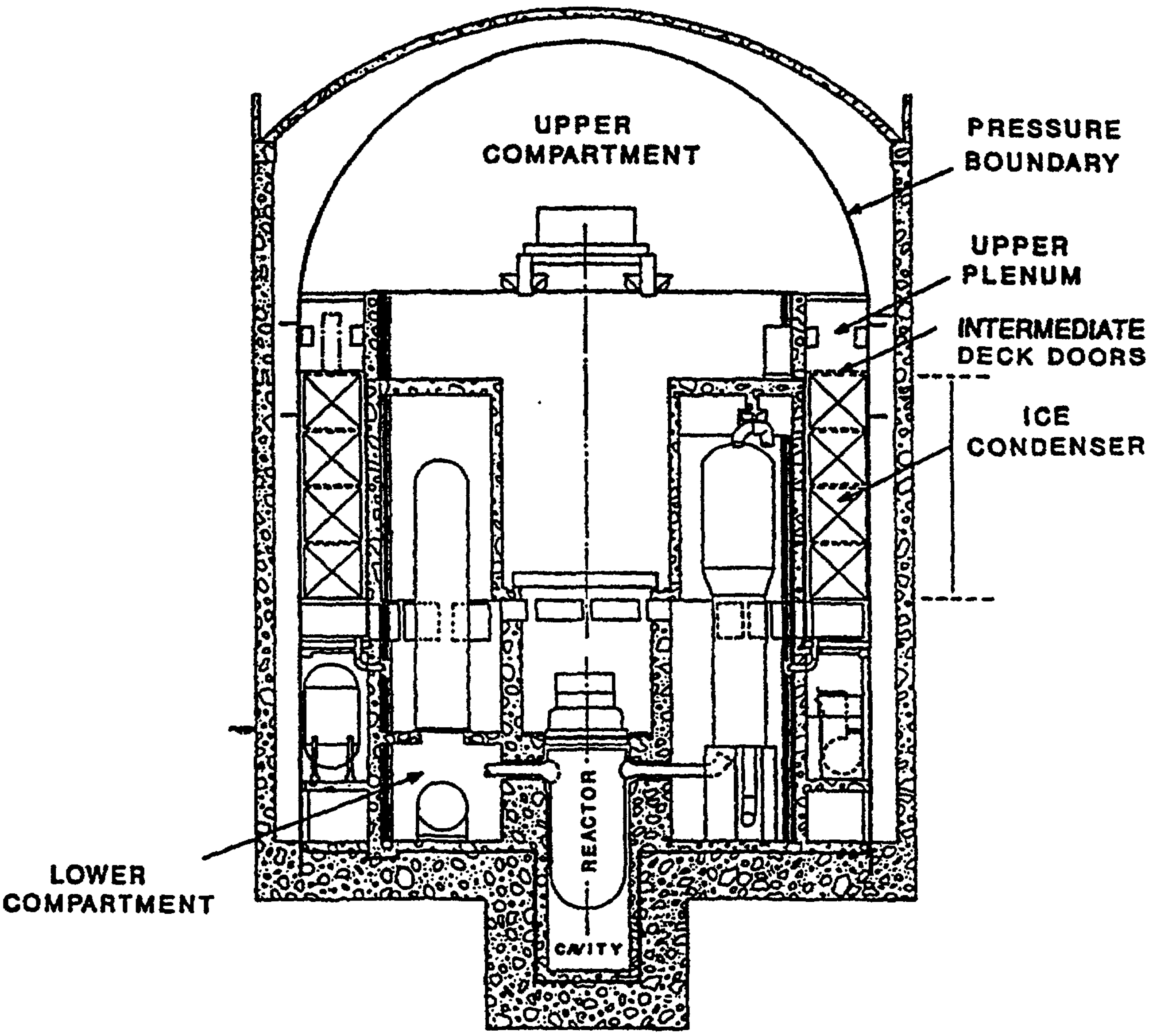


Figure C.4.1 Cross section of Sequoyah containment.

Each panelist was asked to provide a distribution that represented his or her estimate for the probability that a given concentration of combustible gas would ignite in the absence of an operating ignition system. For the Sequoyah containment, separate distributions were elicited for the ice condenser (i.e., region below the intermediate deck doors), the ice condenser upper plenum (i.e., region between the intermediate and top deck doors), and the containment upper compartment. Each of these regions is indicated in Figure C.4.1.

Distributions were elicited only for the outer containment in Grand Gulf (illustrated in Fig. C.4.2). The composition of the gas mixture in the drywell region of the Grand Gulf containment will be noncombustible (i.e., absent of sufficient oxygen or hydrogen to support combustion) prior to vessel breach, thus precluding ignition.

The panel's aggregate distribution (arithmetic average of the panelists' distributions) for Grand Gulf are shown in Figure C.4.3. The frequency of ignition in Grand Gulf is strongly dependent on the initial concentration of hydrogen. The absence of a readily identifiable ignition source in Grand Gulf results in relatively low probabilities of ignition for low hydrogen concentrations. Ignition frequency in the Sequoyah containment (shown in Fig. C.4.4) is less sensitive to initial hydrogen concentration than to the region of containment being considered. The potential for sparks from ice condenser door movement results in generally higher probabilities of ignition in the ice condenser upper plenum than in other areas.

Quantification of Containment Loads Due to Hydrogen Combustion

As indicated in the issue definition, combustion loads are characterized for each of two possible events: deflagration and detonation. Loads accompanying a deflagration are characterized by a distribution for the attendant peak static pressure. Detonation loads are characterized by the maximum impulse (to the drywell wall for Grand Gulf and to the ice condenser walls for Sequoyah). The likelihood that a particular combustion event would result in a deflagration or detonation is described by a conditional probability distribution (i.e., given the occurrence of a combustion event, what is the probability that it takes the form of a detonation?).

An aggregate distribution was generated for each permutation of cases (listed in Section C.4.1) and various ranges of plausible initial hydrogen and/or steam concentration. The ranges of initial conditions considered were:

Grand Gulf:

- Deflagration Loads: <4%(H), 4–8%(H&L), 8–12%(H&L), 12–16%(H&L), >16%(H&L)
- Conditional Probability of Detonation: 12–16%(H), 16–20%(H&L), >20%(L)
- Detonation Loads: >12%(H&L)

Sequoyah:

- Deflagration Loads: Loads calculated in containment event tree*
- Conditional Probability of Detonation: 14–16%, 16–21%, >21%
- Detonation Loads: >14%

Nomenclature: % refers to hydrogen mole fraction
H refers to a "high" steam mole fraction (40–60%)
L refers to a "low" steam mole fraction (20–30%)

Distributions for each of the case/initial-condition permutations are not presented in this report. Selected aggregate distributions are shown and discussed below to illustrate the range of values used in the PRA. The reader is encouraged to review Reference C.4.11 for complete documentation of individual panelists' judgments as well as the panel aggregate for each case.

*Formula used to estimate peak static pressure is based on information provided by expert panelists. This formula is described in Reference C.4.11.

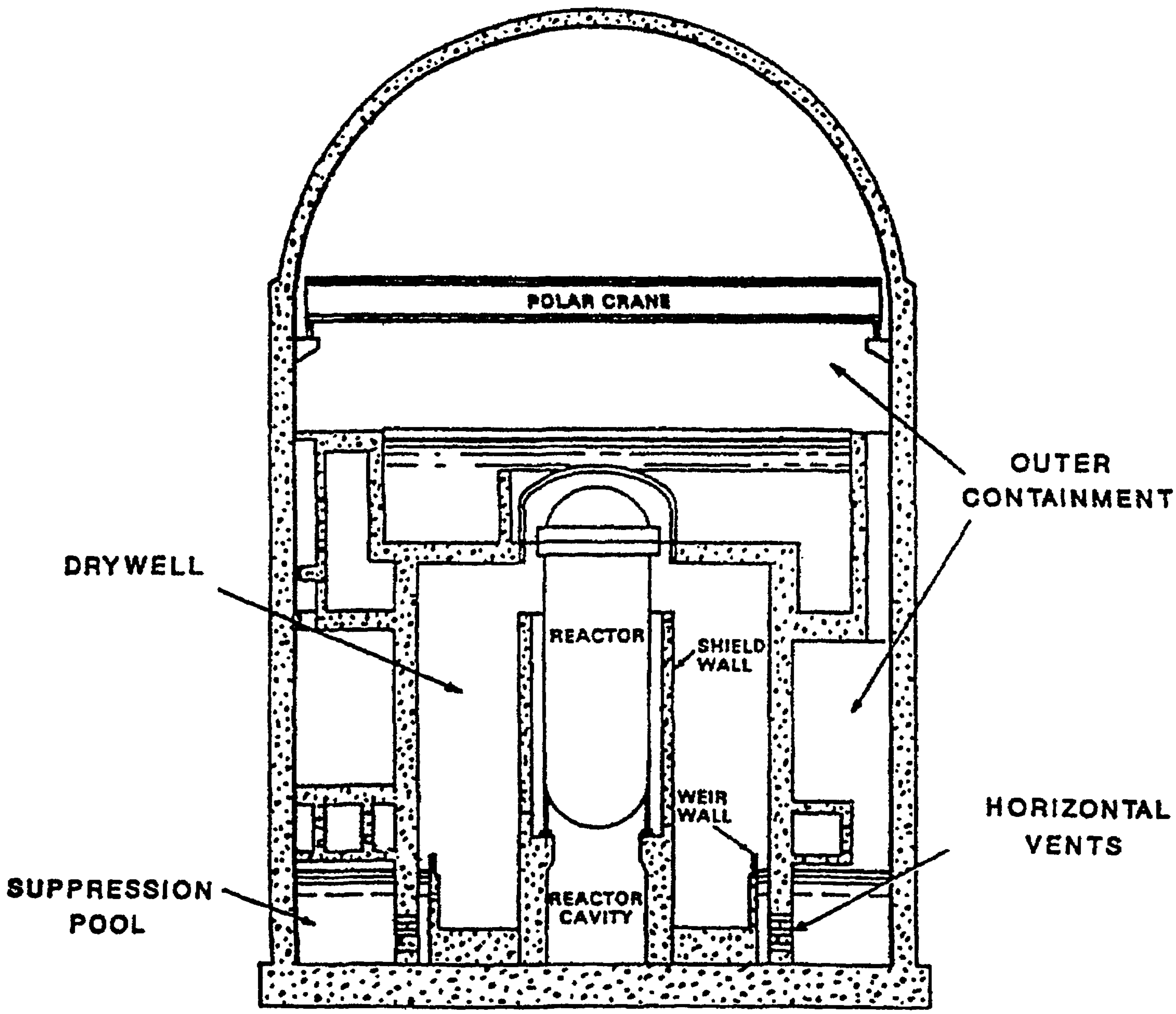


Figure C.4.2 Cross section of Grand Gulf containment.

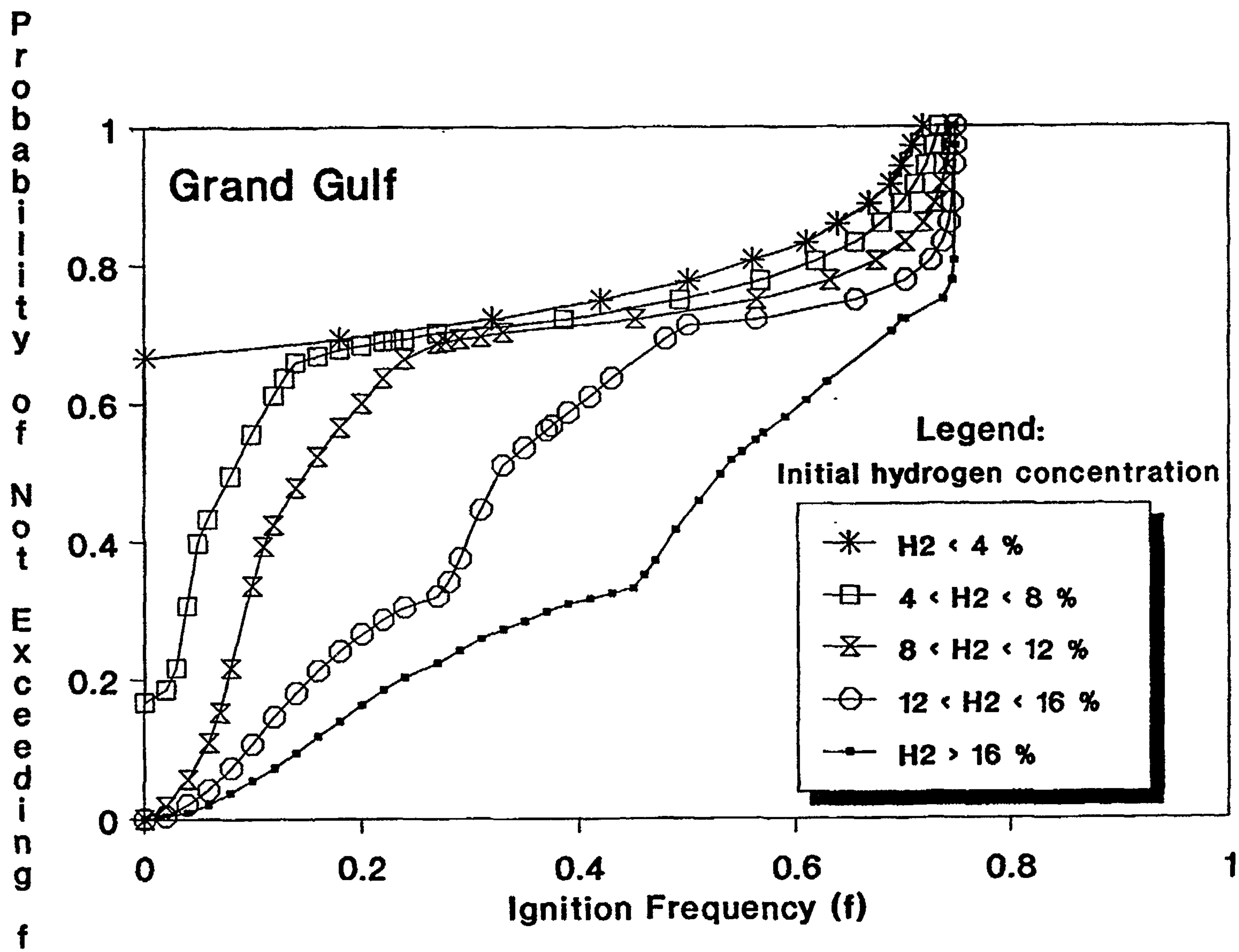


Figure C.4.3 Ignition frequency as a function of initial hydrogen concentration in the Grand Gulf containment building (outer containment-wetwell region for accident progressions in which the RPV is at high pressure).

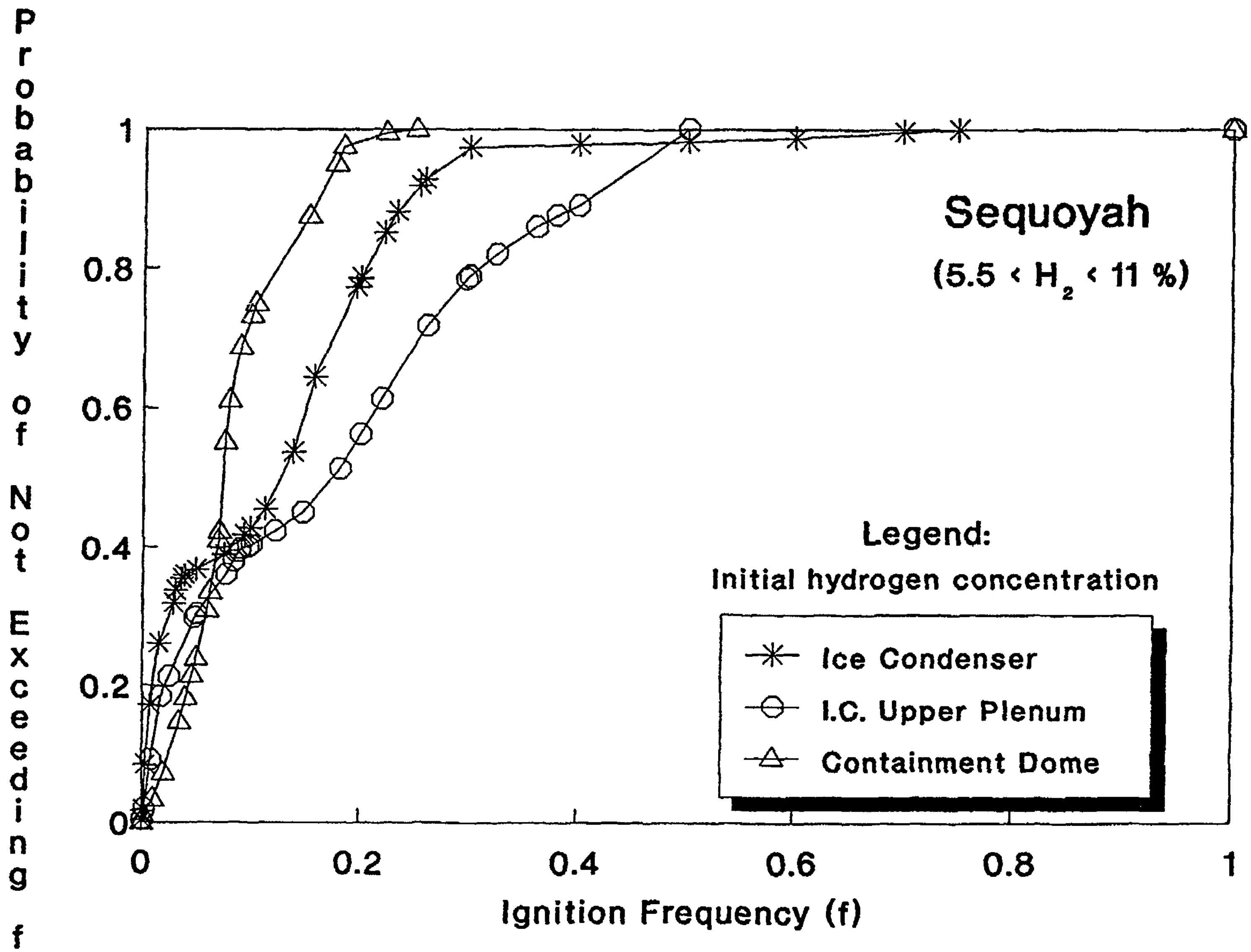


Figure C.4.4 Ignition frequency for various regions of the Sequoyah containment—illustrated for an assumed initial hydrogen concentration between 5.5 and 11 volume percent.

Selected Distributions for Grand Gulf

The range of estimated loads on the Grand Gulf containment generated by a hydrogen deflagration, initiating from various hydrogen concentrations, is shown in Figures C.4.5 and C.4.6. The former represents accident progressions in which the concentration of steam in containment at the time of ignition is relatively high, and the latter represents accident progressions with low initial steam concentrations. The distribution for the estimated strength of the containment pressure boundary and the wall separating the drywell from the outer containment is also indicated on these figures to illustrate the potential for structural failure. The likelihood of coincident failure of the containment shell and drywell wall is discussed in Section C.4.3.

Selected Distributions for Sequoyah

The range of estimated loads on the Sequoyah containment generated by a hydrogen deflagration is shown in Figures C.4.7 and C.4.8. The former illustrates the range of loads generated during fast station blackout sequences, the latter illustrates the range of loads generated during slow blackout sequences with an intermediate-size break in the reactor coolant pump seals. In both figures, distributions for containment loads are shown for cases in which the quantity of hydrogen released to the containment corresponds to that generated from the oxidation of 20, 60, and 100 percent of the core Zircaloy inventory. To illustrate the potential for these loads to cause containment failure, the distribution for the estimated strength of the containment pressure boundary is also indicated on these figures.

Quantification of Hydrogen Detonation Frequency

Detonations can occur if hydrogen is allowed to accumulate to concentrations greater than 12–14 volume percent in the presence of an ignition source and a sufficient concentration of oxygen. The current analysis indicates that the possibility of developing such conditions in either Grand Gulf or Sequoyah is low but cannot be dismissed. Figures C.4.9 and C.4.10 show aggregate distributions for the frequency of hydrogen detonations in Grand Gulf and Sequoyah, respectively (each showing the dependence on initial hydrogen concentration). The frequencies shown in these figures are conditional on hydrogen concentrations exceeding the values shown in the figure legends. The discontinuities in the distributions are a result of averaging the distributions of experts with substantially different judgments regarding detonation frequency, some of whom believe the frequency distribution to have thresholds (i.e., frequency cannot be lower than x or higher than y).

The conditions under which detonations were considered in the two plants are quite different, however. In Grand Gulf, large quantities of hydrogen may accumulate in the outer containment (as a result of substantial in-vessel metal-water reaction) and a global detonation may result. In the case of Sequoyah, a deflagration-to-detonation transition is a more likely means of creating a hydrogen detonation. The configuration of the ice condenser (a vertically oriented enclosed compartment with obstacles in the flow path) can promote flame acceleration and initiate a detonation in upper portions of the ice bed or the upper plenum.

C.4.3 Treatment in PRA and Results

The probability distributions for this issue were implemented in the Grand Gulf and Sequoyah accident progression event trees. These trees (one for each plant) provide a structured approach for evaluating the various ways in which a severe accident can progress, including important aspects of the reactor coolant system thermal-hydraulic response, core melt behavior, and containment loads and performance. The accident progression event tree for each plant is a key element in the assessment of uncertainties in risk by accommodating the possibility that a particular accident sequence may proceed along any one of several alternative pathways (i.e., alternative combinations of events in the severe accident progression). The probability distributions (or split fractions) for individual and combinations of events within the tree provide the "rules" that determine the relative likelihood of various modes of containment failure.

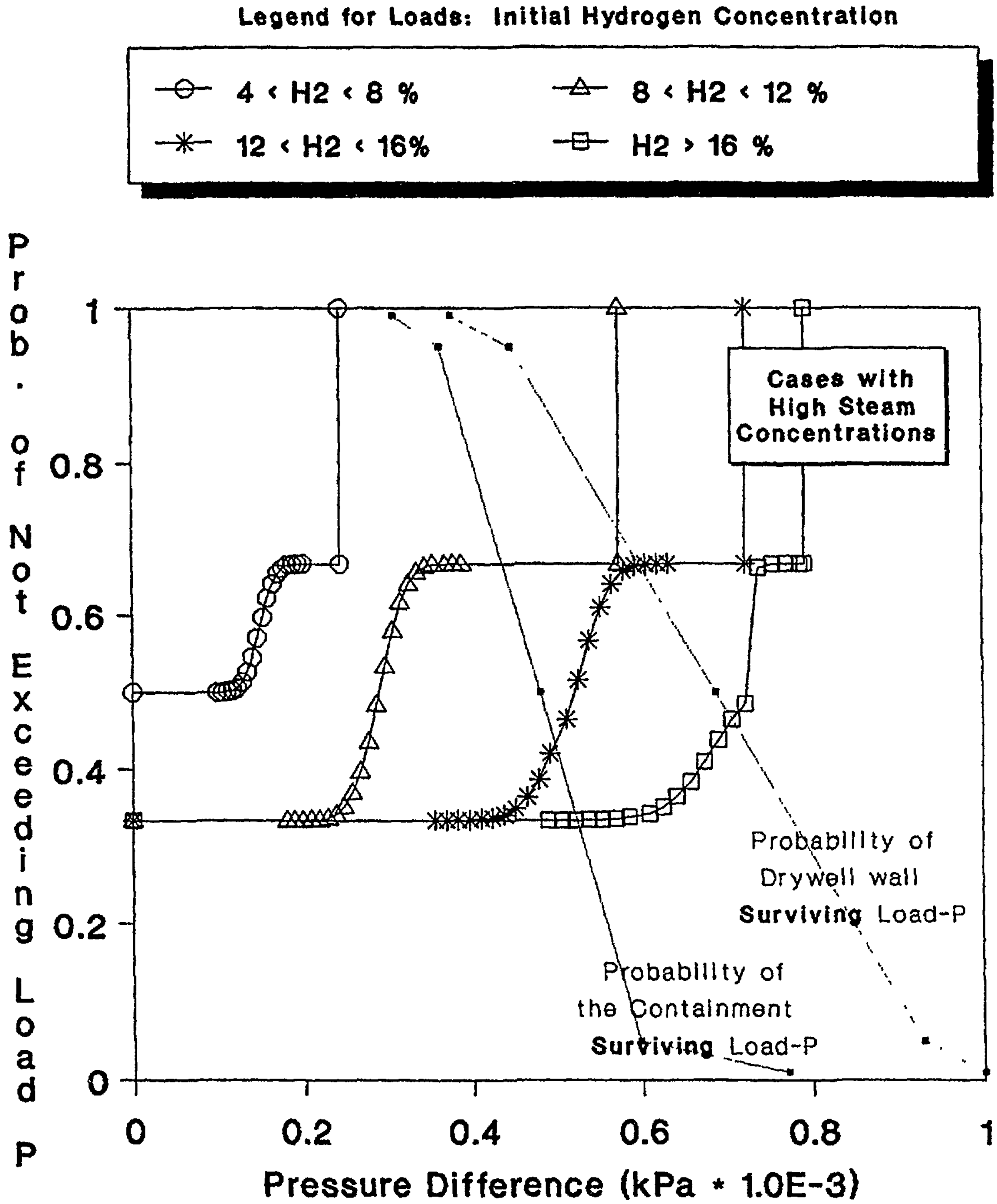


Figure C.4.5 Range of Grand Gulf containment loads in comparison with important structural pressure capacities (various initial hydrogen concentrations and *high* initial steam concentrations).

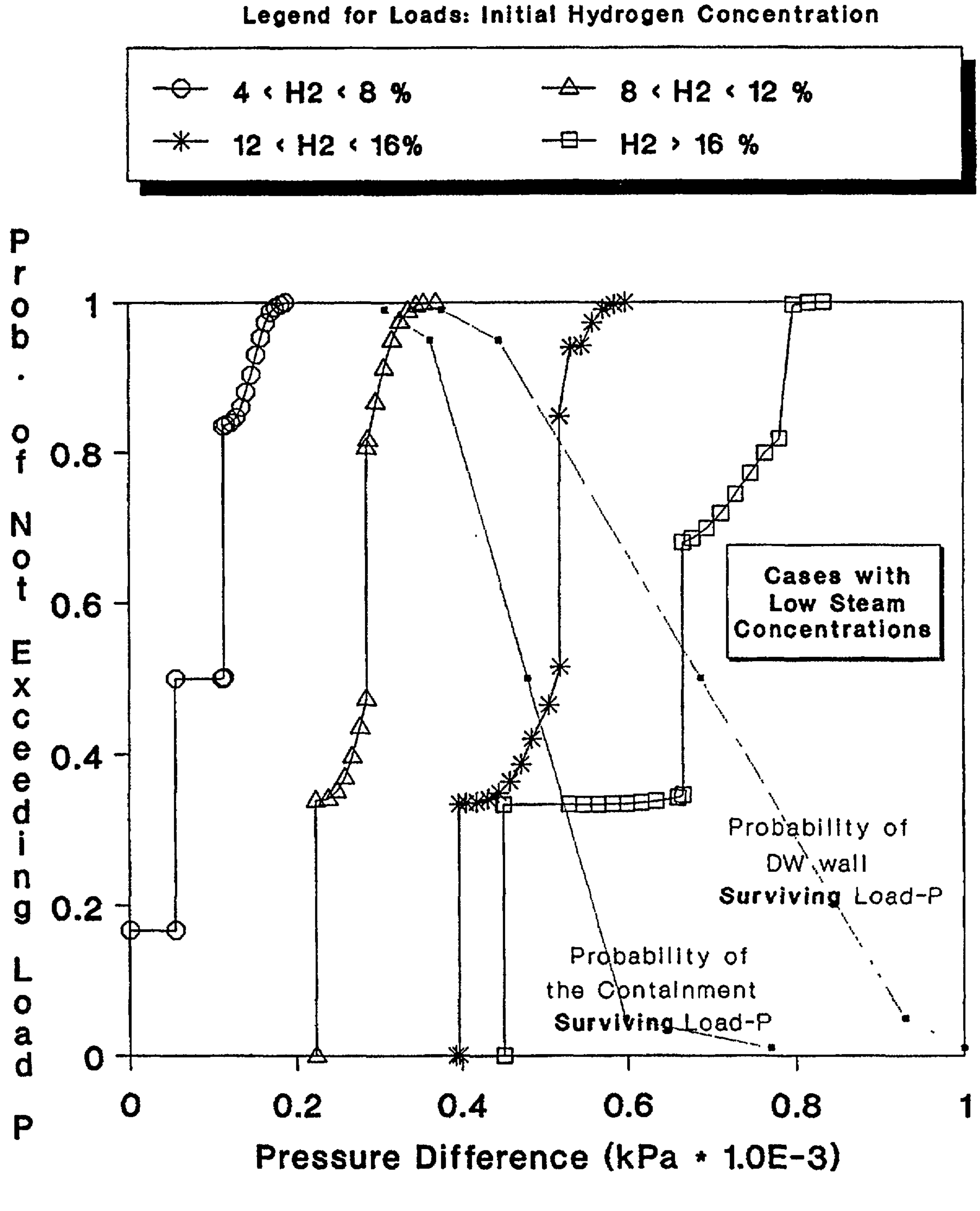


Figure C.4.6 Range of Grand Gulf containment loads in comparison with important structural pressure capacities (various initial hydrogen concentrations and *low* initial steam concentrations).

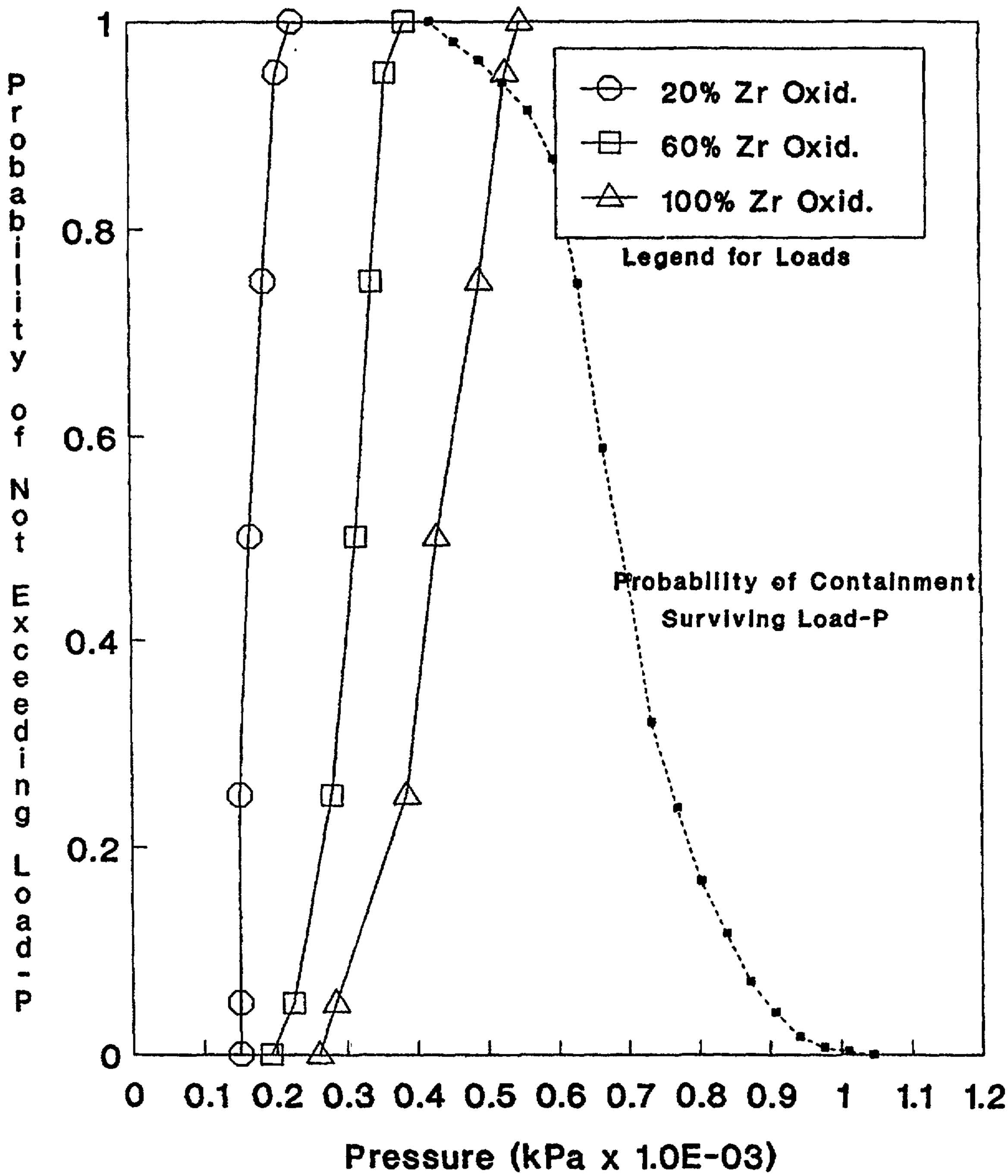


Figure C.4.7 Range of Sequoyah containment loads from hydrogen combustion in comparison with containment pressure capacity (fast station blackout scenarios with various levels of in-vessel cladding oxidation).

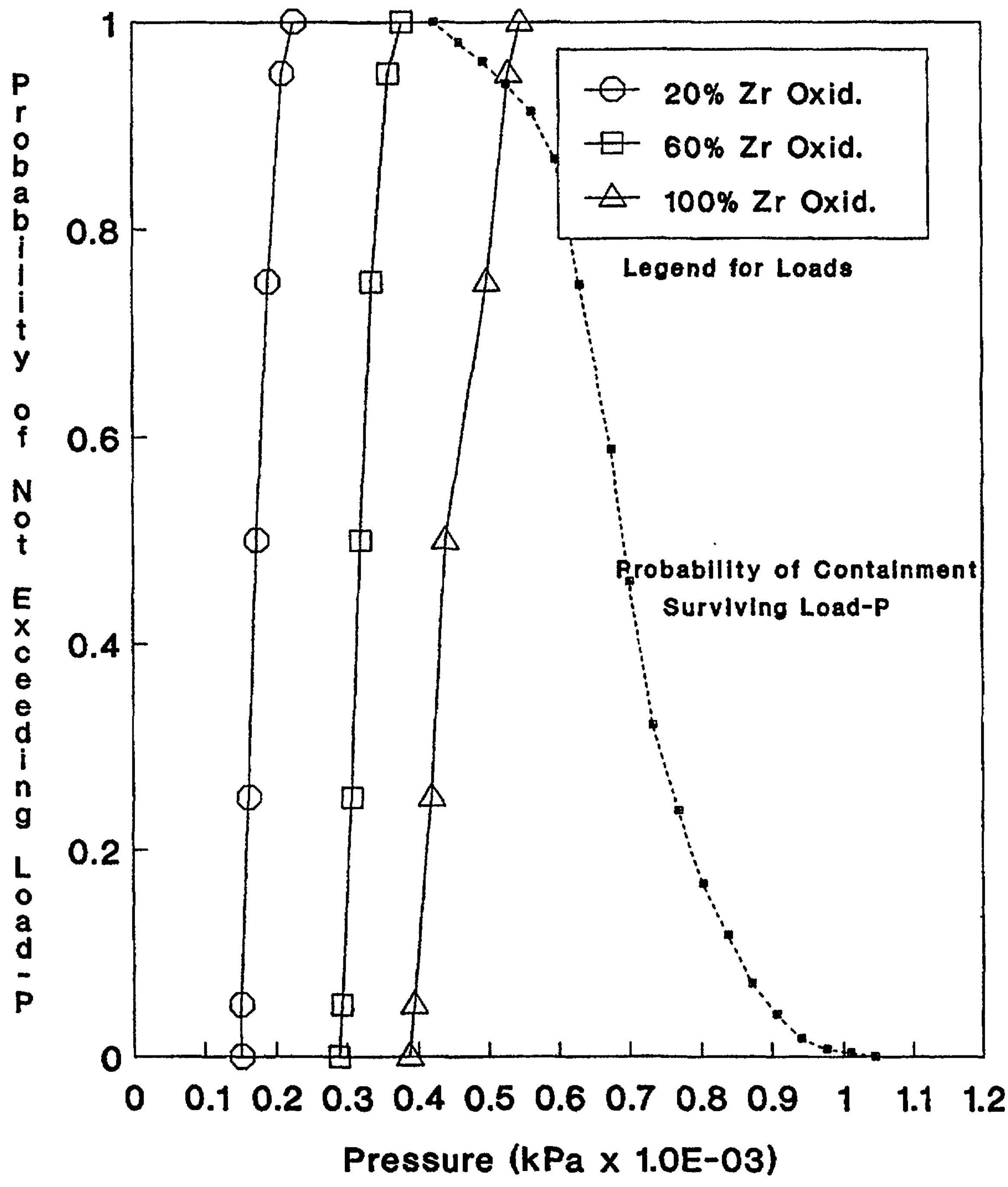


Figure C.4.8 Range of Sequoyah containment loads from hydrogen combustion in comparison with containment pressure capacity (slow station blackout accidents with induced reactor coolant pump seal LOCA and various levels of in-vessel cladding oxidation).

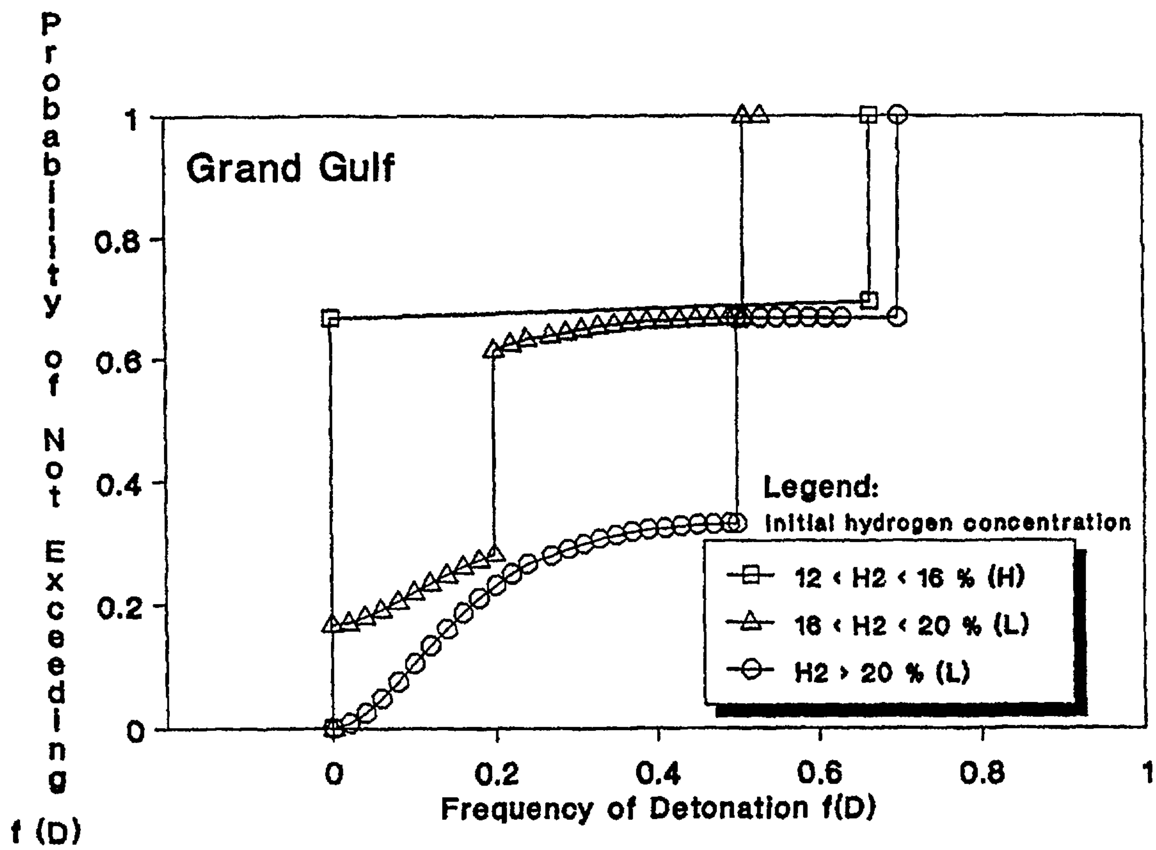


Figure C.4.9 Frequency of hydrogen detonations in Grand Gulf containment (probability of a detonation per combustion event—i.e., given ignition). H and L refer to high and low steam concentrations, respectively.

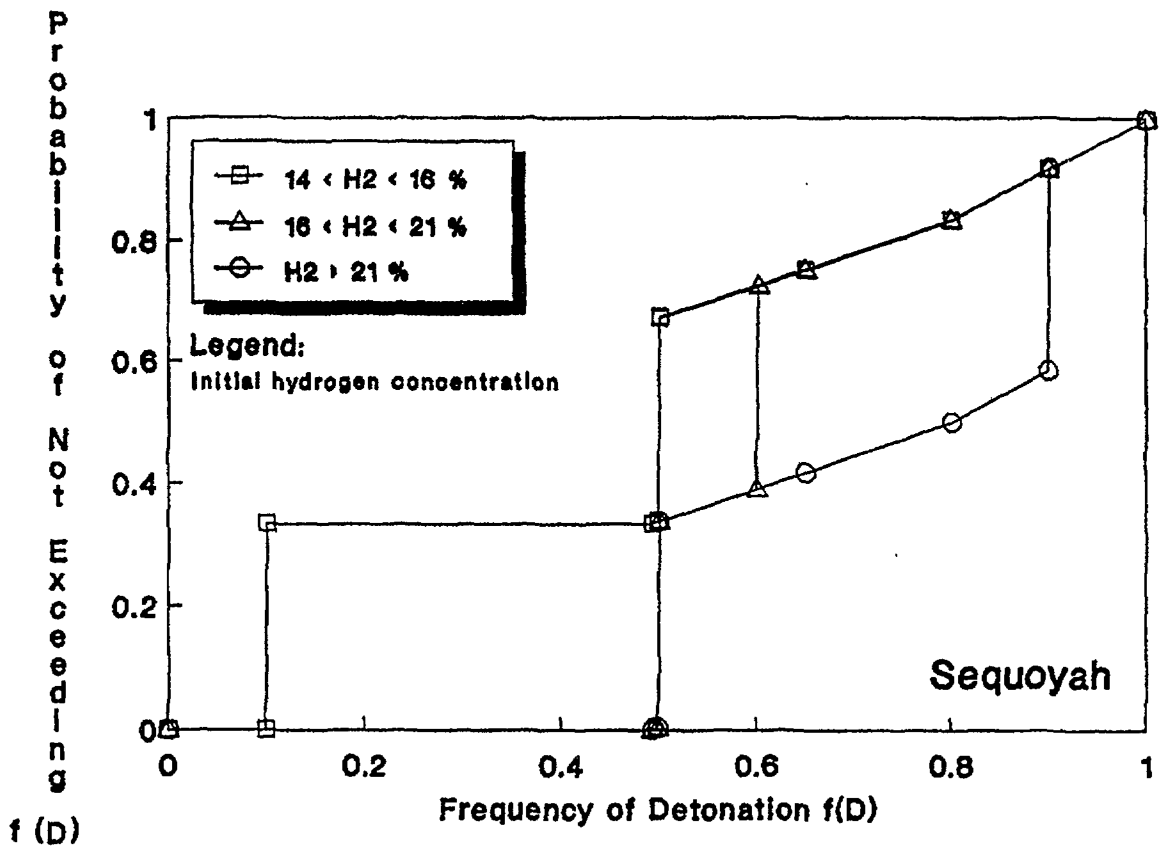


Figure C.4.10 Frequency of hydrogen detonations in Sequoyah ice condenser or upper plenum (probability of a detonation per combustion event).

As mentioned above, distributions for combustion loads were provided as input to the Grand Gulf accident progression event tree directly from the results of expert panel elicitations. An algorithm was created within the Sequoyah tree to calculate combustion loads as a function of "upstream" conditions such as the mass of hydrogen released from the reactor vessel, the distribution of the hydrogen through the containment, the compartment of the containment in which ignition took place, and other variables accounted for in the tree (e.g., burn completeness, potential for flame propagation). For each pass through the event tree in the risk uncertainty analysis, the likelihood of containment failure (and drywell failure for Grand Gulf) was determined by comparing the sampled value of the combustion loads against the sampled value of the containment pressure capacity.

Station blackout dominates the estimated core damage frequency for Grand Gulf, therefore rendering the igniters unavailable for most of the accident sequences important to risk. The attendant potential for hydrogen to accumulate and spontaneously ignite at relatively high concentrations received particular attention in this analysis. Of particular interest was the potential for hydrogen burns to induce a breach of the containment pressure boundary and result in suppression pool bypass. This combination of events could occur if a hydrogen burn were of sufficient magnitude to fail the containment shell and the drywell wall.

The outer containment (wetwell) pressure boundary is not an extremely strong structure in the BWR Mark III design (e.g., the Grand Gulf outer containment design pressure is 15 psig—103 kPa). As with all the pressure-suppression containment designs, heavy reliance is placed on the suppression pool to reduce thermodynamic loads on this structure. The structures forming the drywell, however, are much stronger (design pressure of 30 psid—207 kPa). The present analysis considers the possibility of combustion-generated loads failing either or both structures. If the containment pressure boundary is breached, but the drywell remains intact, the pressure-suppression pool is available throughout the accident to reduce the magnitude of the radioactive release to the environment. If, however, the loads accompanying a hydrogen burn (or some other event) are of sufficient magnitude to damage the drywell walls and allow for suppression pool bypass, the accompanying radioactive release can be substantial.

The contribution of hydrogen combustion to early containment loads in Grand Gulf is evident in the fraction of accident progressions with early containment failure caused by hydrogen burns. These are summarized below for each type of accident sequence that contributes greater than 1 percent of the mean total core damage frequency:

Type of Accident	Fractional Contribution to Mean Total Core Damage Frequency	Fraction Resulting in Early Containment Failure	Fraction of Early Containment Failures Caused by Hydrogen Burn or Detonation
Short-term station blackout	0.94	0.46	0.96
Long-term station blackout	0.02	0.86	0.44
ATWS	0.03	0.85	0.36
Transients	0.01	0.56	0.98

The vast majority of early containment failures for short-term station blackout (the dominant contributor to the Grand Gulf core damage frequency) is shown to be caused by loads generated by hydrogen combustion.

A substantially smaller fraction of the accident progressions in Sequoyah are estimated to result in early containment failure from hydrogen burns. The fraction results in early containment failure for the two most important types of core damage accidents in Sequoyah are summarized as follows:

Type of Accident	Fractional Contribution to Mean Total Core Damage Frequency	Fraction Resulting in Early Containment Failure from Hydrogen Burns
LOCA	0.63	0.001
Station blackout	0.25	0.05

The majority of cases in which hydrogen combustion produces a load sufficiently large to compromise containment integrity involves deflagration (quasistatic) loads, not detonations (dynamic loads).

REFERENCES FOR SECTION C.4

C.4.1 National Research Council, *Technical Aspects of Hydrogen Control and Combustion in Severe Light-Water Reactor Accidents*, National Academy Press, Washington, DC, 1987.

C.4.2 U.S. Nuclear Regulatory Commission, "Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

C.4.3 J.A. Gieseke et al., "Radionuclide Release Under Specific LWR Accident Conditions," Battelle Columbus Laboratories, BMI-2104, Vols. III and IV, July 1984.

C.4.4 R.S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios," Battelle Columbus Division, NUREG/CR-4624, BMI-2139, Vols. 2 and 4, July 1986.

C.4.5 R.S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios: Supplemental Calculations," Battelle Columbus Division, NUREG/CR-4624, Vol. 6, BMI-2139, August 1990.

C.4.6 A.L. Camp et al., "MARCH-HECTR Analysis of Selected Accidents in an Ice-Condenser Containment," Sandia National Laboratories, NUREG/CR-3912, SAND83-0501, January 1985.

C.4.7 S.E. Dingman et al., "Pressure-Temperature Response in an Ice Condenser Containment for Selected Accidents," *Proceedings of the 13th Water Reactor Safety Research Information Meeting* (Gaithersburg, MD), NUREG/CP-0072, February 1986.

C.4.8 S.E. Dingman et al., "MELCOR Analyses for Accident Progression Issues," Sandia National Laboratories, NUREG/CR-5331, SAND89-0072, to be published.*

C.4.9 Technology for Energy Corporation, "Integrated Containment Analysis," IDCOR Task 23.1, 1984.

C.4.10 A.L. Camp et al., "Light Water Reactor Hydrogen Manual," Sandia National Laboratories, NUREG/CR-2726, SAND82-1137, September 1983.

C.4.11 F.T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

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

C.5 PWR Containment Loads During High-Pressure Melt Ejection

During certain severe reactor accidents, such as those initiated by station blackout or a small-break loss-of-coolant accident (LOCA), degradation of the reactor core can take place while the reactor coolant system remains pressurized. Left unmitigated, core materials will melt and relocate to lower regions of the reactor pressure vessel. Molten material will eventually accumulate on the inner radius of the vessel bottom head and attack lower head structures. If the bottom head of the reactor vessel is breached, core debris may be ejected into the containment under pressure.* The blowdown of reactor coolant system gases atomizes ejected molten material and transports the resulting particles through the containment atmosphere. The attendant exothermic oxidation of metal constituents of the molten particles and rapid transfer of sensible heat to the containment atmosphere is referred to as "direct containment heating." The pressure rise in containment induced by high-pressure melt ejection (HPME) can be large enough to challenge containment integrity. Since containment failure immediately following reactor vessel breach can lead to a relatively large environmental release of radionuclides (and proportionally high consequences), uncertainties in the magnitude of containment loads at vessel breach are important contributors to the uncertainty in reactor risk.

A significant rise in containment pressure can be produced by HPME in PWR (large dry, subatmospheric, and ice condenser) containments and BWR (Mark I, II, and III) containments. These loads were, therefore, considered in the analyses for each of the plants examined in this study. The magnitude of the pressure rise depends strongly on details of reactor cavity (PWR)/pedestal (BWR) geometry and is, therefore, highly plant-specific. For the two BWRs examined in the present study, containment loads attributable to high-pressure melt ejection were not found to be the dominant contributor to the likelihood of early containment failure. This is a result of several factors, including the comparatively lower nominal reactor vessel pressure in BWRs, the capability of the suppression pool to attenuate, to some extent, the energy released from HPME, and (in Peach Bottom) the relatively high likelihood of other containment failure mechanisms (e.g., drywell shell meltthrough—refer to Section C.7). The discussion presented in the remainder of this section, therefore, focuses on PWR containment loads during HPME.

The containment loads associated with HPME are generated by the addition of mass and energy to the containment atmosphere from several sources:

1. Blowdown of reactor coolant system steam and hydrogen inventory into the containment.

2. Combustion of hydrogen released prior to and during HPME.

3. Interactions between molten core debris and water on the containment floor.

4. Direct containment heating.

Uncertainties in containment loads at vessel breach arise from the nonstochastic nature of some of these events (e.g., hydrogen burns), as well as a poor understanding of the phenomena governing others (e.g., direct containment heating).

In the preliminary containment response analyses (i.e., published in the February 1987 draft for comment release of this report), each of these contributors to containment loads was treated individually. An estimate of the total rise in containment pressure at vessel breach was generated by the superposition of pressure increments from each contributor. This approach was acknowledged to compromise the synergistic aspects of the phenomena involved but was analytically convenient. Among the motivations for taking this approach was the desire to isolate the uncertainties associated with direct containment heating, a controversial and highly uncertain phenomenon that can have a significant impact on the estimation of risk.

Although more experimental and analytical information regarding direct containment heating has been generated for and incorporated into the final analyses for this study, substantial uncertainties persist and

*In roughly 70+ percent of the PWR accident scenarios during which core degradation begins while the reactor pressure is at elevated pressures, an unisolatable breach in the primary system pressure boundary opens in hot leg piping, the pressurizer surge line, steam generator tubes, reactor coolant pump seals, or via a stuck-open power-operated relief valve. This break is sufficiently large to depressurize the reactor vessel prior to vessel breach. Containment loads during high-pressure melt ejection apply to the scenarios represented by the remaining 30 percent of the cases. The mechanisms for and likelihood of reactor vessel depressurization are treated as a separate uncertainty issue and are discussed in Section C.6.

the phenomenon continues to generate controversy. The motivation for isolating direct containment heating remains valid; however, the arguments in favor of treating containment response during this important stage of severe accident progression in a physically self-consistent manner prevailed. In the current analyses (presented in this document and NUREG/CR-4551 (Refs. C.5.1 through C.5.7)), containment pressure rise at vessel breach is treated as a single issue representing the combined uncertainties associated with the synergism of the four events listed above. As a result, the pressure increment attributable to an isolated phenomenon (e.g., direct containment heating) is not separable.

C.5.1 Issue Definition

This issue characterizes the uncertainties in containment loads that accompany reactor vessel breach. These uncertainties have been characterized over the entire range of possible initial reactor coolant system pressures. The largest loads (and, thus, the most significant challenges to containment integrity), however, are generated when the reactor vessel is breached while at elevated pressures. The following discussion will, therefore, focus on accident scenarios in which reactor vessel breach occurs at pressures between 500 psia (35 bar) and 2500 psia (170 bar). Further, pressure increments are characterized for three PWR containment designs—Surry Unit 1 (subatmospheric), Zion Unit 1 (large, dry), and Sequoyah Unit 1 (ice condenser). Diagrams of the Surry and Zion containments are shown in Figures C.5.1 and C.5.2, respectively. A similar diagram of the Sequoyah containment was discussed in Section C.4 (refer to Fig. C.4.1).

A rise in containment pressure may result from one or more of the four events listed earlier. The blowdown of steam and hot gases from the reactor coolant system into the containment can be calculated with reasonable precision. The pressure rise attributable to this event may be augmented by the steam production accompanying an interaction of core debris and water on the cavity floor, the generation of energy from hydrogen combustion, and energy addition from direct containment heating. Each of these potential contributors to containment loads at vessel breach is subject to physical limitations.

- Potential ex-vessel interactions between core debris and water are of concern only for accident scenarios during which water covers the reactor cavity floor prior to vessel breach. In general, this implies the successful operation of containment sprays or a LOCA or both.

- Combustion of sufficient hydrogen to generate a substantial pressure rise is subject to physical requirements regarding minimum hydrogen concentrations, oxygen availability, and maximum inerting gas concentrations. Hydrogen concentrations in containment prior to vessel breach depend upon in-vessel core melt progression (primarily the fraction of the core Zircaloy oxidized before vessel breach, which was addressed as a separate issue) and the type of accident scenario being considered.

- Direct containment heating is a term that refers to a series of physicochemical processes that have been postulated to accompany the ejection of molten core debris from a reactor vessel under high pressure. If a large fraction of the ejected molten core debris is dispersed into the containment as fine particles, a substantial portion of the debris' sensible heat can be transferred rapidly to the atmosphere. The containment pressure rise accompanying direct containment heating depends on reactor cavity geometry, the mass of material dispersed by reactor vessel blowdown, and several other parameters described later.

The resulting pressure rise can be further supplemented by the release of chemical energy associated with the oxidation of metals in the particulate melt as they are transported through the containment atmosphere. The total energy release, and thus the pressure rise, attributable to these phenomena depend on the fraction of the core (molten mass) participating in the process and the model used to represent the events that accompany reactor pressure vessel breach.

Distinct cases are established to consider separately each plausible combination of debris characteristics and containment initial conditions. The case structure accounts for uncertainties in selected severe accident events and phenomena that precede reactor vessel breach. Uncertainties associated with the containment loads generated by processes and events that occur after the core debris leaves the reactor vessel are represented by the distribution of plausible pressure increments assigned to each case. The specific parameters and range of values used to define the case structure are:

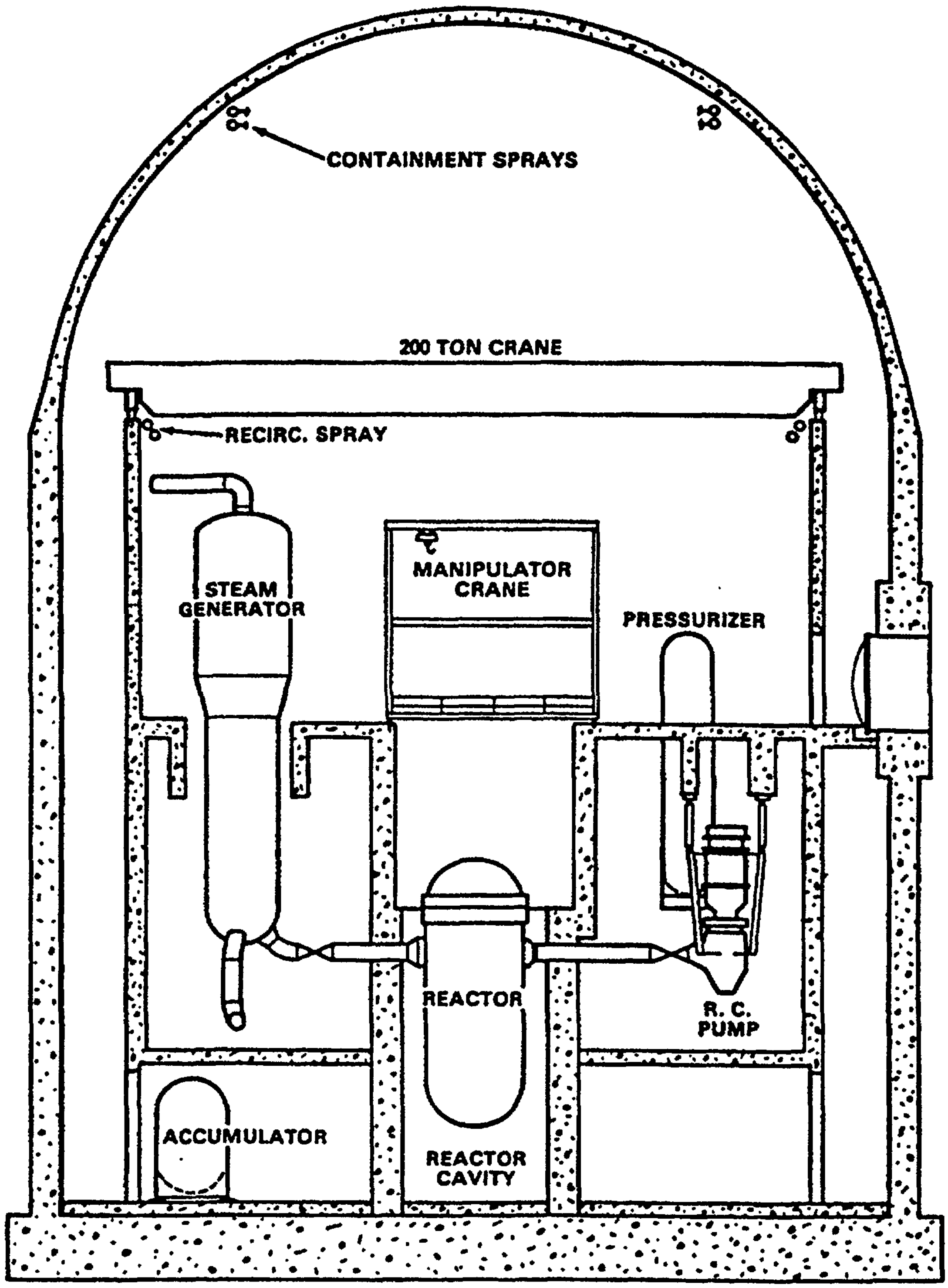


Figure C.5.1 Cross section of Surry Unit 1 containment.

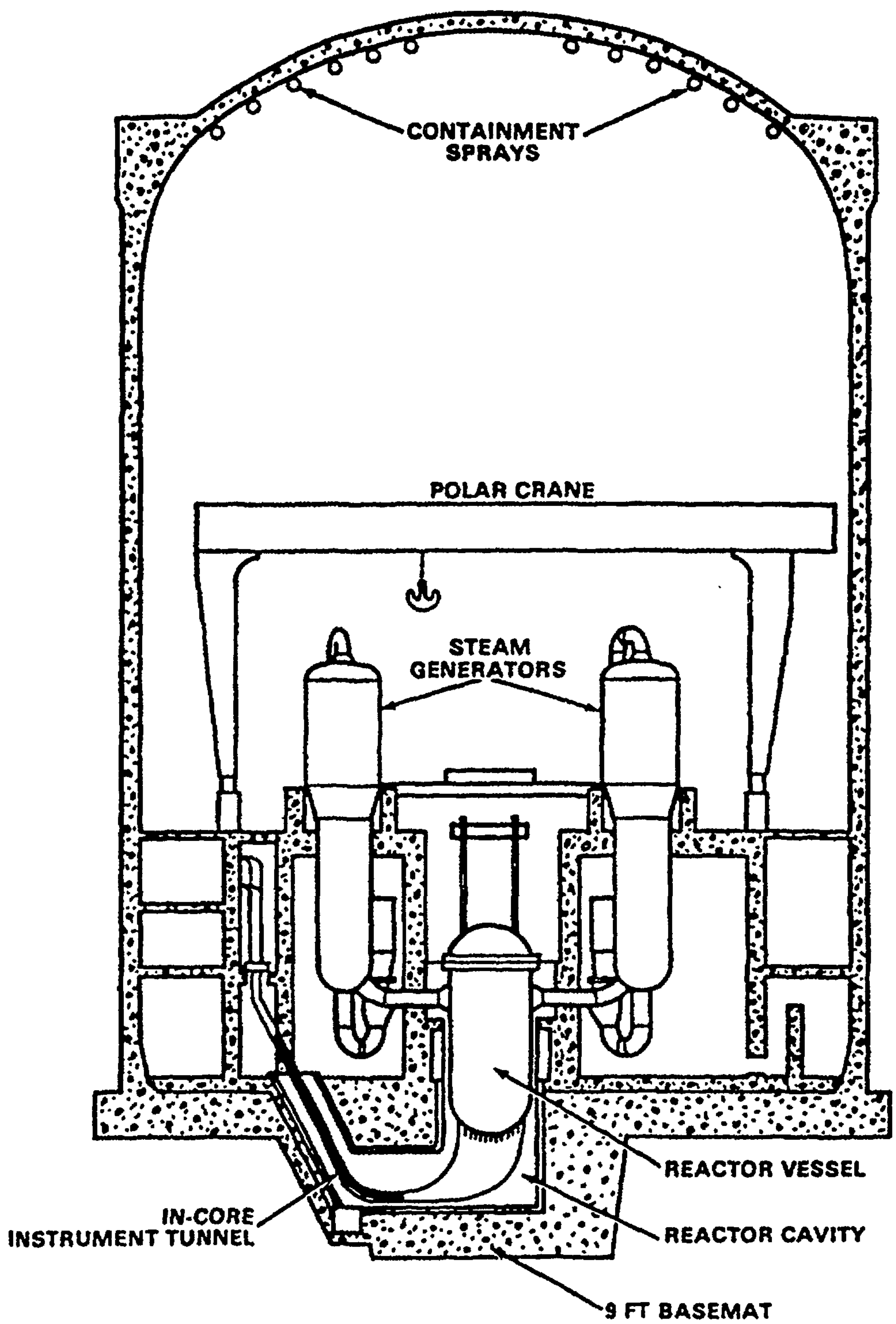


Figure C.5.2 Cross section of Zion Unit 1 containment.

Parameters Defining Case Structure	Values Considered
1. Reactor vessel pressure prior to vessel breach	**High, Medium, Low** p > 1000 psia 500 < p < 1000 psia p < 200 psia
2. Amount of unoxidized metal in melt	**High, Low** 60 percent of initial inventory 25 percent of initial inventory
3. Fraction of molten core debris ejected	**High, Medium, Low** Approx. 50 percent of total Approx. 33 percent of total Approx. 10 percent of total
4. Initial size of hole in reactor vessel lower head when breached	**Large, Small** Approx. 2.0 sq-meters Approx. 0.1 sq-meters
5. Presence (or lack) of water in reactor cavity	**Full, Half-full, Dry**
6. Containment spray operation during HPME	**Yes:** Operating **No:** Not operating

A distribution of values for the incremental rise in containment pressure was generated for each PWR containment type analyzed (large dry, subatmospheric, and ice condenser) and for each combination of parameter values in the case structure. The relative likelihood of the cases was not considered as part of this issue but is determined in the evaluation of the accident progression event tree. Section C.5.3 discusses application of the estimated pressure increments for each case in the risk uncertainty analysis.

A qualitative description of how the above parameters influence containment response follows. The technical bases (experimental evidence, calculational results, or engineering judgment) for quantifying these influences are also indicated.

Reactor Vessel Pressure

Reactor vessel pressure at the time of vessel breach characterizes the internal energy stored in reactor coolant system gases and provides the motive force for core debris dispersal. Higher initial pressures lead to larger pressure increments from reactor coolant system blowdown. Provided the initial reactor vessel pressure is sufficient to transport hot gases and reactive material (molten debris particles) to upper regions of the containment, the pressure rise attributable to direct containment heating is probably insensitive to the initial reactor vessel pressure. Attempts have been made to define a cutoff pressure (pressure below which substantial direct containment heating does not occur); however, the technical basis for a cutoff pressure is weak. In this assessment, direct containment heating is regarded as possible if the reactor vessel pressure at the time of vessel breach is greater than approximately 200 psia (14 bar). The likelihood of, and pressure rise associated with, hydrogen combustion and ex-vessel core-coolant interactions are largely insensitive to initial reactor vessel pressure.

Unoxidized Metal Content in Melt

Among the important contributors to containment loads during high-pressure melt ejection is the energy release associated with the oxidation of unreacted metals (particularly Zircaloy) in the melt. The fragmentation and dispersal of debris throughout the containment atmosphere can significantly enhance the rate of Zircaloy oxidation by exposing a large surface area of unreacted metal to the containment atmosphere. The conceptual picture of hot, unreacted metals being dispersed in air atmosphere might suggest a strong relationship between the total energy released and the mass of unoxidized metal being dispersed. However, CONTAIN calculations (Ref. C.5.8) suggest that containment loads are relatively insensitive to the extent of in-vessel Zircaloy oxidation (mass of Zircaloy consumed in-vessel and, therefore, unavailable for oxidation during melt ejection). These calculations indicate only minor differences in containment pressurization when the unoxidized fraction of metal in the melt (dispersed at vessel breach) is increased from 50 to 70 percent of the initial inventory. It is not clear that a similar trend would be observed if the mass of dispersed metals were less than 50 percent of the initial inventory (i.e., if greater than 50 percent of the metal mass oxidized in-vessel or if a small fraction of the total mass of debris were to be ejected).

Fraction of Molten Debris Ejected

The amount of core material ejected from the vessel depends on the fraction of the core that has melted and collected at the bottom of the reactor vessel at the time of vessel breach. This is governed by the model used to represent in-vessel core melt progression (addressed in other uncertainty issues). Three nominal values were considered in this assessment (0.50, 0.33, and 0.10) to represent large (greater than 40 percent), medium (between 20 and 40 percent), and small (less than 20 percent) fractions of core melted and available for ejection, respectively. The results of parametric studies performed with the CONTAIN computer code (Ref. C.5.8) indicate that the containment pressure increment at vessel breach increases substantially with increasing fraction of melt ejected. Illustrative results of these calculations are shown in Figure C.5.3, which presents the predicted peak pressure for variations of a station blackout accident scenario in which progressively greater fractions of the initial core mass were assumed to be ejected. In these calculations, all the melt ejected is assumed to participate in direct containment heating. Results of scaled high-pressure melt ejection experiments in the Surtsey facility (Refs. C.5.9 through C.5.13) indicate that not all ejected debris may participate, however.

Initial Size of Hole in Vessel When Breached

Alternative conceptual models for in-vessel core melt progression suggest quite different modes of reactor vessel breach. One model assumes that a localized thermal attack of the vessel lower head results in the failure of one or more lower head penetrations. Such a model suggests an initial hole size in the neighborhood of 0.1 m^2. Much larger hole sizes are conceivable, however, particularly if the reactor vessel lower head fails by creep rupture or when the hole is ablated. The primary parameter affected by the initial hole size is the rate at which vessel blowdown occurs (larger initial hole size implying more rapid blowdown and melt ejection). The CONTAIN parametric studies referenced above also examined this sensitivity by varying the length of time required to blow down the reactor vessel. Substantially larger pressure rises were predicted when the blowdown period was shortened from 30 seconds to 10 seconds.

Presence of Water in Reactor Cavity

At least two scenarios are conceivable when water interrupts the pathway for debris dispersal following reactor vessel breach (as it would if the reactor cavity were filled with water). One scenario is that one or more steam explosions will occur after only a fraction of the debris has been injected into the cavity and that the cavity water will then be dispersed ahead of the bulk of the injected debris. Another possibility is that the relatively cold water will be co-dispersed with the debris, exiting the cavity region as small droplets intermixed with the transported debris, steam, and hydrogen. Experiments with water-filled cavities (Ref. C.5.13) have been inconclusive, in part because of the tendency of the experimental facilities to be destroyed by the debris-water interactions. Reality may involve some combination of these two scenarios.

The scenario resulting in co-dispersed water has received considerable attention. Of principal interest is the nature of the interaction between the debris particles and water droplets. The water may continue to quench the debris, mitigating the effects of direct containment heating. However, the fate of the steam generated by this quenching is uncertain. It could simply increase the partial pressure of steam in the containment, thereby producing a moderate addition to containment loads, or it might act as a source of oxygen for unquenched debris and substantially enhance the oxidation of metallic particles. This tradeoff was investigated in some detail in the CONTAIN sensitivity studies referenced above. The effects of co-dispersed water were shown to be quite sensitive to the timing and location of water addition, assumptions regarding droplet-debris reaction kinetics, and the amount of water involved.

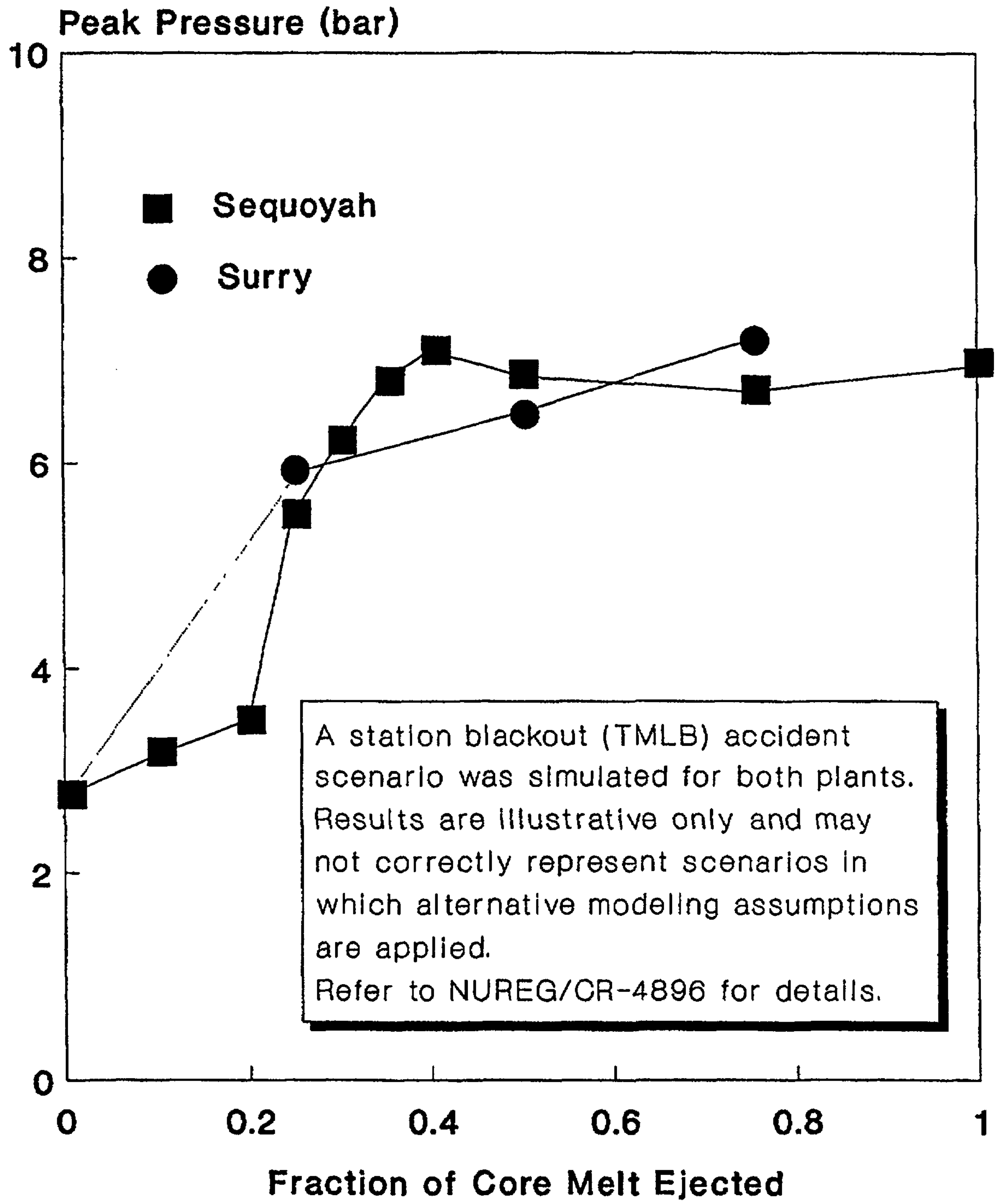


Figure C.5.3 Calculated containment peak pressure as a function of molten mass ejected (Ref. C.5.8).

Containment Sprays and Ice Condenser

The effect of containment sprays on containment loads from reactor vessel blowdown has been relatively well characterized by numerous containment response computer code calculations. In Surry, for example, it has been estimated (Ref. C.5.2) that the operation of containment sprays reduces the pressure rise at vessel breach by approximately 30 to 45 psi (2 to 3 bar). The impact of sprays on dispersed core debris is not well understood and was not explicitly examined in the CONTAIN sensitivity analyses (Ref. C.5.8). For some plants, however, early spray operation may ensure a substantial inventory of water in the reactor cavity. The uncertainties in estimating the effect of this water were described above.

The presence of the ice condenser in the Sequoyah containment introduces significant uncertainties in estimating HPME loads for this plant. There are no experimental data regarding ice condenser performance under conditions representative of those accompanying HPME. In the present study, quantitative assessments of core debris capture and pressure suppression during HPME is largely based on the subjective judgment of experienced containment response analysts. Topics of particular concern include the potential for "channeling" (the preferential melting of a vertical column of ice, creating an early ice bypass pathway) and hydrogen detonations. The possibility of a rapid release of large quantities of hydrogen following reactor vessel breach, accompanied by effective steam condensation as the steam/hydrogen mixture passes through the ice beds, can generate conditions that favor hydrogen detonations in the upper regions of the ice condenser. The dynamic loads generated from such events are not explicitly included in this issue. The reader is referred to Section C.4 for more details on the treatment of hydrogen combustion phenomena in this containment design.

C.5.2 Technical Bases for Issue Quantification

This issue was presented to a panel of experienced severe accident analysts. Six panelists addressed containment loads for the three PWR plants:*

Louis Baker—Argonne National Laboratory,
Kenneth Bergeron—Sandia National Laboratories,
Theodore Ginsberg—Brookhaven National Laboratory,
James Metcalf—Stone & Webster Engineering Corp.,
Martin Plys—Fauske & Associates, Inc., and
Alfred Torri—Pickard, Lowe & Garrick, Inc.

Each panelist provided a distribution of values for containment pressure rise following reactor vessel breach for each of the cases outlined in Section C.5.1. These distributions characterize the panelist's judgment for the range of plausible containment loads for each case and the relative confidence (i.e., degree of belief) that particular values within that range are the "correct" ones for the conditions specified by the issue case structure. Panelists based their judgments on the current body of experimental evidence and analytical information, a sample of which was summarized above.

A summary of the expert panel's judgments is provided below. In most instances, aggregate distributions (arithmetic average among the panelists for a particular case) are presented to illustrate observable trends between cases. Examples of individual panelists' distributions are also shown to illustrate the variance of opinion within the panel. Complete documentation of the elicitation of expert judgment from these analysts is provided in Reference C.5.2.

An estimate of containment loads has little meaning if isolated from a corresponding estimate of static pressure capacity. Therefore, for each plant, the appropriate distribution for its static failure pressure (discussed in detail in Section C.8) is shown on each plot of containment loads. An example display of containment loads versus a reference static failure pressure is shown in Figure C.5.4. The curve for containment loads (base pressure plus the pressure increment accompanying reactor vessel breach) is shown as a cumulative distribution function (CDF). The static failure pressure is shown as a complementary cumulative distribution function (CCDF). The reason for this display format is to allow the reader to perform the following visual exercise:

*A minimum of three panelists addressed each plant. In general, each panelist was responsible for addressing the uncertainties in containment loads for two of the three PWR plants (Surry, Zion, or Sequoyah). If any analyst wished to provide a judgment for more than two, he was free to do so.

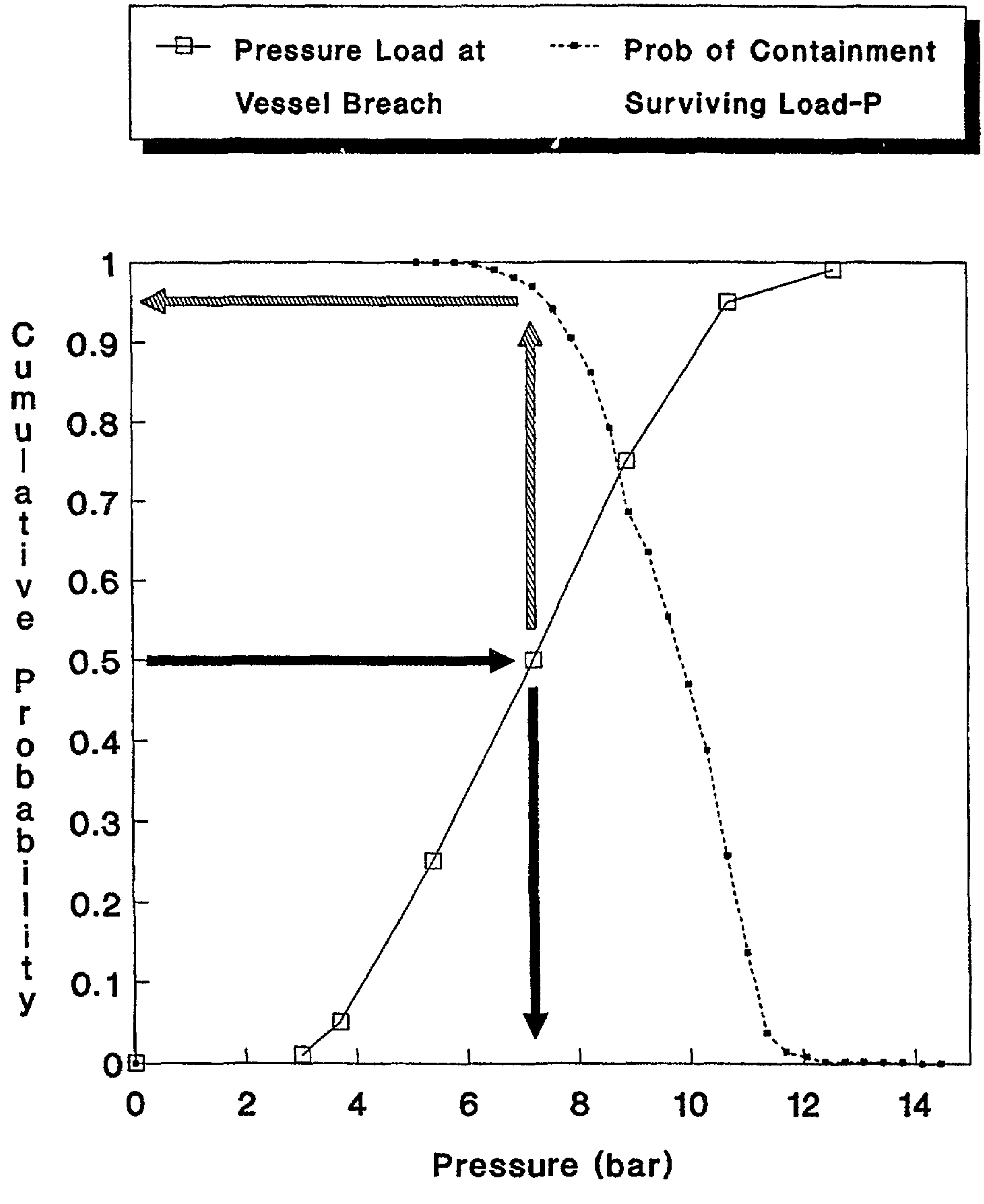


Figure C.5.4 Example display of distributions for containment loads at vessel breach versus static failure pressure.

Select a value (0.5 in the example—the median of the distribution) for the probability that the containment load at vessel breach is equal to or below some level. Read (horizontally) along the selected probability value, and determine the corresponding containment load (approximately 7 bar in the example). This means that the probability that the containment load at vessel breach is 7 bar or less is approximately 0.5. Next, read vertically upward to determine the point on the static failure pressure curve that intersects the same value of pressure, then left, back to the ordinate. This final value of probability (0.95 in the example) is the probability that the containment will survive the imposed load (7 bar in the example).

This format for displaying the containment performance information allows the reader to examine the relative likelihood of the containment surviving a particular load and the corresponding likelihood of that load being produced. In performing this exercise, it must be remembered that the distributions displayed in this manner apply only to the initial and boundary conditions specified by the case structure.

Containment Load Distributions for Surry

An important parameter in characterizing containment loads for high-pressure accident scenarios in Surry is the operation of containment sprays (or more specifically, the presence of water in the reactor cavity). Figures C.5.5 and C.5.6 show the estimated containment loads for HPME cases with sprays operating and not operating, respectively, for vessel failure with the reactor coolant system at high pressure. In each figure, the four curves for containment loads represent the aggregate (arithmetic average) distribution for the expert panel for each of four cases (identified in the legend). The variables that change among these cases are the initial size of the hole in the reactor vessel lower head at vessel breach and the fraction of molten debris ejected. (Each may take on high or low values as indicated in Section C.5.1. Curves are not given for cases with a "medium" fraction of the core ejected.) The largest loads are generated when both parameters take on high values (with or without sprays operating). Significantly lower loads are likely for cases in which containment sprays operate.

The likelihood that the Surry containment would survive the median (50th percentile) loads is greater than approximately 90 percent for all cases in which the sprays operate or provided a small fraction of the core debris is ejected (with or without sprays). It should be noted that accident sequences for which containment sprays are assumed to operate generally result in a cavity at least partially filled with water. The distributions shown in Figure C.5.5, therefore, assume a full cavity; those in Figure C.5.6, likewise, assume a dry cavity.

For Surry, the variance in the estimated containment loads among panelists is comparable to the variance among cases. Figure C.5.7 shows the distributions generated by each panelist for the four cases shown in Figure C.5.5. The range of median values for pressure rise among the panelists spans 30 to 60 psi (2 to 4 bar). This range increases to 120 psi (8 bar) at the distributions' upper bound. This trend is typical of virtually all the Surry cases.

Containment Load Distributions for Zion

Example distributions of containment loads at vessel breach in the Zion Unit 1 containment are shown in Figures C.5.8 and C.5.9 (with and without containment sprays operating, respectively). The boundary conditions represented by the cases illustrated in these figures are the same as those shown in Figures C.5.5 and C.5.6 for Surry. The Zion containment is shown to be able to withstand high-pressure melt ejection loads (even at the upper end of the uncertainty range) with very high confidence.

The variance in the estimated containment loads (among panelists) for Zion is also very similar to that for Surry. The variance indicated in the individual distributions displayed in Figure C.5.7 (for Surry) is representative of that observed for Zion. Individual panelists' distributions for Zion are, therefore, not displayed in this document; the reader is encouraged to review Reference C.5.2 for this information.

Containment Load Distributions for Sequoyah

The case structure for this plant includes an additional variable to account for uncertainties related to ice condenser performance (namely, the fraction of ice remaining at vessel breach). Distributions for Sequoyah containment loads for the cases similar to those displayed previously for Surry and Zion are

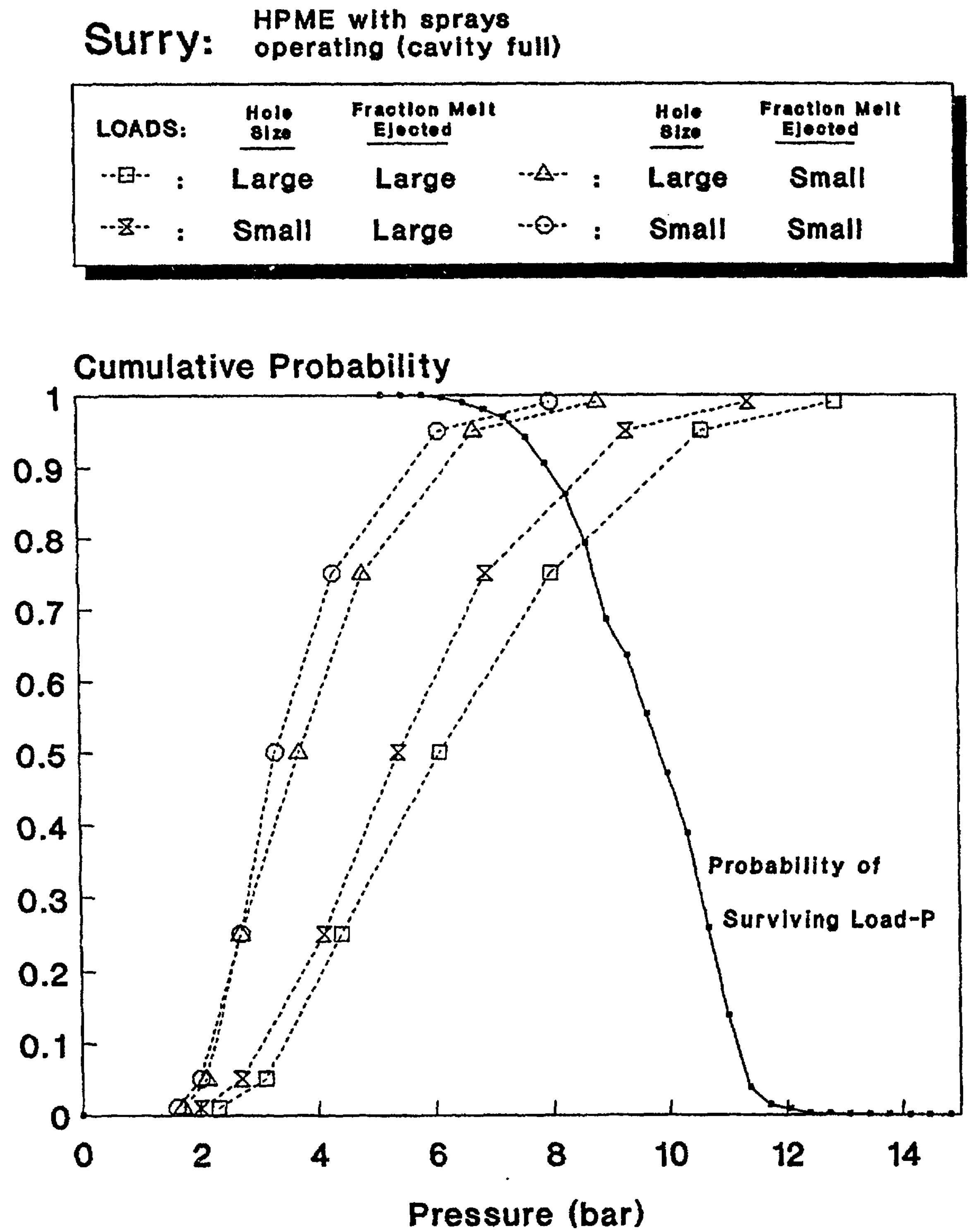


Figure C.5.5 Surry containment loads at vessel breach; cases involving vessel breach at high pressure with containment sprays operating (wet cavity).

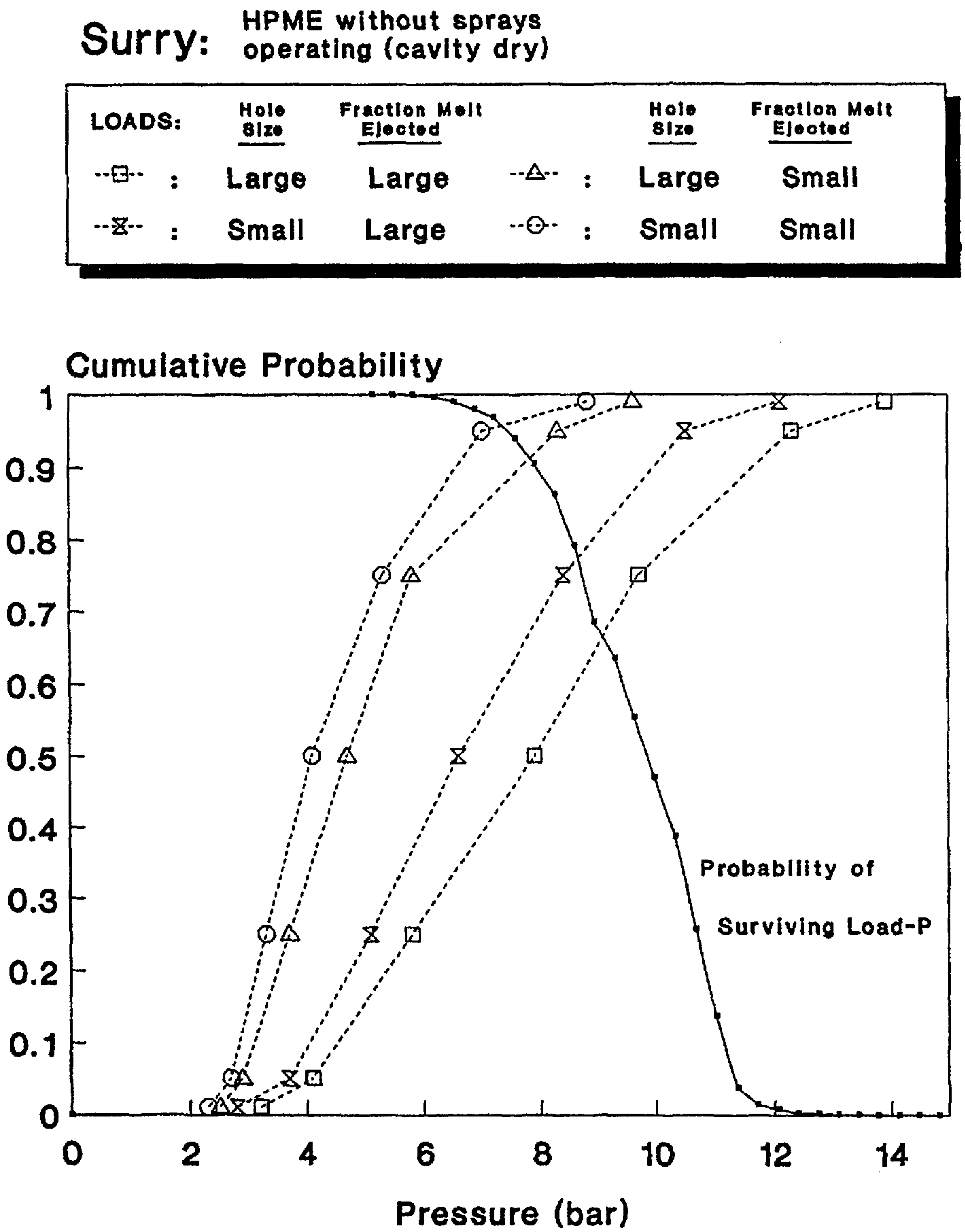


Figure C.5.6 Surry containment loads at vessel breach; cases involving vessel breach at high pressure without containment sprays operating (dry cavity).

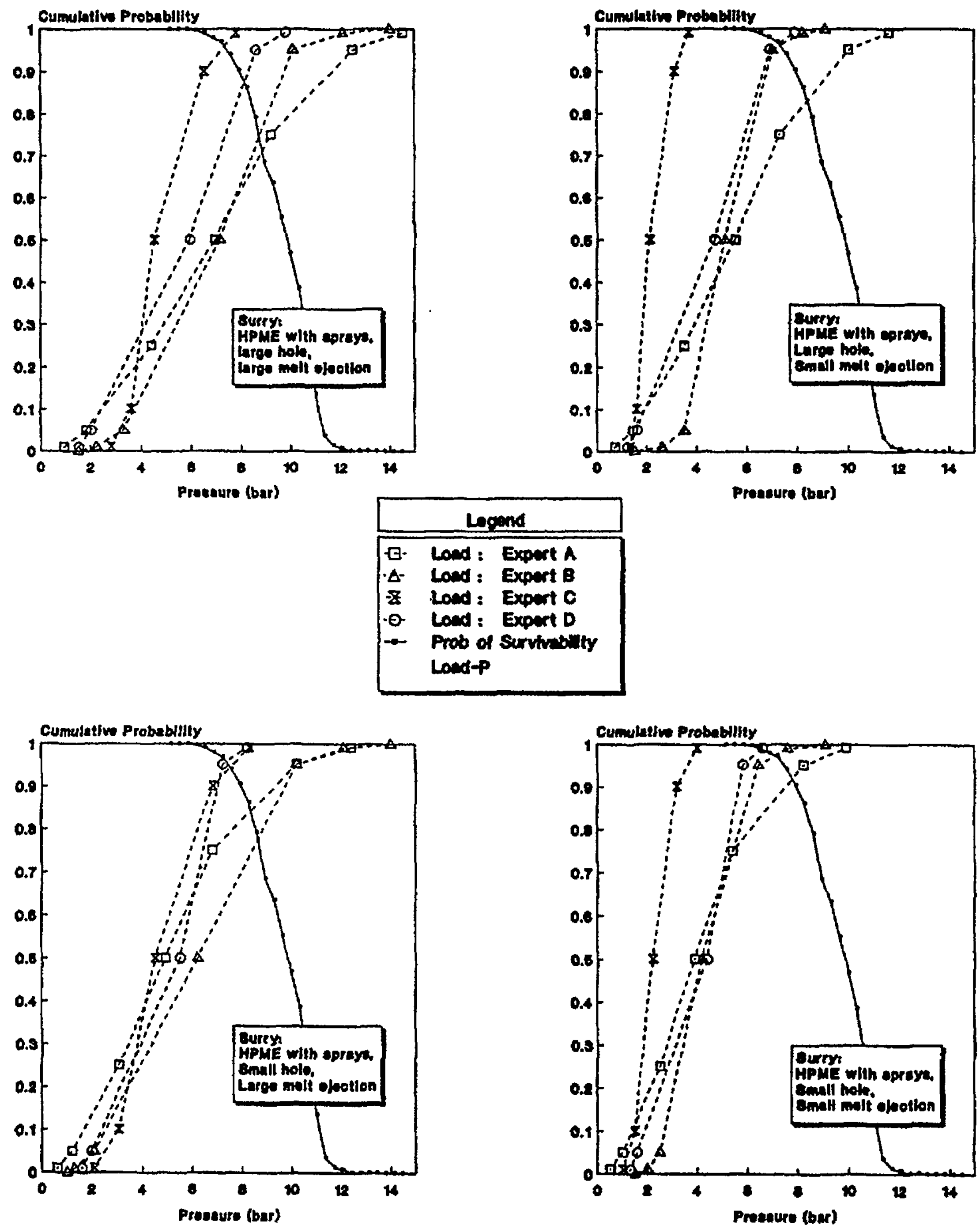


Figure C.5.7 Surry containment load distributions generated by composite of individual experts for each of the cases shown in Figure C.5.5.

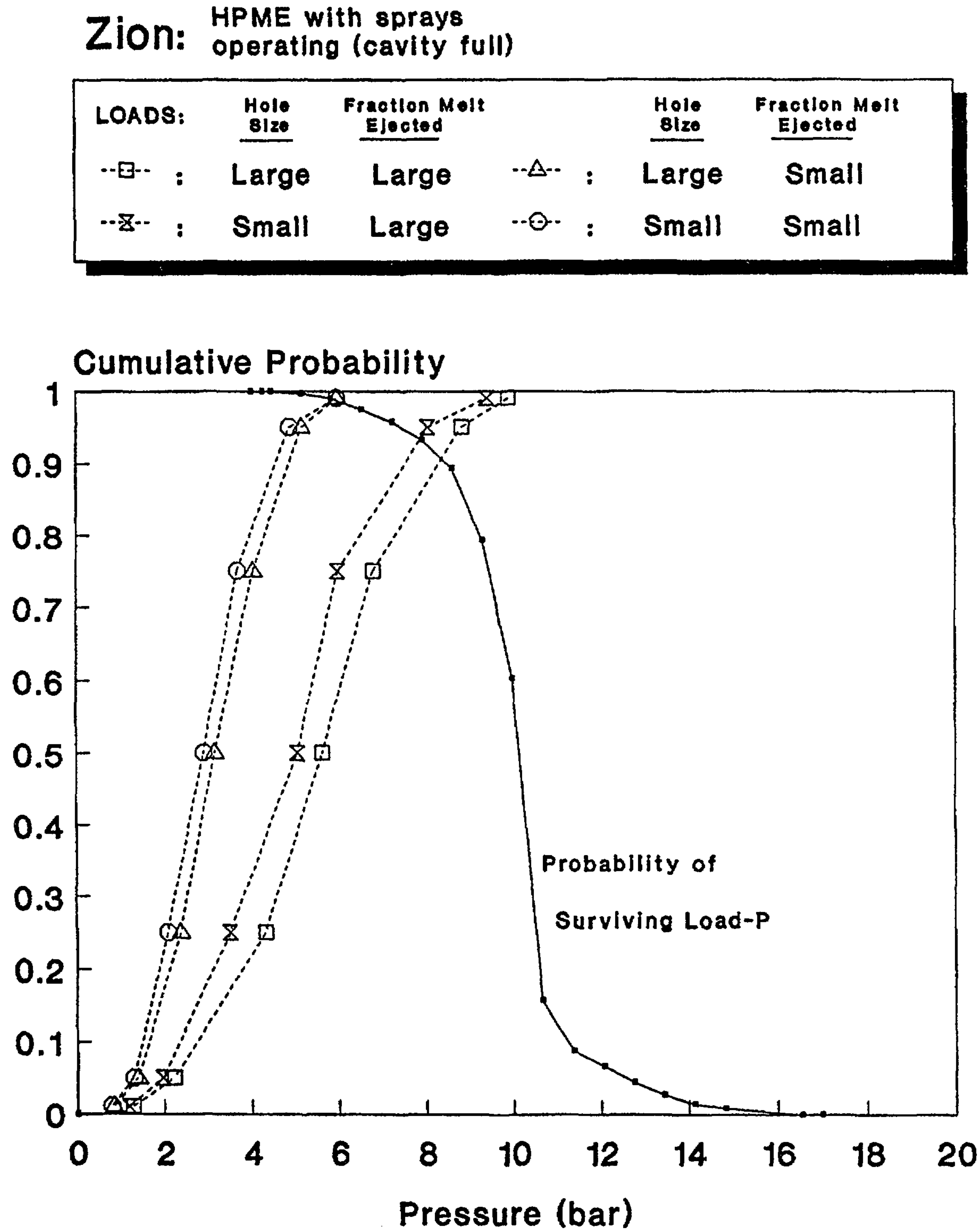


Figure C.5.8 Zion containment loads at vessel breach; cases involving vessel breach at high pressure with containment sprays operating (wet cavity).

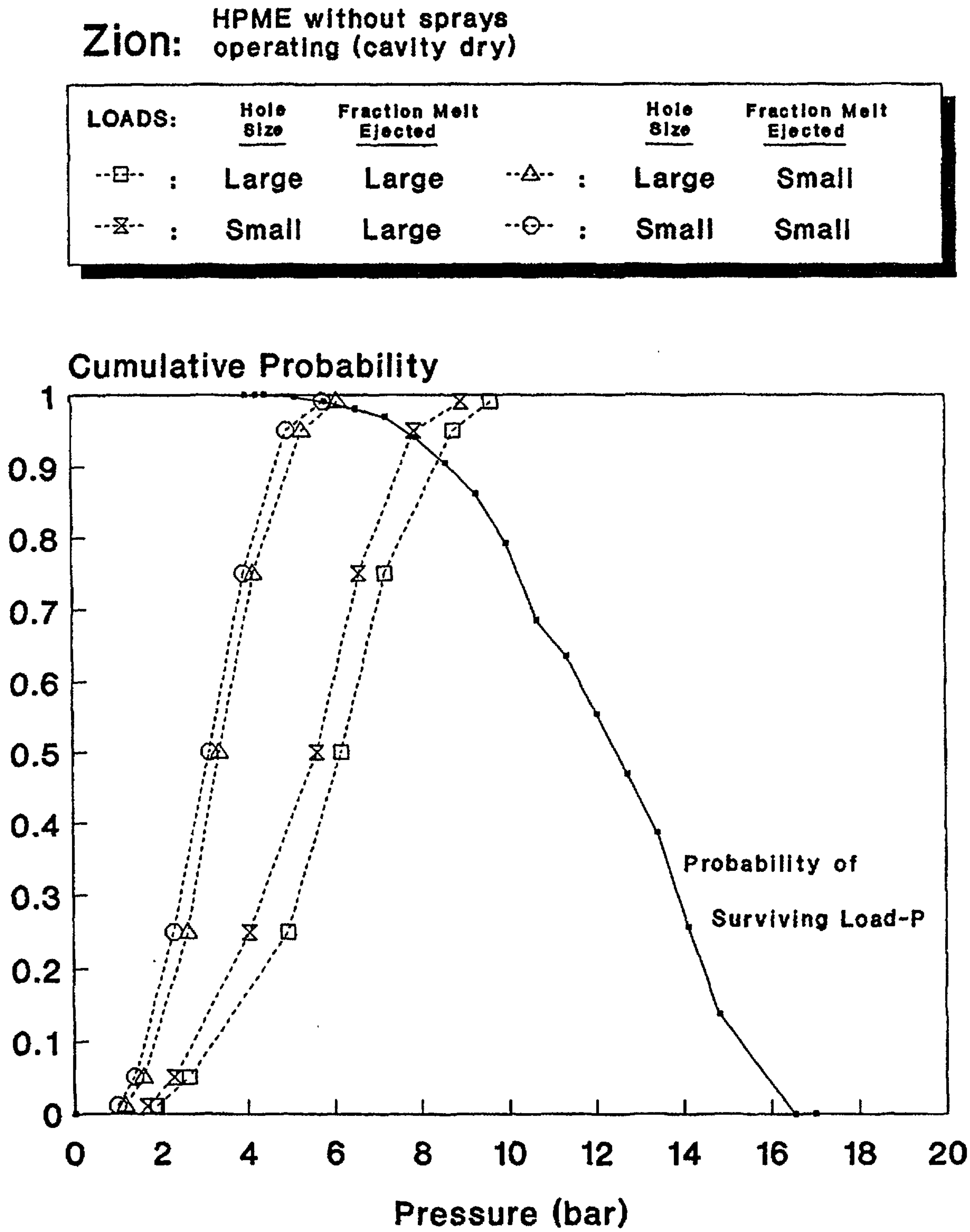


Figure C.5.9 Zion containment loads at vessel breach; cases involving vessel breach at high pressure without containment sprays operating (dry cavity).

shown in Figures C.5.10 and C.5.11 (i.e., they represent the loads for high-pressure accident scenarios with and without containment sprays operating, respectively). Note that, for cases with a wet reactor cavity,* the distributions for containment pressure rise were observed to be relatively insensitive to the assumed size of the hole generated in the reactor vessel bottom head at vessel breach. Separate distributions are, therefore, not displayed in Figure C.5.10 for cases with different assumed hole sizes. At Sequoyah, there are accident progressions when the reactor cavity is deeply flooded** (water level is well above the bottom of the reactor vessel, attaining a level as high as the hot legs). The expert judgment concerning this plausible situation is that pressure rise attendant to HPME is substantially mitigated. Containment loads for the deeply flooded cases were assessed separately from the dry and wet cavity cases and, because the threat to containment integrity is minimal, they are not presented here. The reader is encouraged to consult Reference C.5.2 for full details of the containment loads for the cases with a deeply flooded cavity.

In Figures C.5.10 and C.5.11, the load distributions represent accident situations in which a substantial fraction (greater than 50 percent) of the initial inventory of ice remains in the ice condenser at vessel breach. Such conditions may arise during small-break LOCAs or station blackout. Other accident scenarios, however, may result in substantial ice depletion prior to reactor vessel breach (such as small-break LOCAs with failure of ECCS in the recirculation mode). Representative distributions of containment loads at vessel breach for these cases are shown in Figure C.5.12.

The value of the ice condenser for containment pressure suppression is readily apparent when comparing the distributions in Figure C.5.12 with those in Figures C.5.10 and C.5.11. The Sequoyah containment is considerably more likely to survive the static pressure loads generated at vessel breach if a substantial quantity of ice (i.e., greater than 10 percent of the initial inventory) remains in the ice condenser than when the ice inventory is depleted. The influence of containment spray operation on containment performance is noticeable, but far less dramatic.

C.5.3 Treatment in PRA and Results

The probability distributions for this issue were implemented in the PWR accident progression event trees. These trees (one for each plant) provide a structured approach for evaluating the various ways in which a severe accident can progress, including important aspects of reactor coolant system thermal-hydraulic response, core melt behavior, and containment loads and performance. The accident progression event tree for each plant is a key element in the assessment of uncertainties in risk; it considers the possibility that a particular accident sequence may proceed along any one of several alternative pathways (i.e., alternative combinations of events in the severe accident progression). The probability distributions for individual and combinations of events within the tree provide the rules that determine the relative likelihood of various modes of containment failure.

As mentioned in the introduction to Section C.5, uncertainties in containment loads accompanying high-pressure melt ejection are not major contributors to the overall uncertainty in risk for any of the three PWRs examined in this study. There are two reasons for this. First, comparison of the range of potential loads against the estimated strength of the large, dry containments (Surry and Zion) indicates high confidence that these containments can accommodate the pressure increment accompanying high-pressure melt ejection. A similar conclusion cannot be supported for the Sequoyah containment without additional assurance that some of the containment safety features operate (e.g., a substantial inventory of ice remains at the time of vessel breach). Secondly, accident sequences that have traditionally been considered as "high-pressure" core meltdown accidents (e.g., a fast station blackout***) are estimated to result in a depressurized reactor vessel by the time of reactor vessel breach with a relatively high frequency. Depressurization mechanisms considered in the present analysis include temperature-induced hot leg failure and steam generator tube ruptures, reactor coolant pump seal

*For substantial quantities of water to accumulate on the containment floor and overflow into the Sequoyah cavity, the refueling water storge tank (RWST) inventory must dump onto the containment floor (e.g., via containment sprays) and approximately 25 percent of the ice inventory must melt.

**Deep flooding of the cavity occurs with approximately 50 percent of the ice inventory and transfer of the RWST inventory onto the containment floor.

***A fast station blackout involves the loss of electrical power and failure of steam-driven auxiliary feedwater, thus rendering all decay heat removal systems unavailable.

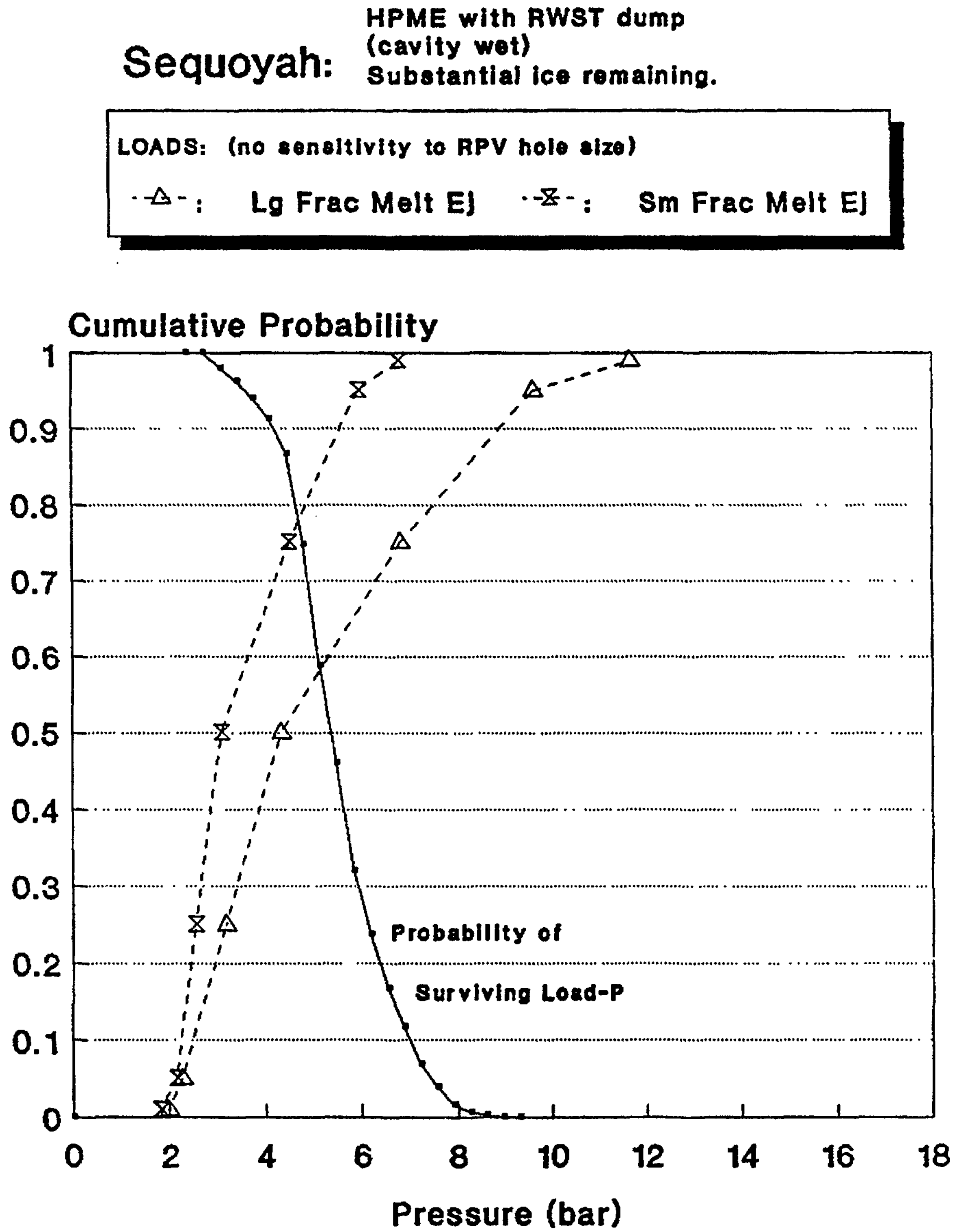


Figure C.5.10 Sequoyah containment loads at vessel breach; cases involving vessel breach at high pressure with containment sprays operating (wet cavity) and a substantial inventory of ice remaining.

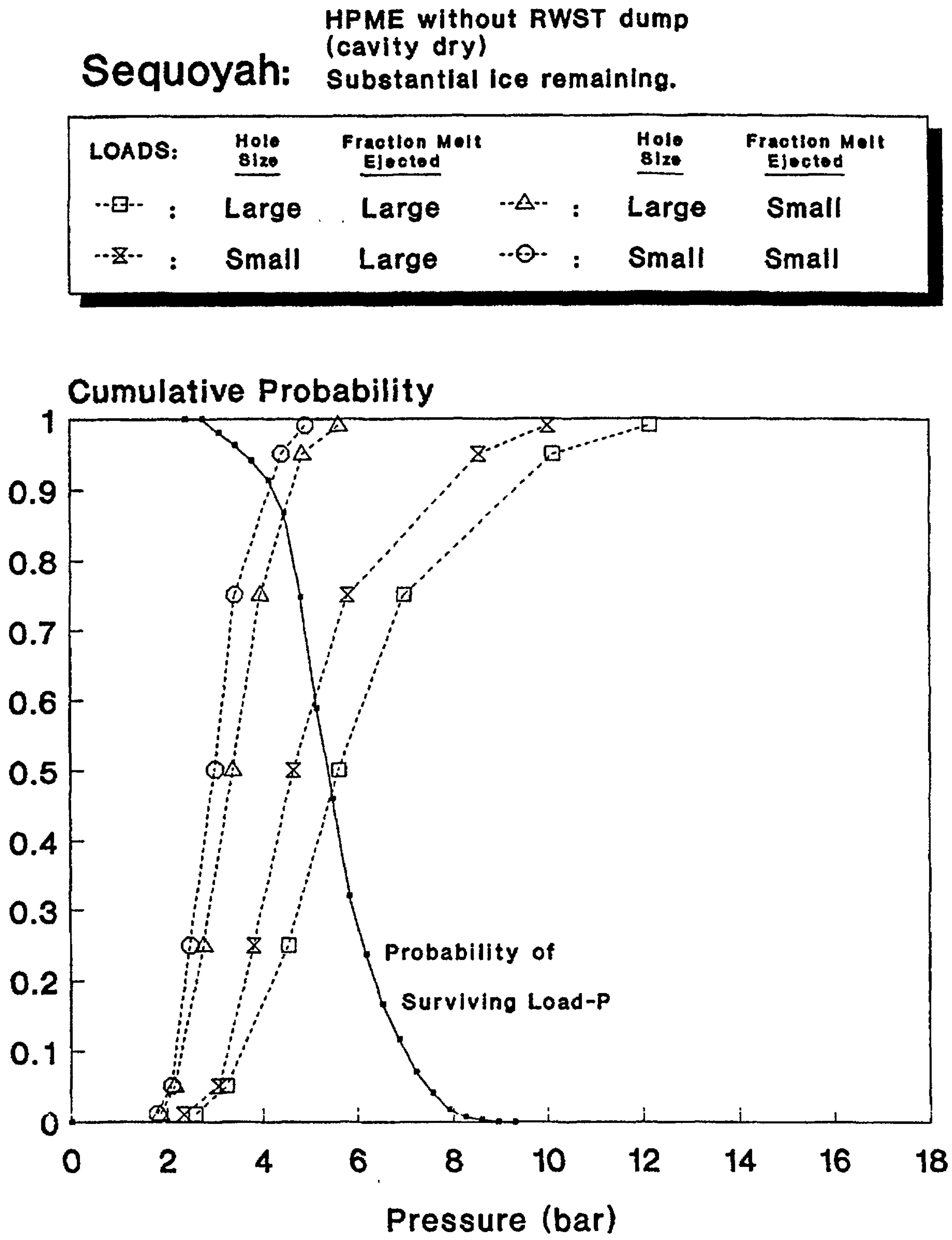


Figure C.5.11 Sequoyah containment loads at vessel breach; cases involving vessel breach at high pressure without containment sprays operating (dry cavity) and a substantial inventory of ice remaining.

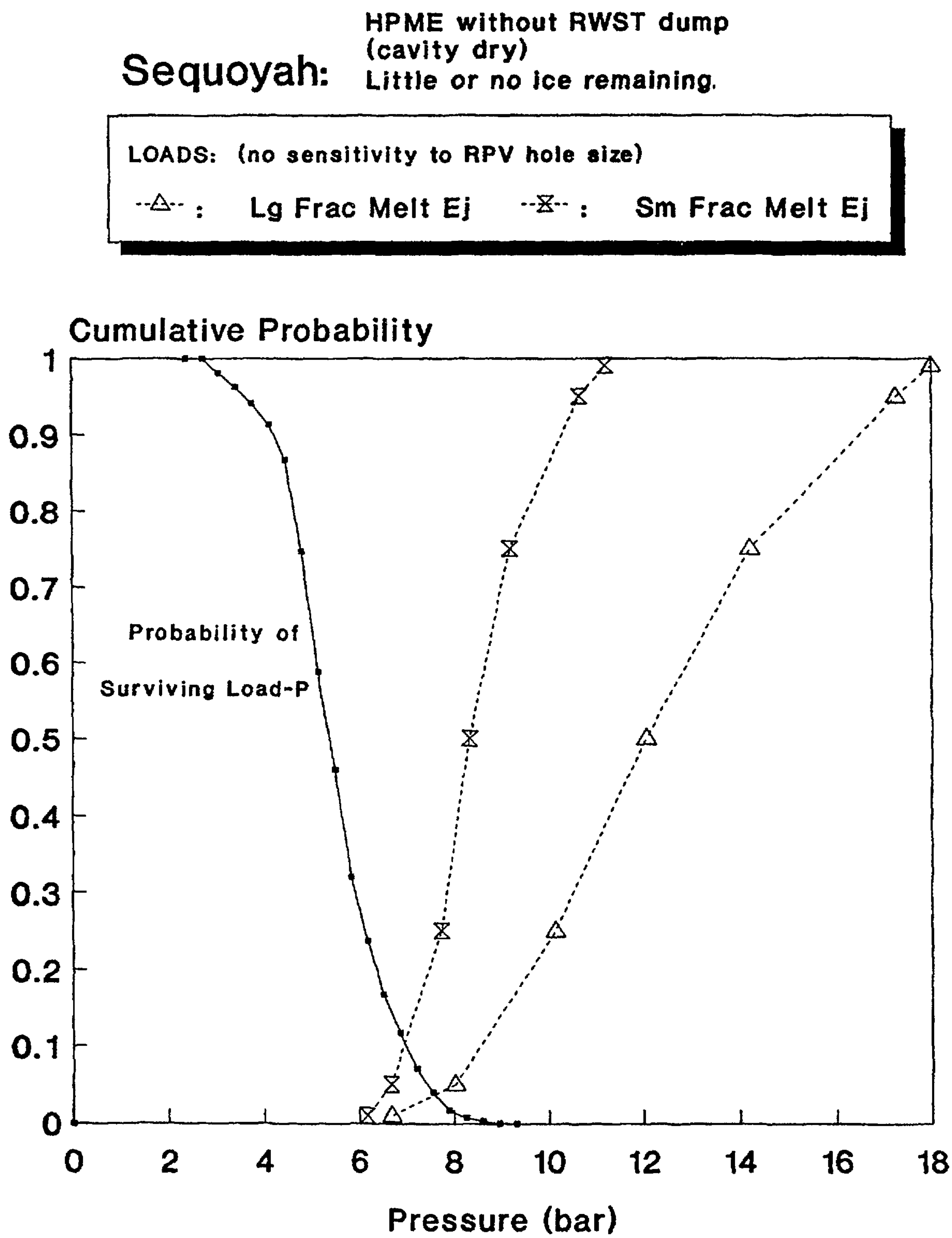


Figure C.5.12 Sequoyah containment loads at vessel breach; cases involving vessel breach at high pressure without containment sprays operating (dry cavity) and a negligibly small inventory of ice remaining.

failures, and stuck-open power-operated relief valves (PORVs). These mechanisms are described in detail in Section C.6. The result of incorporating the potential for reactor vessel depressurization prior to vessel breach is a reduced frequency of high-pressure melt ejection and reduced containment loads at vessel breach. Another potential means of mitigating HPME loads at Sequoyah is deep flooding of the reactor cavity. However, deep flooding introduces a potential for an ex-vessel steam explosion. Challenges to containment integrity from ex-vessel steam explosions are discussed in Section C.9.

As an illustration of the reduced frequency of high-pressure melt ejection and a resulting reduced frequency of early containment failure in the present analysis (from that estimated in the preliminary analyses—published in the February 1987 draft for comment release of this report (Ref. C.5.14)), Table C.5.1 summarizes the relative likelihood of various modes of containment failure for each of the PWRs examined. The numbers shown in this table are frequency-weighted averages (i.e., they are the mean probability of containment failure given core damage). It is important to note that the probability of no containment failure is significant, and the average probability of early containment failure is shown to be low for all three plants.

Table C.5.1 Mean conditional probability of containment failure for three PWRs.

Containment Failure Mode	Surry	Zion	Sequoyah
Early failure with reactor vessel at pressure > 200 psi	0.004	0.02	0.04
Early failure with reactor vessel at pressure < 200 psi	0.0	—	0.02
Late containment failure	<0.01	—	0.04
Containment bypass	0.12	0.006	0.06
Others (alpha,* basemat meltthrough)	0.06	0.22	0.18
No containment failure or arrested core damage with no vessel breach	0.81	0.76	0.66

*Steam explosion-induced containment failure. The analyses supporting the quantification of this mode of containment failure are described in Section C.9.

C.5.4 Differences in Treatment of HPME and DCH Between First and Second Drafts of NUREG-1150

There are important differences in the role played by high-pressure melt ejection/direct containment heating (HPME/DCH) in the second and final versions of NUREG-1150 versus its role in the first draft. In the latter, DCH contributed about 80 percent to the mean early fatality risk at Surry, while the contribution is substantially less in the current version of NUREG-1150, about 17 percent. Similar trends resulted for Zion. For Sequoyah, however, the change was in the opposite direction: HPME/DCH was a very minor contributor in the first draft of NUREG-1150, while it is significant in the current version of NUREG-1150.

Some of the implications of NUREG-1150 may be understood by first examining the reasons for the changes between the final version of NUREG-1150 and the first draft of NUREG-1150 results. A large number of factors are involved, with the following being especially noteworthy:

1. In the final version of NUREG-1150, much higher probabilities were assigned for partial or total depressurization of the vessel prior to vessel breach (VB) because of the occurrence of induced failures of the RCS boundary: hot leg and surge line LOCAs, pump seal LOCAs, and failure of PORVs to reclose. In the Surry analysis for the first draft, for example, the vessel remains pressurized in about 66 percent of the TMLB' accidents, while partial or complete depressurization occurs in 97 percent of the Surry TMLB' accidents in the final version.

2. The final version of NUREG-1150 takes credit for offsite power recovery during the period between the onset of core degradation and the occurrence of VB in station blackout accidents, which are the

principal accident sequences leading to HPME/DCH. In Surry, for example, recovery occurs in 60 percent of the cases, and recovery arrests the accident (preventing HPME/DCH) in 90 percent of the recovered cases. The PWR analyses in the first draft of NUREG-1150 did not consider core damage arrest between onset of core degradation and VB.

3. The high end of the distribution for HPME/DCH loads in large, dry containments was reduced somewhat, relative to the first draft of NUREG-1150, but these changes were not large; containment-threatening loads were still considered quite credible. (In Surry, there was also an increased estimate of containment strength.) In ice condenser plants, however, the assessed threat was substantially increased, both because of increased appreciation of the potential role played by hydrogen in augmenting the DCH threat in these plants and also because of the recognition of the possibility of containment failure due to direct impingement of melt on the containment shell.

4. The ultimate strengths of the containment building under severe accident loads increased somewhat.

5. In the final version of NUREG-1150, steam generator tube ruptures (SGTRs) were included as accident initiators, and they contribute significantly to the role of bypass accidents for some consequence measures (their contribution is minor for early fatality risks, however). The first draft of NUREG-1150 did not consider SGTRs as initiators for severe accidents.

It should be noted that the changes summarized in points 1 and 3 above were heavily influenced by results from the Severe Accident Research Program during the interim period between the first and second drafts of NUREG-1150. Review of the NUREG-1150 expert elicitation documentation (Ref. C.5.2) shows that these research results played an important role in guiding the uncertainty distributions supplied for many important parameters, including both those governing DCH loads and those governing the probability of RCS depressurization.

The results noted above indicate that the most important single reason for the reduced contribution of HPME/DCH to mean risk in the large, dry containments is the perception that HPME/DCH scenarios are more likely to be prevented by the occurrence of unintentional RCS depressurization associated with induced failures of the RCS boundary and, to a lesser extent, by power recovery. In Sequoyah, this effect is more than compensated for by the increase in perceived threat from HPME/DCH phenomenology itself.

The degree to which the reduced importance of HPME/DCH in the Surry analysis depends upon RCS depressurization is especially noteworthy because this conclusion hinges upon such uncontrollable and unplanned factors as the temperature-induced failure of RCS components. This behavior is highly dependent on code predictions of core melt progression and the response of the RCS during degraded core accidents; the predictions have not been validated experimentally. The TMI-2 accident is also interesting in this regard because it provides evidence that degraded core accidents can progress quite far at elevated pressure without approaching temperature-induced failure of RCS components.

REFERENCES FOR SECTION C.5

C.5.1 E.D. Gorham-Bergeron et al., "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration, and Uncertainty Analyses," Sandia National Laboratories, NUREG/CR-4551, Vol. 1, Draft Revision 1, SAND86-1309, to be published.*

C.5.2 F.T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

C.5.3 R.J. Breeding et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, October 1990.

C.5.4 A.C. Payne, Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft Revision 1, SAND86-1309, to be published.*

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

C.5.5 J.J. Gregory et al., "Evaluation of Severe Accident Risks: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, December 1990.

C.5.6 T.D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 6, Draft Revision 1, SAND86-1309, to be published.*

C.5.7 C.K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

C.5.8 D.C. Williams et al., "Containment Loads Due to Direct Containment Heating and Associated Hydrogen Behavior: Analysis and Calculations with the CONTAIN Code," Sandia National Laboratories, NUREG/CR-4896, SAND87-0633, May 1987.

C.5.9 M. Pilch and W.W. Tarbell, "Preliminary Calculations of Direct Heating of a Containment Atmosphere by Airborne Core Debris," Sandia National Laboratories, NUREG/CR-4455, SAND85-2439, July 1986.

C.5.10 W.W. Tarbell et al., "Results from the DCH-1 Experiment," Sandia National Laboratories, NUREG/CR-4871, SAND86-2483, June 1987.

C.5.11 M. Pilch et al., "High Pressure Melt Ejection and Direct Containment Heating in Ice Condenser Containments," *Proceedings of the International Topical Meeting on Operability of Nuclear Power Systems in Normal and Adverse Environments, ANS/ENS* (Albuquerque, NM), September 29-October 3, 1986, SAND87-2141C.

C.5.12 M. Pilch and W.W. Tarbell, "High Pressure Ejection of Melt from a Reactor Pressure Vessel—The Discharge Phase," Sandia National Laboratories, NUREG/CR-4383, SAND85-0012, September 1985.

C.5.13 W.W. Tarbell et al., "Pressurized Melt Ejection into Scaled Reactor Cavities," Sandia National Laboratories, NUREG/CR-4512, SAND86-0153, October 1986.

C.5.14 U.S. Nuclear Regulatory Commission, "Reactor Risk Reference Document," NUREG-1150, Vols. 1-3, Draft for Comment, February 1987.

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

C.6 Mechanisms for PWR Reactor Vessel Depressurization Prior to Vessel Breach

The previous section addressed the range of thermodynamic loads to a PWR containment accompanying penetration of the reactor pressure vessel lower head by molten core debris and subsequent ejection of material into the containment atmosphere. These loads can present a significant challenge to containment integrity if penetration of the reactor vessel occurs at sufficiently high vessel pressure. For the three PWRs examined in this study, however, a substantial fraction of the severe accident progressions that started with the reactor vessel at high pressure depressurized before vessel breach. That is, many of the accident scenarios important to risk result in—by one means or another—a breach in the reactor coolant system (RCS) pressure boundary of sufficient size to reduce reactor vessel pressure below approximately 200 psi before reactor vessel lower head failure. An outcome of this result is that the uncertainties in high-pressure melt ejection loads are observed to have a relatively small impact on the overall uncertainties in reactor risk. This observation is a substantial change in results from those of preliminary analyses published in draft form in February 1987.

Unlike the BWRs examined in this study, the PWRs do not have a system specifically designed to manually depressurize the reactor vessel. Feed-and-bleed operations can effect limited depressurization if the necessary systems are operable. Many of the accident sequences leading to core damage in the three PWRs examined in this study, however, include combinations of failures that render feed-and-bleed operations unavailable. This section addresses the other means by which the reactor vessel pressure may be reduced to levels below which high-pressure melt ejection loads do not threaten containment integrity:

- Temperature-induced failure of steam generator tubes,
- Temperature-induced failure of primary coolant hot leg piping or the pressurizer surge line,
- Failure of reactor coolant pump seals,
- Stuck-open power-operated relief valves (PORVs), and
- Manual (operator) actions to depressurize the RCS.

The estimated frequency of each of these events and their influence on reactor vessel pressure was incorporated in the accident progression analysis for the Surry, Sequoyah, and Zion plants. Manual depressurization was found to be ineffective for most PWR accident sequences because of limitations in the appropriate emergency procedures and the need for ac power to operate relief valves. This mechanism is, therefore, not discussed further. The manner in which the other hypothetical events were considered, the means of quantifying their likelihood, and illustrations of the impact they have on the results are discussed in the following sections.

C.6.1 Issue Definition

The general issue is the frequency with which PWR severe accident progressions involve a breach in the RCS pressure boundary of sufficient size to reduce the reactor vessel pressure below approximately 200 psia. The mechanisms for depressurizing the reactor vessel that are considered in the present analysis are those listed in the introduction above. The first two mechanisms involve temperature-induced (i.e., creep rupture) failures of RCS piping. In both cases, the heat source for such failures is hot gases transported from the core via natural circulation or exiting the RCS through the PORV. The natural circulation pattern may involve an entire RCS coolant loop if water in the loop seals has cleared. If the loop seals have not cleared, a countercurrent natural circulation flow pattern may be established within the hot leg piping, transporting superheated gases and radionuclides from the core region of the reactor vessel to the steam generators. Effective cooling of the steam generator tubes is not available in many of the accident sequences considered in this analysis because of depletion of secondary coolant inventory earlier in the accident. Decay heat from radionuclides deposited in the steam generator inlet plenum and inside the tubes may also contribute to local tube heating. In either case, natural circulation flow (if established) may be interrupted by the frequent cycling of the pressurizer PORV or by the accumulation (and stratification) of hydrogen in the reactor vessel upper plenum and hot legs. The specific parameter to be quantified is the frequency with which creep rupture of hot leg piping or steam generator tubes results from the transfer of heat from the core (via gas circulation) to RCS structures. The temperature-induced failures of interest here are limited to those that occur before reactor vessel failure.

Degradation and failure of reactor coolant pump seals may also result from overheating. In this case, overheating results from the loss of seal cooling water flow or loss of heat removal from the seal cooling water system. A number of potential "seal states" have been identified in reactor coolant pump performance studies, which result in a range of plausible leak rates from the reactor coolant system. The parameters to be quantified are the frequency of pump seal LOCAs, the relative likelihood of various leak rates that result from these failures, and the resulting value of reactor vessel pressure at the time of vessel breach.

The fourth mechanism considered in this analysis, stuck-open PORV(s), may result following the repeated cycling (opening and reseating) of the PORVs during the course of an accident. Such events have been observed (with relatively low frequency) during transient events in which plant conditions never exceed design basis conditions. PORVs have also been tested for their reliability to close after repeated cycles at design basis conditions. This issue considers the effect of beyond design basis conditions on the frequency with which PORVs fail to close after several cycles.

C.6.2 Technical Bases for Issue Quantification

Two of the four mechanisms, temperature-induced hot leg failure and steam generator tube ruptures, were presented to a panel of experienced severe accident analysts. Each panelist was asked to provide a probability distribution representing his estimate of the frequency of each event. Their judgments were to be based on current information, made available to each of the panelists, and their own professional experience. The panelists participating were:

Vernon Denny—Science Applications International Corp.,
Robert Lutz—Westinghouse Electric Corp., and
Robert Wright—U.S. Nuclear Regulatory Commission.

The individual distributions prepared by these panelists were then combined (i.e., an aggregate distribution was generated by averaging those of the three panelists) to develop a single distribution for application in the PRA. The methods used to aggregate individual panelists' distributions are described in Reference C.6.1.

The frequency of reactor coolant pump seal failures was addressed by an expert panel in support of the systems analysis for the PWRs (Ref. C.6.2). This panel's judgments were adopted for use in the accident progression event tree. Very limited data are available to support an assessment of the frequency of PORVs sticking open when subjected to severe accident conditions. A broad distribution was, therefore, assigned to the frequency of stuck-open PORVs. A summary of the technical bases for quantifying the frequency of RCS depressurization for each of these four mechanisms is given below.

Frequency of Hot Leg Failure

A case structure was established to consider a spectrum of plausible severe accident conditions for which the frequency of hot leg failures needed to be quantified. The case structure was formulated around accident sequences that represent a significant contribution to the total core damage frequency. The cases considered were:

Case 1: A classic TMLB'* scenario (station blackout). RCS pressure is maintained near 2500 psia by the continuous cycling of the PORV. The secondary side of the steam generator is at the steam relief valve setpoint pressure (approx. 1000 psia) and is depleted of coolant inventory. Reactor coolant pump seal cooling is maintained at the nominal flow rate.

Case 2: Station blackout sequence during which reactor pump coolant seals fail, yielding a leak rate equivalent to a 0.5-inch-diameter break in each coolant loop. The steam generator secondary coolant inventory is depleted and the auxiliary feedwater system is unavailable.

Case 3: Same as Case 2 except the steam generators maintain an effective RCS heat sink with auxiliary feedwater operating.

*Reactor Safety Study [WASH-1400] nomenclature for accident sequence delineation. The alphabetical characters represent compound failures of plant equipment leading to the loss of plant safety functions. The characters TMLB' represent a transient initiating event, loss of decay heat removal, and loss of all electrical power.

The technical bases used by the panelists for characterizing the frequency of temperature-induced hot leg failures for each case were dominated by calculations performed with various severe accident analysis computer codes and by several different organizations. Those cited by the panelists in their elicitations (Ref. C.6.1) included TRAC/MELPROG calculations of TMLB' scenarios in Surry (Ref. C.6.3), RELAP5/SCDAP calculations of similar accident scenarios (Ref. C.6.4), CORMLT/PSAAC calculations for Surry and Zion (Refs. C.6.5 and C.6.6), and MAAP calculations performed in support of the Ringhals Unit 3 PRA (Ref. C.6.7) and the Seabrook PSA (Refs. C.6.8 and C.6.9). Ringhals Unit 3 is a three-loop plant with an NSSS similar to that of Surry; Seabrook is a four-loop plant with an NSSS similar to those of Sequoyah and Zion.

Only two specific references were cited by the panelists regarding experimental data or other physical evidence of natural circulation and its effect on heating RCS structures. These were the natural circulation experiments sponsored by EPRI (Ref. C.6.10) and the results of post-accident examinations of the Three Mile Island Unit 2 core debris and RCS structures (Ref. C.6.11). Information from neither of these sources is believed to have significantly influenced the panelists' judgments on this issue.

The aggregate distribution for the frequency of temperature-induced hot leg failures are shown in Figure C.6.1 for Cases 1 and 2 outlined above. The probability that Case 3 would result in an induced hot leg failure was judged to be essentially zero. The distributions shown in Figure C.6.1 are displayed in the form of a cumulative distribution function (CDF); that is, the curve displays the probability that the frequency of an induced hot leg failure is not greater than a particular value. The likelihood of an induced hot leg failure, given a station blackout accident during which the reactor vessel pressure remains high (i.e., no reactor coolant pump seal LOCAs, stuck-open PORVs, etc.), is shown to be relatively high; the median frequency is greater than 95 percent. In contrast, lower reactor vessel pressures in Case 2 (with an early pump seal LOCA) make an induced hot leg failure unlikely; there is an 83 percent chance that a hot leg failure will not occur.

Frequency of Induced Steam Generator Tube Ruptures

Essentially the same information (results of several computer code calculations) were used to characterize induced steam generator tube rupture (SGTR) frequency. All three panelists agreed that the likelihood of an induced SGTR is quite low. The three panelists noted that temperature-induced tube ruptures are driven by the same phenomena that drive temperature-induced hot leg failure (natural circulation flow of hot gases from the reactor vessel); therefore, the frequency distributions are correlated. Two of the panelists believed that the frequency of SGTR is very small because of the assumption that the hot leg would fail first, and neither of their distributions for frequency of induced SGTR exceeded a value of 0.0005. The aggregate distribution (shown in Fig. C.6.2) is dominated by a single panelist, whose distribution was strongly influenced by consideration of pre-existing flaws in steam generator tubes, resulting in the assumption that SGTR might occur before hot leg failure.

Frequency of Induced Reactor Coolant Pump Seal LOCAs

The frequency of pump seal LOCAs of various sizes (corresponding to various pump seal states) was considered by a panel of experts as a systems analysis issue. Degradation mechanisms for reactor coolant pumps are highly plant- (or pump-) specific and can be quite complicated. Details of the analyses leading to the characterization of the various pump seal states and the corresponding spectrum of possible leak rates are not provided here but are available in the documentation of the expert panel elicitations (Ref. C.6.2). An indication of the potential importance of modeling pump seal LOCAs, however, can be found by examining the accident progressions for which the reactor vessel pressure remains at or near the system setpoint (e.g., station blackouts with no other breach in the RCS pressure boundary). In the Surry analysis, approximately 71 percent of these accident progressions result in a failure of the seals in at least one reactor coolant pump. Of these, roughly one-third are estimated to result in a large enough leak rate to depressurize the reactor vessel to less than approximately 200 psia prior to reactor vessel breach; another third result in leak rates small enough to preclude any significant depressurization. In the remaining one-third of the cases, the reactor vessel is at intermediate pressure (200–600 psia) at the time of vessel breach (Ref. C.6.12).

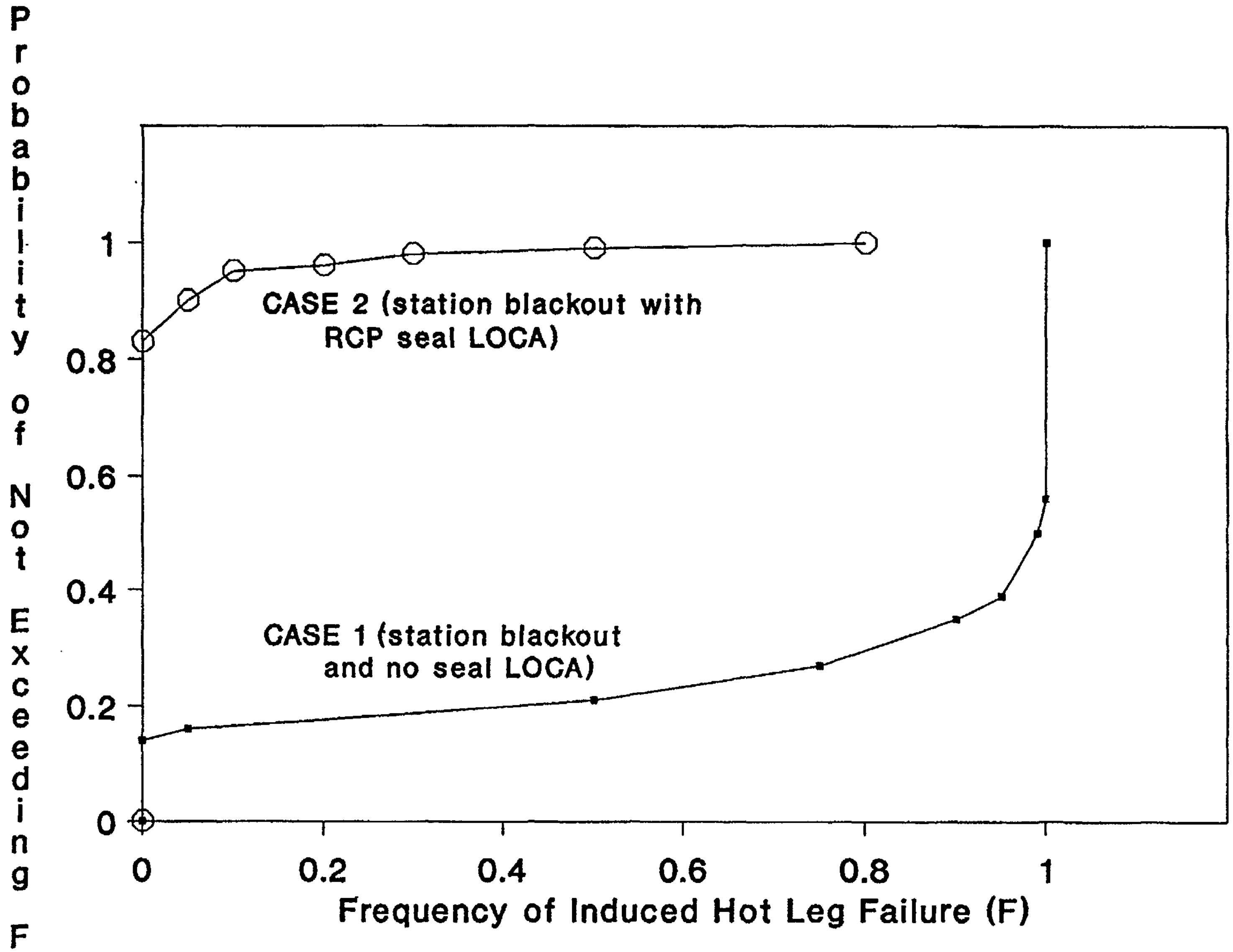


Figure C.6.1 Aggregate distribution for frequency of temperature-induced hot leg failure (Surry, Zion, and Sequoyah).

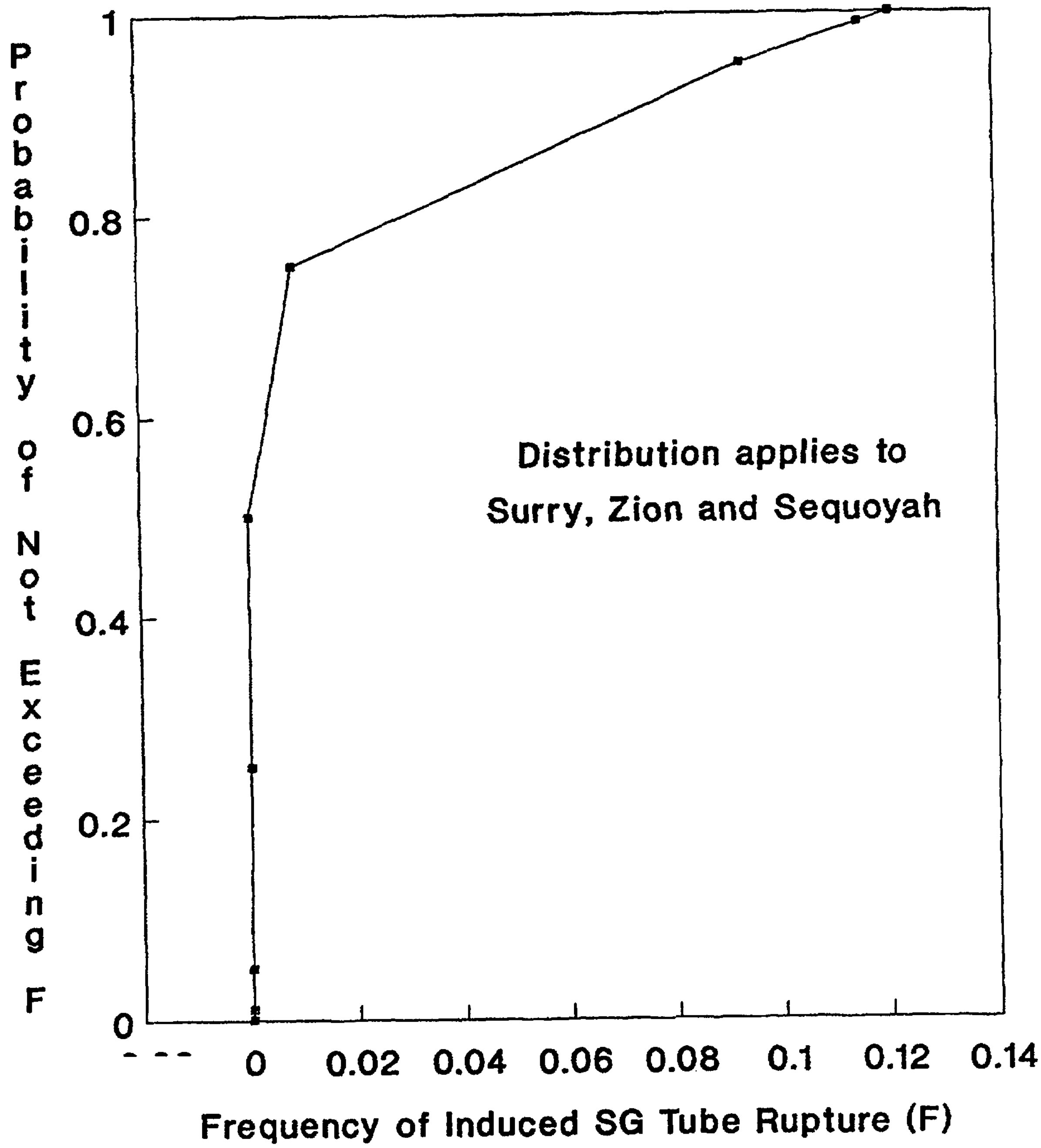


Figure C.6.2 Aggregate distributions for frequency of temperature-induced steam generator tube rupture.

Frequency of Stuck-Open PORVs

This issue was also addressed in the "front-end" analysis as an uncertainty issue (Ref. C.6.2). The RCS conditions under which PORVs will cycle after the onset of core damage, however, are expected to be significantly more severe than those for which the valves were designed and more severe than the conditions under which PORV performance has been tested. In lieu of specific analyses, test data, or operating experience, an estimate of frequency with which a PORV will stick open and an estimate for the resulting RCS pressure were generated as follows:

> The valve is expected to cycle between 10 to 50 times during core degradation and prior to vessel breach. Extrapolation of the distributions for the frequency of PORV failure-to-close from the front-end elicitations indicates an overall failure rate (for 10 to 50 demands) in the neighborhood of 0.1 to 1.0. A uniform distribution from zero to 1.0 was, therefore, used in the Surry and Sequoyah analyses.
>
> TRAC/MELPROG and Source Term Code Package (STCP) analyses were reviewed to characterize the rate at which a stuck-open PORV could depressurize the reactor vessel (Ref. C.6.12). The results of this review resulted in an estimate that there is an 80 percent probability that the reactor vessel pressure at the time of vessel breach will be less than 200 psia; in the remaining 20 percent of the cases, the vessel pressure will be at intermediate levels (200–600 psia).

C.6.3 Treatment in PRA and Results

The probability distributions for this issue were implemented in the PWR accident progression event trees. These trees (one for each plant) provide a structured approach for evaluating the various ways in which a severe accident can progress, including important aspects of RCS thermal-hydraulic response, core melt behavior, and containment loads and performance. The accident progression event tree for each plant is a key element in the assessment of uncertainties in risk; it considers the possibility that a particular accident sequence may proceed along any one of several alternative pathways (i.e., alternative combinations of events in the severe accident progression). The probability distributions for individual and combinations of events within the tree provide the rules that determine the relative likelihood of various modes of containment failure.

For the issue of reactor vessel depressurization, probability distributions for each of the mechanisms discussed above were incorporated in the accident progression event tree to determine reactor vessel pressure prior to vessel breach. As indicated in Section C.5, the containment loads accompanying vessel breach strongly depend on reactor vessel pressure. The load at vessel breach assigned to a particular accident progression, therefore, depends on the outcome of questions in the tree regarding reactor vessel depressurization. Selected results from the accident progression event tree analysis are summarized below.

The pressure history (as determined by the Surry accident progression event tree) for slow station blackout accident sequences* is summarized in Table C.6.1. This table shows the fraction of slow station blackout accident progressions for which the RCS pressure is at the PORV setpoint at high, intermediate, and low levels at the time the core uncovers and the time of reactor vessel breach.

A substantial fraction of the slow blackout accident progressions that start out with the RCS pressure at the PORV setpoint pressure are depressurized by one (or more) of the mechanisms described in Reference C.6.1 and result in a low pressure by the time of vessel breach.

A sensitivity study was performed to examine the effect of neglecting temperature-induced hot leg failure and steam generator tube ruptures on the observed results. Table C.6.2 summarizes the results of this study (presented in an identical format as Table C.6.1).

The results for pressure when the core uncovers are not affected by the change since temperature-induced hot leg failure and steam generator tube ruptures can only occur after the onset of core damage. The elimination of the possibility of these failures does affect the fraction of accident progressions involving reactor vessel breach at high pressure. The occurrence of high-pressure melt ejection is observed to roughly double in frequency.

*Slow station blackout accident sequences contribute more than one-half of the mean total core damage frequency for Surry. The results indicated for this group of accident sequences are not generally applicable to other Surry accident sequences or other plants.

Table C.6.1 **Surry reactor vessel pressure at time of core uncovery and at vessel breach.**

	Fraction of Slow Blackout Accident Progressions With Pressure-P at the Time of:	
RCS Pressure (psia)	**Core Uncovery**	**Reactor Vessel Breach**
2500	0.54	0.06
1000—1400	0.13	0.10
200—600	0.33	0.19
<200	0.0	0.65

Table C.6.2 **Surry reactor vessel pressure at time of core uncovery and at vessel breach (sensitivity study without induced hot leg failure and steam generator tube ruptures).**

	Fraction of Slow Blackout Accident Progressions With Pressure-P at the Time of:	
RCS Pressure (psia)	**Core Uncovery**	**Reactor Vessel Breach**
2500	0.54	0.25
1000—1400	0.13	0.10
200—600	0.33	0.19
<200	0.0	0.46

The increase in accident progressions resulting in vessel breach at high pressure is not observed to significantly affect the likelihood of early containment failure, however. Table C.6.3 shows the fraction of slow blackout accident progressions that results in various modes of containment failure (including no failure) for the Surry base case analysis and for the sensitivity analysis in which induced hot leg failures and steam generator tube ruptures were eliminated.

The insignificant change in results is largely attributable to the strength of the Surry containment and its ability to withstand loads as high as those estimated to accompany high-pressure melt ejection with a relatively high probability (refer to Section C.5).

Qualitatively similar results are observed for Sequoyah. Elimination of the potential for early reactor vessel depressurization by induced hot leg failure or steam generator tube rupture (via a sensitivity analysis) has a noticeable, but not dramatic, influence on the likelihood of high-pressure melt ejection. Table C.6.4 shows the fraction of Sequoyah accident progressions (for two important types of core melt accidents) that results in high-pressure melt ejection* for the base case analysis and the sensitivity analysis. In adjacent columns of this table are the fractions of the time that high-pressure melt ejection occurs and results in containment failure by overpressurization.

*The values shown only account for cases in which high-pressure melt ejection occurs in a cavity that is not deeply flooded. Cases in which the cavity is deeply flooded do not usually generate loads sufficiently large to threaten containment integrity.

Table C.6.3 Fraction of Surry slow blackout accident progressions that results in various modes of containment failure (mean values).

	Fraction of Slow Blackout Accident Progressions Resulting in Containment Failure Mode X	
Containment Failure Mode	**Base Case Analysis**	**Sensitivity Analysis**
Structural Rupture	0.01	0.01
Leak	0.01	0.01
Basemat Meltthrough	0.07	0.06
Containment Bypass	< 0.01	0.0
No Failure*	0.91	0.92

*Included in this category are accident progressions in which core damage is arrested in-vessel, thus preventing reactor vessel breach and containment failure. For Surry, these cases comprise approximately 60-65 percent of the "No Failure" scenarios.

Table C.6.4 Fraction of Sequoyah accident progressions that results in HPME and containment overpressure failure.

	Fraction Resulting in HPME Without a Flooded Cavity		Fraction of Columns (A) Cases in Which Containment Overpressure Failure Occurs	
Type of Core Damage Accident	**(A) Base Case Analysis**	**(A) Sensitivity Analysis**	**Base Case Analysis**	**Sensitivity Analysis**
LOCA	0.11	0.11	0.16	0.16
Station Blackout	0.16	0.21	0.20	0.21

As might be expected, no change is observed for the LOCA accident scenarios. Negligible changes are also observed for station blackout scenarios.

REFERENCES FOR SECTION C.6

C.6.1 F.T. Harper et al., "*Evaluation of Severe Accident Risks: Quantification of Major Input Parameters*," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

C.6.2 T.A. Wheeler et al., "Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation," Sandia National Laboratories, NUREG/CR-4550, Vol. 2, SAND86-2084, April 1989.

C.6.3 J.E. Kelly et al., "MELPROG-PWR/MOD1 Analysis of a TMLB' Accident Sequence," Sandia National Laboratories, NUREG/CR-4742, SAND86-2175, January 1987.

C.6.4 P.D. Bayless, "Natural Circulation During a Severe Accident: Surry Station Blackout," EG&G Idaho, Inc., EGG-SSRE-7858, 1987.

C.6.5 V.E. Denny and B.R. Sehgal, "PWR Primary System Temperatures During Severe Accidents," *ANS Transactions*, 47, (317-319).

C.6.6 B.R. Sehgal et al., "Effects of Natural Circulation Flows on PWR System Temperatures During Severe Accidents," *1985 National Heat Transfer Conference*, (223–234), 1985.

C.6.7 R.J. Lutz, Jr., et al., "Ringhals Unit 3 Severe Accident Analyses to Support Development of Severe Accident Procedures," Westinghouse Nuclear Technology & Systems Division, WCAP-11607, 1987.

C.6.8 M.G. Plys et al., "Seabrook Steam Generator Integrity Analysis," Fauske & Assoc., FAI/86-39, 1986.

C.6.9 K.N. Fleming et al., "Risk Management Actions to Assure Containment Effectiveness at Seabrook Station," Pickard, Lowe and Garrick, PLG-0550, 1987.

C.6.10 W.A. Stewart et al., "Experiments on Natural Circulation Flow in a Scale Model PWR Reactor System During Postulated Degraded Core Accidents," *Proceedings of Third International Topical Meeting on Reactor Thermal Hydraulics* (Newport, RI), ANS, Vol. 1, October 1985.

C.6.11 American Nuclear Society, *Transactions of the ANS/ENS Topical Meeting on the TMI-2 Accident* (Washington, DC), October 30–November 4, 1988.

C.6.12 J.E. Kelly memorandum to R.J. Breeding, Sandia National Laboratories, "RCS Pressure at Vessel Breach," dated January 27, 1989 (with errata dated February 15, 1989).

C.7 Drywell Shell Meltthrough

The potential for early containment failure is an important contributor to the potential consequences of severe accidents and, thus, to risk. In this context, "early" means before or immediately following the time at which molten core debris penetrates the lower head of the reactor vessel. A number of plausible mechanisms for early containment failure were identified in the Reactor Safety Study, many of which have since been determined to be of extremely low probability and are currently considered to be negligible contributors to risk. More recent containment performance studies have identified a few mechanisms that were not considered in the Reactor Safety Study and have been found to be important contributors to the uncertainty in risk for some plants.

An early failure mechanism that has received a great deal of attention is penetration of a BWR Mark I containment resulting from thermal attack of the steel containment shell by molten core debris. The scenario for this mode of containment failure postulates that, as core debris exits the reactor pressure vessel and is deposited on the floor within the reactor pedestal, it flows out of the pedestal region through an open doorway in the pedestal wall and onto the annular drywell floor. For the debris to contact the drywell shell, it must flow across the drywell floor until it contacts the steel containment shell along the line where the shell is embedded in the drywell floor. Figures C.7.1 and C.7.2 show the relevant geometry of the Peach Bottom drywell in vertical and horizontal cross sections, respectively. If hot debris contacts the drywell shell, two failure modes may occur: the combined effects of elevated containment pressure and local heating of the steel shell may result in creep rupture, or, if hot enough, the debris may melt through the carbon steel shell. Both of these plausible modes are considered in this issue and are collectively referred to as "drywell shell meltthrough."

C.7.1 Issue Definition

This issue represents the ensemble of uncertainties associated with the conditional probability of drywell shell meltthrough. (Failure is conditional on core meltdown progressing to the point that core debris penetrates and is discharged from the lower head of the reactor pressure vessel.) This probability is known to depend on the condition of the core debris (physical state, composition, release rate from the reactor vessel, etc.) as it relocates to the reactor pedestal floor. Distinct cases are established to consider separately each plausible combination of debris conditions at vessel breach. The case structure accounts for uncertainties in severe accident events and phenomena that precede the potential challenge to drywell integrity by drywell shell meltthrough.

Uncertainties associated with the processes and events that occur after core debris leaves the reactor vessel are represented by the probability distribution assigned to each case. The parameters considered in the case structure are:

Parameters Defining Case Structure	Values Considered
1. Rate at which core debris flows out of the reactor vessel.	High: R > 100 kg/sec Med: 50 kg/sec > R > 100 kg/sec Low: 50 kg/sec > R
2. Reactor vessel pressure when core debris first begins to exit the vessel.	High: Near 1000 psia Low: < 200 psia
3. Amount of unoxidized metals in melt.	High: 65 percent of initial inventory (representing range: 50–80 percent) Low: 35 percent of initial inventory (representing range: 20–50 percent)
4. Amount of debris superheat (temperature above melting point of debris).	High: > 100K Low: < 100K

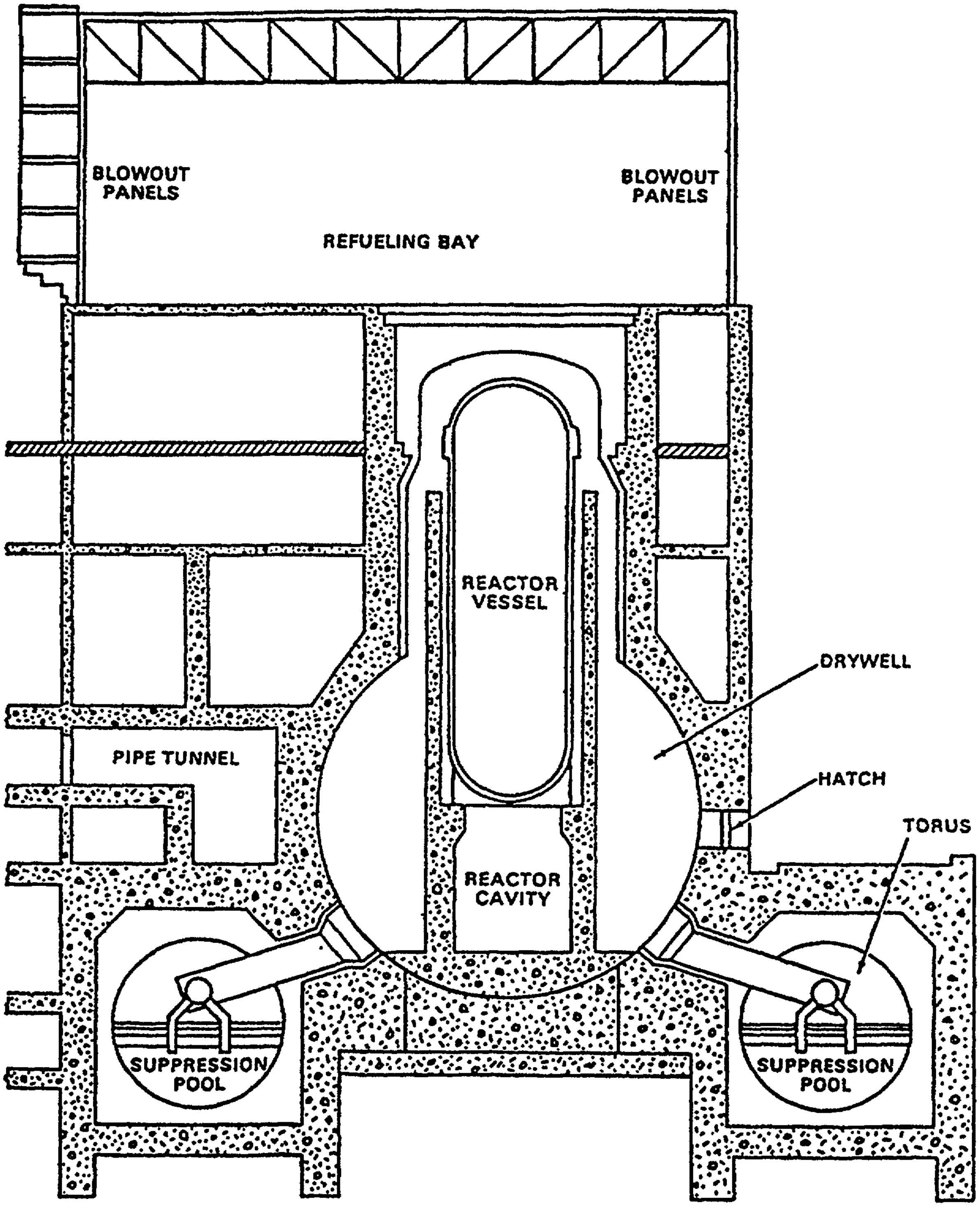


Figure C.7.1 Configuration of Peach Bottom drywell shell/floor—vertical cross section.

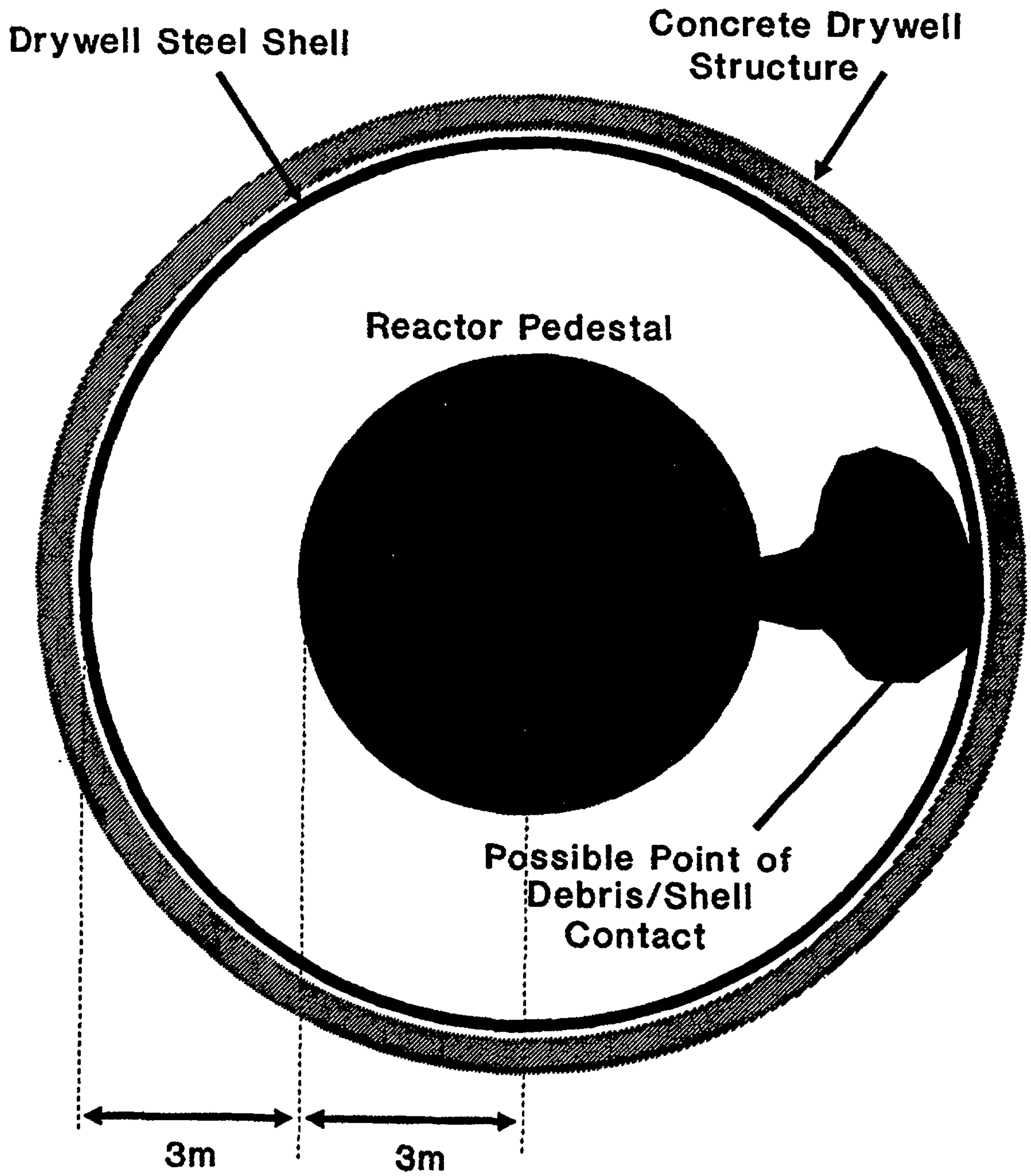


Figure C.7.2 Configuration of Peach Bottom drywell shell/floor—horizontal cross section.

5. Presence (or lack of) water on drywell floor before debris is expelled from reactor vessel.*	Yes: Sufficient to overfill sumps and replenishable No: No water on drywell floor

To account for the possibility that any permutation of these parameters may be important, the conditional probability of containment failure is quantified for each of the 48 cases. For each case, the containment failure probability is allowed to vary with time (after reactor vessel failure).

The uncertainties considered in characterizing the failure probabilities for each case include:

- Heat transfer characteristics of the debris on the concrete floor (e.g., thermal properties of melt and concrete, heat transfer coefficients for competing mechanisms, physical configuration of debris constituents, rate of internal heat generation).
- Heat transfer characteristics of the melt/steel shell interface (e.g., anticipated configuration and composition of debris in contact with steel, mechanism(s) for deterioration of shell thickness, properties of interfaces between debris and steel, and the steel shell and materials outside the shell).
- Debris transport characteristics when flowing across drywell floor (e.g., rheology of molten corium, drywell floor area covered by debris, barriers to flow—sump pits, pedestal wall).
- Structural behavior of the carbon steel drywell shell when in contact with molten material (e.g., formation of eutectics, alternative failure mechanisms).

C.7.2 Technical Bases for Issue Quantification

This issue was presented to a panel of six experienced severe accident analysts:

David Bradley—Sandia National Laboratories,
Michael Corradini—University of Wisconsin,
George Greene—Brookhaven National Laboratory,
Michael Hazzan—Stone & Webster Engineering Corp.,
Mujid Kazimi—Massachusetts Institute of Technology, and
Raj Sehgal—Electric Power Research Institute.

The panelists' individual judgments for the conditional probability of containment failure by drywell shell meltthrough formed the basis for quantifying this issue. Each of the panelists used the results of several published analyses in their deliberations. References C.7.1 through C.7.6 are among those identified by the panelists as having had an influence on their technical judgments. Complete documentation of the elicitation of expert judgment is provided in Reference C.7.7. A summary of the probability distributions that were generated by this process and a description of important areas of agreement and disagreement are presented below.

Each panelist generated a conditional probability distribution for each of the 48 cases outlined above. However, as will be shown, the uncertainties associated with this issue (i.e., the divergence in quantitative judgment among the panelists) are quite large. Although collectively divergent, the panelists' judgments are individually self-consistent. As a result, the collective judgment of the panel for all cases can be reasonably characterized by a handful of aggregate probability distributions. To preserve the true characteristics of the panel's case-by-case judgments, however, the distributions for individual cases are retained in the analysis for Peach Bottom documented in Reference C.7.8 (refer to Section C.7.3). The aggregate distributions are quite useful for illustrating important similarities and differences in the rationales of the panelists and are discussed later.

Before discussing specific topics dominating uncertainty, it should be noted that there are some aspects of this issue on which the technical community appears to agree. The appropriate failure criterion for the drywell steel shell is generally accepted as a shell temperature exceeding 1100–1300K. This range

*This parameter includes the effects of spray operation.

envelopes the uncertainties related to alternative mechanisms for failure of the drywell pressure boundary. At the lower end of this range, breach of the drywell pressure boundary could occur by creep rupture, particularly at elevated containment pressures. Alternatively, localized penetration of the steel shell by molten debris might occur if the debris in contact with the shell is sufficiently hot.

It is also generally agreed that the composition and temperature of the debris exiting the reactor pressure vessel are important properties to characterize when estimating the likelihood of hot debris reaching the drywell floor/wall interface. Specifying these properties, however, is limited by large uncertainties in core melt progression. These uncertainties have been treated parametrically (i.e., they are addressed through the case structure outlined above).*

Unfortunately, the harmony ends here. Most panelists expressed a strong allegiance to a specific technical rationale that, in many cases, was physically inconsistent with rationales expressed by other panelists. These dramatic differences in technical judgment reflect the polarized views on this issue that have developed in recent years. Over the full spectrum of plausible severe accident conditions, roughly half the technical community (as represented by the expert panel) appears to believe with near certainty that the drywell shell will fail following many core meltdown accidents; the other half is equally certain that it will not fail. If drywell shell meltthrough does occur, the containment pressure boundary is most likely to be breached within the first hour following reactor vessel penetration.

The variation of the conditional probability of drywell meltthrough among the 48 cases considered in this issue is not very wide. The net probability (i.e., arithmetic average of all panelists) of drywell shell meltthrough is no smaller than 0.33 and no larger than 0.87. The lowest failure probabilities correspond to cases in which water is assumed to cover the drywell floor; the highest correspond to cases in which the drywell floor is dry and debris flow rate, debris temperature (superheat), and debris unoxidized metal content are all at the high end of their range.

An illustration of the overall result for this issue is shown in Figure C.7.3. As mentioned above, the probability distributions for the 48 cases examined as part of this issue can be aggregated into five classes without introducing substantial error. These are displayed in the figure. Two classes characterize all the cases with low or medium flow rates of debris from the reactor vessel. The only single parameter significantly affecting the failure probability for these cases is the presence (or lack) of replenishable water on the drywell floor (all other parameters are observed to have a relatively minor influence on the probability of failure). Each of the remaining three classes represent "high flow" cases; one represents cases with water covering the drywell floor, and two represent cases without water on the drywell floor. Again, the presence (or lack) of water is the only single parameter significantly affecting the outcome. The values of other parameters (in certain combinations) influenced the panelists' judgments for these cases, however. The highest probability of drywell shell meltthrough is observed for cases in which two of the remaining three parameters take on values at the high end of their range (denoted 2/3H in the figure's legend). Lower values are observed when at least two of the three parameters take on low values (denoted 2/3L).

A conclusion that might be drawn from Figure C.7.3 is that the probability of drywell shell meltthrough is lowered by ensuring the presence of water on the drywell floor. This position is supported by some, but not all, of the panelists. It is important to keep in mind that the values displayed in Figure C.7.3 are averages of individual experts' judgments. As such, they tend to mask the divergence of technical views on many of the topics important to quantifying this issue. In this appendix, we do not intend to elaborate on the details of technical rationales expressed by individual panelists. The reader is encouraged to study Reference C.7.7 to gain a more thorough understanding of the phenomena and modeling assumptions disputed within the technical community. The following example is provided, however, to illustrate the divergence of technical views and explain why the average value of drywell shell meltthrough, displayed in Figure C.7.3, can be misleading.

Figure C.7.4 displays the probability distributions generated by each panelist for four specific cases. The case definition is noted in the legend on each plot. The upper two cases represent scenarios with low core debris flow rates. The case shown in the upper left represents a scenario in which the drywell floor is dry;

*The relative likelihood of one case over another is treated as a separate issue.

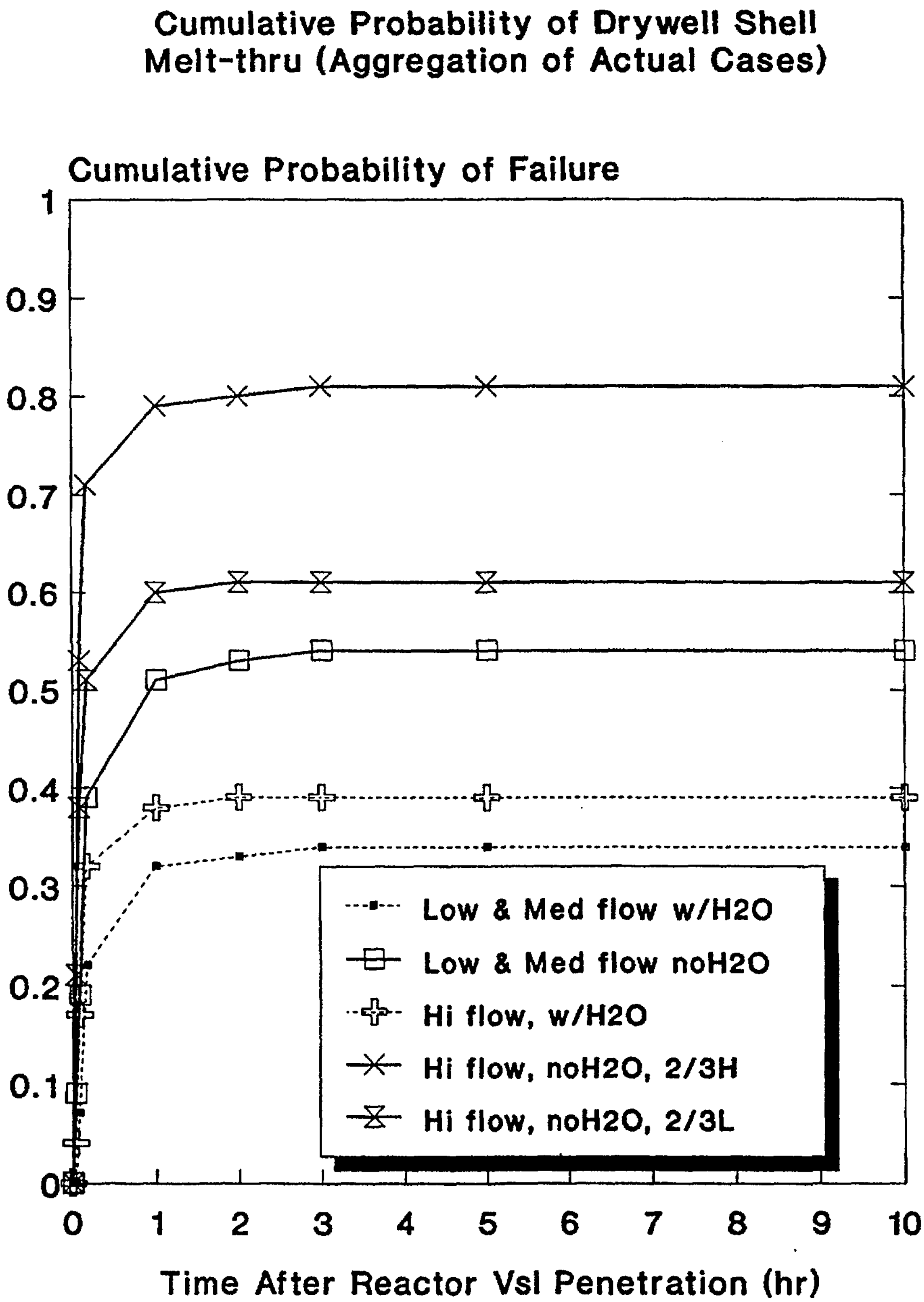


Figure C.7.3 Aggregate cumulative conditional probability distributions for Peach Bottom drywell shell meltthrough.

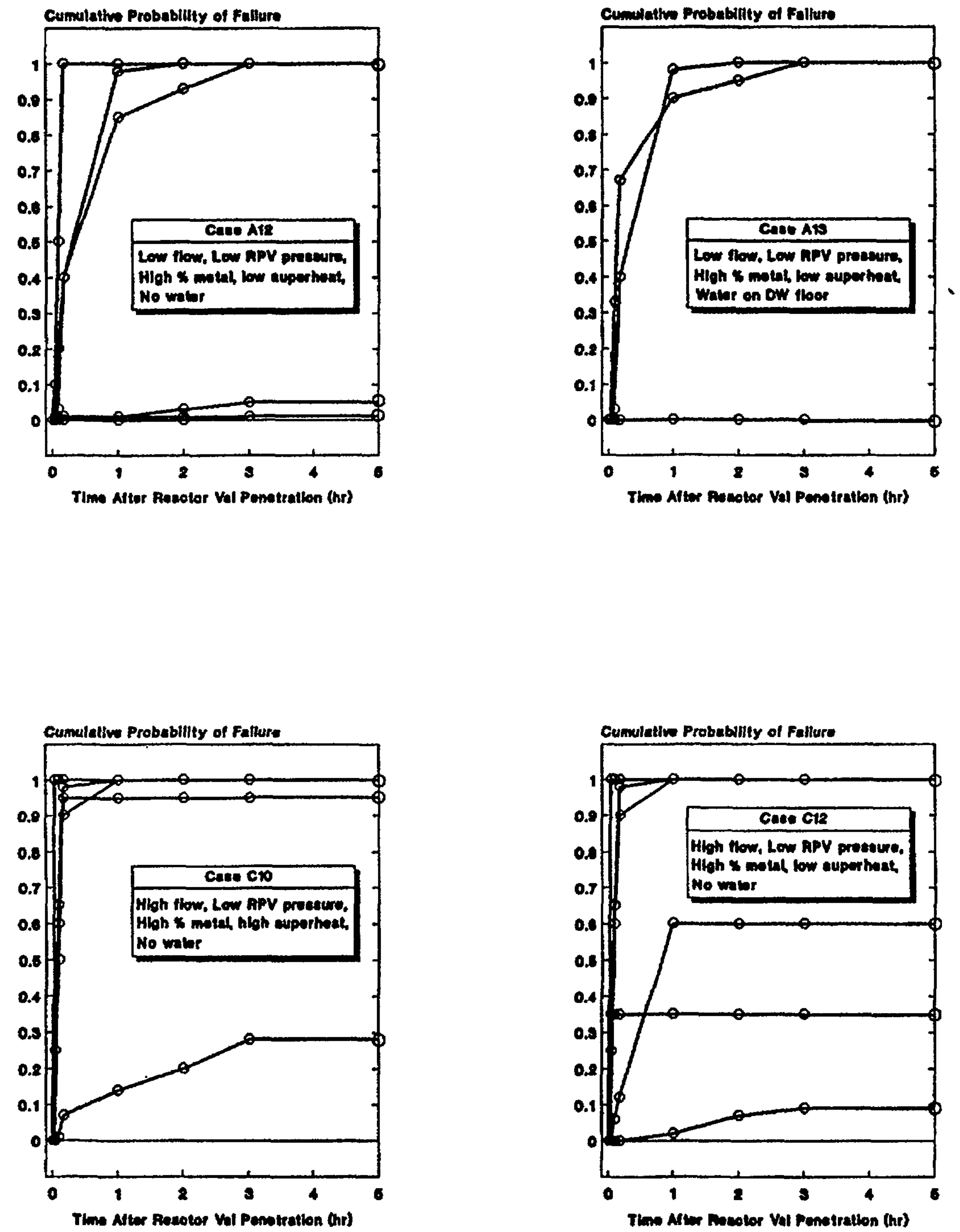


Figure C.7.4 Cumulative probability distributions composite of individuals on expert panel for this issue. (Six panelists (6 curves) are shown for each of four cases.)

the case shown in the upper right represents a scenario in which sufficient water accumulates on the floor to overfill the drywell sump. The lower two plots depict the results for two of the high flow cases (both without water on the drywell floor). One (lower left) assumes a high fraction of unoxidized metals in the melt, high debris temperatures, and low reactor pressure vessel pressure (i.e., class 2/3H in Fig. C.7.3). The plot in the lower right corner differs only by the assumption of low debris superheat.

Two important observations should be noted. First, the individual panelists' judgments appear binary (i.e., taking on values very near zero or unity). The results for the majority of cases appear this way, with at least one panelist providing a judgment at each end of the spectrum. Intermediate values were provided for relatively few cases, such as case C12 shown in the lower right corner of Figure C.7.4. This divergence of quantitative judgment explains why the values for the average probability appear constrained between 0.3 and 0.8. The average value for case A12 (upper left) is approximately 0.5 because three panelists provided values of 1 and three values of 0. In quantifying case A13 (upper right), one of the three panelists who was certain failure would occur in case A12 felt that the addition of water would prevent molten debris from reaching the drywell wall and, thus, changed his judgment from 1 to 0 for case A13. The other two panelists believe that the debris would be largely unaffected by the presence of water on the floor and did not alter their quantitative judgment. The average failure probability, therefore, changed from 0.5 to 0.33. A similar, but less dramatic, effect is observed in cases C10 and C12 (lower two plots) for which the average values changed from 0.87 to 0.68, respectively.

The second observation is that, despite an apparent consensus among panel members that the largest source of uncertainty for this issue is the initial conditions (i.e., state of core debris and drywell floor prior to reactor vessel penetration), dramatically different quantitative judgments were provided for the same initial conditions. Moreover, most panelists were very confident that their judgments were correct for many cases (i.e., a conditional probability of 1.0 or zero was provided for several types of conditions). In reviewing the elicitation of the expert panelists, one difference in rationale appears to have had a strong influence on their judgments. Some panelists believe that the flow of corium across the drywell floor is hydrodynamically limited (i.e., governed by the rheology and transport properties of the flowing core debris/concrete mixture). Others believe that the flow is thermodynamically limited (i.e., governed by the heat transfer characteristics of the mixture).

C.7.3 Treatment in PRA and Results

The probability distributions for this issue were implemented in the Peach Bottom accident progression event tree. This tree provides a structured approach for evaluating the various ways in which a severe accident can progress, including important aspects of RCS thermal-hydraulic response, core melt behavior, and containment loads and performance. The accident progression event tree is a key element in the assessment of uncertainties in risk by accommodating the possibility that a particular accident sequence may proceed along any one of several alternative pathways (i.e., alternative combinations of events in the severe accident progression). The probability distributions (or split fractions) for individual and combinations of events within the tree provide the rules that determine the relative likelihood of various modes of containment failure.

Drywell shell meltthrough is represented as an explicit event in the Peach Bottom accident progression event tree, and the probability assigned to it is dependent upon the path taken through the tree. For example, each path through the tree involves events that imply a particular combination of initial conditions for a potential challenge to drywell integrity. Values for each of the parameters defining the case structure outlined in the previous section are, therefore, established by the outcome of events occurring earlier in the tree. For example, one path may imply conditions of low reactor vessel pressure (perhaps caused by early actuation of the automatic depressurization system), high unoxidized metal content in the debris leaving the reactor vessel (from low in-vessel cladding oxidation), low debris temperature and low flow rates of debris leaving the reactor vessel (from a small hole size in the reactor vessel lower head), and, finally, no water on the drywell floor (failure of drywell sprays). For these conditions, the conditional probability of drywell shell meltthrough is represented by the distributions for case A12 (shown in Fig. C.7.4).

The distributions generated by each of the panelists are given equal weight in the accident progression event tree analysis. This is accomplished by generating an aggregate distribution for each case in the case structure that represents the composite judgment of the expert panel. The aggregate distribution is generated by averaging the distributions prepared by the panelists. This is equivalent to randomly sampling values from each panelist's distribution but constraining the sampling process to ensure that each distribution is sampled an equal number of times. Additionally, drywell shell meltthrough is treated as a nonstochastic event (i.e., it either occurs or it does not occur). Therefore, the conditional probabilities generated by the expert panel are appropriately converted to event tree branch probabilities of 1 or zero. For example, if a conditional probability of failure equal to 0.3 is selected for a particular path in the event tree, a branch point probability of 1 would be assigned to that branch in 30 percent of the sample members. Likewise, a value of zero would be assigned for the other 70 percent of the sample members. Recall that the method used to perform the statistical uncertainty analysis is one involving multiple passes through the accident progression event tree. The values of the branch point probabilities are allowed to change from one pass to the next, generating a distribution of possible accident outcomes.

An indication of the importance of drywell shell meltthrough on the results of the Peach Bottom accident progression analysis is the mean probability that accident sequences contributing to the total core damage frequency are estimated to result in this mode of containment failure. Table C.7.1 shows the mean conditional probability of drywell shell meltthrough for several important core damage accident groups.

Another indication of the importance of drywell shell meltthrough on the Peach Bottom results is the decrease in the mean conditional probability of containment failure (i.e., given a core damage accident) when shell meltthrough is assumed to be impossible. The mean probability of early containment failure (frequency-weighted for all accident sequences*) for the base case analysis (i.e., with the drywell shell meltthrough probabilities outlined above) is 0.56. This value decreases to 0.20 if drywell shell meltthrough is assigned a probability of zero. The remaining (20%) probability of early containment failure results from other containment failure mechanisms such as overpressure failure and ex-vessel steam explosions. The latter are described further in Section C.9.

Drywell shell meltthrough is also an important contributor to each of the measures of risk for Peach Bottom. For example, Peach Bottom severe accident progressions (from all types of accident sequences) that result in drywell shell meltthrough contribute approximately 70 percent of the mean estimate for latent cancer fatality risk and 60 percent of the mean estimate for early fatality risk.

Table C.7.1 Probability of drywell shell meltthrough (conditional on a core damage accident of various types).

Type of Core Damage Accident	Mean Frequency* of Accident Type	Mean Probability of Drywell Shell Meltthrough
LOCAs	1.50E–7	0.32
Transients	1.81E–7	0.32
Station blackout	2.08E–6	0.44
ATWS	1.93E–6	0.42

*These frequencies consider internally initiated events only.

REFERENCES FOR SECTION C.7

C.7.1 M.G. Plys, J.R. Gabor, and R.E. Henry, "Ex-Vessel Source Term Contribution for a BWR Mark-I," International ANS/ENS Topical Meeting on Thermal Reactor Safety (San Diego, CA), February 1986.

C.7.2 L.S. Kao and M.S. Kazimi, "Thermal Hydraulics of Core/Concrete Interaction in Severe LWR Accidents," MIT Nuclear Engineering Dept., MITNE–276, June 1987.

C.7.3 S.A. Hodge, "BWRSAT Approach to Bottom Head Failure," Presentation to NUREG–1150 Review Group, November 1987.

*These frequencies consider internally initiated events only.

C.7.4 D.A. Powers, "Erosion of Steel Structures by High Temperature Melts," *Nuclear Science and Engineering*, 88, 1984.

C.7.5 G.A. Greene, K.R. Perkins, and S.A. Hodge, "Mark I Containment Drywell, Impact of Core/Concrete Interactions on Containment Integrity and Failure of the Drywell Liner," *Source Term Evaluation for Accident Conditions*, paper IAEA-SM-281/36, IAEA, Vienna, Austria, October 26–November 1, 1985.

C.7.6 J.J. Weingardt and K.D. Bergeron, "TAC2D Studies of Mark I Containment Drywell Shell Meltthrough," Sandia National Laboratories, NUREG/CR-5126, SAND88-1407, August 1988.

C.7.7 F.T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

C.7.8 A.C. Payne Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft Revision 1, SAND86-1309, to be published.*

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

C.8 *Containment Strength Under Static Pressure Loads*

Since the containment building of a nuclear power plant constitutes the ultimate barrier between the in-plant environment and the outside atmosphere, its anticipated performance during a severe accident has a substantial impact on the risk characteristics of the plant. Uncertainty regarding the capability of a containment to withstand the challenges associated with severe accidents can, therefore, be an important contributor to the uncertainty in risk.

In the risk models of this study, determining containment performance involves assessing the probability that the containment would be breached under a range of hypothesized severe accident conditions. In addition to the likelihood of failure, other critical factors in the characterization of containment performance are:

- *Failure size:* The larger the hole in the containment, the more rapid the escape of radioactive material in the containment atmosphere to the outside environment. This reduces the time available for radioactive material to deposit within the containment building and also reduces the opportunity for effective offsite emergency response.

- *Location of failure:* The retention of radioactive material by a breached containment building may be highly dependent upon the location of failure relative to containment systems designed to mitigate accident conditions. For example, in an ice condenser containment, failure of the containment in the lower compartment permits radioactive material to bypass the ice compartments while escaping to the outside environment. In contrast, containment failure in the upper compartment, provided the ice condenser is not degraded, requires that radioactive material pass through the ice compartments before escaping to the outside. In the latter case, retention of material by the ice would substantially reduce the radioactive release.

Consideration of these elements of containment performance, in conjunction with assessment of the degree and the timing of potential challenges to the containment, provides the basis for determining the likelihood and the consequences of scenarios involving containment failure.

C.8.1 Issue Definition

This issue addresses the response of each of the five containments to the potential pressure loads associated with severe accident conditions. Other containment failure mechanisms, such as penetration by missile, structural failure due to impulse loads (e.g., from hydrogen detonation), and meltthrough by molten material, are excluded from the scope of this issue. These are discussed elsewhere in Appendix C.

The set of plants evaluated in this study was selected to encompass a broad spectrum of containment designs. Consequently, details of important severe accident conditions and modes of containment response differ substantially among the plants analyzed. For this reason, the issue of containment performance is discussed largely on a plant-specific basis. However, it is possible to characterize broadly the range of qualitatively distinct pressure loads that may result from severe accident conditions. These are:

- *Gradual pressure rises:* Gradual pressurization of the containment building would result from the protracted generation of steam and noncondensible gases through the interaction of molten core material with the concrete floor beneath the reactor vessel. This pressurization process could last from several hours to several days, depending upon accident-specific factors such as the availability of water in the containment and the operability of engineered safety features. An additional mechanism for gradual pressurization in BWR pressure-suppression containments is the generation of steam from the suppression pool in the circumstance that pool heat removal capability is degraded.

- *Rapid pressure rises:* The high-pressure expulsion of molten material from the vessel, the deflagration of combustible gases, and the rapid generation of steam through the interaction of molten fuel with water in the containment are phenomena that could lead to pressure rises in the containment over a period of a few seconds. Such pressure rises may be viewed as rapid in a thermophysical context; however, from a structural perspective, they are effectively static. It is essential, nevertheless, to distinguish between gradual and rapid pressure rises in the characterization of containment performance since the rate of pressure increase may have a significant influence upon determining the ultimate mode of containment failure.

This influence stems from the possibility of multiple containment failures or the development of one failure mode into another more severe mode. For example, where gradual containment pressurization results in containment breach by leakage, the pressure relief associated with the leak prevents further pressurization, and, thus, precludes more severe modes of containment failure. For rapid pressure rises, however, an induced leak would not preclude continued pressurization of the containment and, therefore, a more severe failure of the containment building could ultimately result. Hence, while the distinction between gradual and rapid pressure rises will not influence the pressure at which failure first occurs, it may influence the ultimate severity of that failure.

While the question of containment failure location is largely specific to the individual containment geometries, the approach to characterizing potential failure sizes is more generic. Three possible failure sizes were distinguished in this study: leak, rupture, and catastrophic rupture. Working quantitative definitions of each failure size were based on thermal-hydraulic evaluations of containment depressurization times.

- A leak was defined as a containment breach that would arrest a gradual pressure buildup but would not result in containment depressurization in less than 2 hours. The typical leak size was evaluated for all plants to be on the order of 0.1 ft^2.
- A rupture was defined as a containment breach that would arrest a gradual pressure buildup and would depressurize the containment within 2 hours. For all plants, a rupture was evaluated to correspond to a hole size in excess of approximately 1.0 ft^2.
- A catastrophic rupture was defined as the loss of a substantial portion of the containment boundary with possible disruption of the piping systems that penetrate or are attached to the containment wall.

A panel was assembled to address issues of containment structural performance with severe accident loads. Its members were:

D. Clauss, Sandia National Laboratories,
C. Miller, City College of New York,
K. Mokhtarian, Chicago Bridge and Iron, Inc.,
J. Rashid, ANATECH,
W. Von Riesemann, Sandia National Laboratories,
S. Sen, Bechtel,
R. Toland, United Engineers and Constructors,
A. Walser, Sargent and Lundy,
J. Weatherby, Sandia National Laboratories, and
D. Wesley, IMPELL.

The experts provided distributions that defined the probability of failure as a function of pressure and of the mode and location of failure. The distinction between rapid and gradual static pressurization cases in determining ultimate failure size was treated within the methodological framework of this study (see Section C.8.3). The experts only addressed initial failure sizes (i.e., they did not distinguish between the rapid and gradual pressurization cases).

C.8.2 Technical Bases for Issue Quantification

Detailed structural evaluations of various scopes exist for each of the containments assessed by the expert panel. These evaluations, supplemented by calculations performed by the experts in support of the elicitation process, provided the basis for quantification of this issue for each plant.

Zion

The Zion large, dry containment building, shown in Section C.5, is a concrete cylinder with a shallow-domed roof and a flat foundation slab. The thickness of the concrete is 3.5 feet in the walls, 2.7

feet in the dome, and 9 feet in the basemat. The containment wall and dome are prestressed by a system of tendons, with each tendon composed of 90 1/4-inch-diameter steel wires, while the foundation slab is reinforced with bonded, reinforcing steel. The entire structure is lined with a 1/4-inch welded steel plate attached to the concrete by means of an angle grid, stitch-welded to the liner plate, and embedded in the concrete. The free volume of the containment is approximately 2.6 million cubic feet and its design internal pressure is 47 psig.

The three experts who addressed the Zion containment performance issue had access to existing detailed structural calculations of containment response at Zion reported in References C.8.1 to C.8.3.

Two potential failure locations were identified by the expert panel: in the midsection of the cylindrical wall and at the junction between the basemat and the wall. Based upon source term considerations, the distinction between these failure locations was preserved in the risk analyses since failure in the cylindrical wall permits a direct release of radioactive material to the outside environment while failure at the cylinder-basemat intersection occurs approximately 16 feet below ground level. Some degree of ex-containment fission product decontamination prior to atmospheric release would, therefore, be anticipated in the latter case.

The mechanism for failure in the midsection of the cylindrical wall was assessed to be yielding of the hoop tendons. To develop probability distributions over potential failure pressures, a range of failure criteria of 1 to 2 percent strain in the hoop tendons was assumed. The resulting probability of failure was in the pressure range of 130–140 psig. Leak was anticipated to be the predominant failure size for failure pressures up to 138 psig. For higher failure pressures, breach by rupture was assessed to be more likely. The possibility that failure would first occur at pressures up to 150 psig was also identified, in which case catastrophic rupture was assessed to be the most likely failure size. However, it was believed that for gradual pressurization cases, the prior occurrence of a leak due to a tear in the liner would likely preclude catastrophic failure of the containment. Reference C.8.1 constituted the main analytical basis for evaluation of this failure location.

The alternative containment breach location identified involved the cracking of concrete at the shear discontinuity between the cylinder wall and the basemat. Only one of the three experts assessed this failure mode to be credible based primarily on Reference C.8.2. Uncertainty in the failure pressure associated with this location was broad. The bulk of probability was assigned to the failure pressure range of 110–180 psig. Failure by leak was assessed to be the most likely size of breach at the wall-basemat junction, although containment rupture was identified as a possibility toward the upper end of the failure pressure range.

Figure C.8.1 displays the range of failure pressures for the Zion containment at the 5th–95th percentile levels. This range is 108–180 psig, which includes all failure sizes and locations. It is based on the distribution that resulted from aggregation of the three expert-specific probability distributions.

Surry

The Surry containment building, depicted in Section C.5, is comprised of a vertical right cylinder with a hemispherical roof and a flat foundation slab. It is constructed from reinforced concrete and lined with a 1/4-inch steel plate. The containment atmosphere is maintained during operation at below ambient pressure (at approximately 10 psia), the design internal pressure is 45 psig, and the containment free volume is 1.8 million cubic feet.

Four experts addressed the issue of containment performance at Surry. The limited availability of structural calculations for the Surry containment led the experts to rely partially upon detailed calculations performed for similar containments, such as Indian Point (Ref. C.8.3).

All failure locations identified for Surry provided direct pathways to the outside environment and the distinction between these locations was, therefore, unnecessary from a risk perspective. Yielding of one of the steel hoop bars that reinforce the vertical concrete wall was identified as a likely mode of failure by all the experts. The likely location of failure was assessed to be near the intersection of the wall with the dome. Evaluation of this failure location was based partially upon the analysis of Reference C.8.4. Leakage due to the formation of a tear in the steel liner was also identified as the most likely failure mode

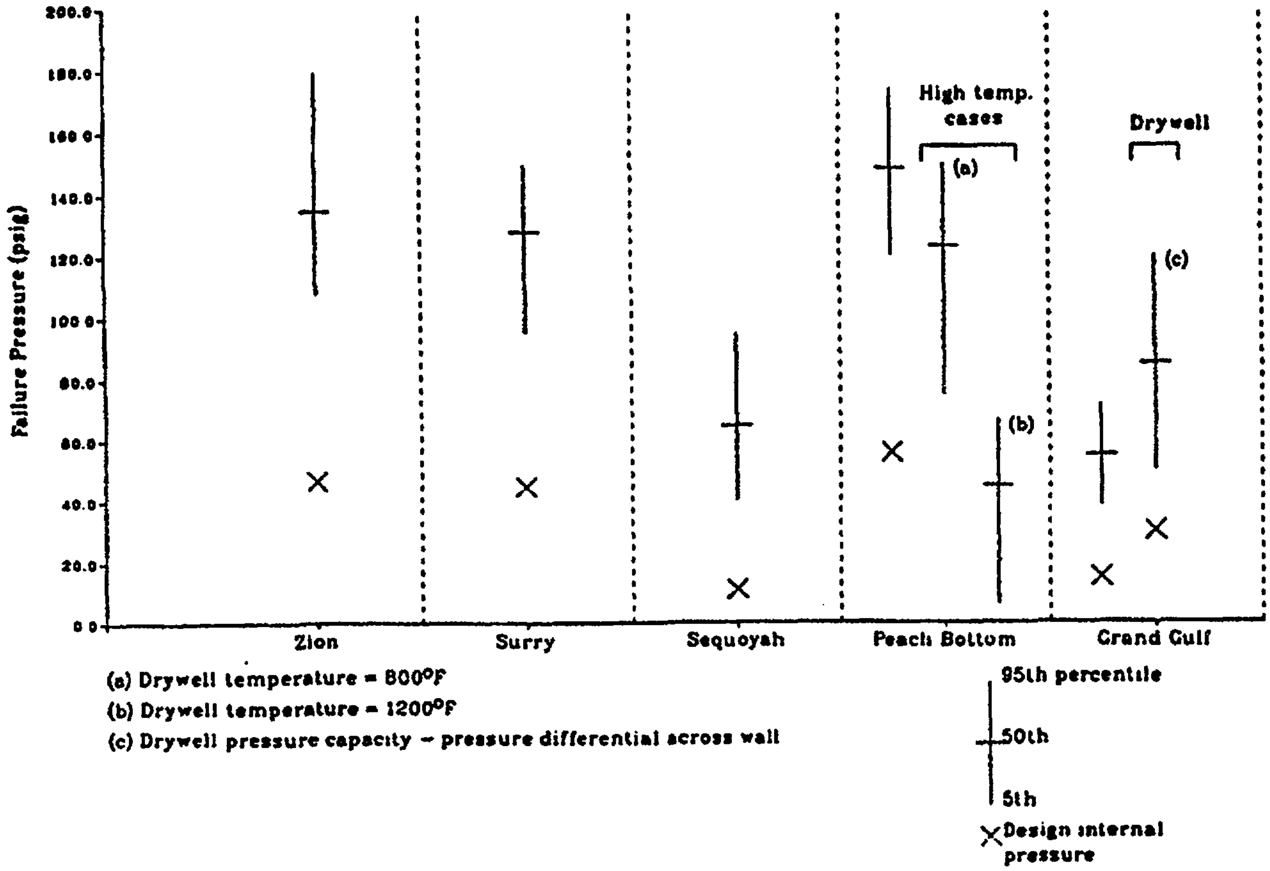


Figure C.8.1 Containment failure pressure.

in light of the results of the 1:6 scale model reinforced concrete containment test performed at Sandia National Laboratories (Ref. C.8.5). Other potential modes of containment breach identified were failure around pipe and electrical penetrations and failure due to distortion of the hatch opening. However, no consensus existed regarding the credibility of these other modes.

The probability distribution over potential failure pressures resulting from aggregating the distributions of each expert is summarized in Figure C.8.1. The 5th–95th percentile range of potential failure pressures extends from approximately 95 psig to 150 psig. Leakage was assessed as the most likely mode of failure for breaches occurring up to 140 psig, while ruptures were the most likely modes of breach for failure pressures in the 140–150 psig range. While higher failure pressures were judged unlikely, the dominant mode of failure beyond 155 psig was assessed to be catastrophic rupture.

Sequoyah

The Sequoyah ice condenser containment, shown in Section C.4, is a freestanding steel structure consisting of a cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete. The cylinder varies from 1-3/8-inch thickness (at the bottom) to 1/2 inch (at the dome junction); the dome varies from 7/16-inch thickness (at the cylinder junction) to 15/16 inch (at the apex); and the bottom liner plate is 1/4-inch thick. Vertical and horizontal stiffeners are provided on the outside of the shell. Three volumes comprise the inside of the containment shell: the lower compartment, the ice condenser, and the upper compartment. The reactor vessel and reactor coolant system are located in the lower compartment. The ice condenser consists of an annular volume that partially lines the containment wall (subtending an angle of 300 degrees at the center of the containment) and is comprised of a series of ice compartments located between an upper and a lower plenum. The ice condenser is lined with corrugated steel insulating panels designed to withstand an outside pressure of 19 psig. The role of the ice condenser is that of a passive heat sink intended to reduce steam pressures generated in design basis accident conditions.

The only significant structure within the upper compartment is the polar crane. Since the containment is equipped with the ice condenser pressure-suppression system, it is designed to an internal pressure of only 10.8 psig, with a free volume of 1.2 million cubic feet. A concrete shield building surrounds the steel shell. However, it is not a significant barrier to fission product release since its pressure capacity is substantially less than that of the shell.

Structural assessment of the Sequoyah containment required certain severe accident considerations specific to the ice condenser design. The location of containment failure relative to the position of the ice condenser and the structural integrity of the ice condenser are crucial factors in determining the concomitant radioactive release. If the route taken by radioactive aerosols to the outside environment involves passage through the ice, significant retention of radioactive material within the containment would be expected. Failure of the containment in the lower compartment or significant disruption of the ice would involve bypass of the ice compartments. Since the ice condenser constitutes the only passageway from the reactor vessel to the upper compartment, failure in the latter will avoid ice condenser bypass. Ultimately, however, the effect of failure location upon the source term is determined by the availability of ice, which in turn is dependent upon the accident sequence under consideration.

A further concomitant of the compartmentalized nature of the ice condenser containment is the possibility of localized accumulations of hydrogen and the possibility of containment failure through hydrogen deflagration or detonation. While the issue of containment response to hydrogen deflagration was subsumed in the case of rapid pressurization, the effects of hydrogen detonation and the associated dynamic loads were considered separately by the experts. Response of the Sequoyah containment to impulse loads is discussed in Section C.4.

Three experts addressed the issue of containment response to pressure loads at Sequoyah based on the detailed analyses of References C.8.1 and C.8.6 and on supplemental hand calculations. Membrane failure in the cylindrical wall of the upper compartment via either rupture or catastrophic rupture was identified as the most likely mode of containment failure. Failure criteria adopted involved a range of strain levels (2–10%) in the shell membrane. Since the ice condenser is attached to the cylindrical wall and subtends an angle of 300 degrees at the central containment axis, 5/6 of the ruptures in the wall of the upper compartment are expected to occur in the ice condenser. Catastrophic rupture of the wall, in contrast, would always fail the ice condenser.

One expert identified the possibility of failure in the lower compartment due to a crack in the weld at the point of embedment. Such a failure would result in ice condenser bypass. Based upon results of the 1:8 scale steel containment pressurization tests (Ref. C.8.7), two experts identified the possibility of containment leakage due to ovalization of the equipment hatch flange. The equipment hatch, a door 20 feet in diameter, is located in the upper compartment, and its failure would not result in ice condenser bypass.

Rupture or catastrophic rupture were assessed to be the most likely modes of failure in the Sequoyah containment. The only failure location associated with leakage involved ovalization of the hatch. While one expert predicted the occurrence of such a leak in the 65–70 psig range, a second expert believed that such a failure would not be likely to occur below 120 psig and that shell rupture would probably occur first. The third expert excluded this failure possibility completely. At the 5th–95th percentile levels, the range of failure pressures resulting from aggregating the individual expert distributions was 40–95 psig.

For all potential failure pressures, the probability of ice condenser bypass exceeded the probability of no bypass by factors of between 5 and 10. This stems from the dominance of the catastrophic rupture failure modes at Sequoyah. Catastrophic rupture is assumed always to fail the ice condenser since the ice compartments subtend most of the containment wall area. As discussed earlier, all failures in the lower compartment bypass the ice. The probability distribution over potential failure pressures at Sequoyah is summarized in Figure C.8.1.

Peach Bottom

The Peach Bottom containment, shown in Section C.7, is of the Mark I pressure-suppression type. The steel "light-bulb-shaped" drywell that contains the reactor vessel is a steel spherical shell intersected by a circular cylinder. The top of the cylinder is closed by a head bolted to the drywell. The pressure-suppression chamber, or the wetwell, is a toroidal steel vessel located below and encircling the drywell. The wetwell and drywell are interconnected by eight circular vent pipes. The containment is enclosed by the reactor building, which also contains the refueling area, fuel storage facilities, and other auxiliary systems. In the event of primary system piping failure within the drywell, a mixture of drywell atmosphere and steam would be forced through the vents into the suppression pool resulting in steam condensation and pressure reduction. The design internal pressure of the containment is 56 psig and the free volumes of the drywell and wetwell are 159,000 cubic feet and 119,000 cubic feet, respectively.

From a source term perspective, the location of containment failure relative to the suppression pool and to structures external to the containment is important. Three experts evaluated containment performance at Peach Bottom. Based on prior detailed analyses of Mark I designs (Refs. C.8.1 and C.8.8) and on calculations supporting the opinion elicitation process, several failure locations were considered credible:

- *Wetwell above the water line.* In this case, the suppression pool would not be bypassed.
- *Wetwell below the water line.* Consequent drainage of the wetwell would effectively result in pool bypass.
- *Drywell near the vent pipes.* This would result in suppression pool bypass although some credit would be taken for fission product decontamination by the reactor building.
- *Drywell head.* In this case, the suppression pool would be bypassed although some credit would be taken for fission product decontamination in the refueling area.

The relative likelihood of leak versus rupture was considered dependent on the failure location and the associated failure mechanisms. For example, one expert assessed the relative likelihood of a leak occurring in the wetwell below the water line to be low since any leak at that location would develop rapidly into a rupture. The predicted failure mechanism was a crack in the hoop direction, which would rapidly unzip, given the absence of a mechanism to arrest the rupture.

Because of the small volume of the Peach Bottom containment, the possible effect upon containment pressure capacity of the high drywell temperatures expected to occur in scenarios involving the attack of concrete beneath the vessel by molten core debris was considered. Such temperatures could be as high as 800°F to 1200°F and might substantially reduce material strengths. One expert assumed, for example, that material strength in the drywell at the vent pipes would be reduced by between 25 percent and 90 percent and that gasket resiliency at the drywell head would be lost. Since wetwell temperatures were assessed to be at saturation, high drywell temperatures were determined to have little impact upon the pressure capacity of the wetwell.

In low-temperature conditions, the range of possible failure pressures for the Peach Bottom containment was determined to be 120–174 psig. This reflects the 5th–95th percentile interval of the probability distribution resulting from aggregating the expert-specific distributions. Conditional upon containment failure in the lower part of this range, 50 percent of probability was associated with leakage at the drywell head while the remaining failure probability was dominated by wetwell leakage above the suppression pool. At the top edge of the failure pressure range, wetwell rupture above and below the suppression pool were each assessed to account for 25 percent of the total failure probability, with catastrophic wetwell rupture accounting for a further 10 percent. Leakage in the drywell (principally at the head) accounted for approximately 25 percent of the conditional failure probability, with wetwell leakage accounting for the remaining 15 percent.

Two high drywell temperature cases were considered by the experts: 800°F and 1200°F. The 5th–95th percentile failure pressure ranges were assessed to be 75–150 psig and 6–67 psig, respectively. With the drywell at 800°F, failure at the lower edge of the pressure range was assessed to be dominated by leakage in the drywell, principally (90% of the probability) through degradation of the head gasket. Toward the higher end of the failure pressure range, wetwell leaks above the suppression pool accounted for 30 percent of the failure probability, drywell leakage for a further 40 percent, and rupture at the drywell head for 12 percent. In the highest drywell temperature regime, i.e., 1200°F, the reduction in material strengths was assessed to ensure failure in the drywell. At the lower boundary of the failure pressure range, leakage in the drywell (principally at the head) was assessed to account for 95 percent of the failure probability while, at the upper boundary of the failure pressure range, drywell ruptures (principally in the main body of the drywell) were assessed to be as likely as leaks. Figure C.8.1 summarizes the aggregated probability distributions over possible containment failure pressures at Peach Bottom.

Grand Gulf

The Grand Gulf containment, shown in Section C.4, is of the Mark III pressure-suppression type. It is constructed from reinforced concrete lined with a 1/4-inch welded steel plate. The circular foundation mat, the cylindrical wall, and the hemispherical dome are 9.5, 3.5, and 2.5 feet thick, respectively. The containment volume is divided into two main compartments: the drywell, which is the central cylindrical volume of the containment and houses the reactor vessel; and the wetwell, which constitutes the outer annular volume and the dome. These compartments are connected at the base of the containment via an annular pool that provides a passive heat sink for steam in design basis accident conditions. The drywell wall, composed of reinforced concrete, is 5 feet thick and lined with 1/4-inch steel plate. Since the Grand Gulf containment is equipped with pressure-suppression features, its nominal design internal pressure is only 15 psig. The drywell has a design internal pressure of 30 psid (i.e., the differential pressure across the drywell-wetwell boundary). The free volumes of the wetwell and the drywell are 1.4 million and 270,100 cubic feet, respectively.

From a severe accident perspective, an important feature of the Mark III design is the relative configuration of the drywell, the suppression pool, and the wetwell. This configuration ensures that, provided the integrity of the drywell wall is not compromised, radioactive material released from the fuel would need to pass through the suppression pool to escape from the containment, if breached. This would result in significant radioactive material retention by the pool. In assessing performance of the Mark III, it is important, therefore, to determine the response to severe accident conditions not only of the outer containment but also of the drywell.

Given the low-pressure capacity of the Grand Gulf containment relative to anticipated pressure loads, the study project team assessed minimal uncertainty to be associated with response of the Grand Gulf containment to severe accident pressurization levels. To use expert resources most efficiently, therefore, the issue of static overpressurization at Grand Gulf was not taken to the expert panel on containment structural performance issues. The required probability distributions were developed by a structural expert at Sandia National Laboratories, who had been a member of the original panel. Detailed structural evaluations of containment performance at Grand Gulf reported in Reference C.8.1 provided a basis for the expert's evaluation. The dominant failure location of the containment due to static overpressurization was assessed to be at the intersection of the cylinder wall and the dome.

The lower bound of the Grand Gulf distribution over failure pressures was assessed to be approximately twice the design internal pressure. The upper distribution bound was identified with the calculated ultimate material strength of the steel-reinforced concrete containment. A distribution between these bounding points was then developed.

Pressure capacity distributions for the drywell were developed in a similar way. Based on the various potential failure locations in the drywell, the wall was assessed to be the weakest structure and therefore the most likely failure point. The expert determined that the failure criterion was, in terms of the pressure differential across the drywell, independent of the direction of the pressure gradient. At the 5th–95th percentile level, the range of potential failure pressures for the Grand Gulf containment and drywell were 38–72 psig and 50–120 psid, respectively. Figure C.8.1 summarizes the underlying probability distributions.

C.8.3 Treatment in PRA and Results

Within the accident progression event trees (APETs) developed in this study, the probability of containment failure associated with each identified accident progression path was determined by comparing the value of the containment load selected from its distribution to the selected value of the load capability. As part of the overall uncertainty analyses, Monte Carlo methods were used to randomly select values for containment loads and load capacity from their distributions (Ref. C.8.9). Among the elements of each sample member were a containment failure pressure, with the corresponding failure size and failure location. For some plants, more than one load and capacity pair may be selected to simultaneously represent alternative challenges to containment integrity. In the Peach Bottom analysis, for example, one combination would dictate the containment pressure capacity, the failure size, and the failure location corresponding to rapid pressure rises at vessel breach. Another combination would characterize containment response to gradual pressure rises.

Each sample member also contained a series of pressure loads corresponding to load-generating events that occur over the time represented by the accident progression. For each accident progression, the loads and the load capacities in the sample member are compared in a time-ordered way. If the first load exceeds the corresponding pressure capacity, the containment is assumed to fail at that time (unless preceded by containment breach due to some other failure mechanism, such as impulse loading or thermal attack). The location and the size of the failure are specified in the sample member. A relative frequency of 1 (i.e., the split fraction) is then attached to the selected failure size and location conditional upon the prior path taken through the containment event tree. If none of the loads exceeded the pressure capacity, then the containment is assumed to retain its integrity unless breached by some other mechanism.

Based upon the findings of the expert panel on containment structural performance issues, the pressure at which a containment first failed was modeled to be independent of the pressurization rate. However, the ultimate size and location of failure was coupled to the pressurization rate in this study. Since the experts focused largely on the issue of the initial mode of containment failure, their distributions over failure size and location could not be used directly in the treatment of rapid pressure rises. For example, if leak were the initial failure size resulting from rapid pressurization, then, since a leak could not halt further pressurization, whether a more severe breach of the containment would occur subsequently was considered.

To generate probability distributions over ultimate failure modes in the case of rapid pressure rises, both the containment failure pressure and the peak pressure load in any one sample member were considered for each case. If the failure pressure exceeded the peak pressure, failure was assumed not to occur. If, however, the peak pressure exceeded the failure pressure, then a probability distribution over potential failure modes was constructed, which accounted for the possibility that containment rupture or catastrophic rupture could occur after a leak developed and before the peak pressurization level was reached. Sampling from this distribution provided the ultimate failure size and location.

In the Grand Gulf analysis, drywell and containment performance were evaluated in similar ways. Structural performance of the drywell became an issue for conditions in which a pressure differential is established across the drywell-wetwell boundary. This occurs in cases of rapid pressurization (e.g., hydrogen deflagration in the wetwell or loads from vessel breach) where the inertia of water in the suppression pool prevents immediate pressure equalization across the boundary. Within each sample member, the pressure differential for each accident progression was compared to the sampled drywell pressure capacity. High correlations were imposed in sampling from probability distributions over the drywell and the containment pressure capacities since the same basic uncertainties were involved in each.

Table C.8.1 summarizes the failure pressure ranges and the likely modes of containment overpressure failure identified by the expert panel for each plant.

The influence of the containment failure pressure on risk and uncertainty in risk for each plant is dependent ultimately on the predicted severe accident pressure loads and the relative likelihood of containment breach by other mechanisms, such as thermal attack. Since scenarios involving failure or bypass of the containment at or before vessel breach were found to be the dominant contributors to offsite risk, the importance of overpressure failure modes to risk may be characterized in terms of their contribution to bypass and early containment failures. Similarly, the importance of uncertainty in the failure pressure can be evaluated in terms of the impact upon the conditional probability of early containment failure of varying the failure pressure within its range of uncertainty. These importance

Table C.8.1 Containment strength under static pressure loads: summary information.

Plant	Containment Free Volume (Millions of Cubic Feet)	Design Internal Pressure (psig)	Failure Pressure Range(a) (psig)	Sizes/Locations Dominant Failure
Zion	2.6	47	108–180	Leak/rupture in cylinder wall or basemat/wall intersection
Surry	1.8	45	95–150	Leak/rupture near dome/wall intersection
Sequoyah	1.2	11	40–95	Gross rupture of the containment or rupture in the lower compartment
Peach Bottom	0.16 (drywell) 0.12 (wetwell)	56	120–174	Leak at drywell head or leak/rupture of wetwell
			high temp case: (b) 75–150	Leak at drywell head or in wetwell above suppression pool
			high temp case: (c) 6–67	Leak at drywell head or rupture of drywell wall
Grand Gulf	0.27 (drywell) 1.4 (wetwell)	15	38–72	Leak/rupture near dome/wall intersection
		Drywell: 30(d)	50–120 (d)	

(a) 5th–95th percentile range
(b) Drywell temperature at 800°F
(c) Drywell temperature at 1200°F
(d) Drywell/wetwell pressure differential in psi

measures are discussed briefly for each plant. For direct comparison among plants, attention is confined to internal initiating events.

Early containment failure scenarios dominate all offsite risk measures at Zion. The robustness of the Zion containment ensures, however, that the mean frequency of early containment breach conditional on core damage is small (approximately 1%). Ten percent of early containment failures are due to overpressure, the remainder being associated with in-vessel steam explosions (see Section C.9) and pre-existing containment isolation failures. Variations in the failure pressure within its range of uncertainty result in a minimal change to the risk profile at Zion, given the high strength of the containment relative to anticipated loads.

Similarly, the Surry containment appears to be extremely robust. Its mean frequency of early failure conditional on core damage is less than 1 percent. Accident scenarios involving bypass of the containment dominate all offsite risk measures while early containment failures contribute approximately one-quarter or less. Less than 60 percent of early failures are associated with containment overpressure. The remainder result from in-vessel steam explosions. As for Zion, variation of the containment failure pressure within its reasonable range of uncertainty would be expected to result in minimal change in the risk profile at Surry because of the high strength of the containment.

At Sequoyah, approximately 73 percent of mean early fatality risk results from scenarios involving bypass of the containment; the remaining mean early fatality risk is due to early containment failures in loss-of-offsite-power sequences. Early containment failures account for the remaining early fatality risk and for approximately one-half of the latent cancer fatality risk. Containment overpressure accounts for approximately 90 percent of early failures, while direct contact of molten debris with the steel containment, impulse loads from hydrogen detonation, in-vessel steam explosions, pre-existing isolation failures, and ex-vessel steam explosions constitute the remaining 10 percent. The mean frequency of containment failure conditional on core damage is approximately 7 percent. Comparison of this value with the early failure frequency of the large, dry containments reflects the lower pressure capacity of the ice condenser design. The overlap between anticipated pressure loads and the range of containment failure pressures at Sequoyah leads to the conclusion that uncertainty in the overpressure criterion may have an impact on uncertainty in risk.

Scenarios involving early failure of the drywell dominate all offsite risk measures at Peach Bottom both because of the high mean conditional frequency of this mode of failure (approximately 50%) and because of the associated bypass of the suppression pool by radioactive material. The dominant mechanism for early drywell failure is attack of the drywell wall by molten debris on the cavity floor. The mean conditional frequency of early wetwell failure at Peach Bottom is approximately 3 percent. This is dominated by overpressure failures. The contribution to risk of early wetwell failure is minor, however (approximately 1% for the mean estimate of early and latent cancer fatality risk). Consequently, uncertainty in the failure pressure level at Peach Bottom has minimal influence upon uncertainty in risk.

At Grand Gulf, overpressurization is the dominant mechanism for failure of the containment and for failure of the drywell. Scenarios involving early failure of both the containment and the drywell are the principal contributors to all offsite risk measures. The mean frequency of these scenarios conditional upon core damage is approximately 20 percent. Variation of the containment failure pressure within its estimated range of uncertainty has a minimal impact on the performance of the Grand Gulf containment, given its low structural strength relative to the anticipated pressure loads, principally from hydrogen deflagration.

Considerable overlap exists between the range of drywell failure pressures and the range of anticipated pressure loads on the Grand Gulf drywell. The assumption that the drywell failure pressure lies toward the lower end of its uncertainty range could, therefore, result in a significant increase in mean offsite risk. The assumption of a high drywell failure pressure would not be expected to decrease risk significantly, however, since additional mechanisms exist for early failure of the drywell wall. These mechanisms involve pedestal collapse at the time of vessel breach, due either to overpressure or an ex-vessel steam explosion (see Section C.9), and subsequent failure of the drywell wall, due to damage incurred by penetrating pipes.

In summary, containment failures due to overpressure are significant contributors to risk at all plants except Peach Bottom. Uncertainty in structural failure pressure has the potential to significantly influence uncertainty in risk only for the Sequoyah containment and the Grand Gulf drywell. For other containment structures, there is limited overlap between the range of anticipated pressure loads and the uncertainty range of failure pressures.

More details of the treatment of containment structural performance issues in this study can be found in Reference C.8.10.

REFERENCES FOR SECTION C.8

C.8.1 IDCOR Technical Report 10.1, "Containment Capability of Light Water Nuclear Power Plants," July 1983.

C.8.2 S. Sharma et al., "Ultimate Pressure Capacity of Reinforced and Prestressed Concrete Containment," Brookhaven National Laboratory, NUREG/CR-4149, BNL-NUREG-51857, May 1985.

C.8.3 T.A. Butler and L.E. Fugelso, "Response of the Zion and Indian Point Containment Buildings to Severe Accident Pressures," Los Alamos Scientific Laboratory, NUREG/CR-2569, LA-9301-MS, May 1982.

C.8.4 W.J. Pananos and C.F. Reeves, "Containment Integrity at Surry Nuclear Power Station," Stone & Webster Engineering Corp., TP84-13, 1984.

C.8.5 R.A. Dameron et al., "Analytical Correlation and Post-Test Analysis of the Sandia 1:6-Scale Reinforced Concrete Containment Test," Fourth Workshop on Containment Integrity (Arlington, VA), June 14-17, 1988.

C.9 Containment Failure as a Result of Steam Explosions

A steam explosion is the result of rapid transfer of thermal energy from a hot liquid to water over a time scale of the order of milliseconds. Industrial experience has revealed that such explosions have the potential to do significant damage.

The possibility that certain severely degraded reactor core conditions, involving the flow of molten core material into a pool of water in the lower plenum of the reactor vessel, could be conducive to the occurrence of a steam explosion was first assessed in the Reactor Safety Study, WASH-1400. The in-vessel steam explosion scenario may be of particular significance in determining the risk profile of a nuclear power plant since, not only does this phenomenon allow the possibility of catastrophic pressure vessel breach, but the concomitant generation of a missile consisting of the upper head of the vessel could lead to failure of the containment building. An in-vessel steam explosion is a phenomenon, therefore, that could breach the last two barriers between fission products in the core and the ex-containment environment virtually simultaneously. Containment failure resulting from an in-vessel steam explosion was termed "alpha-mode failure" in the Reactor Safety Study.

The sequence of events constituting the hypothesized alpha-mode failure scenario is as follows: Because of either failure of the reactor coolant system boundary or loss of the core heat removal function, the core uncovers. The generation of fission product decay heat and the exothermic oxidation of fuel cladding results in core degradation until liquefaction of the fuel occurs. The relocation of liquefied material to cooler locations near the lower core support plate results in freezing of the fuel and the consequent formation of a crust. This crust supports the molten material that is formed. When the mass of molten material reaches a critical limit, the crust can no longer support it, and the material flows coherently into any water remaining in the lower plenum of the vessel. A steam explosion occurs, generating an upward moving slug of water and molten fuel, which lifts the upper head of the vessel. The reactor head then acts as a missile that perforates any structures above the vessel and, ultimately, penetrates the containment building.

The risk impact of a steam explosion is not confined to the in-vessel phase of a severe accident. When water is present in the reactor cavity or pedestal region at the time of vessel failure, the contact of molten core debris with the water may result in a steam explosion. If the containment geometry is such that an ex-vessel water pool could contact the containment wall or could contact structures that, if disrupted, would result in impairment of the containment function, then ex-vessel steam explosions can have potentially significant risk impact.

It should be noted that an ex-vessel steam explosion may result not only in an impulse load, but also in a quasistatic pressure load on containment structures. Indeed, in assessing pressure loads at vessel breach, the expert panel on containment loading issues accounted for the possibility of load contributions from ex-vessel steam explosions in the development of their probability distributions (see Section C.5). The current section, however, focuses upon challenges to containment structures associated uniquely with the dynamic loads resulting from ex-vessel steam explosions.

The consequences associated with in-vessel and ex-vessel steam explosions, as with other scenarios resulting in early breach of the containment building, are potentially significant. Determining the risk posed by steam explosion scenarios, however, demands not only an evaluation of the resultant consequences but also an assessment of their probability of occurrence. Evaluation of this probability is the focus of the steam explosion issue.

C.9.1 Issue Definition

The range of accident scenarios addressed in the containment analyses of this study includes alpha-mode failure and containment breach due to an ex-vessel steam explosion. The current discussion focuses on the specific accident scenarios and plants for which the steam explosion phenomenon is of the greatest potential risk significance.

C.9.1.1 In-Vessel Steam Explosions

For in-vessel steam explosions, attention is confined to large, dry containment PWR reactor systems. While there currently exists no clear basis for the assumption that other reactor/containment types are less vulnerable to alpha-mode failure, different modes of containment breach were identified in this study as dominating at other plants. The alpha-mode probability distributions developed in the resolution of this issue were applied, however, to all the plants studied. The accident scenarios of particular concern relative to alpha-mode failure are those in which core degradation occurs at low reactor coolant system pressures since experiments indicate that high ambient pressures tend to reduce the likelihood of, although not preclude, the triggering of a steam explosion (Ref. C.9.1).

To quantify the accident progression models in these analyses, the in-vessel steam explosion issue was defined in terms of the probability, conditional upon core degradation at low ambient pressures, of the occurrence of alpha-mode failure.

While numerous small and intermediate scale simulant tests have provided substantial data, and related analytical models of steam explosion phenomena exist (Refs. C.9.2 and C.9.3), there is limited agreement within the technical community regarding the probable phenomena that govern the onset and the effects of an in-vessel steam explosion. Much of the uncertainty about steam explosion phenomenology is associated with the applicability of small and intermediate scale test results (typically involving a melt mass of less than 40 lb) to actual reactor scales and geometries (involving molten fuel masses of up to 280,000 lb in a PWR). Additionally, there is no consensus regarding the appropriateness of the various analytical models that have been used to evaluate the phenomena governing the alpha-mode scenario.

The fundamental energetic condition for alpha-mode failure is that, of the thermal energy contained in the reactor fuel, the amount converted ultimately into the kinetic energy of an upward moving missile is sufficient to permit penetration of the containment building. The maximum total heat content of the fuel elements of a typical PWR is of the order of 105 megajoules (MJ). The energy required to fail the reactor vessel is of the order of 103 MJ, while energies sufficient to fail a large, dry containment building are also of the order of 103 MJ. Simple energy balance considerations, therefore, cannot provide a basis for excluding the alpha-mode failure scenario. Hence, the crucial questions surrounding the alpha-mode issue relate to the ultimate partition of energy, both with respect to its form (thermal, kinetic, strain, and gravitational) and with respect to the mechanical elements of the system in which it resides (molten debris, in-vessel water, reactor internals, containment building shield, and upward and downward moving missiles).

The energy partition question can be addressed through decomposition of the alpha-mode failure scenario into several phases. The first phase involves the transfer of thermal energy from the molten fuel to water in the lower plenum of the reactor vessel. Subissues relating to this phase are the availability of molten core debris and water for interaction over explosive time scales, the geometry of the debris (since this determines the efficiency of the thermal interactions), and the existence of a steam explosion trigger. The second phase involves the generation of an upward moving slug (water, melt, and structural materials) within the vessel. Subissues relating to this phase are the fraction of thermal energy involved in the steam explosion that is converted to kinetic energy, possible failure of the lower head of the vessel thereby relieving the in-vessel explosive pressures, and the distribution of kinetic energy between the upward moving slug and a downward moving slug.

The third phase involves failure of the vessel upper head. Related subissues are the fraction of the initial energy of the slug that is dissipated as strain energy in the upper internal structural components of the vessel (e.g., upper core support plate, control rod drives) and, if the upper head of the vessel fails, the energy of the missile thereby generated. The fourth phase involves the impact of the missile upon the vessel shield where the relevant issue is the associated degree of energy dissipation. The final phase involves failure of the containment building. The crucial issue here is, given the loss of kinetic energy by the missile associated with its ascension to the containment boundary, the capability of the missile to penetrate the containment.

In 1984, a panel of experts was convened to summarize current understanding of steam explosion phenomena and to assess the likelihood of alpha-mode failure. The 13-member Steam Explosion Review Group (SERG) represented substantial cumulative experience in the experimental investigation and the analytic modeling of severe accident phenomena. Findings of the panel were published in June 1985 (Ref. C.9.4). The mandate of the panel also included review and assessment of analytical work undertaken by Berman et al. (Ref. C.9.5), which addressed the likelihood of alpha-mode failure. To encapsulate the spectrum of expert views on the steam explosion issue, this study drew upon both the findings of SERG and the judgment provided by the primary author of the work reviewed by SERG. The views of each participating expert were then weighted equally in arriving at the final characterization of uncertainty in the likelihood of alpha-mode failure.

Prior to their use in this study, members of the SERG panel reviewed the probability distributions relating to alpha-mode failure that, based upon the earlier findings of SERG, had been developed at Sandia National Laboratories. Through consideration of the way in which their findings had been interpreted and of relevant information acquired since publication of the SERG report, the same panel members modified the distributions tentatively developed. These modified distributions provided the basis for this study.

C.9.1.2 Ex-Vessel Steam Explosions

Each plant evaluated in this study was screened for potential vulnerabilities to ex-vessel steam explosions. The containment design assessed to display the most significant vulnerability was Grand Gulf. The scenario of concern in the Mark III design is one in which a steam explosion impulse is delivered to the reactor pedestal through water on the drywell floor. The likelihood of a deep water pool in the drywell at Grand Gulf is high during the course of a severe accident. A dominant mechanism for this is the expulsion of water from the suppression pool as a result of pressurization of the wetwell through hydrogen deflagration. Upon receiving the explosion impulse, the pedestal collapses, resulting in failure of the drywell wall due either to impact by the unsupported vessel or damage by the penetrating steam line and feedwater pipes. Loss of the drywell wall then permits bypass of the suppression pool by fission products in the event of pre-existing or subsequent containment failure. The potentially significant risk impact of drywell failure at Grand Gulf stems from the relatively high likelihood of containment overpressure either prior to or following vessel breach.

The Zion and Surry containments were not assessed to have significant vulnerability to impulse loads from ex-vessel steam explosions since water in the cavity would not directly contact structures that are both vulnerable and essential to the containment function. Initial assessment of the Peach Bottom and Sequoyah containment designs identified potential vulnerability to ex-vessel steam explosions, associated principally with pedestal collapse (Peach Bottom) and seal table disruption or vessel dislocation (Sequoyah). However, scoping shock wave hydrodynamics calculations and application of underwater impulse correlations (based on Ref. C.9.6) revealed minimal threat to these containments from ex-vessel steam explosions.

Attention was focused therefore on the Grand Gulf containment. The ex-vessel steam explosion issue was couched in terms of three parameters:

- The likelihood (conditional frequency) of an ex-vessel steam explosion occurring conditional upon the presence of water in the cavity at vessel breach.
- The likelihood of pedestal failure conditional upon the occurrence of a steam explosion.
- The likelihood of drywell failure due to collapse of the pedestal.

Evaluation of these parameters was based upon impulse loading calculations performed at Sandia and upon the elicitation of judgments from the expert panel on containment structural performance issues (see Section C.8).

C.9.2 Technical Bases for Issue Quantification

C.9.2.1 In-Vessel Steam Explosions

The approach adopted by most of the experts in determining the probability of alpha-mode failure was decomposition of the scenario into a sequence of events and the assignment of a probability, or a range of probabilities, to each constituent event (Ref. C.9.4). These events constituted elements of the four phases of alpha-mode failure defined in the previous section. The product of event-level probabilities was then equated with the probability of alpha-mode failure. Other experts adopted variant approaches in which probabilistic judgment was exercised directly at the level of the compound-event alpha-mode failure, or in which probability distributions reflecting uncertainty in relevant physical parameters were propagated through deterministic models to determine the probability of alpha-mode failure. These latter approaches do not permit straightforward extraction of the probabilities associated with the primary-level events, and direct comparison of the event likelihoods assessed by all the experts is therefore difficult. The range of views in each phenomenological area is described here, and direct comparisons of expert-specific probabilities are made where possible.

Initial Conditions

For the energy involved in a steam explosion to be commensurate with the energies associated with vessel and containment failure, sufficient amounts of molten material and water must be available to participate. Additionally, a coherent pour of melt into the water is required to ensure maximal participation of the melt available over explosive time scales. While some experts assigned a low likelihood to the required initial conditions, based on the premise that the meltdown process would involve the incoherent dripping of molten material into the lower plenum, others assigned probabilities in the range of 0.75 to 1.0 for the occurrence of the required conditions. These higher probabilities were based generally upon identification of a scenario in which a crust of refrozen melt at the lower core support plate breaks suddenly, permitting the coherent release of molten material into the lower plenum. Substantial uncertainty was identified regarding the process of the core degradation and fuel relocation.

Molten Core/Water Mixing

The degree of interpenetration between the melt and water in the lower plenum determines the efficiency of thermal interaction between the two media. Currently no widely accepted model of molten fuel/water mixing under severe accident conditions exists. While efficient mixing has been observed in several small and intermediate scale tests (Ref. C.9.7), various experts argued that scaling effects prevent the conclusion that efficient mixing would occur in full-scale reactor geometries. One analytical model involves a process in which hydrodynamic instabilities break up the fuel jet as it pours into the lower plenum. Rapid steam production (although not as rapid as that associated with a steam explosion) then expels water from the mixing region, thereby severely limiting the potential for effective mixing.

This process of jet fragmentation and fluidization as a result of hydrodynamic instabilities was not accepted by all participating experts, however. A process in which a blanket of steam forms around the jet body, thus limiting access by water to the jet, was the basis for an alternative model. This model confines fragmentation of the melt to the leading edge of the jet, thereby reducing the potential for mixing.

The initial mixing of melt and water as a condition for large-scale steam explosions was questioned by some experts. It was observed that, even for an initial configuration involving minimal fragmentation of the melt, the occurrence of small steam explosions sufficient to disrupt the melt could create boundary conditions conducive to the onset of a larger explosion. While no model of the net effects of preliminary steam explosions upon in-vessel melt/water configurations exists, the observation of multiple steam explosions in small-scale tests prevents the exclusion of such scenarios from consideration. Where probabilities were assigned specifically to the event of large-scale mixing conditional upon a coherent melt pour, they ranged from 10^{-2} to 0.3.

Explosion Trigger

While large-scale mixing of melt with water provides boundary conditions required for significant thermal interactions between the two media, the question of whether that interaction takes the form of a steam explosion is dependent upon whether a trigger is present. The mechanism for triggering a steam explosion is poorly understood. One model involves the onset of oscillations in the steam film barrier isolating a molten fuel fragment from water. Where these oscillations permit direct contact of fuel with water, a trigger occurs. Experiments reveal that the existence of a trigger is extremely sensitive to initial conditions (Ref. C.9.1) and that triggering becomes less likely at high ambient pressures. For low reactor coolant system pressure scenarios, those experts who provided a probability relating specifically to the occurrence of a trigger assessed it to be equal to unity.

Explosion Efficiency

Fundamental thermodynamic factors limit the efficiency with which the thermal energy involved in a steam explosion may be converted into kinetic energy. While this theoretical, maximum conversion ratio is in the range of 40–50 percent (based upon Hicks-Menzies conditions), the value appropriate for a reactor configuration depends upon the constraints provided by the internal vessel geometry. The experts agreed that conversion ratios of up to 15 percent are possible.

Slug Formation and Vessel Head Impact

The primary mechanism for transmittal of the kinetic energy generated by the steam explosion to the upper head of the vessel is the formation of an upward-moving slug composed of molten fuel, water, and structural material. The resultant impulse upon the upper head could also be supplemented by the transmittal of an impulse from the lower core support plate, through the core barrel, to the upper head flange. The possibility also exists that the pressures generated by the steam explosion could result in failure of the lower head, thereby venting explosive pressures in the vessel and reducing the energy delivered to the upper head. Much of the uncertainty associated with the likelihood of upper head failure relates to the distribution of material within the vessel. For example, uncertainty regarding the fraction of the molten fuel and water inventories above the trigger location was taken into account explicitly by some of the experts in determining the possible mass and composition of the upward-moving slug. Uncertainty was identified also in estimating the fraction of the slug's kinetic energy dissipated as strains within the upper internal components of the vessel. Expert-specific point-value probabilities assigned to the event of upper head failure and the generation of a missile capable of failing containment (conditional upon significant thermal energy conversion) ranged from 10^{-2} to 1, although one expert provided a probability range extending from 10^{-4} to 1.

Vessel Head Failure and Containment Breach

Following bolt failure at the upper head flanges, development of the alpha-mode failure scenario involves impact of the dislocated vessel head against the missile barrier positioned above the vessel. The barrier is perforated, thus attenuating the energy of the missile. The missile continues to rise, expending kinetic energy to acquire gravitational potential energy, and ultimately impinges upon the containment wall. Some of the experts based their assessments of the likelihood of containment failure upon detailed structural calculations. Those experts who assigned probabilities to individual events within the alpha-mode scenario generally absorbed the structural uncertainties into their assessment of the probability of vessel head failure by defining such a failure as one that generates a missile capable of penetrating the containment.

C.9.2.2 Ex-Vessel Steam Explosions

The scenario of concern at Grand Gulf is one in which molten debris is released from the breached vessel into a deep water pool (about 7 meters) on the drywell floor. A steam explosion is triggered, which delivers an impulse load to the reactor pedestal. The pedestal collapses leaving the reactor vessel unsupported. The drywell wall then fails as a result of damage caused by the penetrating vessel piping or by direct contact by the vessel. The accident progression models developed for this study decompose the ex-vessel steam explosion scenario into three phases: (1) occurrence of the steam explosion, (2) subsequent failure of the pedestal, and (3) subsequent failure of the drywell wall.

Occurrence of Steam Explosion

The parameter evaluated for this phase of the scenario is the fraction of occasions upon which a steam explosion would be triggered conditional upon the release of molten debris from the vessel to an underlying water pool. An estimate of 0.86 for this parameter was based upon intermediate-scale experimental results in which 86 percent of tests involving the release of molten thermite into water at low ambient pressures resulted in a significant steam explosion (Ref. C.9.7).

Failure of Pedestal

Assessment of the dynamic load capacity of the pedestal was based upon adaptation of information elicited from the expert panel on containment structural performance issues regarding the strength of the drywell wall and upon supplementary information provided by a structural expert from Sandia, who had been a member of the original panel. The expert aggregate probability distribution over potential failure impulse levels for the Grand Gulf drywell wall extends from 3.5 to 18 psi-s. Since the pedestal and the drywell wall at Grand Gulf are both composed of reinforced concrete of similar thickness, the dynamic load capacity of the pedestal was assumed to be comparable to that of the drywell wall. While the impulse capacity range for the drywell wall was based on the assumption of loads associated with hydrogen detonations, the similarity in impulse duration between steam explosion and gas detonation loads (typically milliseconds) was assessed to warrant adoption of the range, given its broadness. Supplementary evaluations of the pedestal impulse capacity by an internal Sandia expert on containment structural performance issues confirmed the appropriateness of this range.

Estimation of the impulse delivered to the pedestal conditional upon the occurrence of a steam explosion was based on the Similitude Equations (Ref. C.9.6). These equations, reflecting the correlation between underwater explosion size, distance from the explosion center, and impulse level, were adopted to determine the relationship between the mass of debris participating in the steam explosion and the impulse to the pedestal. Calculations revealed that, with less than 1 percent of the core participating in the explosion, the impulse to the pedestal would reach the lower edge (i.e., 3.5 psi-s) of the uncertainty range over pedestal failure loads. The upper edge of the range (18 psi-s), based upon shock wave hydrodynamics calculations, would be reached if 10 percent of the core participated in the explosion. It was noted that release from the vessel of 10 percent of the core corresponds to the 90th percentile level of the aggregate distribution over core release levels at BWR vessel breach.

It was concluded that, conditional upon the trigger of a steam explosion in the Grand Gulf drywell, failure of the pedestal is credible. To reflect maximal uncertainty regarding the fraction of ex-vessel steam explosions that would result in pedestal failure, a uniform probability distribution was assigned to the interval between the fraction zero and the fraction 1.

Failure of Drywell Wall

The final parameter associated with the ex-vessel steam explosion issue at Grand Gulf is the fraction of occasions that collapse of the pedestal results in failure of the drywell wall. This question was addressed by two members of the expert panel on containment structural performance issues. Based upon their engineering judgment and supporting hand calculations, each expert provided a single point estimate of the required parameter. These two point estimates were averaged to generate a single estimate for input to the Grand Gulf risk model. This average was 0.17.

C.9.3 Treatment in PRA and Results

C.9.3.1 In-Vessel Steam Explosions

Of the 14 experts (13 in SERG and one additional expert as discussed in Section C.9.2) participating in the steam explosion evaluation process, 12 provided probabilities for alpha-mode failure conditional upon core damage in a PWR at low reactor coolant system pressure. Two of these experts collaborated in the generation of probabilities; thus, their results reflected a single approach. Eleven independent sets of probabilities were ultimately provided.

The extreme sensitivity of the onset of a steam explosion to prevailing physical boundary conditions (Ref. C.9.7) provides a strong basis for treating alpha-mode failure as a stochastic phenomenon. This means that, conditional upon a specified plant damage state, the associated range of possible physical conditions within the vessel results in the situation that alpha-mode failure would occur only on some fraction of occasions. The probabilities assigned by the participating experts were therefore interpreted as estimates of the relative frequency of alpha-mode failure, i.e., as the fraction of severe accidents resulting in failure of the containment building due to an in-vessel steam explosion. In conformance with the approach to uncertainty characterization used in this study, probability distributions were constructed over these relative frequencies on an expert-specific basis. Aggregation over the expert distributions then provided the net representation of uncertainty regarding the frequency of alpha-mode failure.

Construction of the expert-specific distributions was based upon identification of a best-estimate frequency with the median of the distribution, identification of an upper estimate with the 95th percentile, and identification of a lower estimate with the 5th percentile. Use of an entropy-maximization algorithm (Ref. C.9.8) in conjunction with these distribution constraints ensured that uncertainty was appropriately preserved in formulation of the expert-specific probability distributions. Details of this approach to the use of probabilistic information were reviewed by each participating expert.

Figure C.9.1 displays the cumulative probability levels that bound the expert-specific distributions. These bounds exclude two experts who assessed the likelihood of alpha-mode failure to be so low that they assigned zero probability to the scenario. This figure also displays the final aggregate distribution based upon the 11 expert-specific distributions (including the two experts who assigned zero probabilities). Note that the aggregate distribution is not completely encapsulated by the bounding distributions because of the effect of the two experts who assigned zero probabilities. It can be seen that the median relative frequency of alpha-mode failure for the aggregate distribution is approximately 4E–5. That is, there is equal net probability that less than, or more than, four in a 100,000 core damage scenarios occurring at low ambient pressure will result in alpha-mode failure of the containment building. It can be seen also that the greatest median frequency proposed by any one expert was of the order of one in a 100, while the smallest finite median frequency was of the order of one in 100,000. Hence, while no consensus existed regarding details of the phenomenology of steam explosions, the conclusion that, at the median level, alpha-mode failure is unlikely was shared by each participating expert. In each plant study, the aggregate distribution displayed in Figure C.9.1 provided the basis for sampling alternative values of the frequency of alpha-mode failure conditional upon core degradation at low pressure. The frequency of alpha-mode failure conditional upon core degradation at high reactor coolant system pressure was set at one order of magnitude below the frequency associated with the low-pressure case in each sample member. This reflects the experimental observation that high ambient pressures tend to reduce the likelihood of a steam explosion trigger (Ref. C.9.1).

Although the median relative frequency of the alpha-mode scenario is low, the high relative likelihood of core degradation at low reactor coolant system pressures in PWRs (see Section C.6) has the effect of highlighting the alpha-mode scenario as a mechanism for early failure of the containment, especially at Surry and Zion for which other early failure mechanisms are of low likelihood. At Surry, for example, while about half of the mean frequency of early containment failure conditional upon core damage is

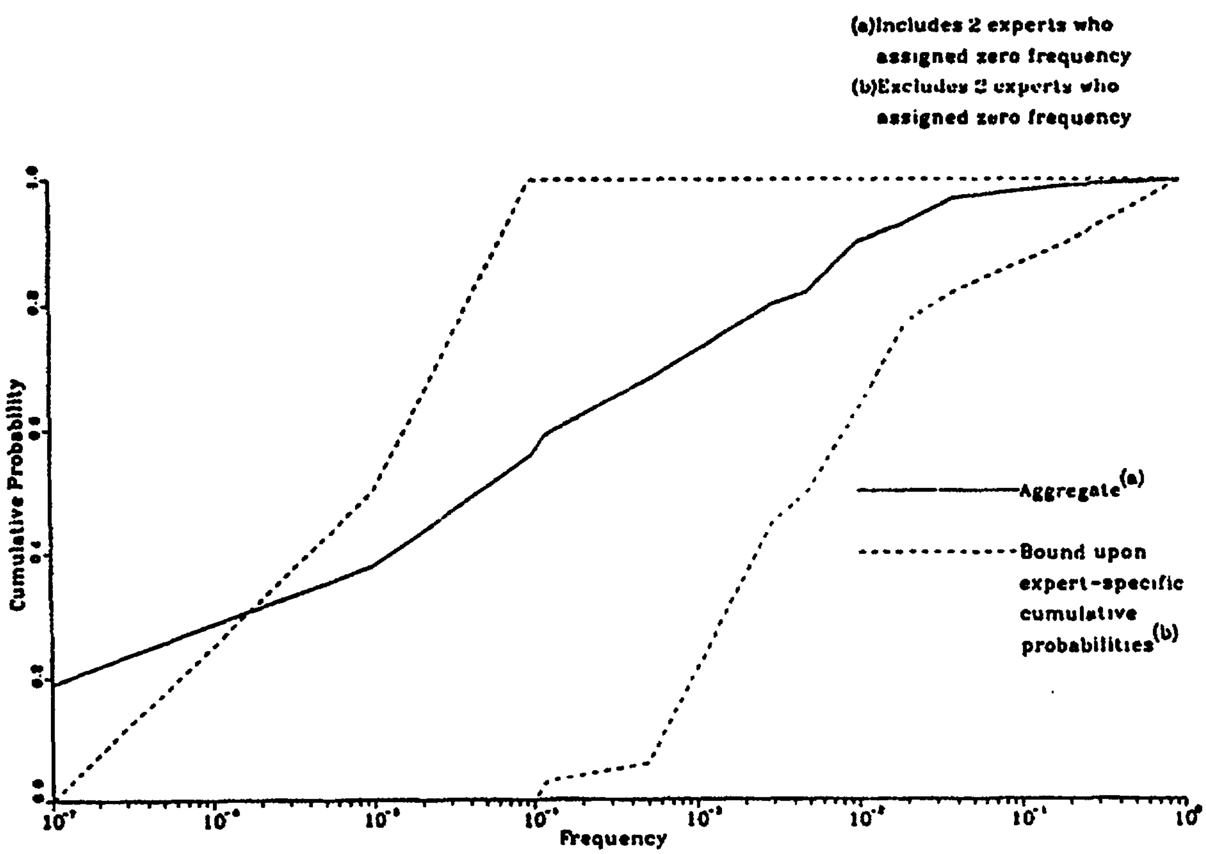


Figure C.9.1 Frequency of alpha-mode failure conditional upon core damage.

associated with the alpha-mode scenario, this total mean frequency is less than 10^{-2}. For the pressure-suppression containment types, the likelihood of alpha-mode failure is low relative to other containment failure mechanisms.

C.9.3.2 Ex-Vessel Steam Explosions

The three parameters required to characterize the likelihood of drywell failure due to ex-vessel steam explosions at Grand Gulf are defined in Section C.9.2. Each of these was an input to the Grand Gulf accident progression event tree. Conditional upon vessel breach and wet cavity conditions, the fraction of occasions upon which the drywell fails as the result of an ex-vessel steam explosion was equated with the product of these three parameters. The fraction of steam explosions leading to pedestal failure was identified as the most significant source of uncertainty because of uncertainty regarding the amount of molten material that would participate in the explosion. In the Monte Carlo uncertainty analysis, this parameter was sampled from a uniform probability distribution over the interval of fractions zero to 1. Based upon the parameter values described in the Section C.9.2, it can be deduced that the mean fraction of occasions on which failure of the vessel, in wet cavity conditions, results in breach of the drywell wall as a result of a steam explosion is approximately 0.07.

While Grand Gulf is the containment for which the threat posed by ex-vessel steam explosions is the most significant, the relative importance of this mechanism for drywell failure compared to others is small. Conditional upon core damage at Grand Gulf, less than 10 percent of early drywell failures result from an ex-vessel steam explosion. The dominant causes of drywell failure are associated with pressurization of the drywell atmosphere at the time of vessel breach.

More details of the treatment of steam explosions in this study can be found in Reference C.9.9.

REFERENCES FOR SECTION C.9

C.9.1 D.E. Mitchell, M.L. Corradini, and W.W. Tarbell, "Intermediate Scale Steam Explosion Phenomena: Experiments and Analysis," Sandia National Laboratories, NUREG/CR-2145, SAND81-0124, October 1981.

C.9.2 T.G. Theofanous et al., "An Assessment of Steam-Explosion-Induced Containment Failure," *Nuclear Science and Engineering*, 97, 259, 1987.

C.9.3 M. Berman, "A Critique of Three Methodologies for Estimating the Probability of Containment Failure Due to Steam Explosions," *Nuclear Science and Engineering*, 96, 173, 1987.

C.9.4 Steam Explosion Review Group, "A Review of the Current Understanding of the Potential for Containment Failure from In-Vessel Steam Explosions," USNRC Report NUREG-1116, June 1985.

C.9.5 M. Berman et al., "An Uncertainty Study of PWR Steam Explosions," Sandia National Laboratories, NUREG/CR-3369, SAND83-1438, July 1984.

C.9.6 R.H. Cole, *Underwater Explosions*, Princeton University Press, 1948.

C.9.7 B.W. Marshall et al., "Recent Intermediate-Scale Experiments on Fuel-Coolant Interactions in an Open Geometry," *Proceedings of the ANS/ENS International Topical Meeting on Thermal Reactor Safety* (San Diego, CA), February 1986.

C.9.8 S.D. Unwin, "IMPAGE. An Information Theory-Based Probability Assignment Generator. Brief Code Description and User's Guide," Brookhaven National Laboratory, Technical Report A-3829, August 1987.

C.9.9 F. T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

C.10 Source Term Phenomena

The magnitude and timing of release of radioactive material from a nuclear reactor in a severe accident depend on a variety of thermal, hydraulic, and mechanical processes, as well as the chemistry and physics of fission product release and transport. Uncertainties in core melt progression, containment loads, and containment performance produce uncertainties in the release to the environment. In this study, however, each accident progression bin represents a particular state of melt progression and containment performance. Thus, uncertainties associated with how and when the containment fails are reflected as uncertainties in the likelihoods of the accident progression bins rather than in the parameters that describe the release to the environment. This section addresses the phenomena that affect the magnitude of release of the elemental groups in an accident progression bin, not its likelihood. These phenomena relate directly to the chemistry and physics of fission product release and transport. Some uncertain aspects of core melt progression (e.g., the time-temperature history of the fuel as fission products are being released) are included implicitly in the assessment of uncertainties, however.

C.10.1 Issue Definition

Following the Three Mile Island accident, the NRC established an Accident Source Term Program Office to evaluate the realism with which the analytical methods available at that time could predict severe accident source terms. In 1981, the NUREG-0772 report, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents" (Ref. C.10.1), reviewed the state of the art and identified research needs in a number of areas. In response to these needs, the NRC undertook a substantial effort to direct severe accident research toward development of improved methods of analysis supported by a more comprehensive data base (Ref. C.10.2).

In 1986, the NRC published the NUREG-0956 report, "Reassessment of the Technical Bases for Estimating Source Terms" (Ref. C.10.3). One of the purposes of the present study was to develop a perspective on how changes in source term methodology, as represented in NUREG-0956, affect estimated risk to the public from severe accidents. In their review of NUREG-0956, the American Physical Society (APS) identified the principal areas of uncertainty in severe accident analysis and made recommendations for future research. The research needs that relate directly to the release and transport of radioactive materials are listed in Table C.10.1. Additional research has been performed in each of these areas subsequent to the APS review, and the results have been incorporated into the current study.

Table C.10.1 APS recommendations for source term research (Ref. C.10.3).

1. Vaporization of low volatility fission products
2. Release of refractory materials in core-concrete interaction
3. Transport of radionuclides through reactor
4. Tellurium behavior
5. Release of volatile forms of iodine
6. Generation mechanisms for aerosols
7. Effectiveness of suppression pools and ice beds
8. Growth and deposition of aerosols
9. Change of sequence by fission product heating
10. Intercomparison of aerosol codes
11. Aerosol deposition on pipes
12. Integrated severe accident code

The simplified source term codes, XSOR,* which were developed to support the uncertainty analysis for this study, represent source term processes as integral parameters such as release fractions, decontamination factors, and transmission factors. The chemistry and physics of these processes are contained in the mechanistic codes against which the XSOR parameters are benchmarked. The same parametric representation of source terms will be used in this section to discuss source term uncertainties:

*A separate code was written for each of the plants: SURSOR, SEQSOR, ZISOR, PBSOR, and GGSOR.

1. Fraction of initial inventory of species released from the fuel prior to vessel breach,

2. Fraction of release from fuel that transports from the reactor vessel into the containment,

3. Fraction of initial core inventory released during core-concrete interaction,

4. Fraction of source term to the containment atmosphere that subsequently is released to the environment,

5. Decontamination factors for engineered safety features or water pools,

6. Fraction of species deposited in reactor coolant system that is subsequently revaporized, and

7. Fraction of iodine in suppression pools or water pools that is subsequently evolved.

C.10.2 Technical Bases for Issue Quantification

The status of each of the major areas of severe accident uncertainty identified by the APS has been reviewed in Appendix J of draft NUREG-1150 (Ref. C.10.4). A "Review of Research on Uncertainties in Estimates of Source Terms from Severe Accidents in Nuclear Power Plants," (Ref. C.10.5), was also undertaken by a panel of eminent scientists under the leadership of Dr. H. Kouts. These reports discuss areas of source term uncertainties qualitatively and identify needs for further research.

In the current study, it was necessary to develop a quantitative characterization of the uncertainties in source term phenomena. A panel of experts in source term phenomena was assembled to develop the uncertainty distributions for the most important phenomena. Table C.10.2 identifies the experts and lists the issues elicited. The bases on which the experts made their judgments differed. In each case, computer analyses and experimental data were available to the experts. In many instances, the experts performed their own calculations. Each expert provided documentation on the rationale supporting his elicitation. The variety of considerations by the experts is too broad to reproduce in this appendix for the source term issues. The reader is referred to Reference C.10.6 for a detailed discussion of the bases for the elicitations.

C.10.3 Treatment in PRA and Results

In-Vessel Release

Within the range of uncertainties, the experts were not able to distinguish between the magnitude of release for different accident sequences other than for different degrees of zirconium oxidation. Thus, distributions for only four cases were developed: PWR-high oxidation, PWR-low oxidation, BWR-high oxidation, and BWR-low oxidation. The results for the four cases are similar. Figure C.10.1 illustrates the distribution obtained for the PWR case with low zirconium oxidation. The uncertainty range* for the release of iodine and cesium is from approximately 10 percent to 100 percent of the initial core inventory, for tellurium from 1 percent to 90 percent, for barium and strontium from very small to 50 percent, and for the involatiles from very small to a few percent.

In-Vessel Retention

Four cases were considered for the PWRs: setpoint pressure (2500 psia), high pressure (1200-2000 psia), intermediate pressure (150–600 psia), and low pressure (50–200 psia). Three cases were considered for the BWRs: fast (e.g., short-term station blackout), high pressure; fast, low pressure; and slow (e.g., long-term station blackout), high pressure.

In all cases 100 percent of the noble gases were assumed to escape from the reactor coolant system. For the PWR, the estimated fractional releases at setpoint pressure are typically small, as illustrated in Figure C.10.2. For all species the range is from 0.001 percent to 80 percent, with median fractional release of 9

*5th percentile and 95th percentile values are used to characterize the range.

Table C.10.2 Source term issues.

Technical Experts
P. P. Bieniarz, Risk Management Associates
A. Drozd, Stone & Webster Engineering Corp.
J. A. Gieseke, Battelle Columbus Division
R. E. Henry, Fauske and Associates
T. Kress, Oak Ridge National Laboratory
Y. H. Liu, University of Minnesota
D. Powers, Sandia National Laboratories
R. C. Vogel, Electric Power Research Institute
D. C. Williams, Sandia National Laboratories
Source Term Issues Elicited
1. In-vessel fission product release
2. Ice condenser DF—Sequoyah
3. Revolatilization (from RCS/RPV) after vessel breach
4. Core-concrete interaction (CCI) release
5. Release from containment
6. Late sources of iodine—Grand Gulf and Peach Bottom
7. Reactor building DF—Peach Bottom
8. Releases during direct containment heating

percent for iodine and 3 percent for the bulk of the other aerosols. As illustrated in Figure C.10.3, at low pressure the range for fractional release is 12 percent to 99 percent for the iodine, with a median of 50 percent, and from 4 percent to 99 percent for the bulk of the other aerosols. The high and intermediate pressure cases fall between the system setpoint case and low-pressure case.

The distributions for the BWR cases are similar to those described for the PWR cases. The distributions for the two high-pressure cases are similar to the distribution for the PWR setpoint pressure case in that the majority of the distribution indicates a small release, but at the upper end of the range the release is substantial (~80 percent). The distribution for the low-pressure case is similar to that of the PWR low-pressure case.

Core-Concrete Release

Distributions were obtained for 16 different cases. Zion, Sequoyah, and Surry were each treated separately. A common distribution was obtained for Peach Bottom and Grand Gulf because the same type of concrete was used in the construction of the two plants. For each plant, four scenarios were considered for a wet or dry cavity and high or low zirconium oxidation.

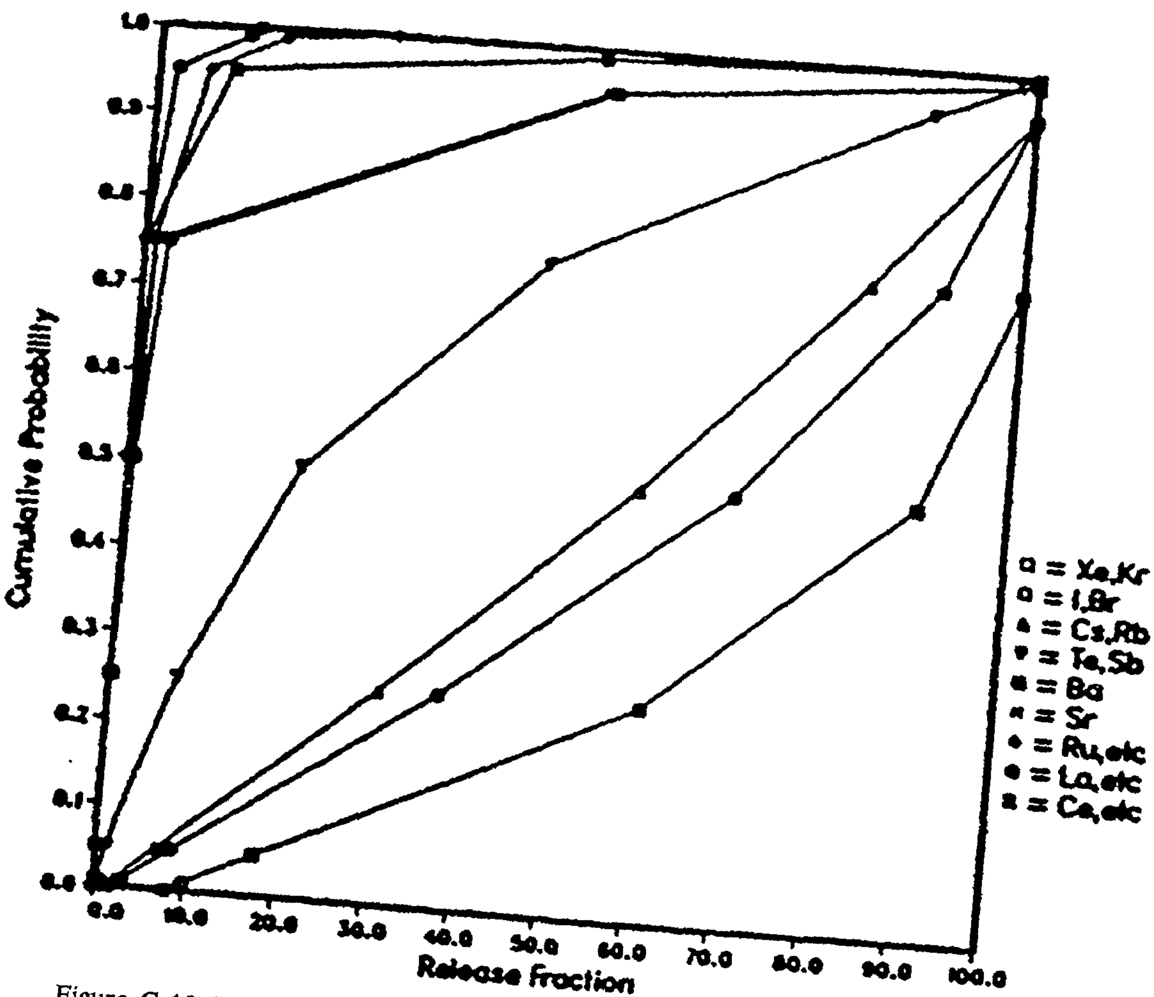


Figure C.10.1 In-vessel release distribution, PWR case with low cladding oxidation.

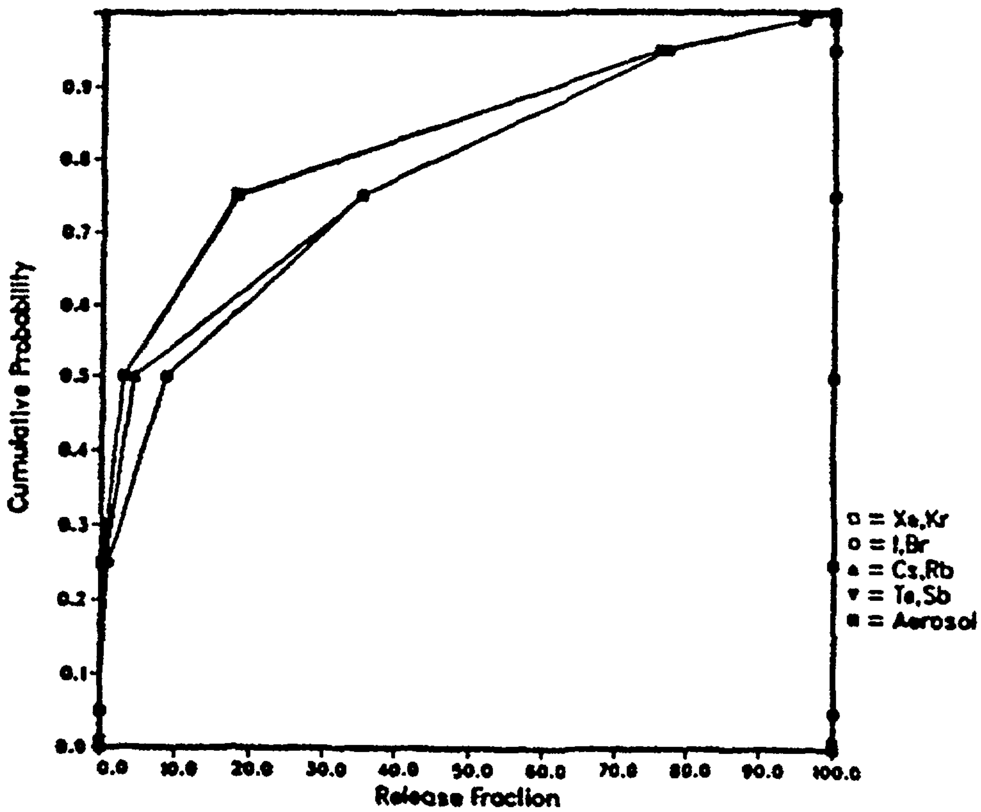


Figure C.10.2 RCS transmission fraction, PWR case at system setpoint pressure.

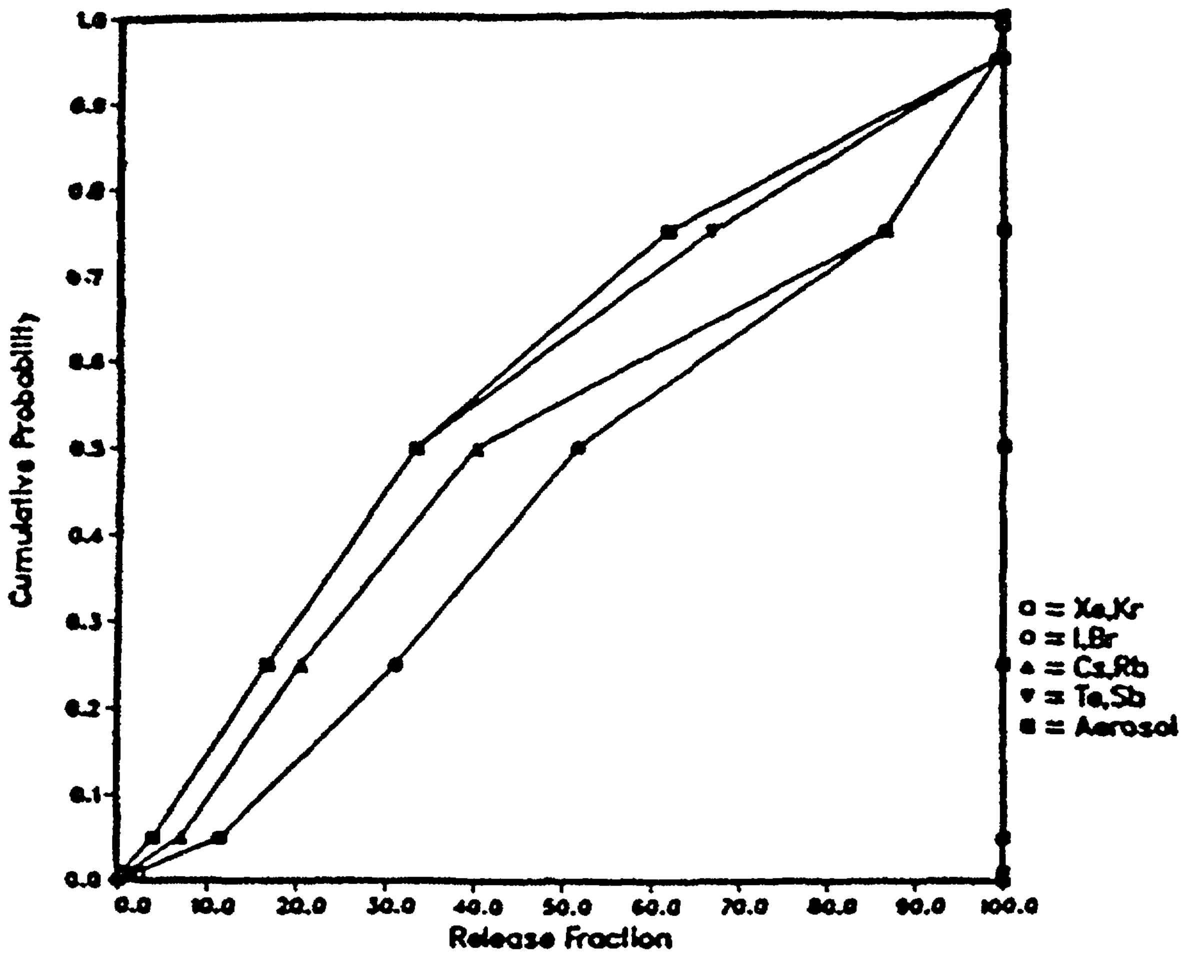


Figure C.10.3 RCS transmission fraction, PWR case with low system pressure.

Release fractions for five elemental groups were evaluated: tellurium, strontium, barium, lanthanum, and cerium. The uncertainty distributions obtained from the experts indicate broad ranges of uncertainty. The fractional release of tellurium is likely to be quite large. Median values of the release fraction for the different cases typically are approximately 50 percent, and the upper bound release is approximately 90 percent. The lower bound release fractions vary from 2 percent to 10 percent. Barium and strontium are indicated to be substantially less volatile than tellurium but at the upper end of the uncertainty range could also lead to substantial release. Median values for the release of these species vary from 2 percent to 5 percent for the different cases. The release fractions for the lanthanum and cerium groups are substantially smaller. Median values are typically less than 0.1 percent. Upper bound values are typically less than 10 percent of the inventory, except for the cerium group release in the BWR cases, which extends to 20 percent.

Containment Release Fraction

This factor is defined as the fraction of radioactive material released to the containment atmosphere that eventually leaks to the environment. Eighteen different distributions were developed associated with different plant types, whether the release was from fuel in-vessel or ex-vessel, the timing of containment failure, the mode of containment failure, and in some cases whether the suppression pool was saturated.

It is difficult to generalize the results because of the variety of containment conditions analyzed. In some early failure cases, however, the transmission fraction is quite high for the entire range of uncertainty. In an early containment failure case for the Sequoyah plant, in which the failure leads to bypass of the ice bed, the fractional release of radioactive material ranges from 25 percent to 90 percent of the material released from the reactor coolant system.

Decontamination Factors for Engineered Safety Features

Distributions for decontamination factors (DFs) were developed for suppression pools, ice condensers, overlaying water pools (for the core-concrete interaction release), containment sprays, and for the Peach Bottom reactor building. Only the ice condenser and Peach Bottom reactor building DFs were submitted to the panel of experts for quantification. The NUREG-1150 analysis staff developed the distributions for the other factors.

Although the range of uncertainty in the water pool DFs is large, water pools are sufficiently effective in decontamination that the resulting source terms are not dominant contributors to the plant risk. In comparison to the suppression pool DF, the ice condenser is not as effective in the removal of radioactive material. Four cases were considered by the experts:

Case 1: Air-return fans on, delayed containment failure, multiple passes through ice bed, no direct containment heating, low steam fraction.

Case 2: Air-return fans on, early containment failure, single pass through ice bed, no direct containment heating, low steam fraction.

Case 3: Air-return fans off, single pass through ice bed, no direct containment heating, high steam fraction.

Case 4: Direct containment heating, single pass through ice bed, high gas velocity, high steam fraction.

For a typical case, with multiple passes through the bed, low steam content, and without high-pressure melt ejection, the range of DF was from 1.2 to 20 with a median value of 3.

Distributions were developed for six cases for the DF of the Peach Bottom reactor building. The variations in conditions were associated with combinations of the mode of drywell failure (rupture, shell meltthrough, or head seal leakage) and whether the suppression pool was saturated. For the head seal leakage cases, the leak is into the refueling bay rather than into the reactor building and smaller DFs were assessed. For a typical case involving drywell rupture with the suppression pool subcooled, the range of DF is 1.1 to 10 with a median value of 3.

Revolatilization from Reactor Coolant System

Radioactive material deposited on surfaces within the BWR reactor vessel and PWR reactor coolant system can be reevolved after vessel failure because of the self-heating of the radioactive material. For the PWRs, two cases were considered: one hole in the reactor coolant system or two holes in the reactor coolant system. The latter case offers the opportunity of a chimney effect and a greatly different environment. For the BWRs, two cases were also considered: high drywell temperatures or low drywell temperatures. Variations to these cases were considered by some of the experts.

Distributions were developed for three elemental groups: iodine, cesium, and tellurium. In all cases, the fractional release is greatest for the iodine and least for the tellurium. Figure C.10.4 illustrates the distribution for the release of iodine for the PWR case with two holes in the reactor coolant system. The range for this case, which produced the greatest release, is from 0 percent to 70 percent, with a median release of 20 percent. Median releases of iodine for the other cases varied from 3 percent to 10 percent. Cesium release fractions were comparable to the iodine values but slightly reduced. The median release of tellurium was 0 percent in all cases, but the upper bound varied from 20 percent to 60 percent. This skewed distribution is indicative of a general belief by the experts that there will be little or no revaporization of tellurium but with a recognition that substantial revaporization cannot be ruled out.

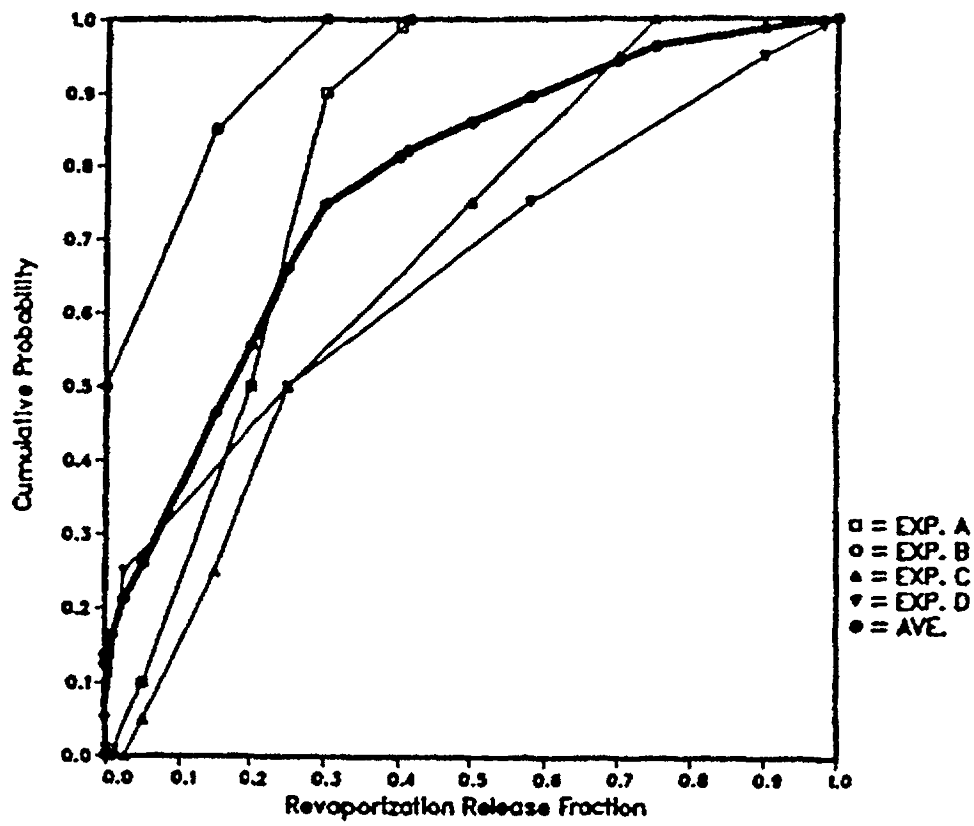


Figure C.10.4 Revaporization release fraction for iodine, PWR case with two holes.

Late Release of Iodine

This issue addresses the potential for the long-term release of iodine from suppression pools and reactor cavity water. Four cases were considered: a subcooled suppression pool, a saturated suppression pool, a flooded drywell, and a limited supply of water in the pedestal that mostly boils away.

The release from the subcooled suppression pool is limited. The upper bound of the distribution is 10 percent and the median is 0.1 percent. Release from the saturated pool is somewhat greater. The median release is 0.5 percent and the upper bound release is 80 percent.

The releases from the flooded cavity cases are substantially larger. For the case with a large volume of water, the range of release is from 10 percent to 90 percent, with a median of 50 percent. The limited water supply case has a range of 20 percent to 100 percent, with a median of 80 percent.

Summary of Results

By examining the ranges used to represent the source term factors (e.g., release from fuel in-vessel and fractional release from reactor coolant system), it is evident why the ranges of the environmental release fractions are large. To begin with, the source distributions for the release of radionuclides from fuel in-vessel and ex-vessel are very broad. Even radionuclides that are typically not considered to be volatile, such as barium and strontium, have ranges of uncertainty that extend as high as 50 percent for in-vessel release. Similarly, the decontamination factors that are applied to these release terms can vary over a range of three orders of magnitude. In some instances, the separation between the mean and the median of the source term distribution for the environmental release of a radionuclide can be as large as three orders of magnitude. These very broad distributions are the result of sampling from multiplicative factors, each of which has a wide distribution. No specific source term issues stand out as dominating the uncertainty because there are a number of contributors.

It is important to recognize that the origin of the uncertainties in the source term issues does not completely arise from uncertainties in the chemistry of fission product interactions or the physics of aerosol transport. Uncertainties in core melt progression, in thermal-hydraulic behavior within the reactor coolant system, and in thermal-hydraulic behavior within containment (and secondary buildings) also have a major effect on the uncertainties in calculated source terms.

REFERENCES FOR SECTION C.10

C.10.1 USNRC, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," NUREG-0772, June 1981.

C.10.2 J. T. Larkins and M. Cunningham, "Nuclear Power Plant Severe Accident Research Plan," NUREG-0900, January 1983.

C.10.3 M. Silberberg et al., "Reassessment of the Technical Bases for Estimating Source Terms," NRC Report NUREG-0956, July 1986.

C.10.4 USNRC, "Reactor Risk Reference Document," NUREG-1150, Appendix J of Vol. 3, Draft for Comment, February 1987.

C.10.5 H. Kouts, "Review of Research on Uncertainties in Estimates of Source Terms from Severe Accidents in Nuclear Power Plants," Brookhaven National Laboratory, NUREG/CR-4883, BNL-NUREG-52061, April 1987.

C.10.6 F.T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

C.11 Analysis of Seismic Issues

Since the first attempt in the Reactor Safety Study in the mid–1970's to quantify seismic risk, some 25 plant-specific seismic PRAs have been completed. These later PRAs have shown that seismic events can be significant contributors both in terms of core damage frequency and the potential for releases of radioactive material. There are many reasons for these findings. The foremost reason is that like many other external events, a seismic event not only acts as an initiator but can also compromise mitigating systems because of its common-cause effects. Secondly, the large uncertainties associated with the rare (particularly for Central and Eastern United States sites) but large seismic events that are significant to risk analyses result in large uncertainties in the outcome of the risk study. Uncertainties in these risk measures also make seismic events significant contributors when risk indices, such as mean frequencies, are used as measures. Table C.11.1 (reproduced from Ref. C.11.1) shows results of some seismic PRAs. Prior to discussing the issue of the uncertainty in the seismic hazard and its impact on PRA results and interpretation, a brief overview of both the seismic risk analysis procedure and seismic hazard methods is described in the following section.

C.11.1 Issue Definition

The elements of the seismic risk analysis procedure can be identified (Ref. C.11.2) as analyses of (1) the seismic hazard at the site, (2) the response of plant systems and structures, (3) component fragilities, (4) plant system and accident sequences, and (5) consequences. The results of these analyses are used as inputs in defining initiating events, in developing system event trees and fault trees, in quantifying accident sequences, and in modifying the accident progression event trees and consequence models to reflect the unique features of seismic events. There are uncertainties associated with each step of the risk analysis procedure. However, a number of studies (Ref. C.11.3), including the seismic risk studies performed in connection with NUREG–1150 (Parts 3 of Refs. C.11.4 and C.11.5), have shown that the uncertainties in the first element, the seismic hazard analysis, dominate the uncertainties in the overall results.

As shown in Figure C.11.1, the major steps involved in performing the site-specific hazard analysis are as follows:

- Identification of the sources of earthquakes, such as faults (F1, F2) and other seismotectonic sources (A1, A2, A3);
- Evaluation of the earthquake history of the region to assess the frequencies of occurrence of earthquakes of different magnitudes or epicentral intensities (recurrence) and determination of maximum magnitude;
- Development of attenuation relationships (including random uncertainty) to estimate the intensity of earthquake-induced ground motion (e.g., peak ground acceleration) at the site (attenuation); and
- Integration of all the above information to generate the frequencies with which different values of the selected ground-motion parameter would be exceeded (seismic hazard).

Because of the brevity of the historical record and lack of full understanding of earthquake processes and their effects in much of the United States, considerable uncertainties are associated with each of the above steps, resulting in the large uncertainties in the seismic hazard estimates. An accepted procedure for including the uncertainties of the parameters in the hazard analysis is to postulate a set of hypotheses (e.g., specific source configuration, specific value of slope parameters in recurrence relation). A probability value is assigned to each of these hypotheses, based on the analyst's expert judgment. A seismic hazard curve representing the annual frequency of exceeding a specified peak ground acceleration is generated from each hypothesis resulting in a family of different hazard curves, each representing probability of exceedance (see Fig. C.11.1(d)). Such a family of hazard curves has generally been used in past PRA applications.

Two major programs have been undertaken in the past few years to develop methods and data banks to estimate the seismic hazard at all locations of the United States east of the Rocky Mountains. The first program conducted by the Lawrence Livermore National Laboratory (LLNL) under the auspices of the NRC is entitled the Seismic Hazard Characterization Project (Ref. C.11.6). The method used in the LLNL project embodies the four steps described above. In order to capture the uncertainties, both the random (physical) uncertainty and the modeling (knowledge) uncertainty, expert judgment was used to

Table C.11.1 Seismic core damage and release frequencies from published probabilistic risk assessments.

Plant	Type	SSE (g)	Seismic Core Damage Frequency (mean) per Year	Seismic Release Frequency (mean) per Year	% of Total Core Damage	Rank of Release Sequence	Dominant Earthquake Level (g)
Zion 1 & 2	PWR	0.17	5.6E-6	———	3	1	>0.35
Indian Point 2	PWR	0.15	1.4E-4 (rev. 4.8E-5)	1.4E-4	30	1	>0.30
Indian Point 3	PWR	0.15	3.1E-6 (rev. 2.5E-5)	2.4E-6	1	8	>0.30
Limerick	BWR	0.15	4.0E-6	2.0E-7	—	1	>0.35
Millstone 3	PWR	0.17	9.4E-5	———	68	3	>0.3
Seabrook	PWR	0.25	2.9E-5	———	13	30	>0.3
Oconee 3	PWR	0.15	6.3E-5	6.0E-5	25	1	>0.15

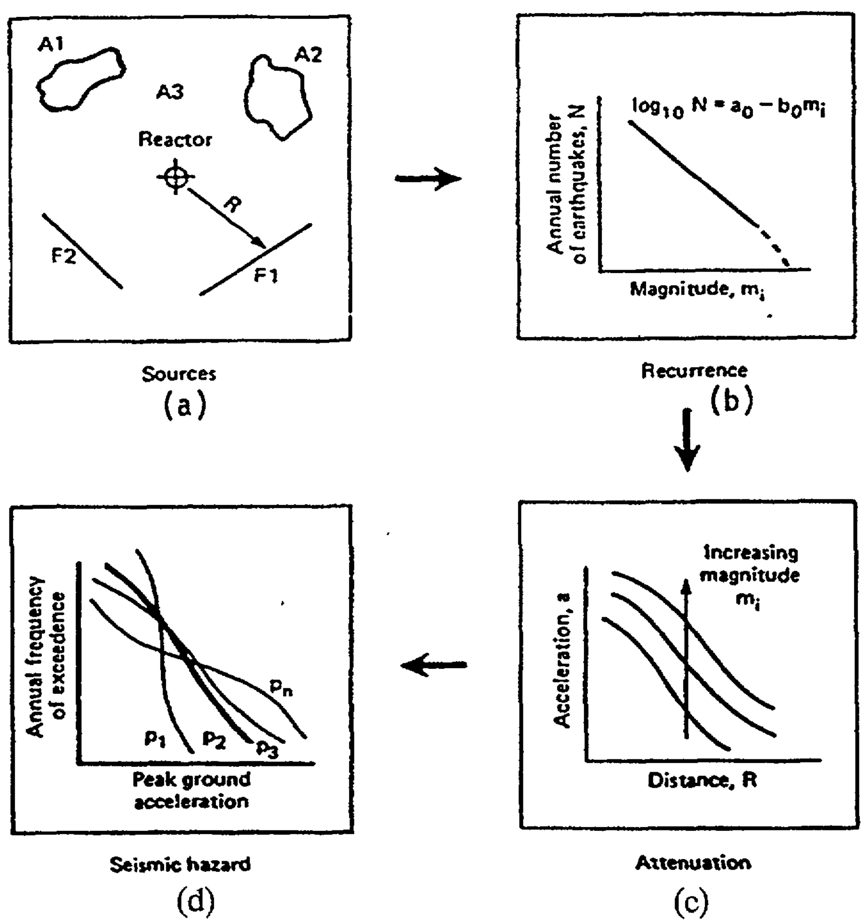


Figure C.11.1 Model of seismic hazard analysis.

assess parameter values and models in the fields of seismicity modeling and ground-motion prediction modeling. To this end two panels were formed. The S-Panel was made up of experts on seismicity and zonation; experts on ground-motion prediction formed the G-Panel. The independence of the experts was promoted by encouraging them, if they so preferred, to use their own information and data bases. The method developed was not intended to lead to some kind of artificial consensus, but rather the display of the full range of judgments was to be retained. The judgments of the experts were elicited through a series of written questionnaires, feedback meetings, and feedback questionnaires.

To propagate uncertainties in parameter values and models and develop a probability distribution of the hazard, an approach based on simulation was used. Using a Monte Carlo approach, each of the parameters was sampled a large number of times from its respective probability distribution, which described its uncertainty. With each hazard curve resulting from a given simulation was associated a weight, or probability of being the true hazard curve, which was calculated as the product of the probabilities or weights of each of the random parameter values used in that simulation. For each pair of seismicity and ground-motion experts (respectively S-Expert and G-Expert) described earlier, a typical simulation was carried out as follows:

- Draw a map from the distribution of maps for this S-Expert,
- For each one of the seismic sources in a sample map, draw a set of seismicity parameters from their respective distribution, i.e.:
  - A value for the a parameter of the recurrence law (see Fig. C.11.1(b)),
  - A value for the b parameter of the recurrence law (b is allowed to have three levels of correlation with a, as specified by the S-Expert, Fig. C.11.1(b)), and
  - The value of the upper magnitude (or intensity) cutoff;
- Draw a ground-motion model from the distribution of models; and
- Draw a value for the random uncertainty parameter, which is associated with the selected ground motion, for the appropriate Eastern United States region (Northeast, Southeast, North Central, or South Central).

The hazard was calculated for each of the seismic sources and combined for all sources. Each simulation gives a possible hazard curve. For each site, typically 2,750 curves (50 simulations for each of the possible combinations of 5 G-Experts and 11 S-Experts) were developed. Percentiles, usually the 15th, 50th, and 85th, were then used to describe the uncertainty in the hazard. Typical hazard curves are presented in Figure C.11.2 for the Peach Bottom site.

In addition to the hazard curves for 69 nuclear plant sites east of the Rocky Mountains, the LLNL project also generated uniform hazard spectra for various return periods for each site. Uniform hazard spectra for the Peach Bottom site are presented in Figure C.11.3.

The second program was undertaken by the Electrical Power Research Institute (EPRI) (Ref. C.11.7) with similar objectives to the LLNL program. While the LLNL and EPRI approaches have many similarities, that is, they rely upon expert judgment, there are significant differences in the manner in which the expert judgment was solicited and in the treatment of ground motions (Ref. C.11.8). EPRI's major effort was aimed at developing a structured approach to the delineation and characterization of seismic sources. On the other hand, LLNL's approach was that of the solicitation of expert judgment from individuals, among whom there was a moderate amount of interaction, while EPRI relied upon the use of expert teams, among whom there was a great deal of interaction through workshops and meetings devoted to specialized seismological and tectonic topics. For example, instead of the 11 individuals, (primarily seismologists) upon whom LLNL relied for the seismic zoning input, EPRI used six teams, each of which spanned the disciplines of geology, seismology, and geophysics. After discussion and interaction, each team agreed upon a common input. In order to ensure uniformity in data assumptions, EPRI compiled a common geological, geophysical, and seismological data base. With respect to seismicity recurrence parameters, a good deal of effort was expended in defining uniform statistical techniques for estimating these parameters. The teams had the option, based on these studies, of allowing variations within the

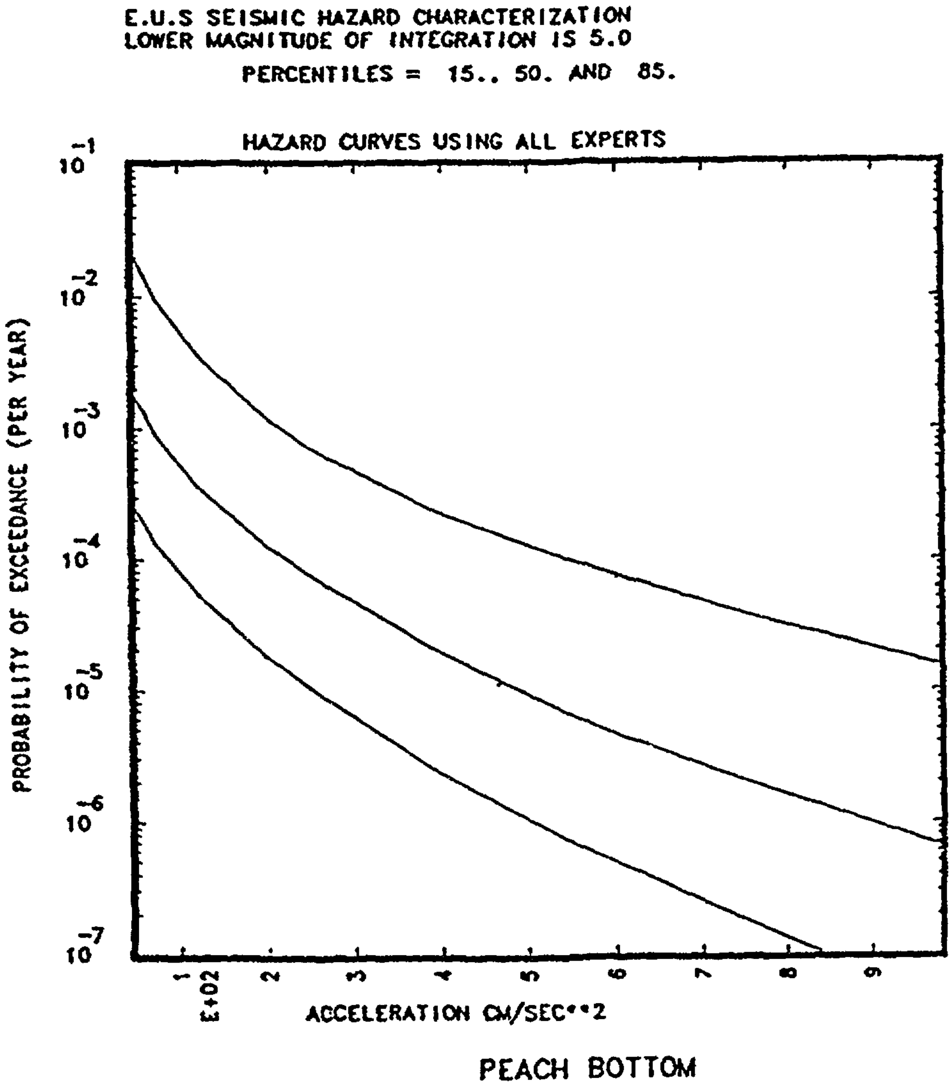


Figure C.11.2 LLNL hazard curves for Peach Bottom site.

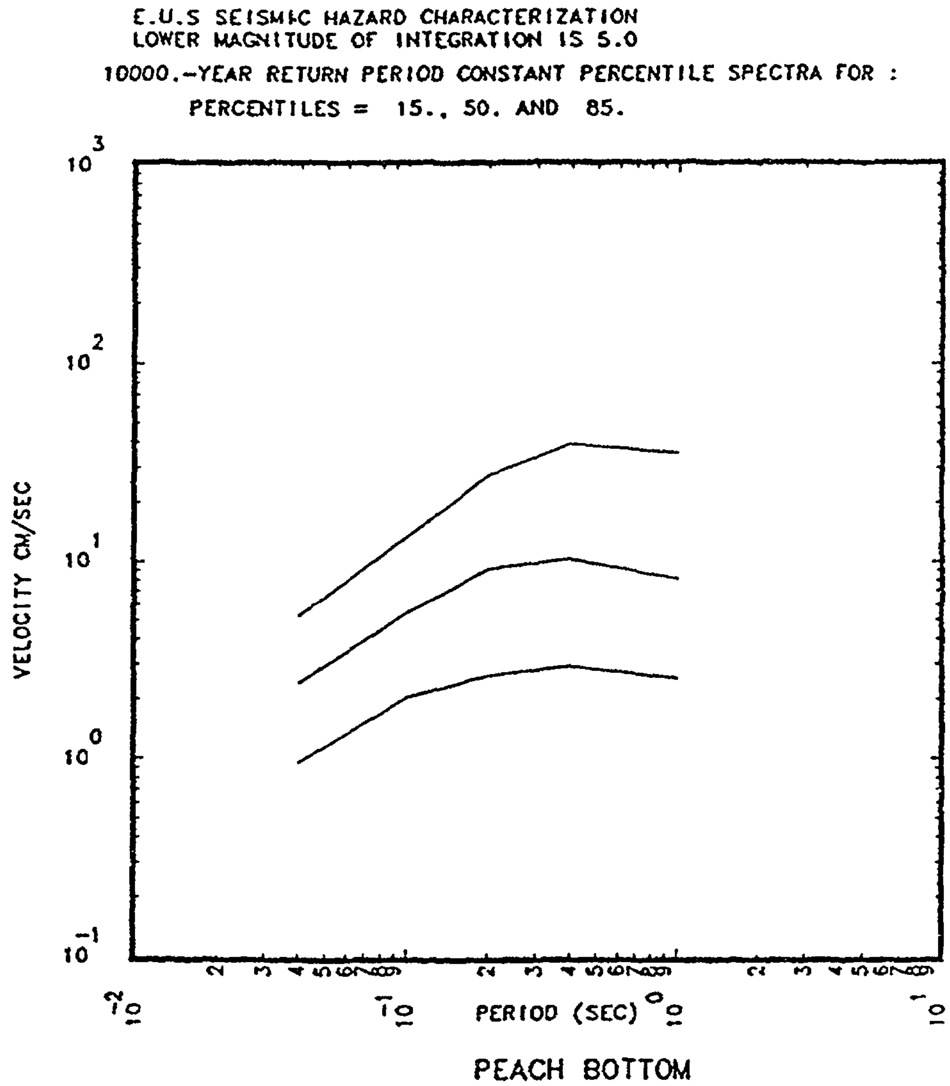


Figure C.11.3 10000-year return period uniform hazard spectra for Peach Bottom site.

seismic sources themselves. Instead of employing an expert panel for the ground motion, EPRI used three models to reflect uncertainties in the ground-motion estimates. EPRI felt that ground-motion model development is less subjective and fairly well defined and any needed evaluation could be done by its consultants (Ref. C.11.9). In the EPRI approach, in contrast to the Monte Carlo approach used by the LLNL, uncertainties in seismic sources, seismicity parameters, maximum magnitude, and ground-motion models are propagated through and represented in a logic-tree format (Fig. C.11.4). Each level of the tree represents one source of uncertainty; each terminal node represents one "state of nature." Corresponding to each terminal node, there is a hazard curve. The probability associated with a terminal node (and with the corresponding hazard curve) is the product of the probabilities associated with all intermediate branches in the path from the root to the terminal node. Results for the Peach Bottom site from the EPRI program are shown in Figure C.11.5. The uniform hazard spectra obtained from the EPRI program, in general, exhibit similar characteristics to the LLNL results. According to Reference C.11.10, which compared preliminary results of both studies of nine test sites, the most significant differences in the results of the LLNL and EPRI studies that primarily affect the uncertainty distributions are (see also Ref. C.11.11):

- A larger number of ground-motion models, encompassing a large range of opinions, are used in the LLNL project than in the EPRI study; and

- The EPRI study has less uncertainty in the seismicity parameters, leading to lower uncertainty in the estimate of the hazard.

Thus, there are now two sets of hazard curves available for use at sites east of the Rocky Mountains. Issues associated with the use of these two sets of hazard curves, associated uncertainties, and consideration of uniform hazard spectra in the PRA applications are discussed below.

The primary issue in the seismic risk analysis is the large uncertainty associated with the computed results and use of these results in decisionmaking. The uncertainties, as discussed earlier, largely stem from uncertainties in hazard estimates. In addition to the issue of the uncertainty in hazard, publication of uniform hazard spectra will also have an impact on the PRA application.

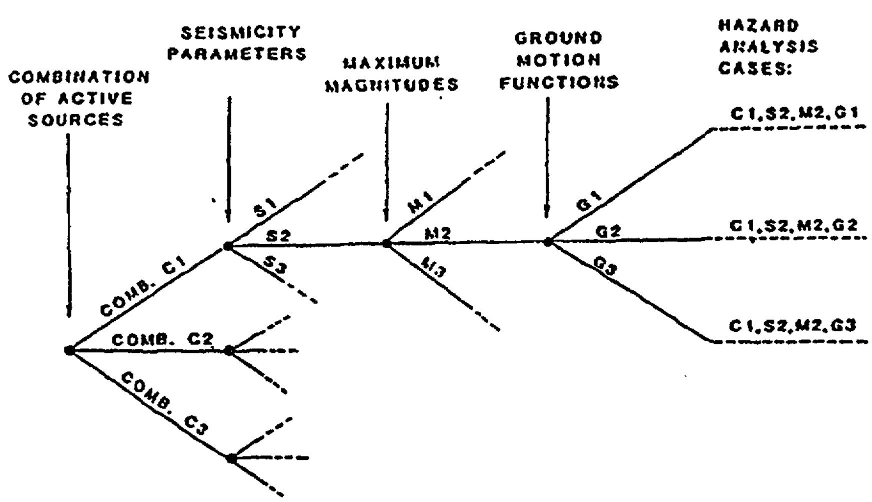


Figure C.11.4 Example of logic-tree format used to represent uncertainty in hazard analysis input (EPRI program).

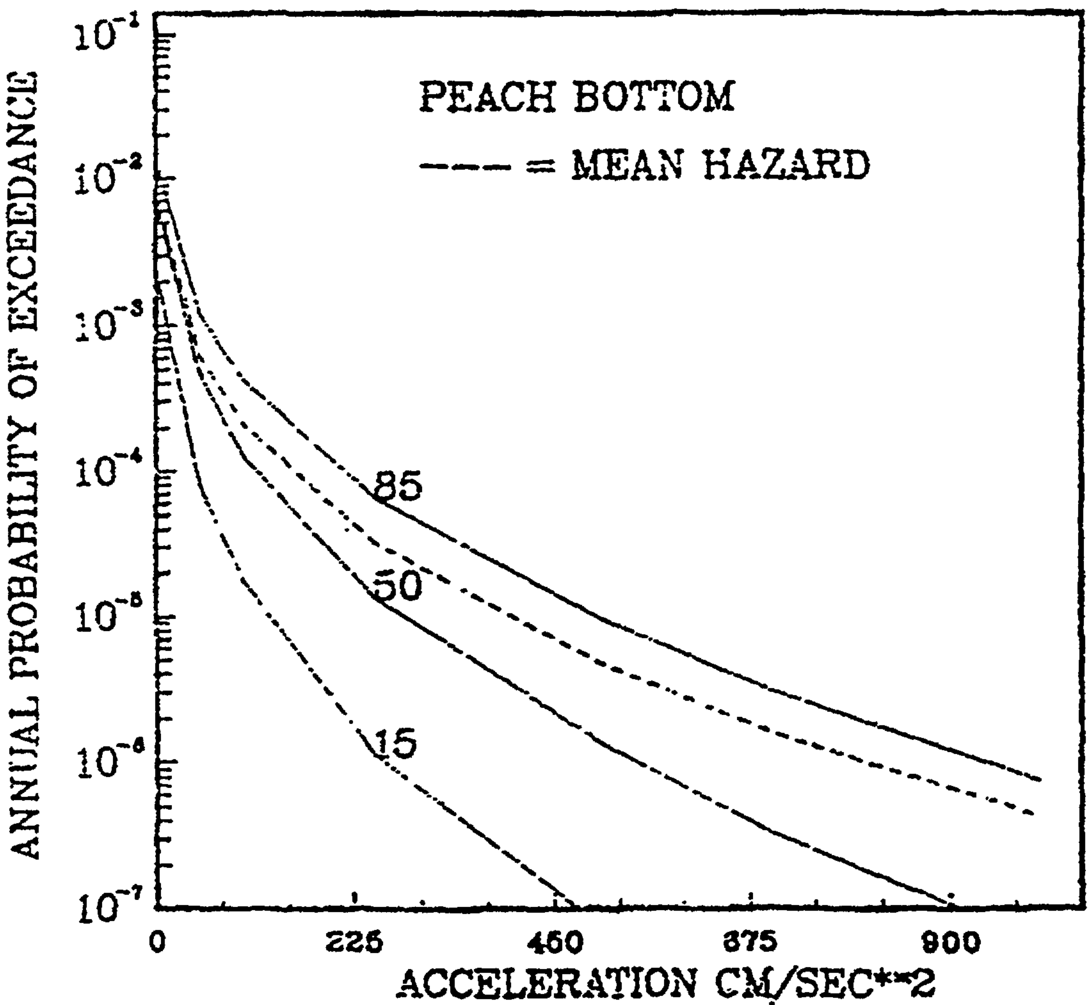


Figure C.11.5 EPRI hazard curves for Peach Bottom site.

- *Uncertainty in Hazard Estimates:* As seen in Figure C.11.2, from the large spread between the 15th percentile and 85th percentile of the hazard calculations, it is evident that seismic hazard estimates are associated with substantial uncertainties. In terms of ground-motion parameters, the LLNL results (Ref. C.11.12) indicate that, for a fixed annual probability of exceedance, the difference between the 15th and 85th percentile curves corresponds to approximately a factor of 4 or larger in both peak ground acceleration (PGA) and spectrum-related ground-motion estimates. When the probability of exceedance at the 15th and 85th percentile levels are compared at a fixed PGA, large differences, ranging from a factor of 40 to 100, can be observed, depending upon the PGA level. Sensitivity studies have shown that the largest contribution to modeling uncertainty is caused by the uncertainties associated with the models relating ground motion to distance and magnitude. It should also be noted that the mean hazard, because it is sensitive to a highly skewed distribution, can lie above the 85th percentile of the hazard. Median hazards are not strongly affected by the extreme values of the probability distributions.

  Sensitivity studies on the LLNL results indicate that individual expert judgment can, under certain conditions, dominate the hazard calculation. Specifically, if a site in question is a rock-based site where distant large earthquakes are the major contributors to the overall seismic hazard, the inclusion or exclusion of the input from one ground-motion expert (G-Expert 5) leads to significant differences in the hazard. This effect is particularly evident at the 85th percentile and mean hazard estimates.

  The widely recognized difficulties in the estimation of the likelihood of rare events are compounded in the case of seismic hazard estimation by the lack of knowledge with respect to basic causes and future locations of earthquakes in the Eastern United States. This is clearly illustrated by the results of two independent studies, the LLNL and the EPRI studies. These studies represent the most comprehensive efforts of their kind undertaken to date. Although attempts have been made (Ref. C.11.10) and studies are under way to understand and reduce the differences between the results of these studies, the methods of each of these studies should be viewed as valid. Because of the inherent uncertainties, results from both sets of hazard curves should be included in a risk study (Ref. C.11.12). Reducing the combined range of uncertainties to a single point estimate ignores the fundamental message. Enveloping the uncertainty is also inappropriate in that the least well-defined aspects are the upper and lower bounds. Therefore, in NUREG-1150 studies, both sets of curves are used independently and results are presented side by side (Figs. C.11.6 and C.11.7). The use of these two sets of hazard curves and associated risk analysis results are discussed in Section C.11.2.

- *Uniform Hazard Spectra:* As discussed earlier, the LLNL and EPRI studies have also provided estimates of uniform hazard spectra for each Eastern and Central United States site. As described in Section C.11.2, the NUREG-1150 studies used LLNL and EPRI hazard curves but did not use uniform hazard spectra. One of the major findings of both the LLNL and EPRI studies is that the estimated uniform hazard spectra for eastern earthquakes are higher at high frequencies and lower at low frequencies compared to standard broadband spectra (e.g., spectra given in Ref. C.11.13) based on recorded western earthquakes. The spectra used in the NUREG-1150 analyses were developed using primarily western records. Implications of the differences between these spectra on the risk analysis will be studied in detail in a later study but the inference for Surry is quite clear. For all structural frequencies of interest, the uniform hazard spectra are below the median spectra used in the analysis. This should result in smaller plant response for a given PGA. The overall effect, with all else remaining the same, will be a reduction in the core damage frequency. For the Peach Bottom site, the impact on the mean core damage and distribution is also not expected to be significant since, at the frequencies of important structures, both response spectra are similar. These issues will be addressed in additional staff studies. Additional issues associated with the use of uniform hazard spectra are related to partitioning of uncertainties among hazard curves, spectra, and fragility analyses. Both the PGA hazard curves and uniform spectra reflect uncertainty in the underlying seismological parameters as well as randomness in the ground-motion estimates. In addition, peak-to-valley variability and random variability between horizontal components may also be included in these two estimates. Many times these uncertainties are included in the fragility analysis. Thus, clear understanding of uncertainties is needed to avoid double counting or underestimation.

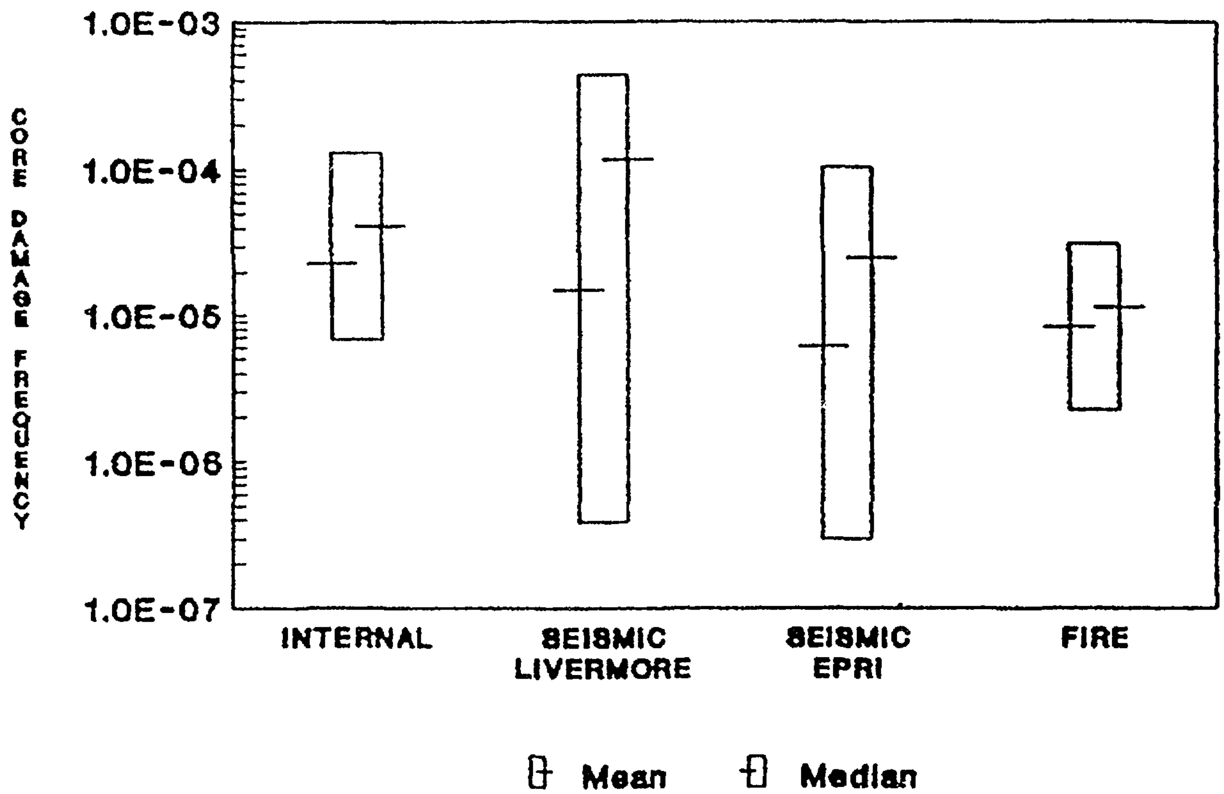


Figure C.11.6 Surry external events, core damage frequency ranges (5th and 95th percentiles).

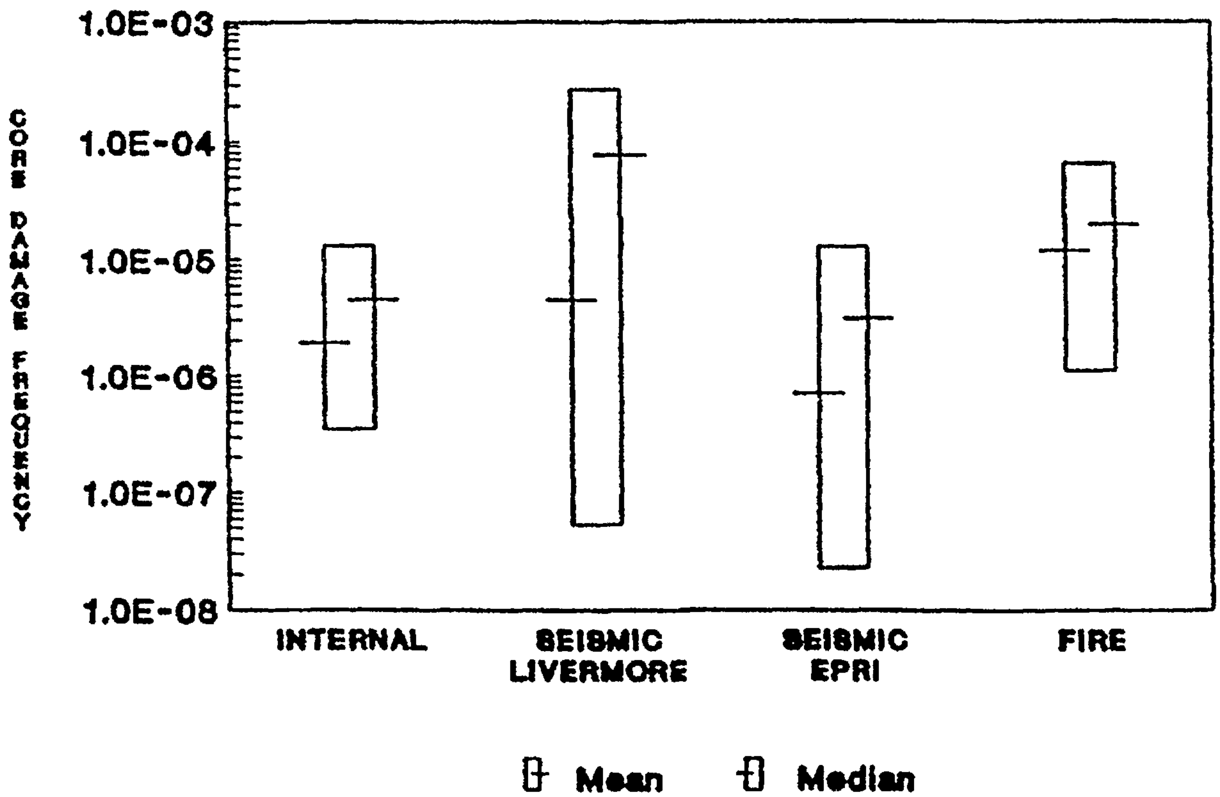


Figure C.11.7 Peach Bottom external events, core damage frequency ranges (5th and 95th percentiles).

C.11.2 Treatment in PRA and Results

In the NUREG–1150 seismic analyses, both the LLNL and EPRI hazard estimates have been used. These estimates, as shown in Figures C.11.2 and C.11.5, are given at selected confidence levels of 15 percent, 50 percent, and 85 percent and mean curves (it is also possible to get results for 5 percent and 95 percent confidence level). In principle, one can use the entire set of 2,750 hazard curves that were generated for each site in the LLNL study. However, since the results are presented in the form of Figure C.11.2, one has to resort to fitting a distribution to the hazard uncertainty at any given PGA and then discretize the distribution into a chosen set of hazard values in order to obtain a discrete set of hazard curves. This approach was used in Reference C.11.14, which also used the LLNL input. In NUREG–1150, it was assumed that the seismic hazard could be approximated by a lognormal distribution that fit the calculated 50th percentile and mean. While the lognormal assumption is a good approximation, the actual distributions vary. The differences can be seen by comparing the actual calculated hazard curves at a different percentile from the LLNL study with that determined from the lognormal fit. The EPRI hazard results can deviate more from a lognormal distribution. In the NUREG–1150 studies, the full lognormal distribution was used in drawing samples for the Monte Carlo analyses. Therefore, no discretization was necessary.

Sensitivity studies have been carried out for both Surry and Peach Bottom analyses to understand the impact of the lognormal assumption as well as to assess the potential effect of contributions from the tail of the assumed distribution. (See individual plant studies (Parts 3 of Refs. C.11.4 and C.11.5) for further discussions.) Since the distribution is derived by fitting it to the mean, there should be minimal impact on the mean core damage frequency from this approximation.

The necessity of the above approach of fitting a distribution to the uncertainty will also result in not simulating the real nature of hazard curves that may intersect each other. The correlation of hazard values at different accelerations arising out of the same source/ground-motion hypothesis model cannot also be consistently treated. The major contributions to the hazard cannot *a priori* be correlated with the size of the contributing earthquakes. For example, the study results (Ref. C.11.6) indicate that although some plant sites in the New England area exhibit relatively high seismic hazards, the contribution to the overall hazard from earthquakes with magnitudes of 6.5 or larger is significantly less than the contribution from these large earthquakes to plant sites near New Madrid, Missouri, or those near Charleston, South Carolina. Such a deaggregation of hazard curves and uniform hazard spectra can be extremely useful in understanding the relative contribution from large magnitude (potentially damaging) events versus low magnitude (less damaging) events. It is very important to understand that PGAs have not been shown to be good indicators of the damage potential of an earthquake for ductile structures/components since low magnitude events can produce a large PGA but little damage (e.g., Ref. C.11.15). (This concern has been alleviated to some extent in the LLNL and EPRI studies by the use of a minimum magnitude of 5.0, the magnitude below which damage to the engineered structures is considered unlikely.) To better characterize damage potential, information with respect to the frequency content of the motion and duration are vital. The fragility analysis used in the NUREG–1150 studies takes into account, to some extent, the earthquake magnitude effects by using the concept of an effective ductility (see Parts 3 of Refs. C.11.4 and C.11.5 for detailed discussions). However, detailed building/component response analysis (including nonlinear effects, if necessary) using magnitude-dependent spectral shapes can be used to remove further conservatisms, if any, included in the plant response/fragility analysis. Consequences of a building failure can also be evaluated more realistically.

While such understanding of hazard curves will not necessarily result in less uncertainty or changes in core damage frequencies, perspectives into which magnitude earthquakes and characteristics of the associated ground motion that contribute to these frequencies can be extremely useful. Recovery actions, not usually considered acceptable in seismic PRAs, may be feasible for the lesser magnitude events. In the additional staff studies, the deaggregation of the hazard curve into various magnitude ranges will be considered to the extent possible.

The seismic risk analysis method used in NUREG–1150 requires the use of earthquake time histories to determine the vibratory motion within the nuclear power plant. Peak ground accelerations from the seismic hazard curves were used to anchor a set of real earthquake records (time histories) for each site. These scaled earthquake records were then used to perform a probabilistic response analysis. The seismic hazard studies, while not providing time histories, do, along with defining peak accelerations, define seismic hazard also in terms of uniform hazard spectra. They are based upon the limited data available from Eastern United States earthquakes and the most current models prepared by seismological experts in the field. They are different from the response spectra of the time histories that were used, which were derived from Western United States earthquakes. The seismic hazard spectra based on eastern earthquakes are higher at high frequencies and lower at low frequencies. Issues associated with the uniform hazard spectra and their use in the PRA application were discussed in Section C.11.1 along with the possible impact on the results of NUREG-1150 studies. This issue will be addressed in additional staff studies as discussed above.

After the establishment of the hazard curves and the spectra for use in the plant response calculations, the remaining steps of the seismic risk analysis (that is, plant system and accident sequence analysis and quantification) are described in Appendix A to NUREG-1150 and in plant-specific external-event analyses (Parts 3 of Refs. C.11.4 and C.11.5). Estimates of the core damage frequencies for the Surry and Peach Bottom plants are reproduced in Figures C.11.6 and C.11.7. Values of mean, 5th, 50th, and 95th percentile estimates are also given in Table C.11.2.

Sensitivity studies conducted specifically for Surry and Peach Bottom in this program and other sensitivity studies have shown that the uncertainty in seismic hazard curves dominates the uncertainty in the seismic core damage frequency. Table C.11.3 shows results of sensitivity analysis performed for Peach Bottom to ascertain the relative contribution of the hazard curve, the seismic response, and the seismic fragility modeling uncertainties to the overall core damage frequency.*

The base case mean, 95th percentile, and 50th percentile core damage frequencies are shown in the first column. The second column shows the corresponding values with the hazard curve fixed at its median value (i.e., with no modeling uncertainty). This results in an error factor of 3.5 versus the error factor of 30.1 for the base case. Clearly, the hazard curve is contributing the vast majority of the uncertainty to the base case results. Note also that the mean core damage frequency is reduced by a factor of 6.1 when no hazard curve uncertainty is included. The third column shows a case wherein all the fragility and response modeling uncertainties are simultaneously set to zero. The error factor for this case is 25.5, which shows that the reduction in response and fragility uncertainties has little effect on the overall core damage uncertainty. For this case, the mean core damage frequency is reduced only by a factor of 1.9. Thus, the fragility and response uncertainties play little role in determining the mean core damage frequency. However, conservatisms associated with the fragilities (median values and uncertainties) and assumed consequences, given a failure of a certain component, may have significant impact on core damage

Table C.11.2 Core damage frequencies.

	5th	Median	95th	Mean
Surry				
LLNL	3.92E-7	1.48E-5	4.38E-4	1.16E-4
EPRI	3.00E-7	6.12E-6	1.03E-4	2.50E-5
Fire	5.37E-7	8.32E-6	3.83E-5	1.13E-5
Internal	6.75E-6	2.30E-5	1.31E-4	4.01E-5
Peach Bottom				
LLNL	5.33E-8	4.41E-6	2.72E-4	7.66E-5
EPRI	2.30E-8	7.07E-7	1.27E-5	3.09E-6
Fire	1.09E-6	1.16E-5	6.37E-5	1.96E-5
Internal	3.50E-7	1.90E-6	1.30E-5	4.50E-6

*Note that the hazard curves used in this sensitivity analysis are not the same as ones used in the final calculation, however, the conclusions are the same. See Part 3 of Reference C.11.5 for the final results.

Table C.11.3 Comparison of contributions of modeling uncertainty in response, fragility, and hazard curves to core damage frequency.

P_{cm}	Base Case	No Modeling Uncertainty in Hazard	No Modeling Uncertainty in Response, Fragility
Mean	1.55E−5	2.25E−6	8.13E−6
95%	5.78E−5	6.53E−6	3.50E−5
50%	1.92E−6	1.87E−6	1.37E−6
$\frac{P_{cm}\ (95\%)}{P_{cm}\ (50\%)}$	30.1	3.5	25.5
$\frac{E[P_{cm}\ \text{base case}]}{E[P_{cm}]}$	1.0	6.1	1.9

frequencies if a single loss dominates the contribution. Dominant sequences and their contributions can also be affected by fragility/consequence assumptions.

These results show quite clearly that the uncertainty in the hazard curve is the dominant factor in both the mean value of core damage frequency and in the uncertainty of the core damage frequency. Further, as discussed in the plant-specific analyses (Part 3 of Ref. C.11.5), it is the mean hazard curve that drives the mean estimate of core damage frequency. Again, this shows the dominant influence of the hazard curve uncertainty (which determines the mean hazard curve) in determining the mean core damage frequency.

Sensitivity studies are also conducted in the plant-specific analyses to examine the importance of the basic seismic failure events to the estimates of mean core damage frequencies. These studies show that for a dominant component, if no failure is assumed, the percentage reduction in the core damage frequency would be in the range of 40 percent. Note that given large uncertainties associated with the seismic results, a change in the mean core damage frequency by a factor of two or so may not be that significant. However, the fragility of the plant (conditional failure probability at a given PGA) may improve appreciably.

In Table C.11.4, dominant sequences and their contributions to the mean core damage frequency are listed for the Peach Bottom plant for both the LLNL and the EPRI hazard curves. Similar studies for the Surry plant are also discussed in the plant-specific studies. Observations discussed here are equally valid for both plants. As seen from this table, although the numerical values are quite different for each sequence when the LLNL or the EPRI hazard curves are used, the order is the same and the relative contributions are slightly varied. This is not surprising since the two mean curves do not intersect each other or indicate drastically different characteristics, such as one being truncated at some acceleration value. If this were the case, then one might expect ranking of sequences and elimination or addition of some sequences. Since the order of dominant sequences remains the same for both the hazard curves, perspectives gained as to the dominant components are also robust. Perspectives as to dominant sequences and components could also be affected if the spectral shapes associated with the different hazard curves were quite dissimilar. However, the uniform hazard spectra from both the LLNL and EPRI studies seem to exhibit similar characteristics.

For the Peach Bottom plant, Figure C.11.8 shows contributions from the different earthquake ranges to the mean core damage frequencies resulting from the use of both the LLNL and the EPRI hazard curves. For both the hazard curves, the majority of contributions is coming from an earthquake range between 0.45g to 0.75g consistent with the Peach Bottom mean plant fragility curve (Fig. C.11.9) for which roughly the 50 percent conditional core damage frequency occurs around 0.6g. Thus, there is a relative robustness as to which earthquake range contributes to the mean core damage frequencies.

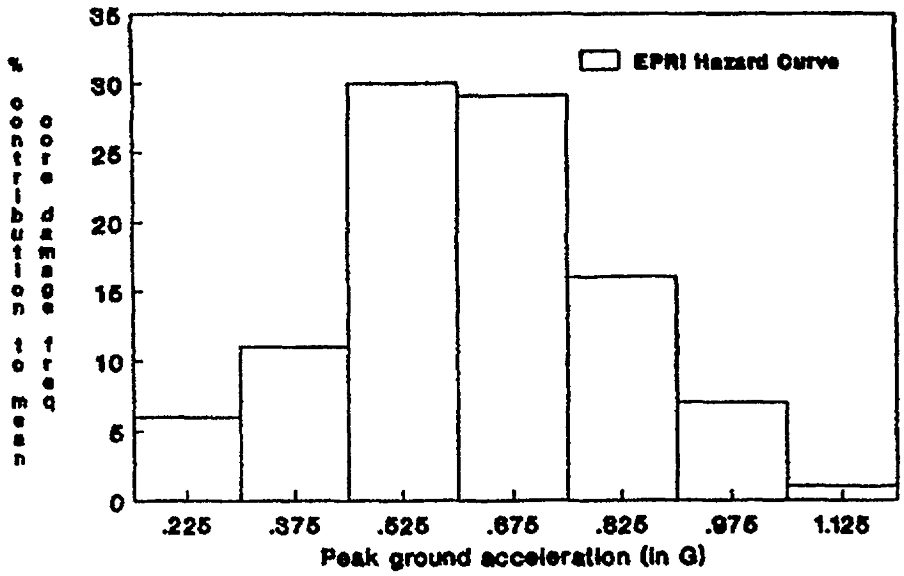
Table C.11.4 Dominant sequences at Peach Bottom.

Total Mean P_{cm} = 7.66E-5 (LLNL), 3.09E-6 (EPRI)

Dominant Sequences	**LLNL**		**EPRI**	
T_1-33	3.69E-5	(48%)	1.61E-6	(52%)
ALOCA-30	1.84E-5	(24%)	6.70E-7	(21%)
RVR-1	8.92E-6	(11%)	3.27E-7	(11%)
S_1LOCA-70	6.67E-6	(9%)	1.85E-7	(6%)
RWT-1	2.76E-6	(4%)	1.75E-7	(6%)
S_2LOCA-42	1.20E-6	(2%)	4.90E-8	(2%)
By Sequence	**LLNL**		**EPRI**	
Transients (LOSP)	3.69E-5	(48%)	1.61E-6	(52%)
LOCAs	2.59E-5	(34%)	9.04E-7	(29%)
Vessel Rupture	8.92E-6	(11%)	3.27E-7	(11%)
RWT Bldg Failure	2.76E-6	(4%)	1.75E-7	(6%)

Several long-term studies are planned to better understand some of the issues associated with the hazard definition; however, it is clear that results of seismic risk analysis will have large uncertainties associated with them. From examination of Table C.11.2, it is evident that conflicting conclusions can be obtained when point estimates are used as risk indices. For example, if the median estimates are used to determine the relative importance between the seismic initiators and the internal initiators, one will conclude that for the Surry analysis, with either the LLNL hazard or the EPRI hazard estimates, the contribution to the total core damage frequency is larger from the internal initiators than the seismic initiators. Conversely, if the means are used, one would conclude that, based on the use of the LLNL hazard curves, the contribution from the seismic initiators is much larger than that from the internal initiators. Based on the results from the EPRI hazard curve, the conclusion would be that internal initiators contribute more than seismic initiators. These findings illustrate the confusion that can result if single estimates are used to characterize the risk.

One clear conclusion is that the distribution of the seismic-induced core damage frequencies are more uncertain than the internal frequencies and their distribution overlaps with the distribution from other initiators and cannot be ignored. In light of the large uncertainties, any decisionmaking should take into account the full range of uncertainty as well as engineering insights and understanding obtained regarding the integrated plant response to a seismic event. Some of the robust findings, such as perspectives regarding dominant sequences and components, were discussed in the preceding paragraphs and can be used to evaluate the need to further refine the analysis or enhance safety by improving the plant procedures or implementing cost-effective fixes.



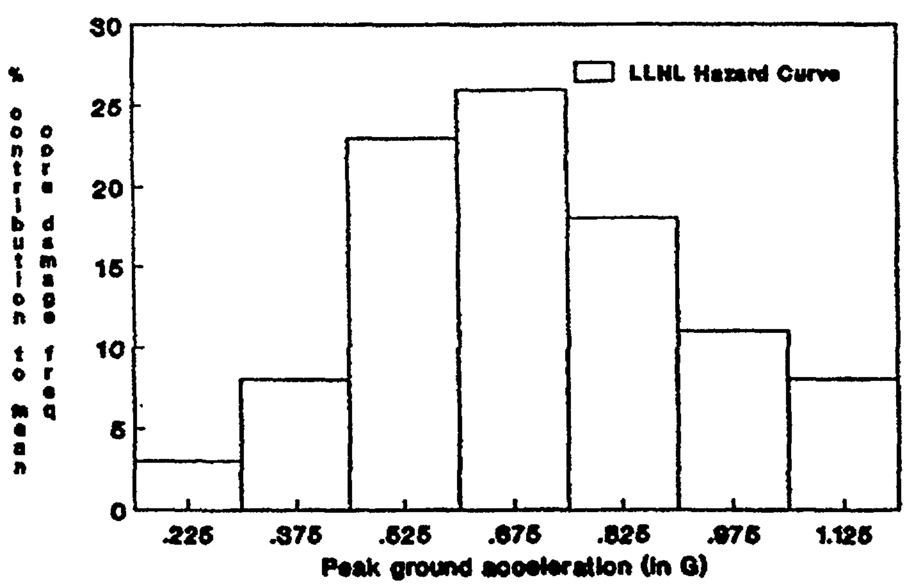


Figure C.11.8 Contribution from different earthquake ranges at Peach Bottom.

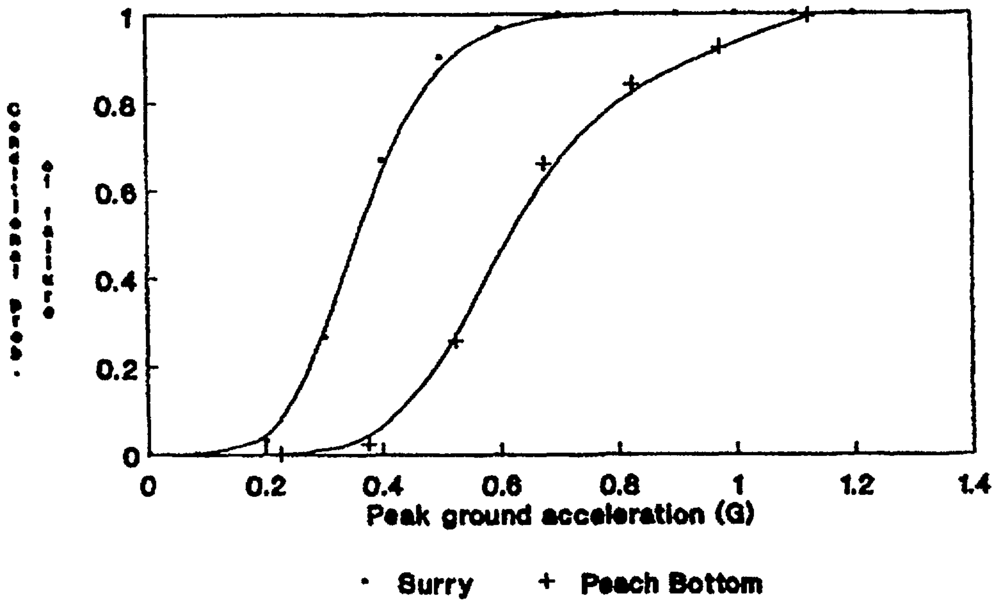


Figure C.11.9 Mean plant level fragilities.

REFERENCES FOR SECTION C.11

C.11.1 P. G. Prassinos, "Evaluation of External Hazards to Nuclear Power Plants in the United States: Seismic Hazard," Lawrence Livermore National Laboratory, NUREG/CR-5042, Supplement 1, UCID-21223, April 1988.

C.11.2 J. W. Hickman et al., "PRA Procedures Guide. A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," American Nuclear Society, NUREG/CR-2300, Vols. 1 and 2, January 1983.

C.11.3 M. K. Ravindra et al., "Sensitivity Studies of Seismic Risk Models," Electric Power Research Institute (EPRI), EPRI NP-3562, June 1984.

C.11.4 R. C. Bertucio and J. A. Julius, "Analysis of Core Damage Frequency: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Revision 1, SAND86-2084, April 1990.

C.11.5 A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

C.11.6 D. L. Bernreuter et al., "Seismic Hazard Characterization of 69 Nuclear Power Plant Sites East of the Rocky Mountains," Lawrence Livermore National Laboratory, NUREG/CR-5250, Vols. 1-8, UCID-21517, January 1989.

C.11.7 Seismicity Owners Group (SOG) and EPRI, "Seismic Hazard Methodology for the Central and Eastern United States," EPRI NP-4726, July 1986.

C.11.8 L. Reiter, "Current Trends in the Estimation and Application of Probabilistic Seismic Hazard in the United States," *Proceedings of IAEA Specialists Meeting on Earthquake Ground Motion* (Moscow, USSR), March 24–28, 1986.

C.11.9 Letter from L. Reiter, NRC, to R. A. Thomas, SOG, Review Comments on EPRI Ground Motion Models for Eastern North America, dated August 3, 1988.

C.11.10 D. L. Bernreuter et al., "Seismic Hazard Characterization of the Eastern United States: Comparative Evaluation of the LLNL and EPRI Studies," Lawrence Livermore National Laboratory, NUREG/CR-4885, UCID-20696, May 1987.

C.11.11 Letter from J. E. Richardson, NRC, to R. A. Thomas, SOG, Safety Evaluation Review of the SOG/EPRI Topical Report, "Seismic Hazard Methodology for the Central and Eastern United States," EPRI NP-4726, dated September 20, 1988.

C.11.12 Memorandum from G. Bagchi, NRC, to L. Shao, NRC, Publication of NUREG/CR-5250, "Seismic Hazard Characterization of 69 Nuclear Power Plant Sites East of the Rocky Mountains," dated March 21, 1989.

C.11.13 U.S. Atomic Energy Commission, Regulatory Guide 1.60, "Design Response Spectra for Nuclear Power Plants," Revision 1, December 1973.

C.11.14 P. G. Prassinos et al., "Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants," Lawrence Livermore National Laboratory, NUREG/CR-5176, UCID-21425, January 1989.

C.11.15 J. W. Reed et al., "A Criterion for Determining Exceedance of the Operating Basis Earthquake," EPRI NP-5930, July 1988.

C.12 Analysis of Fire Issue

Based on plant operating experience over the last 20 years, it has been observed that typical nuclear power plants will have three to four significant fires over their operating lifetime. Previous probabilistic risk assessments (PRAs) have shown that fires are a significant contributor to the overall core damage frequency, contributing anywhere from 7 percent to 50 percent of the total, considering contributions from internal, seismic, flood, fire, and other events (Refs. C.12.1 and C.12.2). There are many reasons for these findings. The foremost reason is that like many other external events, a fire event not only acts as an initiator but can also compromise mitigating systems because of its common-cause effects. An overview of the fire risk analysis procedure is described in the following sections. These sections will give the necessary background for the subsequent discussion of the fire issues.

C.12.1 Analysis Procedure for NUREG-1150 Fire Analysis

As described in detail in Reference C.12.3, the elements of a fire risk analysis procedure can be identified as (1) screening and (2) quantification of the remaining unscreened fire areas.

The screening analysis is comprised of:

1. Identification of relevant fire zones. Those Appendix R (Ref. C.12.4) identified fire zones that had either safety-related equipment or cabling for such equipment were determined to require further analysis.

2. Screening of fire zones based on probable fire-induced initiating events. Determination of the fire frequency for all plant locations and determination of the resulting fire-induced initiating events and "off-normal" plant states were made.

3. Screening of fire zones based on both order and frequency of cut sets.

4. Numerical evaluation and culling of remaining fire zones based on frequency.

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

1. Determine temperature response in each fire zone.

2. Compute component fire fragilities. The latest version of the fire growth code, COMPBRN III (Ref. C.12.5), with some modifications is used to calculate fire propagation and equipment damage. These fire calculations are only performed for the fire areas that survived the screening analysis.

3. Assess the probability of barrier failure for all remaining combinations of fire areas. A barrier failure analysis is conducted for those combinations of two adjacent fire areas which, with or without additional random failures, remain after the screening analysis.

4. Perform a recovery analysis. In a fashion similar to that of the internal-event analysis, recovery of non-fire-related random failures is addressed. Appropriate modifications to recovery probabilities are addressed as necessary.

5. Perform an uncertainty analysis to estimate error bounds on the computed fire-induced core damage frequencies. As in the internal-event analysis, the TEMAC code (Ref. C.12.6) is used in the uncertainty analysis.

Additional detail on the fire analysis methods used in NUREG-1150 may be found in Reference C.12.3.

C.12.2 PRA Results

Tables C.12.1 and C.12.2 provide the core damage frequency results for the Surry and Peach Bottom fire risk assessments, respectively (Refs. C.12.7 and C.12.8). All fire areas that survived the screening process are listed. When comparing fire-induced core damage frequency with the total from all other initiators (including seismic, using the LLNL seismic hazard curves), fire is 7 percent of the total for Surry and

Table C.12.1 Dominant Surry fire area core damage frequency contributors (core damage frequency/yr) (Ref. C.12.7).

Fire Area	Mean	5th Percentile	Median	95th Percentile
Emergency Switchgear Room	6.1E-6	3.9E-9	3.1E-6	2.0E-5
Control Room	1.6E-6	1.2E-10	4.7E-7	6.9E-6
Cable Vault/Tunnel	1.5E-6	6.5E-10	7.0E-7	5.8E-6
Auxiliary Building	2.2E-6	5.3E-7	1.6E-6	5.6E-6
Charging Pump, Service Water Pump Room	3.9E-8	1.4E-10	5.7E-9	1.6E-7
Total	1.1E-5	5.4E-7	8.3E-6	3.8E-5

Table C.12.2 Dominant Peach Bottom fire area core damage frequency contributors (core damage frequency/yr) (Ref. C.12.8).

Fire Area	Mean	5th Percentile	Median	95th Percentile
Emergency Switchgear Room 2A	7.4E-7	4.6E-10	1.6E-7	3.0E-6
Emergency Switchgear Room 2B	3.6E-6	3.5E-9	2.0E-6	1.3E-5
Emergency Switchgear Room 2C	4.7E-6	4.2E-9	2.2E-6	1.7E-5
Emergency Switchgear Room 2D	7.4E-7	4.6E-10	1.6E-7	3.0E-6
Emergency Switchgear Room 3A	7.4E-7	4.6E-10	1.6E-7	3.0E-6
Emergency Switchgear Room 3B	7.4E-7	4.6E-10	1.6E-7	3.0E-6
Emergency Switchgear Room 3C	7.4E-7	4.6E-10	1.6E-7	3.0E-6
Emergency Switchgear Room 3D	8.1E-7	5.3E-10	1.7E-7	3.3E-6
Control Room	6.2E-6	4.2E-10	1.4E-6	8.0E-6
Cable Spreading Room	6.7E-7	9.1E-9	1.7E-7	2.3E-6
Total	2.0E-5	1.1E-6	1.2E-5	6.4E-5

19 percent of the total for Peach Bottom. These fire areas grouped by sequence are given in Tables C.12.3 and C.12.4 for Surry and Peach Bottom, respectively.

The overall fire-induced mean core damage frequency for Surry Unit 1 was found to be 1.1E-5 per reactor year. The dominant contributing plant areas are the emergency switchgear room, auxiliary building, control room, and cable vault/tunnel. Fires in these four areas comprise 99 percent of the total fire core damage frequency. In the cases of the emergency switchgear room, cable vault/tunnel, and the auxiliary building, a reactor coolant pump seal loss-of-coolant accident (LOCA) leads to core damage. The fire itself fails cabling for both the high-pressure injection and component cooling water systems resulting in a seal LOCA. For the control room, a general transient with a subsequent stuck-open power-operated relief valve leads to a small LOCA. Failure to control the plant from the auxiliary shutdown panel results in core damage.

The overall fire-induced mean core damage frequency for Peach Bottom Unit 2 was found to be 1.9E-5 per reactor year. The dominant contributing plant areas are the control room, emergency switchgear room 2C, and emergency switchgear room 2B. Fires in these three areas comprise 75 percent of the total fire core damage frequency. In the case of the control room, a general transient occurs with smoke-induced abandonment of the area. Failure to control the plant from the remote shutdown panel results in core damage. For the two emergency switchgear rooms, a fire-induced loss of offsite power and failure of one train of the emergency service water (ESW) system occurs. Random failure of the other two ESW trains results in station blackout and core damage.

Detailed tracing of control and power cabling revealed that fires can damage the most plant safety functions in cable "pinchpoint" areas. Because of this added detail (cable location identification) as compared with past fire PRAs, many more fire scenarios arose in the initial phases of screening. However, most of these scenarios could be screened from further study by allowing for operator recovery of random failures. The final number of fire areas that survived screening and the total fire-induced core damage frequency for both Peach Bottom and Surry is similar to that found in previous fire PRAs but for dissimilar reasons involving plant-specific cabling configurations. These cable pinchpoint areas were also found in most cases to fail all containment cooling and spray systems. A review of past fire PRAs revealed a similar conclusion with respect to containment systems failure.

C.12.3 Issue Definition and Discussion

The critical fire risk issues and their effect on core damage frequency estimates will be discussed in this section. Uncertainty analysis results (documented in Refs. C.12.7 and C.12.8) highlighted the importance of most of the following issues.

A fire analysis must rely on a partitioning scheme (for fire frequency) to determine fire occurrence frequencies for any given plant area. Additional partitioning within most plant areas will occur because cable "pinchpoint" areas have typically been found to occur in 10 percent or less of these critical plant fire zones. This additional partitioning, based on fire propagation code calculations, is used to determine not only the area of influence of a fire but also the size of the fire required to damage the critical components or their cabling. In older fire PRAs (from the early 1980's), the COMPBRN I code (Ref. C.12.9) was used to determine fire zone specific area ratios and fire size estimates. Larger area ratios and longer time-to-damage estimates were calculated in the NUREG-1150 fire PRAs. As was discussed previously, the modified COMPBRN III code (Ref. C.12.5) was used for the Surry and Peach Bottom fire propagation predictions. For NUREG-1150, small fires were not found to cause damage in most cases. Therefore, fire frequencies were decreased to take credit for the fact that most fires in the data base would not have yielded damage. For plant areas where hot gas layer formation was not predicted, area ratios of 10 percent or less also led to reduced fire frequencies to reflect the situation that a postulated fire could only cause damage in a very specific area within a fire zone. These factors (area ratio and fire size estimate) reduced core damage frequency estimates from what would have been previously predicted (using COMPBRN). However, a competing effect also occurred. Shorter time to damage estimates had the effect of allowing less credit for manual suppression of these fire scenarios (with this impact being overshadowed by the other impacts noted above). Fire testing and code experts walked down each of the Peach Bottom and Surry fire areas to determine the reasonableness of the COMPBRN III predictions. It was their opinion, for the situations that occurred in NUREG-1150, that COMPBRN III yielded reasonable predictions. An analysis of the potential effect of COMPBRN results on fire-induced core

Table C.12.3 Dominant Surry accident sequence core damage frequency contributors (Ref. C.12.7).

Sequence	Fire Area	Mean Core Damage Frequency/yr
$T_3D_3WD_1$	Emergency Switchgear Room	6.1E–6
	Auxiliary Building	2.2E–6
	Cable Vault/Tunnel	1.5E–6
T_3QD_1	Control Room	1.6E–6
	Charging Pump, Service Water Pump Room	3.9E–8

KEY

Symbol	Definition
D_1	Failure of the charging pump system in the high-pressure injection mode.
D_3	Failure of the charging pump system in the seal injection flow mode.
Q	Failure of the SRVs/PORVs to close after a transient.
T_3	Transient consisting of a turbine trip with main feedwater available.
W	Failure of component cooling water to thermal barriers of all reactor coolant pumps.

Table C.12.4 Dominant Peach Bottom accident sequence core damage frequency contributors (Ref. C.12.8).

Sequence	Fire Area	Mean Core Damage Frequency/yr
$T_1BU_1U_2$	Emergency Switchgear Room 2A	7.4E–7
	Emergency Switchgear Room 2B	3.6E–6
	Emergency Switchgear Room 2C	3.6E–6
	Emergency Switchgear Room 2D	7.4E–7
	Emergency Switchgear Room 3A	7.4E–7
	Emergency Switchgear Room 3B	7.4E–7
	Emergency Switchgear Room 3C	7.4E–7
	Emergency Switchgear Room 3D	8.1E–7
$T_3U_1U_2X_1U_3$	Control Room	6.2E–6
	Cable Spreading Room	6.7E–7
$T_1BU_1W_1X_2W_2$ $W_3U_4V_2V_3Y$	Emergency Switchgear Room 2C	8.1E–7
$T_1BU_1W_1X_2W_2$ $W_3U_4V_2V_3Y$	Emergency Switchgear Room 2C	2.7E–7

Table C.12.4 (continued)

KEY

Symbol	Definition
B	Failure of all ac power (station blackout).
T_1	Loss of offsite power transient.
T_3	Transient consisting of a turbine trip with main feedwater available.
U_1	Failure of the high-pressure coolant injection (HPCI) system.
U_2	Failure of the reactor core isolation cooling (RCIC) system.
U_3	Failure of the control rod drive (CRD) system (2 pump mode).
U_4	Failure of the CRD system (1 pump mode).
V_2	Failure of the low-pressure core spray (LPCS) system.
V_3	Failure of the low-pressure coolant injection (LPCI) system.
W_1	Failure of the suppression pool cooling mode of the residual heat removal (RHR) system.
W_2	Failure of the shutdown cooling mode of the RHR system.
W_3	Failure of the containment spray mode of the RHR system.
X_1	Failure to depressurize the reactor coolant system via SRVs or the automatic depressurization system.
X_2	Failure to depressurize the reactor coolant system to allow the shutdown cooling mode of the RHR system to operate.
Y	Failure of primary containment venting (including makeup to the pool as required).

damage frequency predictions was conducted as part of the Fire Risk Scoping Study (Refs. C.12.10 and C.12.11). This analysis concluded that up to a factor of 20 difference was possible for some scenarios.

A second critical issue was the probability of operator recovery from the remote shutdown panel. In the NUREG-1150 fire analysis, this recovery action was quantified on a consistent basis with recovery probabilities calculated for the internal-event analyses. However, since no detailed development of control and actuation circuits was performed, complex control system interactions were not analyzed. If these types of interactions exist for either Surry or Peach Bottom, the effect would be to decrease the credit for recovery from the remote shutdown panel and to increase the fire-induced core damage frequency.

In two studies where detailed models were developed, interactions between the remote shutdown panel and the control room were found. These interactions existed even though the remote shutdown panel was found to be electrically independent of the control room.

The control systems interaction and fire code prediction issues are two of six issues addressed in the Fire Risk Scoping Study. Four other issues that will not be covered in detail here but also have the potential to have a significant effect on fire-induced core damage frequency are:

1. Manual fire brigade effectiveness,
2. Total environment survival,
3. Fire barrier effectiveness, and
4. Fixed fire suppression system damage effects.

Since the Fire Risk Scoping Study was being conducted concurrently with the NUREG-1150 fire analyses, these issues were not addressed in the Surry and Peach Bottom studies. However, any of these issues has the potential to increase the fire core damage frequency when included in an assessment.

A final issue is the Bayesian updating of generic fire initiating event frequencies to determine plant-specific fire frequencies. Several studies have found fire frequency on a plant-wide basis of 0.15 per reactor year. Therefore, any average plant will have only three to four significant fires over its operational lifetime. As was discussed previously, the NUREG-1150 fire analyses used a Bayesian updating procedure to determine plant-specific fire frequencies (for both Surry and Peach Bottom). The plant areas updated were:

1. Control room,
2. Cable spreading room,
3. Auxiliary building, and
4. Electrical switchgear room.

When comparing the Bayesian updated frequencies with the generic fire frequencies for these areas, no more than a factor of 3 difference typically occurred. These plant-specific frequencies were found to be both higher and lower than the average depending on a given plant's operating experience. Since this affects every fire cut set, no more than a factor of 3 difference (as compared with generic estimates) in core damage frequency estimates also occurred.

REFERENCES FOR SECTION C.12

C.12.1 Public Service Company of New Hampshire and Yankee Atomic Electric Company, "Seabrook Station Probabilistic Safety Assessment," Section 9.4, December 1983.

C.12.2 Philadelphia Gas and Electric Company, "Severe Accident Risk Assessment Limerick Generating Station," Chapter 4, Main Report, Report #4161, April 1983.

C.12.3 M. P. Bohn and J. A. Lambright, "Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150," Sandia National Laboratories, NUREG/CR-4840, SAND88-3102, November 1990.

C.12.4 U.S. Code of Federal Regulations, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to Part 50, "Domestic Licensing of Production and Utilization Facilities," of Chapter I, Title 10, "Energy."

C.12.5 V. Ho et al., "COMPBRN III—A Computer Code for Modeling Compartment Fires," Oak Ridge National Laboratory, NUREG/CR-4566, ORNL/TM-10005, April 1986.

C.12.6 R. L. Iman and M. J. Shortencarier, "A User's Guide for the Top Event Matrix Analysis Code (TEMAC)," Sandia National Laboratories, NUREG/CR-4598, SAND86-0960, August 1986.

C.12.7 J. A. Lambright et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2 External Events," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Part 3, SAND86-2084, December 1990.

C.12.8 J. A. Lambright et al., "Analysis of Core Damage Frequency: Surry Unit 2 External Events," Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Part 3, SAND86-2084, December 1990.

C.12.9 N. O. Siu, "COMPBRN—A Computer Code for Modeling Compartment Fires," University of California at Los Angeles, NUREG/CR-3239, UCLA-ENG-8257, May 1983.

C.12.10 V. F. Nicolette et al., "Observations Concerning the COMPBRN III Fire Growth Code," Sandia National Laboratories, SAND88-2160C, April 1989.

C.12.11 J. A. Lambright et al., "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," Sandia National Laboratories, NUREG/CR-5088, SAND88-0177, January 1989.

C.13 Containment Bypass Sequences

Accident sequences that involve bypass of the containment, thereby having a direct release path to the environment, were assessed to be important at all three PWRs and were risk dominant at the Surry and Sequoyah plants. The two categories of bypass sequences that were found to be important to the NUREG-1150 PWRs are interfacing-system loss-of-coolant accidents (ISLOCAs) and steam generator tube ruptures (SGTRs). The ISLOCA sequences can lead to a direct release path to the environment if failures occur that connect the high-pressure reactor coolant system (RCS) to a low-pressure interfacing system and subsequent failure of the low-pressure boundary occurs outside containment. SGTR sequences can lead to a direct release path to the environment if an SGTR event causes subsequent failure of the secondary side pressure boundary, usually through a stuck-open relief valve. Many different factors influence the initiation and progression of these accident sequences. Some of those factors were evaluated by expert panels, while others were evaluated by the analysis teams. Additional details on these analyses may be found in References C.13.1 and C.13.2.

C.13.1 ISLOCAs—Accident Sequence Issues

This section discusses those issues that arose in the accident sequence analysis of the interfacing-system loss-of-coolant accidents.

C.13.1.1 Issue Definitions

ISLOCAs are sequences in which the pressure isolation valves that separate low-pressure systems from the RCS fail, thus subjecting the low-pressure system piping and components to full RCS pressure. In many cases, the low-pressure system will subsequently fail, and there is a high probability that the failure will occur outside containment, usually in the auxiliary building or the safeguards area. As the RCS loses coolant inventory, makeup will be provided from the refueling water storage tank (RWST). When the water in the RWST is depleted, recirculation cooling is not possible because the water has been lost outside containment. Core damage will then occur if an alternative water supply is not provided. At issue in the accident sequence analysis is the frequency of check valve failures that represent the initiating event for this sequence. This issue was addressed by members of the accident frequency analysis expert panel, including:

Dennis Bley—Pickard, Lowe, and Garrick, Inc.,
Gary Boyd—SAROS, Inc.,
Robert Budnitz—Future Resource Associates, Inc.,
Karl Fleming—Pickard, Lowe, and Garrick, Inc., and
Garreth Parry—NUS Corporation.

C.13.1.2 Technical Basis for Issue Quantification

Three check valve failure scenarios were identified for the Surry and Sequoyah plants. (ISLOCAs were not found to be important at Zion because of the plant-specific configuration of the high-to-low pressure system interfaces.) The three scenarios considered by the expert panel were:

1. Random independent rupture (catastrophic leakage) of the two check valves in series.

2. Failure of one check valve to reclose upon RCS repressurization, followed by random rupture of the other valve.

3. The undetected failure of one valve during operation (between test periods), followed by rupture of the other valve. The undetected failure can be caused by opening of the disc or severe deterioration of the seat. This failure is not detected because the other valve is holding pressure. When the other valve fails, the ISLOCA is initiated.

The panel provided an ISLOCA initiating event frequency distribution considering the three failure modes listed above along with possible common-cause failures affecting valves in the same line. The model for Sequoyah was slightly different from the one for Surry because the accumulators inject between the two check valves at Sequoyah, thus making failures of the outboard check valve detectable. The expert panelists considered information from valve specialists and available data bases in developing their positions. A high degree of uncertainty is evident in the distributions because of the lack of directly applicable data.

C.13.1.3 Treatment in PRA and Results

The results of the interfacing-system LOCA expert elicitation were used as the primary basis for determining the frequency of the ISLOCA initiating event. The plant analysts provided information on the check valve testing frequency and procedures that influenced the models. Additionally, for Sequoyah some ISLOCAs can be isolated, depending upon the break location. The plant analysts provided the probability that the operators would isolate the break, if possible. Except for the probability of isolating the break at Sequoyah, the ISLOCA events are assessed to proceed to core damage with unity probability, given the initiating event. The probability distributions for the initiating event that resulted from the expert elicitation are shown below. These distributions are for a single high-to-low pressure interface. At Surry, there are three such interfaces, while Sequoyah has four.

Percentile	Frequency per Reactor Year Surry	Frequency per Reactor Year Sequoyah
5th	1.3E-11	1.2E-11
50th	1.6E-8	9.5E-9
95th	1.7E-6	1.2E-6
Mean	3.8E-7	2.7E-7

C.13.2 ISLOCAs—Source Term Issues

This section discusses those issues that arose in the source term analysis of ISLOCAs. In the source term analysis, this accident was usually referred to as Event V.

C.13.2.1 Issue Definition

A number of factors influence the source terms that result from ISLOCA sequences. Because the check valve failures result in large leakage paths, the RCS will be at low pressure. The in-vessel behavior is expected to be similar to that of other, non-bypass, low-pressure sequences. Releases from the auxiliary building are strongly influenced by the location of the break in the auxiliary building and whether or not the break is ultimately covered by water, thereby leading to pool scrubbing. Fire suppression sprinklers in the auxiliary building may also act to reduce the releases. Some of these issues were addressed by the source term expert panel, some by an expert panel for the first draft of NUREG-1150, and some by the analysis teams.

C.13.2.2 Technical Basis for Issue Quantification

The releases for ISLOCA are calculated with the regular early and late release equations in the XSOR codes (see the description of the XSOR codes in Appendix A and in Ref. C.13.3). There are specific distributions for ISLOCA for the factor that accounts for the release from the vessel and for the factor that accounts for the release from the building. For ISLOCA, the vessel release factor, which represents the fraction released from the vessel to the containment in non-bypass accidents, is redefined to represent the release fraction from the vessel to the auxiliary building through the high-pressure and low-pressure piping. Because ISLOCA is a low-pressure sequence, the distributions for the ISLOCA vessel release factor are similar to those for non-bypass low-pressure accidents. For ISLOCA, the building release factors, which represent the fraction released from the containment building in non-bypass accidents, are redefined so that they account for the release fraction from the auxiliary building to the environment. These factors do not include the effects of pool scrubbing or auxiliary building sprays; reductions of the release due to mitigating effects such as these are accounted for by a separate decontamination factor. The distributions for the ISLOCA building release factor are similar to those for an early rupture of containment.

At Surry, the releases from an ISLOCA will be reduced by passage through a water pool if the pipe break location is under water. Pool scrubbing at Surry is treated by a specific pool scrubbing distribution that is used for the decontamination factor. At Sequoyah, the ISLOCA releases will be reduced if the thermally activated auxiliary building fire spray system operates. Spray scrubbing at Sequoyah is treated by a specific spray scrubbing distribution that is used for the decontamination factor. The decontamination factor is applied in addition to the building release factor. Some of the distributions used in determining the ISLOCA release were determined by the source term expert panel, some by an expert panel for the first draft of NUREG-1150, and some by the project staff.

C.13.2.3 Treatment in PRA and Results

The ISLOCAs were treated similarly to the non-bypass accidents in the PWR XSOR codes insofar as possible. The vessel release factor and the building release factor were redefined for application to ISLOCAs.

The vessel release factor is denoted FVES in SURSOR and SEQSOR. Distributions for this factor for the non-bypass accidents were determined by the source term expert panel. Distributions for FVES for ISLOCA were determined by the project staff; they were based on the low-pressure non-bypass FVES distributions of the source term expert panel. It was determined that the Ba, Sr, Ru, La, and Ce radionuclide classes would all behave as aerosols, so the distributions for these classes are identical. Three points on the distributions for the nine radionuclide classes are given below for Surry and Sequoyah. NG denotes the noble gases. In SURSOR and SEQSOR this accident is called Event V.

FVES distributions for Event V for Surry and Sequoyah

Percentile	NG	I	Cs	Te	Ba	Sr	Ru	La	Ce
5	1.0E+0	1.6E-1	1.5E-6	6.4E-2	1.0E-1	1.0E-1	1.0E-1	1.0E-1	1.0E-1
50	1.0E+6	6.1E-1	6.0E-1	2.5E-1	3.5E-1	3.5E-1	3.5E-1	3.5E-1	3.5E-1
95	1.0E+0	9.0E-1	9.7E-1	9.3E-1	9.6E-1	9.6E-1	9.6E-1	9.6E-1	9.6E-1

There are two building release factors in SURSOR and SEQSOR. FCONV is the fraction of the fission products in the auxiliary building from the RCS release that is released to the environment in the absence of mitigating factors. FCONC is the fraction of the fission products in the auxiliary building from the core-concrete interaction (CCI) release that is released to the environment in the absence of mitigating factors. The RCS release is sometimes called the early release, and the CCI release is sometimes referred to as the late release. Distributions for FCONV and FCONC for ISLOCA were determined by the project staff.

It was determined that, for the escape from the containment for the early release, one distribution could be applied to all radionuclide classes except the noble gases. FCONV is unity for the noble gases since they are not held up or absorbed in the reactor vessel. The values for the 5th, 50th, and 95th percentiles for the Surry and Sequoyah are:

FCONV distributions for Event V for both Surry and Sequoyah

Percentile	FCONV
5	1.6E-1
50	5.0E-1
95	8.6E-1

For the escape of fission products released during CCI, it was concluded that iodine and cesium would behave similarly, and all the other radionuclides except noble gases would behave as aerosols. The values for the 5th, 50th, and 95th percentiles for Surry and Sequoyah are given below.

FCONC distributions for Event V for Surry and Sequoyah

Percentile	NG	I	Cs	Te	Ba	Sr	Ru	La	Ce
5	1.0E+0	1.1E-1	1.1E-1	9.3E-2	9.3E-2	9.3E-2	9.3E-2	9.3E-2	9.3E-2
50	1.0E+0	5.0E-1	5.0E-1	4.7E-1	4.7E-1	4.7E-1	4.7E-1	4.7E-1	4.7E-1
95	1.0E+0	7.7E-1	7.7E-1	7.7E-1	7.7E-1	7.7E-1	7.7E-1	7.7E-1	7.7E-1

At Surry, much of the low-pressure piping that will be exposed to full RCS pressure in Event V is located near the floor in an area of the auxiliary building that will retain water. Since the contents of the RCS and the RWST will escape out the break before the onset of core damage, there is a good chance the pipe break will be under several feet of water when the release commences. Whether the pipe break is under water is determined in the accident progression analysis, based on the results of a panel convened for the first draft of NUREG-1150 to consider this specific question. The conclusion of this panel was that the mean probability that the Event V low-pressure pipe break at Surry will be under water is 0.85.

At Sequoyah, the pipe break is likely to be in an area of the auxiliary building that has heat-activated fire sprays (unlike Surry, which has no such sprays). There is no possibility at Sequoyah that the water escaping from the low-pressure pipe break in Event V will collect so that the pipe break will be under several feet of water when the release commences. It was decided by the project staff that there would be a high probability that the releases would be subject to scrubbing by fire sprays (the mean probability was assessed to be about 0.80).

The source term expert panel decided that the uncertainty in the distribution used for the decontamination factor for sprays or passage through a water pool was less important than the uncertainty in whether the sprays were on or whether the pipe break was under water. The decontamination factors for both sprays and pool scrubbing were therefore determined by the project staff. There is a single decontamination factor distribution that applies to all radionuclide classes except the noble gases, which are not affected by sprays or a passage through water.

Decontamination factor (DF) distribution for Event V at Surry (pool scrubbing) and at Sequoyah (sprays)

Percentile	DF
5	4.1E+3
50	6.2
95	1.8

C.13.3 SGTRs—Accident Sequence Issues

This section discusses those issues that arose in the accident sequence analysis of the steam generator tube ruptures.

C.13.3.1 Issue Definition

A steam generator tube rupture is a unique initiator because it represents a failure of the RCS boundary into the secondary side. As a result of the tube rupture, the secondary side may be exposed to full RCS pressures. These pressures are likely to cause relief valves to lift on the secondary side. If these valves fail to reclose, an open pathway from the vessel to the environment can result. Thus, this accident has some of the same characteristics as an ISLOCA. In determining the frequency of SGTR sequences, there are a number of important issues, including the SGTR initiating event frequency, the likelihood that the operators will depressurize the RCS to a pressure lower than in the secondary side so as to terminate the loss of coolant, and the response of the secondary side safety valves. The plant analysts addressed all the issues except those dealing with the safety valves; the latter were addressed in an elicitation process involving the project staff. The nature and response of safety valves to various types of demands were the focus of the elicitation process.

C.13.3.2 Technical Bases for Issue Quantification

The consideration of secondary side safety valve behavior considered the following four questions:

1. What is the probability of a secondary safety valve demand?

2. What is the probability that, given a safety valve demand, liquid or two-phase flow through the safety valves occurs?

3. What is the probability of failure to reclose, given that the safety valves pass liquid or a two-phase mixture?

4. How many demands of the secondary safety valves will occur?

The evaluation of these questions was based on consideration of plant emergency operating procedures, thermal-hydraulic calculations, and data from actual SGTR events, such as at North Anna. The likelihood of a secondary side safety valve demand depends largely on the status of the atmospheric dump valves (ADVs) and their block valves. If the ADVs are operating in an unblocked mode, their operation may preclude demands of the safety valves. On the other hand, if the ADVs are blocked, demand of the safety valves is likely. The passage of liquid through the valves depends on the ability of the operators to prevent filling up of the faulted steam generator. The operators need to control the RCS pressure and reduce the flow from the high-pressure injection system in order to preclude liquid flow through the safety valves. Given that liquid flow occurs and that multiple demands are considered likely, it was assessed that there was a high likelihood that the safety valves would stick open.

C.13.3.3 Treatment in PRA and Results

For the PRAs, the plant analysts identified the possible situations where safety valve demand might be of concern. The mean probabilities of having a stuck-open safety valve were assessed for the eight cases identified below. Cases 2 and 6 were assessed to be inapplicable because it is impossible to control RCS pressure if safety injection is not also controlled.

Table C.13.1 Secondary side safety valve failure probabilities.

	Case							
	1	2	3	4	5	6	7	8
Primary Pressure Controlled	Y	Y	N	N	Y	Y	N	N
Safety Injection Controlled	Y	N	Y	N	Y	N	Y	N
ADVs Blocked	Y	Y	Y	Y	N	N	N	N
Mean Probability of Stuck-Open Safety Valve	.28	NA	1.0	1.0	0.0	NA	.15	.15

Cases conditional upon the failure of the operators to depressurize were assessed to be the most likely to lead to safety valve demand and subsequent failure. The plant analysts then combined this information with the analysis of operator actions to depressurize and take other mitigative steps in order to determine an overall frequency of SGTR core damage events. These frequencies were estimated to be quite low at the three PWRs in NUREG-1150; however, they remain important relative to other sequences because of the possibly large source term that can result from a bypassed containment.

C.13.4 SGTRs—Source Term Issues

This section discusses those issues that arose in the source term analysis of the steam generator tube ruptures.

C.13.4.1 Issue Definition

The magnitude of the source term from an SGTR accident depends on the integrity of the secondary system and the containment. If the integrity of both is maintained, the releases may be quite small. On the other hand, if the safety valves on the secondary system stick open, then a direct path from the vessel to the environment is created and the releases may be very high. If the safety valves on the secondary system do not stick open, the releases depend on the time at which the containment fails (if at all) as in non-bypass accidents. Thus, there are two pathways from the reactor vessel to the environment. The one pathway is through the broken steam generator tube and out through the safety valves on the secondary system. The second pathway is the same as the pathway for non-bypass transient accidents: into the containment through a safety or relief valve (if open) until the vessel fails and directly after the vessel fails; and from the containment to the environment through the failure in the containment (if it fails). If the safety valves on the secondary system stick open, the first pathway dominates. This accident is of considerable concern because of the potentially large releases that might result. If the safety valves on the secondary system do not stick open, the releases by the second (non-bypass) pathway are more important.

C.13.4.2 Technical Basis for Issue Quantification

The equation used to determine the early release for SGTR accidents differs from the usual early release equation in that it contains an additional term for the release through the secondary system and the secondary system safety valves. If the secondary safety valves are stuck open, this bypass term accounts for the bulk of the release. For the non-bypass or normal pathway to the environment via the containment, the SGTR accidents are treated no differently than similar non-bypass accidents. Only the vessel release factor has a specific distribution for the SGTR accidents. This distribution was determined by the project staff.

The additional (bypass) term for the RCS release for SGTR accidents consists of two factors. One represents the fraction released from the core that enters the secondary side of the steam generator; the second is the fraction entering the secondary side of the steam generator that is released to the environment through the safety valves. No removal factors, such as building or spray decontamination factors, are applied to this release path. For the SGTRs where the secondary system safety valves reclose, the distributions for these two factors were determined by the project staff. For the SGTRs where the secondary system safety valves stick open, the distributions for these two factors were determined by an expert panel convened specifically to consider the release from this type of SGTR.

C.13.4.3 Treatment in PRA and Results

For the non-bypass pathway to the environment via the containment, the radioactive material released from the PWRs are computed using the same equation used for non-bypass accidents. The only factor for which specific distributions are defined is the vessel release factor, FVES, which represents the fraction passing from the vessel to the containment. It cannot be the same as in a non-bypass accident since some of the fission products escape from the vessel into the steam generator. Distributions for FVES for SGTRs were determined by the project staff. In this situation, the Ba, Ru, La, and Ce radionuclide classes would all behave as aerosols so the distributions for these classes are identical. Strontium is somewhat less likely to be released.

FVES distributions for SGTRs for Surry and Sequoyah

Percentile	NG	I	Cs	Te	Ba	Sr	Ru	La	Ce
5	1.0E+2	2.4E-2	2.4E-2	8.8E-2	1.3E-2	9.4E-3	1.3E-2	1.3E-2	1.3E-2
50	1.0E+0	1.7E-1	1.7E-2	3.2E-1	5.8E-2	4.4E-2	5.8E-2	5.8E-2	5.8E-2
95	1.0E+0	8.7E-1	8.2E-1	9.5E-1	7.3E-1	6.7E-1	7.3E-1	7.3E-1	7.3E-1

In SURSOR and SEQSOR, FISG represents the fraction released from the core that enters the steam generator; and FOSG represents the fraction entering the steam generator that is released from the steam generator to the environment. For SGTRs in which the secondary system safety valves reclose, the distributions for FISG and FOSG were determined by the project staff and separate distributions were defined for each of these factors. It was determined that the Ba, Sr, Ru, La, and Ce radionuclide classes would all behave as aerosols so the distributions for these classes are very similar.

FISG distributions for SGTRs with the secondary safety valves reclosing for Surry and Sequoyah

Percentile	NG	I	Cs	Te	Ba	Sr	Ru	La	Ce
5	2.5E-1	1.1E-1	1.0E-1	2.9E-1	1.6E-1	1.7E-1	1.7E-1	1.7E-1	1.6E-1
50	5.8E-1	2.9E-1	2.8E-1	5.6E-1	3.3E-1	3.4E-1	3.4E-1	3.4E-1	3.3E-1
95	9.5E-1	8.5E-1	7.7E-1	1.0E+0	9.0E-1	9.0E-1	9.0E-1	9.0E-1	9.0E-1

FOSG distributions for SGTRs with the secondary safety valves reclosing for Surry and Sequoyah

Percentile	NG	I	Cs	Te	Ba	Sr	Ru	La	Ce
5	2.9E-1	2.0E-1	2.0E-1	2.6E-1	2.6E-1	2.5E-1	2.5E-1	2.6E-1	2.6E-1
50	6.7E-1	5.3E-1	5.4E-1	5.0E-1	5.2E-1	5.3E-1	5.3E-1	5.3E-1	5.3E-1
95	1.0E+0	1.0E+0	1.0E+0	9.2E-1	1.0E+0	1.0E+0	1.0E+0	1.0E+0	1.0E+0

For SGTRs in which the secondary system safety valves stick open, the distributions for FISG and FOSG were determined by a special source term expert panel that considered only the releases from this type of SGTR accident. The panel declined to provide separate distributions for FISG and FOSG and provided distributions for the product FISG * FOSG. Because the bypass term for the SGTR release consists of just the product of FISG and FOSG, and these factors appear nowhere else in the release equation, this was acceptable. The experts provided separate distributions for iodine, cesium, tellurium, and aerosols and specified that they applied to all the PWRs. There is no retention in the steam generators for the noble gases. The aerosol distribution applies to the Ba, Sr, Ru, La, and Ce radionuclide classes. The panel concluded that there was a very good chance that there would be little retention of radionuclides in the steam generator and the piping.

FISG * FOSG distribution for SGTRs with the secondary safety valves stuck open for Surry and Sequoyah

Percentile	NG	I	Cs	Te	Ba	Sr	Ru	La	Ce
5	1.0E+0	2.0E-5	2.8E-7	3.8E-5	3.8E-5	3.8E-5	3.8E-5	3.8E-5	3.8E-5
50	1.0E+0	2.7E-1	2.6E-1	1.7E-1	2.4E-1	2.4E-1	2.4E-1	2.4E-1	2.4E-1
95	1.0E+0	8.0E-1	7.8E-1	7.7E-1	7.6E-1	7.6E-1	7.6E-1	7.6E-1	7.6E-1

As implemented in SURSOR and SEQSOR, rather than change the equation for the SGTRs in which the secondary system safety valves stick open, the distributions for the product FISG * FOSG were placed in the data array for FISG and the data array for FOSG was set to 1.0.

REFERENCES FOR SECTION C.13

C.13.1 T. A. Wheeler et al., "Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation," Sandia National Laboratories, NUREG/CR-4550, Vol. 2, SAND86-2084, April 1989.

C.13.2 F. T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

C.13.3 H. N. Jow et al., "XSOR Codes User's Manual," Sandia National Laboratories, NUREG/CR-5360, SAND89-0943, to be published.*

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

C.14 Reactor Coolant Pump Seal Failures in Westinghouse Plants After Loss of All Seal Cooling

C.14.1 Issue Definition

The ability of reactor coolant pump seals to remain intact during loss of all seal cooling is a concern for PWRs. As the three PWRs studied for NUREG-1150 are all Westinghouse plants, results of the analysis of this study are only applicable to Westinghouse plants.

One of the potential concerns during a station blackout is the possibility of a loss-of-coolant accident (LOCA) created by failure of the reactor coolant pump (RCP) seals to maintain a restricted flow between the reactor coolant system and the containment.

The Westinghouse RCP shaft seal is a three-stage seal assembly, as shown in Figure C.14.1. The number one seal is a film-riding controlled leakage seal, whereas the number two and three stages are rubbing-face type seals. The leakage across the number one seal cools the seal assembly. The high-pressure subcooled leakage is supplied by an injection system upstream of the number one seal. Part of the injection water flows through the seal assembly. The remainder flows into the water coolant system as makeup water. Backup cooling is provided by a water-to-water heat exchanger parallel to the labyrinth seal. During a station blackout, both injection and cooling water would be lost.

High-temperature reactor coolant water would then flow up the shaft into the seal system. The shaft and the seal assembly would experience abnormal temperature distributions. This condition will affect the angle between the faceplates of the RCP seals and the gap between the faceplates of the number one film-riding seal.

The gap between the number one seal faceplates is determined by a force balance, which can be affected by flow rate, the angle between the faceplates of the seal ring and runner, enthalpy, and inlet pressure. The fluid pressure profile between the seal faceplates determines the opening forces on the seal. The closing force on the seal is proportional to the differential pressure across the seal and acts on the upper surface of the seal ring. If these forces are unbalanced, the gap will increase or decrease as necessary until the forces are balanced.

The number two seal stage is designed to withstand full system pressure without loss of integrity in the event that number one seal stage fails. In the event that both number one and number two seals fail, the number three seal stage is not expected to limit leakage. The size of a resultant leak rate is dependent on the combinations of seal ring failures and o-ring failures in the various seal stages.

At issue is the leak rate in gallons per minute as a function of time, due to seal failure caused by loss of cooling to the pump shaft, which is expected during station blackout. The hypothesized failure modes involve loss of the seal ring geometry and degradation of the elastomer material. (Additional detail on this analysis may be found in Ref. C.14.1.)

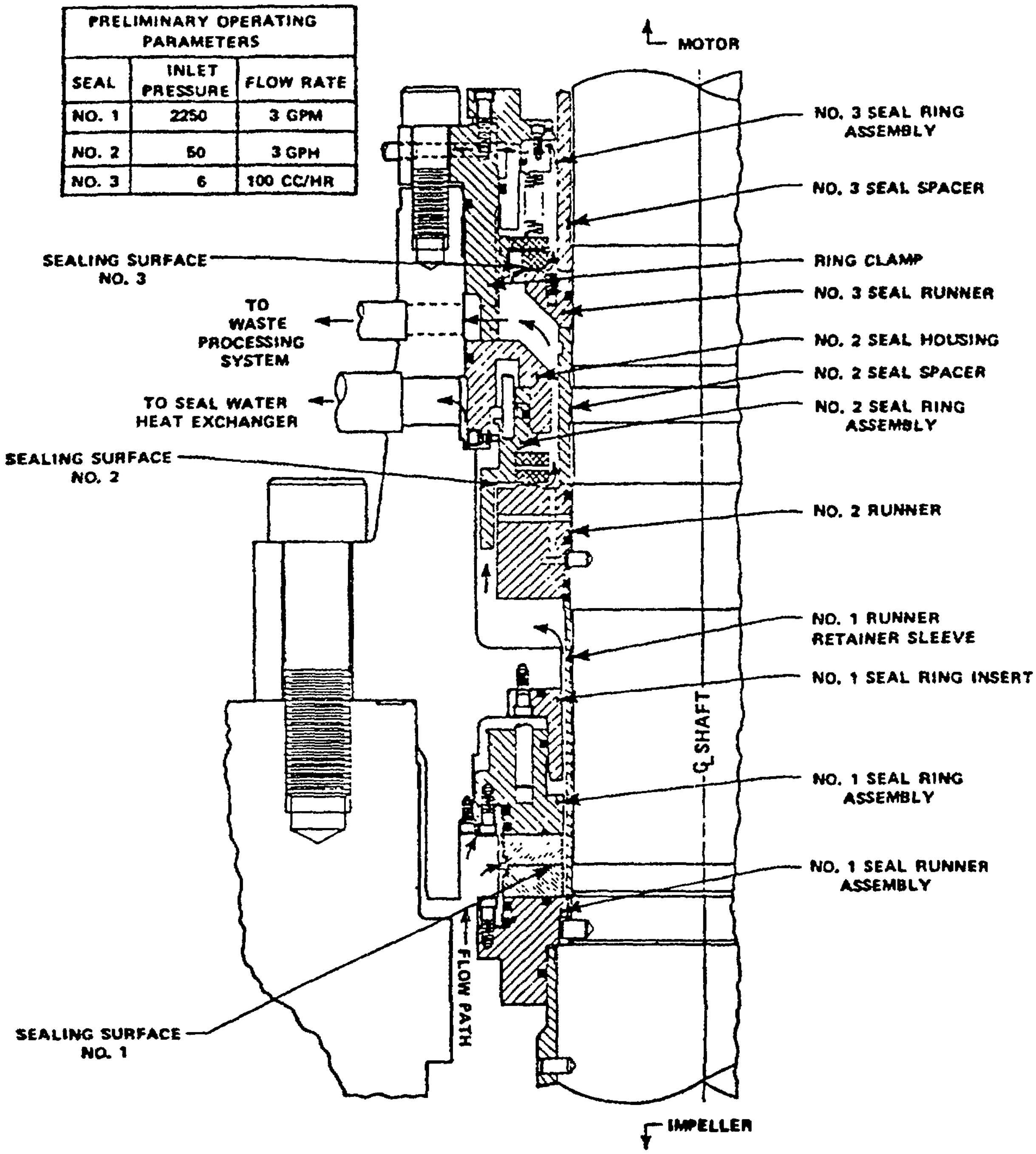
C.14.2 Technical Bases for Issue Quantification

For this issue, three experts participated in these evaluations:

M. Hitchler—Westinghouse Electric Corporation,
J. Jackson—Nuclear Regulatory Commission, and
D. Rhodes—Atomic Energy of Canada Limited

These experts developed a consensus structure with which to analyze the issue. The result was a logic tree that describes the possible failure combinations of seal rings and o-rings and the resultant leak rates for a single pump. This logic tree is shown in Figure C.14.2. The 21-gpm leak per pump is considered to be a successful restriction of flow.

The experts then provided assessments for the failure probabilities for the four events of the logic tree. Assessments were made for two different o-ring elastomer materials. The old o-ring material, which is currently in use at Surry, Sequoyah, and Zion, has exhibited significant degradation in some experiments.



PRELIMINARY OPERATING PARAMETERS

SEAL	INLET PRESSURE	FLOW RATE
NO. 1	2250	3 GPM
NO. 2	50	3 GPH
NO. 3	6	100 CC/HR

Figure C.14.1 Westinghouse RCP seal assembly (Ref. C.14.1).

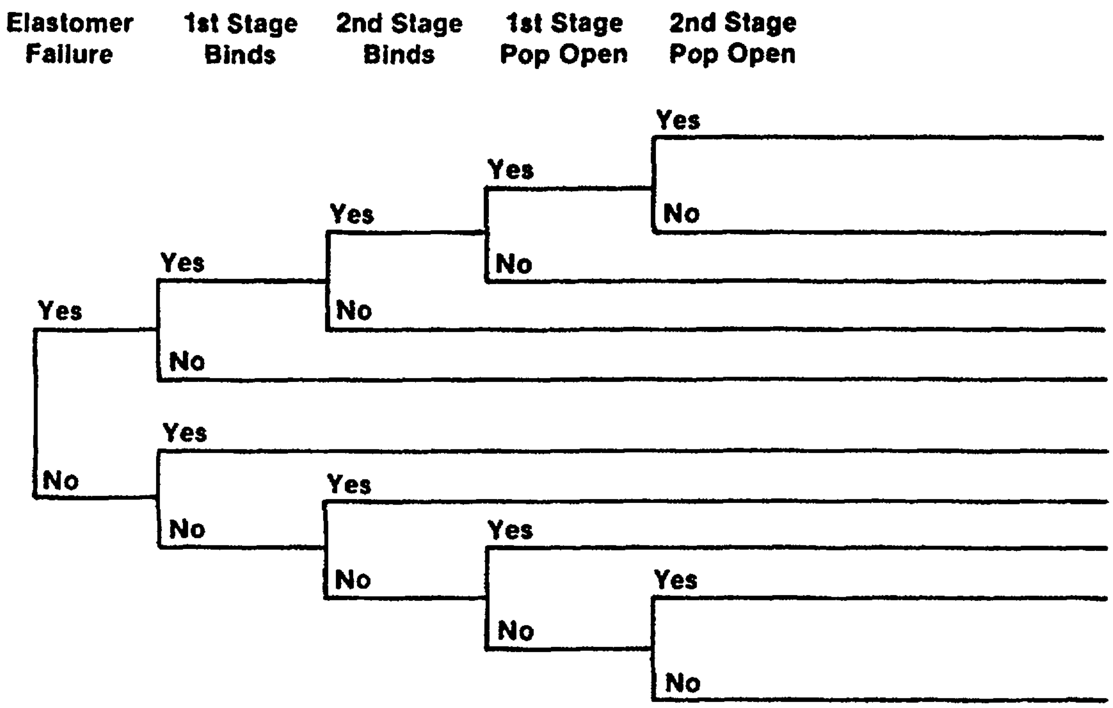


Figure C.14.2 Decision tree (Ref. C.14.1).

The new o-ring elastomer has been shown to be much less susceptible to degradation when subjected to similar experimental conditions. In addition to the type of o-ring material in the RCP seals, elicitations were taken for pressurized and depressurized reactor coolant system (RCS) status. Successful cooldown and depressurization will subject the pump seals to cooler, less harsh conditions.

Total RCS leak rate is based on leak rates from all RCP pumps (3 for Surry, 4 for Sequoyah and Zion). Therefore, in addition to developing an assessment for a single RCP, it was necessary to postulate the degree of correlation between similar failures in different pumps. That is, the assessment included consideration of whether the seals fail independently or have the same leakage.

C.14.3 Treatment in PRA and Results

In order to evaluate the station blackout event tree, it is necessary to have a relationship between leak rate and probability of leak rate versus time. The results for each expert were averaged to calculate aggregate leak rates and their probabilities. The calculations were done for the following three cases:

1. Old o-ring material—with RCS cooldown
2. Old o-ring material—without RCS cooldown
3. New o-ring material—without RCS cooldown

The experts developed very different assessments for correlating component failures between pumps. They reached a consensus on this point: if two similar components in two pumps failed (e.g., first stage seal rings in two different pumps), then the same component could be assumed to fail in all other pumps. This assumption simplified the problem of calculating total RCP seal leak rate probabilities for determining the possible failure combinations for a two-pump model for each expert.

The results are shown in Table C.14.1 for a three-loop Westinghouse plant and Table C.14.2 for a four-loop plant. These tables show the probability of various leak rates at different points in time after loss of cooling to the seals. In order to incorporate this information into the sequence models, it was necessary to calculate a time of initial seal failure and a core uncovery time for each possible failure scenario. A

Table C.14.1 Aggregated RCP seal LOCA probabilities for Westinghouse three-loop plant.*

Leak Rate (gpm)	Old O-Rings Time (h)					New O-Rings Time (h)				
	1.5	2.5	3.5	4.5	5.5	1.5	3.6	3.5	4.5	5.5
63	.306	.290	.274	.274(.258)**	.274(.241)	.817	.816	.814	.812	.811
103	-	-	-	-	-	7.7E-3	7.7E-3	7.7E-3	7.7E-3	7.7E-3
183/224***	.148	.037	.050	.048(.064)	.047(.079)	.014	.014	.016	.017	.019
294	-	-	-	-	-	1.9E-3	1.9E-3	1.9E-3	1.9E-3	1.9E-3
372	8.5E-3	5.0E-3	4.5E-3	3.7E-3	3.3E-3	4.5E-4	5.0E-3	5.3E-3	5.7E-3	6.0E-3
516/526/546	3.5E-4	3.4E-4	3.2E-4	3.2E-4	3.2E-4	.145	.145	.145	.145	.145
602/614	.001	0	0	0	0	4.7E-4	4.7E-4	4.7E-4	4.7E-4	4.7E-4
750	.530	.660	.660	.660	.660	7.7E-3	7.7E-3	7.7E-3	7.7E-3	7.7E-3
1440	4.3E-3	4.3E-3	4.3E-3	4.3E-3	4.3E-3	5.0E-3	5.0E-3	5.0E-3	5.0E-3	5.0E-3

*This table is Table 5.4-1 of Reference C.14.1.
**Parentheses denote calculations that change if no depressurization is assumed. All other probabilities are for depressurized conditions.
***Similar leak rates have been lumped together. These values are the probabilities of being at a particular leak rate at a particular time.

Table C.14.2 Aggregated RCP seal LOCA probabilities for Westinghouse four-loop plant.*

Leak Rate (gpm)	Old O-Rings Time (h)					New O-Rings Time (h)				
	1.5	2.5	3.5	4.5	5.5	1.5	3.6	3.5	4.5	5.5
	1.5	2.5	3.5	4.5	5.5	1.5	3.6	3.5	4.5	5.5
84	.302	.286	.271	.274(.255)	.271(.239)**	.810	.809	.809	.807	.805
244/245***	.148	.038	.053	.051(.067)	.049(.081)	.014	.016	.017	.0198	.020
313	–	–	–	–	–	.010	.010	.010	.010	.010
433	.011	.012	.028	9.9E-3	9.3E-3	6.0E-4	6.0E-4	6.0E-4	6.0E-4	6.0E-4
480	1.3E-3	1.3E-3	1.3E-3	1.3E-3	1.3E-3	–	–	–	–	–
543	–	–	–	–	–	2.6E-3	2.6E-3	2.6E-3	2.6E-3	2.6E-3
688/698/728	1.2E-3	1.2E-3	1.1E-3	1.1E-3	.146	.146	.146	.146	.146	.146
796	–	–	–	–	–	2.7E-3	2.7E-3	2.7E-3	2.7E-3	2.7E-3
1000/1026	.530	.659	.659	.665	.666	8.3E-3	8.3E-3	8.3E-3	8.3E-3	8.3E-3
1230	1.6E-6	1.6E-3	1.6E-3	1.6E-3	1.6E-3	–	–	–	–	–
1920	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3

*This table is Table 5.4-2 of Reference C.14.1.

**Parentheses denote calculations that change if no depressurization is assumed. All other probabilities are for depressurized conditions.

***Similar leak rates have been lumped together. These values are the probabilities of being at a particular leak rate at a particular time.

series of individual scenarios were defined that identify the time of seal failure, the initial leak rate, the progression of leak rate, and the probability of the scenario.

For Sequoyah (a four-loop plant), a total of 17 scenarios were identified and are shown in Table C.14.3. They include the initial leak rate, the time of initial seal failure, any increases in leak rate, the time at which the leak rate increases, and the probability. These 17 scenarios were used to develop point estimate probabilities (i.e., no uncertainty stated) for seal failure and core uncovery. These values were not used in the uncertainty analysis. In order to calculate the probability distributions for seal LOCA sequences, the 17 scenarios were consolidated into seven states. There are six failure states and one success rate (the 84 gpm state is a success state). The seven seal states are summarized below:

Leak Path (gpm)	Time to Transfer (hours)	Probability
1000*	1-1/2	.5298
240 - 1000	2-1/2	.1253
240*	1-1/2	.049
433 - 1000	2-1/2	.0051
1920*	1-1/2	.0042
433*	1-1/2	.0042
84* (success)		.2704

*Constant leak rate

This probability distribution is interpreted as a representation of the experts' collective degrees of belief in the seven states that represent possible outcomes for seal LOCA. The occurrence of a seal LOCA is therefore treated as a modeling uncertainty with respect to time and size. These states were sampled as either a zero or unity probability in the TEMAC uncertainty analysis.

The following discussion illustrates how the seal LOCA model was integrated into the station blackout event trees. (Two constraining criteria were applied to this task: (1) the event heading for nonrecovery of ac power would be separate from the event heading for seal LOCA, and (2) a minimum number of headings would be used.)

The conclusion of the expert panel was that, at 90 minutes after loss of all cooling, the seal temperatures would have increased enough to be at risk of failure. Prior to 90 minutes, there is no risk of seal failure. After 90 minutes, with no cooling, the seal may fail or may remain intact. Leaks may slowly develop and increase with time, or they may have a constant leak rate. If a seal LOCA occurs, core uncovery can be averted if ac power is restored, thus enabling restoration of safety injection flow.

The calculation of the core damage frequency due to seal LOCA is based on a weighted average of the 17 seal failure scenarios. The conditional probability of core damage, given a station blackout, is calculated as follows:

$$P'_{CD} = P_1 * P_2 * P_3$$

where

P'_{CD} = conditional probability of core damage,
P_1 = probability of being at risk for an SLOCA,
P_2 = probability that a SLOCA occurs, and
P_3 = probability of not recovering ac power prior to core damage.

Table C.14.3 Sequoyah RCP seal LOCA model scenarios.*

Leak Path (gpm)	Time to Transfer (hours)	Probability
84**	—	.2707
240**	—	.0490
240–1000	2-1/2	.1253
240–1000	3-1/2	.0021
240–1000	4-1/2	.0021
240–1000	5-1/2	.0010
433**	—	.0042
433–1000	2-1/2	.0051
433–1000	3-1/2	.0013
433–1000	4-1/2	.0006
433–1230	2-1/2	.0013
700**	—	.0011
700–1000	2-1/2	.00007
700–1000	3-1/2	.00007
1000**	—	.5298
1230**	—	.0016
1920**	—	.0042
	TOTAL	.9995

*This is Table D.5-2 in Reference C.14.2.
**Constant leak rate

The probability of being at risk for a seal LOCA is the probability that ac power has not been restored within 90 minutes of a loss of a seal cooling. The probability of a seal LOCA is given by the results of the expert elicitation. There are 17 seal LOCA scenarios, each with a characteristic uncovery time and a specific probability. All seal failure scenarios are assumed to start at 90 minutes from loss of cooling. The probability of not recovering is just the probability of nonrecovery of ac power prior to the characteristic core uncovery time associated with each seal scenario. Note that the nonrecovery term must be conditional on nonrecovery of ac power in the first 90 minutes. The conditional probability of core damage can be written as:

$$P_{CD} = \sum_{i=1}^{17} [P(t)_{NRAC} \, * \, P(t)_{fsl(i)} * P'(t+\beta_i)_{NRAC}]$$

where

i = seal LOCA scenario index, and t, in this case, equals 90 minutes,

β_i = core uncovery time associated with the ith scenario,

$P(t)_{NRAC}$ = probability of nonrecovery of ac power by time t, given loss of power at t = 0.

$P(t)_{NRAC}$ = $1 - P'(t)_{NRAC}$ where P" is the cumulative probability of recovery of ac power,

$P(t)_{fsl(i)}$ = probability of ith seal LOCA scenario, and

$P'(t+\beta_i)_{NRAC}$ = conditional probability of nonrecovery of ac power by time $t + \beta_i$, given no recovery at time t.

$$P'(t+\beta_i)_{NRAC}] = \frac{P(t+\beta_i)_{NRAC}}{P(t)_{NRAC}} = \frac{1 - P'(t+\beta)_{NRAC}}{1 - P'(t)_{NRAC}}$$

Recognizing the form for $P'_{NRAC'}$, the equation reduces to:

$$\sum_{i=1}^{17} P(t)_{fsl(i)} * P(t+\beta_i)_{NRAC}$$

The values for $P(t)_{fsl(i)}$ and $P(t+\beta_i)_{NRAC}$ are shown in Table C.14.4. Core uncovery times were calculated for the case with and without secondary depressurization.

Table C.14.4 Sequoyah RCP seal LOCA model.*

Leak Path (gpm)	Time to Transfer (hours)	Prob.	Time to CU (hr)	Time to RAC (hr)	Prob. NRAC	Prob. CD	Time to CU (hr)	Time to RAC (hr)	Prob. NRAC	Prob. CD
			(with secondary depressurization)				(without secondary depressurization)			
84**	—	.2707	19	20.5	.046	—	8.9	10.4	.046	—
240**	—	.0490	8.1	9.6	.046	.00225	4.6	6.1	.056	.00274
240–1000	2-1/2	.1253	2.2	3.7	.10	.01253	2.15	3.65	.103	.01291
240–1000	3-1/2	.0021	3.06	4.56	.08	.00017	2.91	4.4	.084	.00018
240–1000	4-1/2	.0021	3.9	5.4	.067	.00014	3.7	5.2	.068	.00014
240–1000	5-1/2	.0010	4.7	6.2	.054	.00005	4.4	5.9	.058	.00006
433**	—	.0042	4.2	5.9	.058	.00024	3.1	4.6	.079	.00033
433–1000	2-1/2	.0051	2.06	3.56	.108	.00055	1.96	3.46	.110	.00056
433–1000	3-1/2	.0013	2.73	4.23	.088	.00011	2.52	4.0	.093	.00012
433–1000	4-1/2	.0006	3.41	4.91	.073	.00004	3.1	4.6	.079	.00005
433–1230	2-1/2	.0013	1.86	3.36	.112	.00015	1.78	3.28	.115	.00015
700**	—	.0011	2.6	4.1	.090	.00010	2.0	3.5	.108	.00012
700–1000	2-1/2	.00007	1.8	3.3	.115	—	1.7	3.2	.118	—
700–1000	3-1/2	.00007	2.17	3.67	.103	—	2.0	3.5	.108	—
1000**	—	.5298	2.07	3.6	.105	.05563	1.85	3.35	.115	.06093
1230**	—	.0016	1.1	2.6	.135	.00022	1.1	2.6	.135	.00022
1920	—	.0042	.75	2.25	.162	.00065	.75	2.25	.162	.00068
	Total	.9995				.07283				.07919

*This is Table D.5-3 of Reference C.14.2.
**Constant leak rate.

REFERENCES FOR SECTION C.14

C.14.1 T. A. Wheeler et al., "Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation," Sandia National Laboratories, NUREG/CR-4550, Vol. 2, SAND86-2084, April 1989.

C.14.2 R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

C.15 Zion Service Water and Component Cooling Water Upgrade

In April 1989, the licensee for Zion Unit 1, Commonwealth Edison Co., made commitments to the NRC to make plant and procedure changes to address the major contributor to the core damage frequency, the development of an unmitigated reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) caused by a loss of the component cooling water (CCW) or service water (SW) system (Ref. C.15.1). The status of these commitments was provided in another letter to the NRC (Ref. C.15.2) in which the status was reported as:

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

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

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

C.15.1 Issue Definition

The scope of this issue is the evaluation of the plant and procedure changes that provide an auxiliary source of cooling water to the charging pump oil coolers. The base case Zion analysis, as documented in Reference C.15.3, models the failure of the component cooling water (CCW) or service water (SW) systems as a challenge to the integrity of the RCP seals. The failure of CCW causes overheating of the charging pumps that supply seal injection water. CCW also provides the RCP seal cooling water. The CCW system uses the SW system as its heat sink.

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

C.15.2 Issue Analysis

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

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

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

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

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

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

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

C.15.3 Issue Quantification and Results

The 203 highest frequency accidents and 58 plant damage states were requantified and a new Latin hypercube uncertainty analysis was performed using the failure probability data described above. Table C.15.1 shows the comparison of the plant damage state results between the base case and this sensitivity analysis. The change in the core damage frequency from a mean value of 3.4E-4 to 6.2E-5 per reactor year represents a decrease of about 81 percent.

Table C.15.1 Plant damage state comparison.

	Base Case		Sensitivity Case	
Plant Damage State	**Frequency per reactor year**	**%**	**Frequency per reactor year**	**%**
LOCAs	3.1E-4	93.2	3.9E-5	62.7
ATWS and transients	7.5E-6	4.0	1.4E-5	21.9
Station blackouts	9.3E-6	2.8	9.3E-6	15.0
Bypass	2.6E-7	—	2.6E-7	0.4
Total	3.4E-4	100.0	6.2E-5	100.0

C.15.4 Impact of Issues on Risk

The impact of the sensitivity analysis described above was a significant reduction in the mean core damage frequency, which was obtained by reducing the plant damage states involving CCW and SW induced seal LOCAs. Other plant damage states remained unchanged. Thus, in the sensitivity study, CCW and SW induced seal LOCAs contribute only 24 percent to the mean core damage frequency compared with 86 percent in the base case. This reduction in LOCAs means that other plant damage states such as bypass and station blackout (SBO) become larger contributors to the lower mean core damage frequency. The contribution of SBO accidents increased from about 2 percent to over 10 percent in the sensitivity study. Bypass accidents contributed 2.7 percent to the mean core damage frequency in the sensitivity study compared with 0.5 percent in the base case. As station blackout and bypass accidents tend to be more challenging (in terms of containment performance) than LOCAs, the risk estimates should not be reduced by as large a fraction as the mean core damage frequency. Table C.15.2 presents new mean risk estimates based on the sensitivity study and compares them with original base case risk. The results in Table C.15.2 indicate that the risk measures did in fact decrease by smaller fractions than the mean core damage frequency. The early fatality risk decreases by 75 percent and the latent cancer fatality risk by 67 percent.

The new mean risk estimates were not obtained by performing a completely new uncertainty analysis as was done for the accident frequency analysis (as described in Section C.15.3). The mean risk estimates in Table C.15.2 were obtained by using mean risk values conditional on the occurrence of the various plant damage states. The mean risk estimates that were used are given in Reference C.15.4. The conditional mean risk measures were simply multiplied by the new mean frequencies in Table C.15.1 and summed to obtain the new risk estimates. This approach is not as rigorous as a complete requantification of the uncertainty analysis in which new distributions for the risk measures would have been obtained. However, the approach is straightforward and gives a reasonable estimate of how the mean risk estimates would be reduced by the changes made by Commonwealth Edison Company to eliminate the common-cause failure mechanism represented by the CCW and SW systems.

Table C.15.2 Comparison of mean risk values.

Risk Measure (per reactor year)	Base Case	Sensitivity Case
Early Fatalities	1.1E–4	2.0E–5
Total Fatalities—Entire Region	2.4E–2	8.1E–3
Total Fatalities—50 Miles	1.1E–2	3.3E–3
Individual Early Fatalities	1.0E–8	2.0E–9
Individual Latent Cancer Fatalities	1.1E–8	2.5E–9
Frequency of One Fatality	6.4E–7	1.2E–7
Population Dose (person-rem)—50 Miles	5.5E+1	1.7E+1
Population Dose (person-rem)—Entire Region	1.4E+2	4.6E+1

REFERENCES FOR SECTION C.15

C.15.1 Letter from Cordell Reed, Commonwealth Edison Co. (CECo) to T. E. Murley, NRC, dated March 13, 1989.

C.15.2 Letter from R. A. Chrzanowski, CECo, to NRC Document Control Desk, August 24, 1990.

C.15.3 M. B. Sattison and K. W. Hall, "Analysis of Core Damage Frequency: Zion Unit 1," Idaho National Engineering Laboratory, NUREG/CR-4550, Vol. 7, Rev. 1, EGG-2554, May 1990.

C.15.4 C. K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

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

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** (See instructions on the reverse)	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-1150 Vol. 2
2. TITLE AND SUBTITLE Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants Appendices A, B, and C Final Report	3. DATE REPORT PUBLISHED MONTH: December YEAR: 1990 4. FIN OR GRANT NUMBER
5. AUTHOR(S)	6. TYPE OF REPORT Final Technical 7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

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 which provide the detailed discussion of the methods used and results obtained in these risk studies.

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

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
severe accidents risk analysis probabilistic risk analysis	Unlimited
	14. SECURITY CLASSIFICATION (This Page) Unclassified (This Report) Unclassified
	15. NUMBER OF PAGES
	16. PRICE

NRC FORM 335 (2-89)

THIS DOCUMENT WAS PRINTED USING RECYCLED PAPER.

NUREG–1150
Vol. 3

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

Appendices D and E

Final Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW, Lower Level, Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the Code of Federal Regulations, and Nuclear Regulatory Commission Issuances.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. Federal Register notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Information Resources Management, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG-1150
Vol. 3

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

Appendices D and E

Final Report

Manuscript Completed: January 1991
Date Published: January 1991

Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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 3 of this report contains two appendices. Appendix D summarizes comments received, and staff responses, on the first (February 1987) draft of NUREG-1150. Appendix E provides a similar summary of comments and responses, but for the second (June 1989) version of the report.

CONTENTS

TABLES

APPENDIX D

RESPONSES TO COMMENTS ON FIRST DRAFT OF NUREG-1150

CONTENTS

D.1 Introduction*

The previous draft of NUREG-1150, "Reactor Risk Reference Document," was issued as a draft report for public comment in February 1987. At that time, a notice was published in the *Federal Register* announcing the availability of the report and requesting comment (Ref. D.1). Distribution was made to approximately 850 people or organizations in the United States and abroad.

To assist readers of the document, a 2-day seminar was held in April 1987 on the methods used in the risk analyses of draft NUREG-1150. A notice of this seminar was sent to all persons receiving the draft report and published in the *Federal Register* (Ref. D.2). The seminar took place in Rockville, Maryland, and was attended by 173 people from various organizations, including Federal agencies, State agencies, utilities, architect/engineering firms, and consulting firms.

In response to the request for comments, the NRC staff received 55 letters from 45 authors totaling approximately 800 pages. The authors of these letters and their affiliations are listed in Table D.1. All letters received are available for inspection in the NRC Public Document Room.

In addition to these reviews and comments, draft NUREG-1150 was reviewed by three formal peer review committees. Two of these reviews were initiated by the NRC; the third review was initiated by the American Nuclear Society. Also, as part of the normal review process within the NRC, the staff discussed the methods and results of draft NUREG-1150 with the Advisory Committee on Reactor Safeguards on several occasions (Ref. D.3).

1. Review by Kouts Committee

One of the major advances of the risk analyses discussed in draft NUREG-1150 was the performance of quantitative uncertainty analyses. The specific approach used to perform these uncertainty analyses was reviewed by a panel of five experts, chaired by Dr. Herbert Kouts of Brookhaven National Laboratory. The members of this committee are listed in Table D.2. The committee performed its review from April to October of 1987. Its findings were published as Reference D.4 in December 1987.

2. Review by Kastenberg Committee

The NRC invited Professor William Kastenberg, University of California at Los Angeles, to form and chair a committee to peer review the entire breadth of risk analyses, as documented in draft NUREG-1150 and supporting contractor reports. Lawrence Livermore National Laboratory was funded by NRC to provide technical and administrative support. The members of the committee, listed in Table D.3, were selected by Professor Kastenberg. The committee performed its review from June 1987 to March 1988, with its findings published as Reference D.5 in May 1988.

3. Review by American Nuclear Society

The American Nuclear Society (ANS) chartered a special committee chaired by Dr. Leo LeSage of Argonne National Laboratory to study and critique draft NUREG-1150. The members of this committee are listed in Table D.4. The committee started its work in the fall of 1987 and published its findings as Reference D.6 in April 1988.

4. Overview of Comments and Responses

It is the nature of reviews of documents such as draft NUREG-1150 that extensive comments are received and that most of the comments are critical. Before discussing the principal (negative) comments, it is worth describing the principal positive comments that were expressed in letters and committee reports:

- It is believed that the NUREG-1150 study is the first comprehensive treatment of both modeling and data uncertainty in risk. In prior PRAs, the accounting of uncertainty has been limited to data uncertainties.

*This appendix was published in the second draft of NUREG-1150, reflecting comments on the first draft report (1987). It has not been modified for this (final) version of NUREG-1150 except for updating the reference list.

Table D.1 Authors of public comment letters.

Abe, K.	Japan Atomic Energy Research Institute, Japan
Artigas, R.	General Electric, San Jose, CA
Booker, J. E.	Gulf States Utilities Company, St. Francisville, LA
Boyer, V. S.	Philadelphia Electric Company, Philadelphia, PA
Brons, J. C.	New York Power Authority, White Plains, NY
Butterfield, L. D.	Commonwealth Edison, Chicago, IL
Caisley, J.	Organisation for Economic Co-operative Development/Nuclear Energy Agency (OECD/NEA), Paris, France
Campbell, R. M.	Massachusetts Voice of Energy, Boston, MA
Chubb, Walston	Murrysville, PA
Cogne, F.	Institut de Protection et de Surete Nucleaire, Commissariat a l'Energie Atomique, France
Colvin, J. F.	Nuclear Management and Resources Council, Washington, DC
Cullingford, M.	International Atomic Energy Agency, Vienna, Austria
Edwards, D. W.	Yankee Atomic Electric Company, Framingham, MA
Gardner, R.	Stone & Webster Engineering Corporation, Boston, MA
Gridley, R. L.	Tennessee Valley Authority, Chattanooga, TN
Hayns, M. R.	United Kingdom Atomic Energy Authority, Culcheth, United Kingdom
Hiatt, S. L.	Ohio Citizens for Responsible Energy, Inc., Mentor, OH
Hintz, D. C.	Wisconsin Public Service Corporation, Green Bay, WI
Hobbins, R. R.	Idaho Falls, ID
Hockenbury, R. W.	Rensselaer Polytechnic Institute, Troy, NY
Hoegberg, L., et al.	Swedish Nuclear Power Inspectorate, Sweden
Janecek, R. F.	BWR Owners' Group, Chicago, IL
Khobare, S. K.	Bhabha Atomic Research Centre, Bombay, India
Kingsley, O. D., Jr.	System Energy Resources, Inc., Jackson, MS
Kowalski, S. J.	Philadelphia Electric Company, Philadelphia, PA
Kranzdorf, R.	San Luis Obispo, CA
Langley, J. R.	Mark III Containment Hydrogen Control Owners' Group, St. Francisville, LA
Lash, T. R.	Illinois Department of Nuclear Safety, Springfield, IL
Levenson, Milton	Bechtel Western Power Corporation, San Francisco, CA
Lewis, M. I.	Philadelphia, PA
Liu, K. C.	Atomic Energy Council, Taipei, Taiwan
McNeill, C. A., Jr.	Public Service Electric and Gas Company, Hancocks Bridge, NJ
Myers, R.	Clean Air Council, Philadelphia, PA
Newton, R. A.	Westinghouse Owners' Group, Pittsburgh, PA
Reiman, L.	Finnish Centre for Radiation and Nuclear Safety, Finland
Sholly, S. & Harding, J.	San Jose, CA
Soda, K.	Japan Atomic Energy Research Institute, Japan
Spangenberg, F. A., III	Illinois Power Company, Clinton, IL
Speelman, J. E.	Netherlands Energy Research Foundation, Netherlands
Stewart, W. L.	Virginia Electric and Power Company, Richmond, VA
Taylor, J.	Electric Power Research Institute, Palo Alto, CA
Tucker, H. B.	Duke Power Corporation, Charlotte, NC
Vaughan, J.	Department of Energy, Washington, DC
Warman, E. A.	Stone & Webster Engineering Corporation, Boston, MA
Zaffiro, C.	Energia Nucleare e delle Energie Alternative, Rome, Italy

Table D.2 Members of Kouts Committee.

Herbert Kouts, Chairman	Brookhaven National Laboratory
Allen Cornell	Stanford University
Reginald Farmer	Consultant, United Kingdom
Steven Hanauer	Consultant, Technical Analysis, Inc.
Norman Rasmussen	Massachusetts Institute of Technology

Table D.3 Members of Kastenberg Committee.

William Kastenberg, Chairman	University of California, Los Angeles
George Apostolakis	University of California, Los Angeles
John Bickel	Northeast Utilities
Roger Blond	Science Applications International, Inc.
Simon Board	Central Electricity Generating Board, United Kingdom
Michael Epstein	Fauske and Associates, Inc.
Peter Hoffman	National Nuclear Research Center, Federal Republic of Germany
Frank King	Ontario Hydro Company, Canada
Simon Ostrach	Case Western University
John Reed	Jack R. Benjamin and Associates, Inc.
Robert Ritzman	Electric Power Research Institute
John Stetkar	Pickard, Lowe and Garrick, Inc.
Theofanis Theofanous	University of California, Santa Barbara
Raymond Viskanta	Purdue University

Table D.4 Members of ANS Special Committee on Reactor Risk Reference Document.

Leo LeSage, Chairman	Argonne National Laboratory
Edward Warman, Vice Chairman	Stone and Webster Engineering Corporation
Richard Anoba	Carolina Power and Light Company
Ronald Bayer	Virginia Power Company
R. Allan Brown	Ontario Hydro, Canada
James Carter III	International Technology Corporation
J. Peter Hosemann	Paul Sherrer Institute, Switzerland
W. Reed Johnson	University of Virginia
Walter Lowenstein	Electric Power Research Institute
Nicholas Tsoulfanidis	University of Missouri
Willem Vinck	Consultant, Belgium

- The methods developed during the study are desirable because uncertainty can be quantified, the contribution of various sources of uncertainty can be determined, the net impact of hypothetical design changes can be assessed, and important technical issues can be identified.

- It is believed that the basic methods used to generate the risk distributions are sound.

- It is believed that important advances made in severe accident analysis since the last major risk study, the Reactor Safety Study (Ref. D.7) done in 1975, are reflected.

- The detailed models of events and phenomena that load a containment and the response of the containment structure using event trees place added emphasis on the importance of the containment function. This seems desirable.
- It is believed that a reasonable approach was used to represent the capacity of a containment with technical issues about the failure pressure, failure location, and failure size and including interdependencies among the various phenomena.
- Engineering judgment was used when data were unavailable. Gaps in the understanding of severe accident phenomena were represented with technical issues and used as a means of investigating various hypotheses of severe accidents. This is believed to be acceptable and may be the only way to advance a risk assessment.
- Past risk assessments were reviewed to identify previously uncovered subtle interactions among components with formal investigations thereafter. This is a desirable practice because it takes advantage of previous work.

Most of the critical comments on draft NUREG-1150 were on four broad subjects. Some of them were attributable to technical deficiencies in the risk analyses while others were related to inadequate documentation. The four major areas of concern pertained to: (1) methods that were considered to be inadequate for obtaining and using expert judgments; (2) information that was considered to be outdated; (3) calculations that were considered to be inscrutable; and (4) results that were considered to be improperly presented or displayed. These areas are discussed in Sections D.3.2, D.3.3, D.4.1, and D.4.3 below.

It should be noted that some comments addressed potential new and long-term research programs, especially in the area of severe accident phenomenology. Such comments are not discussed here.

In the following sections, the comments received on draft NUREG-1150 have been grouped into seven major topics: (1) objectives and scope; (2) overall methods; (3) tracing and documenting calculations; (4) accident frequency analysis; (5) containment loads and structural response; (6) source terms and consequences; and (7) regulatory uses of NUREG-1150. With the large number of comments received in each of these areas, it was not possible to list and respond to individual comments. As such, individual comments on similar subjects have been paraphrased and responses then made to these paraphrased comments.

D.2 Objectives and Scope

D.2.1 Objectives

Comment: Clear objectives should be established and explained for NUREG-1150, and the report should be focused on those objectives.

Response:

The objectives of NUREG-1150 have been reviewed and clarified in response to comments on the draft report. These objectives are outlined in Chapter 1 of this report; they 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.

Comment: The manner in which the results of NUREG-1150 are to be used in the regulatory process should be discussed.

Response:

Since the publication of draft NUREG-1150, the NRC staff has developed an integration plan for regulatory closure of severe accident issues (Ref. D.8). In this plan, the role of NUREG-1150 is described as one of the principal supporting resource documents to this regulatory closure process. Further discussion of the uses of NUREG-1150 is provided in Chapter 13 of NUREG-1150.

D.2.2 Scope

Comment: The scope of NUREG-1150 is narrowly defined, making the risk study incomplete. Many types of accident initiators are unaccounted for, including earthquakes, floods, and other external events; reactor coolant pump seal failure; steam generator tube ruptures; and instrument air losses. Other phases of plant operation need to be considered in addition to normal full-power operation, including power ascension and descension; shutdown; and operation with Mark I containment buildings de-inerted. Accidents in spent fuel pools should be taken into account.

Response:

The scope of the current version of NUREG-1150 has been expanded to reflect comments made on the draft report. An improved reactor coolant pump seal LOCA model has been developed, and steam generator tube ruptures have been explicitly considered. As in the draft report, the effect of failures in supporting systems (ac and dc power, instrument air, auxiliary cooling water systems) has been included in the system fault trees. In addition, external events have been included in the analyses for two of the plants (Surry and Peach Bottom) to determine the core damage frequency and containment performance associated with a range of external initiators. The NRC staff intends to evaluate the significance of external events at the remaining plants in a later study, currently scheduled for completion in FY 1990. With these changes, the staff believes that NUREG-1150 presents an adequate representation of the risk associated with the five plants analyzed, subject to the constraints established by the state of the art of probabilistic risk analysis.

To confirm that the scope is appropriate, the NRC is initiating a separate study of the risk associated with low power and shutdown conditions for two of the plants studied in NUREG-1150. The results are expected to be available in FY 1990. The risk associated with spent fuel pool accidents is being assessed separately in studies responding to NRC's Generic Issue 82, "Beyond Design Bases Accidents in Spent Fuel Pools." When completed, these will be examined to determine if further efforts are advisable.

Because of the small fraction of time that the BWR Mark I containment is de-inerted during startup and the approach to shutdown conditions, compared to the length of the operating cycle, and the small frequency of accidents occurring in these times, the NRC does not intend to further study the risk implications associated with de-inerting.

Comment: NUREG-1150 should take credit for accident management strategies to reduce the likelihood of a core damage accident or to mitigate its consequences.

Response:

Both the draft and present versions of NUREG-1150 explicitly consider the effects of plant operational procedures to provide water and cooling to a reactor core to prevent its damage. Procedures for performing such actions are obtained from the specific plant under study. These are reviewed to assess the probabilities of successful use in associated accident scenarios. These probabilities are then incorporated into the accident frequency analyses for that plant.

The present version of NUREG-1150 also incorporates the effects of plant operational procedures on mitigating the consequences of core damage accidents. Procedures in place at the individual plants were used to assess the probability of successful operator action.

Comment: Accident sequences with frequencies below 1E-7 per year have not been considered. Events at 1E-8 or 1E-9 per year might be significant, either individually under particular conditions or cumulatively when many such sequences are excluded.

Response:

Given a set of initiating events, event tree/fault tree analysis like that performed for NUREG-1150 permits the examination of logical accident sequences. The trees were quantified by calculating each branch to its end; when a branch frequency fell below a cutoff frequency, it, and hence the accident sequences it represents, were excluded from the analysis. The cutoff frequency used in the present version of NUREG-1150 is 1E-8 per year for internally initiated events, which is sufficiently low to retain more than 95 percent of the accident sequences contributing to the core damage frequency. However, in several instances accident sequences with a frequency below these levels were explicitly included to ensure adequate representation of a wide spectrum of accidents.

D.3 Overall Methods

D.3.1 Uncertainty Analysis

Comment: The treatment of uncertainty is questionable and inherently biased. The large uncertainty bands and high risk estimates may result from the methods used; uncertainty in variables may have been combined without compensating to prevent many variables from realizing worst case values. Other methods exist for quantifying uncertainty, such as the Optimistic/Central/Pessimistic (OCP) method and sensitivity studies.

Response:

The Latin hypercube sampling (LHS) method used in NUREG-1150 is a form of Monte Carlo sampling, an algorithm for sampling from individual parameter distributions and, based on the risk model, combining these individual distributions into a single distribution. More generally, such sampling is known as mathematical experimentation and is used in other disciplines (Ref. D.9). This mathematical experimentation is used to propagate uncertainty through large mathematical functions that preclude propagating uncertainty analytically. The validity of the results is dependent upon the validity of the assumptions made about the distributions of the variables. The displays of risk may give the impression that the risk estimates are rigorous classical statistics but they must be taken in context, realizing the strengths and the weaknesses of the methods used to compute the estimates and the underlying data base.

LHS was favored over the OCP approach on the basis of theoretical grounds and the potential of each method. Probabilistic techniques constitute a theoretically sound and standard technique for combining uncertainties; the LHS framework contains a basis for implementing such a technique. In contrast, the OCP framework requires that a series of pessimistic or optimistic assumptions be combined without quantifying the likelihood that the combination of assumptions could arise.

The LHS algorithm for generating Monte Carlo samples is one that has undergone extensive review in the open literature and is used both in the United States and abroad. The estimators of LHS are unbiased when the variables are uncorrelated. Bias may occur when the variables are correlated, but this is a characteristic of Monte Carlo sampling in general and not a characteristic of LHS.

Some specific aspects of the use of LHS in draft NUREG-1150 had the potential to introduce bias. In the draft report, the uncertainty in individual parameters was represented in a discrete manner. Because such use may cause bias, the discrete parameters were replaced with continuous parameters for the present version of NUREG-1150. Because the methods used to obtain expert judgments and to formulate distributions based on the judgments can introduce bias, formal methods designed to minimize bias were used in the present risk analyses. This is discussed in greater detail in Section D.3.2.

As noted in the Introduction, the treatment of uncertainty in draft NUREG-1150 was reviewed by a peer committee; their findings are published as Reference D.4, and all their major comments have been incorporated in the present analyses.

Comment: The conclusion that the uncertainties have increased over a dozen years of research does not seem correct. NUREG-1150 does not portray the progress made in hardware modifications, operator experience, and research since the publication of the Reactor Safety Study (Ref. D.7) in 1975. Furthermore, the conclusions in draft NUREG-1150 are similar to those of the Reactor Safety Study, meaning that draft NUREG-1150 is inconsistent with almost every study published in the 1980's, all of which show a trend of lower estimates of risk. After a decade of severe accident research, it is unsettling to see risk results spread over several orders of magnitude.

Response:

It is clear that the technical information base on the frequencies and consequences of severe reactor accidents is substantially better now than in 1975 when the Reactor Safety Study (Ref. D.7) provided its analysis of risk and the uncertainty in risk.

In the Reactor Safety Study, the rigorous quantification of uncertainty was performed only for uncertainties in component failure rates. The overall uncertainties shown in consequences were developed by subjective judgments at a very coarse level. The peer review of the final version of the Reactor Safety Study, known as the "Lewis Report" (Ref. D.10), concluded that these uncertainty estimates were significantly underestimated.

The subjective uncertainty estimates developed and used in NUREG-1150 address uncertainties at a rigorous level and make extensive use of experimental and calculated results developed since 1975. Both the level and the basis improve the realism of the uncertainty analysis. These improvements are a direct result of an improved knowledge base (including the effects of phenomena not known in 1975) that permits a more accurate treatment and characterization of unknown parameters and modeling of physical processes. The large uncertainties simply reflect the state of knowledge in the severe accident area.

Comment: Uncertainty in offsite consequences was not factored into the risk estimates. This leads to misleading risk estimates.

Response:

With the exception of the variability of site meteorology, uncertainties in the consequence analysis have not been included in either the draft or present version of NUREG-1150 because of time constraints. The NRC staff recognizes that there are significant uncertainties in the consequence estimates due to uncertainties in modeling and in input data. Best estimate values of the model parameters for natural processes (plume behavior, deposition, etc.) have been used in Version 1.5 of the MACCS code

(Ref. D.11) for the current NUREG-1150 analysis. While the lack of accounting of consequence uncertainties can have an influence on overall risk results, it does not prevent the development of important perspectives on plant design and operation.

Some of the key parameters in the uncertainty of offsite consequence analyses relate to post-accident protective actions (e.g., emergency response and long-term countermeasures). That is, offsite consequences can be affected by the effectiveness of emergency response of the local population and by the radioactive contamination levels above which crops and land are removed (condemned) from public use. The sensitivity of consequences and risks to the protective actions is discussed in Volume 1 of NUREG-1150 and in References D.12 through D.16.

In addition to the risk studies of the five plants discussed in this report, the NRC staff is supporting the risk analysis of the LaSalle plant, a BWR-5, Mark II plant. It is planned that this risk analysis will include an analysis of the uncertainties in offsite consequences and the effect on risk estimates. This work is scheduled for completion in mid-FY 1990.

D.3.2 Expert Judgment

Comment: The protocol for obtaining expert judgment is not rigorous and yielded judgments with unsound bases. Technical subject areas should be kept separate, and experts should work within those areas (i.e., within their field of expertise). Panels should be composed of experts from all portions of the nuclear industry. The experts should be given a less restricted role in selecting and identifying uncertainty issues. Expert groups should interact to ensure that consistency is maintained throughout the analysis.

Response:

The protocol used to elicit and to aggregate expert judgment has been substantially improved for the present version of NUREG-1150 and is discussed in Section 7 of Appendix A. Standard and rigorous techniques were used; among the developers and the reviewers of the protocol were experts in diverse areas of uncertainty analysis and survey methods, including decision analysts, social psychologists, and statisticians from national laboratories, private companies, and universities. Seven groups of experts were established, each group working within a specific technical area. Each group included representatives from industry, academia, and the national laboratories.

For the current version of NUREG-1150, the experts were allowed to add issues to or delete issues from the list of issues presented to them. The context in which the issues entered the analyses was explained to the experts. The experts were encouraged to modify the statements of the issues to improve technical clarity and to define interdependencies among issues and between issues and other parameters in the analysis. The exchange of information between panels was effected primarily by the project analysis staff. The information exchanged concerned requirements for specific phenomenological information about an earlier phase of a postulated accident sequence that was needed to answer a question about a later phase of the accident.

Comment: Expert judgments are requested for inappropriate portions of the risk analyses. When adequate data exist to define uncertainty, expert judgment should not be used as a substitute. When little or no data exist, experts should not be asked to guess at distributions; instead, the particular uncertainty variable should not be included in the uncertainty analysis. No attempt should be made to quantify technical issues with expert judgments unless there is an adequate basis for that judgment.

Response:

For the present version of NUREG-1150, expert judgments were not used when available data were adequate to provide the required information. In some cases, such data became available during expert panel meetings, and issues were dropped from consideration by the panels.

While it is inappropriate to ask experts to simply guess at issue distributions, it is also inappropriate to exclude issues from consideration because of the scarcity of relevant data. Potentially important issues should be considered, even if the data are scarce and the basis for engineering judgment is very limited, because it is often the paucity of data that renders an issue an important contributor to uncertainty.

In technical areas where significant data existed, expert judgment still played an important role in ascertaining the relevance of the data to a particular application in the risk analysis. As an example, large uncertainties inherent in the models of containment performance precluded an accurate prediction of the location of failure under specific accident loadings. Various calculations of containment response were sometimes found to be conflicting, even when similar analytical methods were used. In 1987, at the Sandia National Laboratories, a scale model of a concrete containment was tested under conditions simulating a slow pressurization from steam and noncondensible gas generation from a severe accident (Ref. D.17). Organizations from the United States and abroad tried to predict the failure location and pressure. Only one of these predictions was close to the test results. For the purposes of NUREG-1150, the available data on both the experimental result and the reasons why calculated results differed were key to the assessment.

Comment: When distributions were assigned to variables, often high weights were assigned to the extremes of these distributions. This is unusual and should be justified. Integrated analyses based on models benchmarked against data would show that extremes could not be realized.

Response:

Documentation of the rationale for uncertainty distribution development, including the results of relevant code calculations and experimental results, was an important step in the elicitation process used in this report. This documentation is provided as Reference D.18. If high weights were assigned to the extremes of the distributions, then the associated documentation should provide the rationale. Discrete distributions were replaced with continuous distributions to permit a better characterization of the uncertainties, particularly in the tails of the distributions.

D.3.3 Quality Assurance, Consistency, and State of the Art

Comment: A thorough review of the draft NUREG-1150 study is needed. The study does not appear to have been checked for inconsistent and meaningless results because some of the results are questionable or contradict results reported elsewhere in the same documents. The computer codes used in the risk analyses should be properly validated, documented, and peer reviewed, which also appears not to have been done.

Response:

The review of NUREG-1150 has been performed at two levels: an external level, including peer review and public comment on the draft report; and an internal review by the various organizations involved in the plant risk analyses. For the specific issues identified in this comment, internal review processes were used to ensure consistency and validity. This internal review had the following elements:

- **QA/QC Review of Principal Analysis Areas:** For each major area of analysis performed for the present version of NUREG-1150 (accident frequency, containment performance and source terms, offsite consequences), a QA team was established and the analyses were reviewed. This process for the accident frequency analysis is documented in References D.19 through D.23. Approximately 25 percent of the resources were spent on reviewing the work. The method was initially reviewed by a Senior Consultant Group, and then more detailed reviews were conducted by a Quality Control Team. These latter reviews occurred periodically over more than 2 years. In addition to the reviews that NRC and its contractors sponsored, the utilities involved have performed reviews and provided extensive comments that have been incorporated, as appropriate, into the analyses. This formal review process continued throughout the reanalysis effort. The modeling of core melting phenomena, source terms, and consequences as well as the risk analysis in the current version of NUREG-1150 were subject to a quality assurance review that is discussed in Appendix A to NUREG-1150.

- **Review of Computer Models:** A large number of computer codes were used in the performance of the risk studies described in this report. A number of them have been reviewed or benchmarked in other contexts and will not be discussed in detail here. These include: Source Term Code Package (Ref. D.24); CONTAIN (Ref. D.25); MELCOR (Ref. D.26); and MELPROG (Ref. D.27). Other codes were, however, developed or first used for this study. For such codes, various types of quality assurance checks were specifically performed as part of the NUREG-1150 study. The LHS code was reviewed in 1984; a user's manual was written (Ref. D.28) and the code has been released to the National Energy Software Center at Argonne National Laboratory. LHS has been in use at Sandia National Laboratories for several years. Since the draft analyses, the EVNTRE and PSTEVNT codes (Ref. D.29) were subject to line-by-line scrutiny and a series of functional tests. User's manuals were written for these codes, and the codes are being released to the National Energy Software Center. The XSOR codes were subject to a line-by-line review by project staff at the Sandia National Laboratories (SNL) and to an independent validation/verification study done at the Battelle Columbus Division and reported as Reference D.30. The PARTITION code (Ref. D.31) was subject to a functional review by the project staff. Its user's manual will be published in 1989. The RISQUE code was functionally reviewed by the SNL staff and elsewhere (Ref. D.29); a list of this code is included as an appendix to Reference D.32. Benchmarking and verification of the current version of the MACCS code, Version 1.5, is now under way. A review has been performed of the chronic exposure pathway modeling (Ref. D.33).

- **Cross-Plant Reviews:** For the accident progression, source term, and consequence modeling, general consistency in phenomenological assumptions and the level of treatment of severe accident phenomena was achieved across the five-plant analyses. This was accomplished primarily through a series of informal interactions among plant analysts and review of the treatment of specific phenomena by the project leader. For example, the Surry accident progression event tree (APET) was used as a base to build the Zion and Sequoyah APETs. The Grand Gulf and Peach Bottom analysts worked jointly to adapt parts of the Grand Gulf APET to Peach Bottom. The treatment of hydrogen (important to the Sequoyah and Grand Gulf analyses) was derived jointly by the Sequoyah and Grand Gulf analysts.

- **Utility Reviews:** An important element of a risk study of a nuclear power plant is the assurance that the risk model is an accurate and up-to-date representation of that plant. For the four plants in this study for which an essentially new risk analysis was performed (Surry, Sequoyah, Peach Bottom, and Grand Gulf), contact with the appropriate utility was maintained throughout the conduct of the study. For the Zion plant, where the accident frequency analysis was a modification of an existing PRA (Ref. D.34), the analysts met with the utility to discuss plant design and operational changes that had occurred since the performance of that PRA.

The present version of NUREG-1150 will undergo a multifaceted review in the near future. This will be a critical review of all important aspects of the document through a formal peer review, university research, professional society discussions, and a public workshop. The emphasis in these forums will be on the responsiveness of the present version of NUREG-1150 to comments on the draft report as well as on how the technology for assessing risk can be improved.

Comment: The analyses of draft NUREG-1150 are not state of the art. The most advanced theoretical and analytical techniques are not always used. Some data are even outdated while other data are ignored or are inappropriately applied.

Response:

A discussion of the methods that were used in the current version of NUREG-1150 is provided in Appendix A. The broad diversity of experts who interpreted published data for the current analyses ensured that the data were up to date and correctly applied. These data reflect the design and operational status of the five plants as of March 1988.

Comment: There must be a consistent and distinct use of terms such as randomness and uncertainty, frequency and probability. In the draft NUREG-1150 report, terminology is sometimes used loosely.

Response:

The consistency of terminology used in the present version of NUREG-1150 has been improved.

Comment: There is a general disregard for technical rigor in the draft NUREG-1150 risk analyses.

Response:

It appears that such a general conclusion was reached based on specific deficiencies in the draft risk analyses, including the process for obtaining expert judgments, the apparent lack of quality assurance reviews, the use of unreviewed and undocumented computer codes, and the reliance on severe accident information from NRC contractors to a greater extent than on that from other sources. Each of these specific issues is discussed elsewhere in this appendix. The NRC staff and its contractors believe that the present version of NUREG-1150 is based on analyses with appropriate technical rigor.

D.3.4 Other Comments on Methods

Comment: The NUREG-1150 results do not appear to be reproducible. It appears that the results seem dependent on the particular experts whose judgments were factored into the analyses. A different selection of experts would make different judgments that would lead to different risk estimates. This point goes beyond just the subjective judgment of experts and extends into the analytical techniques. If another random number generator were to be used in the sampling scheme for generating uncertainty estimates, different uncertainty bands and hence risk estimates would result.

Response:

The selection of experts will have an effect on the results of the risk analyses discussed in this report, as well as in any other circumstance where expert judgment is used. However, given the necessity of using expert judgment, the formal procedures used for the present version of NUREG-1150 offer the following advantages: the expert panels are established using experts from a wide spectrum of interests, minimizing the potential impact of any one group; the use of judgments is explicitly acknowledged; and the rationales underlying judgments are documented.

With any analysis involving a Monte Carlo process, it is inevitable that the results will vary somewhat, depending on the details of the sampling algorithm and the way it is implemented. The variability associated with the sampling process has been investigated as part of the analysis process and found to be small (Ref. D.32). As discussed in Chapter 2, the reader should recognize that the estimated mean values can vary by no more than a factor of two, depending on the Monte Carlo sample that is used. This variability can also impact the relative contribution of factors (e.g., plant damage state frequencies) to the mean, particularly when there are a small number of contributors.

Comment: The methods used to calculate risk are complex and subjective, which is in part because of the performance of uncertainty analysis. The risk-dominant issues should be quantitatively defined with detailed calculations or experimental evidence.

Response:

The present version of NUREG-1150 has made extensive use of mechanistic computer code calculations and pertinent experimental results available from both NRC-sponsored research and that sponsored by the nuclear industry. However, the spectrum of accident conditions is wide, precluding mechanistic calculations for all conditions. For potentially important uncertainty issues, such as containment loads at vessel breach, expert judgment was obtained for a variety of generalized conditions such as vessel breach at high pressure, intermediate pressure, or low pressure, with a flooded reactor cavity or a dry reactor cavity. Typically, the experts could base their judgments on the results of mechanistic code calculations or experiments of only some of these conditions. Parametric codes were used to predict the source terms for a wide range of sequence variations.

Comment: The practice of evaluating complex physical and chemical phenomena in nuclear reactor accidents is not well conceived. This evaluation should be done using such classical methods as scaling analysis, zeroth order estimates, and ideal model simulations.

Response:

Subsequent to the accident at Three Mile Island on March 28, 1979, the NRC undertook a major research effort to develop an improved understanding of severe accident behavior (Ref. D.35). The focus of this effort has been the development and validation of computer codes that estimate the variety of complex processes that can occur in a severe accident. A two-tiered approach to code development has been followed. At one level, detailed mechanistic codes have been developed that analyze a specific aspect of severe accident behavior, such as the use of the CORCON code to analyze the attack of concrete by hot core debris. The second level involves the development of codes that treat all aspects of a severe accident but in less phenomenological detail. Both of NRC's codes of this type, the Source Term Code Package and MELCOR, were used in the source term estimation in this risk study. In the current version of this study, greater use was made of the detailed mechanistic codes than in the draft report. The general approach to the development of the suite of NRC severe accident codes has been reviewed previously by a number of peer committees and is responsive to the recommendations of the review by the American Physical Society (Ref. D.36). These codes are supported by a range of experiments to obtain fundamental data, separate effects, and integral confirmation.

Comment: Various aspects of the draft risk studies lack consistency within each risk study and among the risk studies leading to an unevenness in the overall approach. The level of detail of modeling in each analysis varies, lacking in some portions and extremely detailed in others. A given technical issue is treated differently at different plants. The analysis of the Zion plant is less detailed than the analysis of the other plants.

Response:

With the exception of the Zion accident frequency analysis, the NUREG-1150 methods have been applied consistently for all five plants. Within a specific plant analysis, issues were treated at varying levels of detail, with additional consideration given to potentially more important issues. This is not an unusual practice in PRAs. Issues common to more than one plant were analyzed using the same methods for each plant. However, the resulting outcome (e.g., the impact on core damage frequency) can vary among plants because of plant design and operational differences.

The Zion accident frequency analysis was indeed different from that performed for the other four plants. This difference is a result of the availability of a relatively recent PRA performed for the utility (Ref. D.34) and extensively reviewed by the NRC staff and its contractors (Ref. D.37). For the present version of NUREG-1150, this PRA (as reviewed) was updated to reflect plant design and operational characteristics as of March 1988. The accident frequency analysis methods in NUREG-1150 used for the Zion plant are discussed in Appendix A, Section A.2.2, to this report.

Comment: Because many aspects of severe accidents cannot be quantified, assumptions are made about the aspects to account for them in the risk calculations. Too often such assumptions lack a firm basis. Furthermore, a given assumption varies from one plant to the next. The assumptions are made conservatively to ensure safety but in doing so make the risk estimates unrealistic. An example is the short battery life assumed in station blackout sequences.

Response:

Efforts were made to make reasonable assumptions in all parts of both the previous draft and the present version of the NUREG-1150 analyses; the bases for the assumptions have been thoroughly documented in the present version of NUREG-1150.

In the analysis of Peach Bottom for draft NUREG-1150, the batteries were assumed to be depleted in 6 hours during a station blackout. The assumption was based on information from the Philadelphia Electric Company (PECo), the utility operating that plant. After additional review by PECo and accounting for operator actions for load shedding, the assumption was changed for the present version of NUREG-1150 to 12 hours. The Grand Gulf analysis also assumed a 12-hour battery life.

Comment: There is a tendency to overemphasize the numerical aspects of probabilistic risk assessment. While the quantitative aspects are important, it is also important as a structured and comprehensive framework for safety analyses.

Response:

The intended uses of NUREG-1150 are discussed in detail in Chapter 13, "NUREG-1150 As a Resource Document." These uses do not focus on the bottom-line quantitative results but on the perspectives gained from the development and application of the complex logic models used to calculate the risk estimates.

D.4 Tracing and Documenting Calculations

D.4.1 Tracing Calculations

Comment: The document is inscrutable. It is nearly impossible to follow the development of the results through the calculations. Intermediate results at key points in the calculations would have been useful in understanding the risk estimates. Some conclusions are unsubstantiated and cannot be traced back to their supporting calculations. Although technical issues are delineated, how they affect the results is not discernible.

Response:

The present version of NUREG-1150 has been extensively restructured, relative to the draft report, to improve its clarity. In particular, the report has been more explicitly described as a summary report, written for people not expert in PRA and, as appropriate, directing the reader to sections of the supporting contractor reports for additional detail.

The risk calculations are very complicated, requiring extensive computer calculations to perform the analyses. Documentation of these analyses is found in supporting contractor reports; this documentation has been restructured to improve the traceability of the work and to expand the discussion of the underlying rationale. An example calculation has been developed for the reader seeking details of the risk analyses, with a description provided of both the individual steps of the risk analysis process and the products of the individual steps. The example calculation is provided in Appendix B.

Comment: Because the sequences are grouped into plant damage states, there is difficulty in connecting what follows core damage with containment failure. This could lead to gaps in the development of specific scenarios and detract from those situations where accident management could be effective.

Response:

PRAs such as those conducted in support of NUREG-1150 consider hundreds of thousands of distinct failure combinations leading to severe accident sequences. It is a practical necessity that grouping of these sequences be performed. Plant damage states have been used in many recent PRAs to accomplish such a grouping.

D.4.2 Completeness of Documentation

Comment: NUREG-1150 represents a large amount of information but many topics are insufficiently discussed, such as descriptions of models, treatment of processes of a severe accident, underlying assumptions, and uses of the risk estimates. The expert judgments and the methods used to obtain those judgments must be fully documented; each judgment should be attributed to the particular expert who gave it, together with the basis and the reasoning for the judgment.

Response:

NUREG-1150 is a summary of large and complex risk studies of five nuclear power plants. All aspects of the study cannot be conveyed in such a summary report. The methods used, supporting rationale, and results are discussed in detail in the set of supporting documents (Refs. D.12 through D.16, D.19 through D.23, D.32, D.38, and D.39). Other supporting information, such as on the principal accident analysis codes used in the study, is described in other available documents.

As discussed in Section D.3.2, the process of obtaining and using expert judgments has been substantially improved for the present version of NUREG-1150. Extensive documentation of the bases for these judgments is a major aspect of the new elicitation process. This documentation is provided in References D.18 and D.39.

D.4.3 Display of Results

Comment: Traditional methods of displaying uncertainty, such as cumulative distribution functions, probability density distributions, best estimates, and central estimates should be provided. The presentation of ranges alone without means or other best estimates tends, as in draft NUREG-1150, to focus excessive attention on the extremes and obscures the advances made in nuclear safety since the Reactor Safety Study.

Response:

The present version of NUREG-1150 uses traditional displays of uncertainty. These displays include probability density functions (approximated by histograms) with mean, median, and 5th and 95th percentile values shown. Complementary cumulative distribution functions are used to convey results of source term and offsite consequence results. Other displays, such as bar charts and pie charts, are used to convey supplemental information.

D.5 Accident Frequency Analysis

D.5.1 Logic

Comment: No thermal-hydraulic analyses were done to define what constitutes a successful operation of a given system. Instead, success and failure were defined from previous studies on other plants.

Response:

Numerous thermal-hydraulic studies were available from which conclusions could be drawn relative to safety systems performance under a range of plant conditions. In addition, information was obtained from knowledgeable personnel at the plant sites to better understand system responses under abnormal conditions and some plant-specific thermal-hydraulic calculations (Refs. D.20 and D.22) using the MELCOR (Ref. D.11) and the LTAS (Ref. D.40) codes. The combination of general analyses and plant-specific information is believed to be adequate to define success criteria.

Comment: Support systems and the activation and control of various systems by other systems were not taken into account in the accident frequency analysis.

Response:

Support systems and their impact on emergency core cooling, containment safeguard systems, and other front-line systems were explicitly considered in both the draft and the present version of NUREG-1150. Detailed system-by-system analyses were performed to determine the potential impact of support systems. Those dependencies that were critical to the functioning of a system were then included in the models.

Actuation and control dependencies between systems were taken into account, although a detailed study of each actuating and control device was not performed. Instead, these dependencies were represented with generic failure rates with significant uncertainty bounds. This approach is considered adequate because such failures have not been found important in reviewing the results of other PRAs.

Comment: Prior PRAs, from which much information was derived, were used even though many changes in plant hardware and operation have occurred since the PRAs were performed that are not reflected by these PRAs.

Response:

Prior PRAs provided a basis from which to start the risk studies. Plant design and operational information, obtained from the individual utilities, was obtained and used to perform the actual risk studies.

Comment: The mathematical treatment of common-cause failure (CCF) is more consistent and detailed than in many previous studies. Nonetheless, the importance of CCF dictates that a more comprehensive and quantitative treatment of the factors affecting it be undertaken. The CCF modeling should be improved as several examples illustrate: For a station blackout, the notion of a CCF of the batteries is difficult to accept because the batteries are monitored, are in use, and are checked daily; for a loss of component cooling water (CCW), the CCF of the CCW pumps is difficult to accept because some pumps are normally operating while others are kept in a standby mode; the fuel supply to diesel generators is not mentioned as a potential CCF, etc. In PRAs, CCF must be modeled realistically.

Response:

The NUREG-1150 study treated common-cause failure in as realistic a way as presently possible. The objective was to estimate risk using the best available information and tools given the limitations of available data.

The analyses reported in the supporting documentation had the following characteristics with respect to CCF:

- System interdependencies were modeled in the fault tree analysis and common-mode failures were included as appropriate.
- Common-mode failures of pumps, valves, batteries, diesel generators, and other hardware were explicitly considered.
- To the extent possible, the current analyses used plant-specific data. However, where the plant-specific information was inadequate to generate new CCF models, such failures were treated with realistic generic models, including recent advances in the methods for creating such models.
- Common-cause failures induced by earthquakes were considered when two plants were analyzed; the external-event analysis examined other potential sources of common-mode failures such as fire.

Nevertheless, it is recognized that there are a number of things that are not done in these analyses. These excluded activities were:

- Unique CCFs that might be postulated as a result of faulty construction or counterfeit parts are not modeled, except as they may appear in the common-cause failure data base.
- Detailed examination of the root causes for CCFs was not made.
- The CCF analysis inherently lacks reliable and identifiable data. Under these circumstances, it is often necessary to rely heavily upon engineering judgment, leading to the possibility for disagreements about the outcomes.

While present CCF models are believed to be reasonable, it is also clear that improvements can be made. To this end, the NRC has ongoing programs for developing improved models for CCF analysis.

Comment: Inappropriate application of models of human reliability focused on procedural errors and resulted in low human error contributions to core melt frequency.

Response:

Human error contributed less to the core melt frequency than expected, but, in most cases, there are reasons for the lower values. These include:

- The low probabilities due to human errors were not necessarily a consequence of a simple analysis. For example, low human error probabilities were produced for the BWR ATWS sequence using an extremely detailed human reliability analysis (Ref. D.41).

- Some small values reflect the availability (at the plant) and consideration (in the analysis) of symptom-based procedures. With such procedures, an operator responds to an accident treating conditions that are indicated on the control panel, such as ensuring that the reactor is tripped, the turbine is tripped, the vital electrical buses are energized, and so on. These conditions are treated without recognizing the sequence. The use of such procedures improves the performance of the operators and likewise reduces human error values.

- In some circumstances, low operator error values are the result of the combination of probabilities for several independent actions. When such circumstances occurred, additional checks were performed to ensure the reasonableness of the results obtained. In the current analyses, all combinations of human errors less than 1E-4 required additional analysis and justification; few of these cases occurred.

D.5.2 Quantification

Comment: Generic failure data are used in the risk studies, yet each study is claimed to be plant specific. Furthermore, generic data are sometimes used even when plant-specific data are available.

Response:

Plant-specific information has been obtained (where available) and used for key systems in each plant. Where such information is inadequate for these key systems and for less important systems, generic data have been employed. As a result there is a mix of information sources underlying the analysis. This is discussed in more detail in Section A.2.1 of Appendix A.

Comment: Calculations by the industry indicate that it is important to thoroughly examine the probability that the automatic depressurization system (ADS) in boiling water reactors would not fail when it is assumed that dc power fails.

Response:

Analysis performed for the boiling water reactors indicated that the ADS is dependent upon dc power in that both the logic for control and the valve power come from dc sources. The logic system is failed if the primary dc supply bus and the switching relay to the backup bus are lost. Two dc buses would have to fail to lose all valve power. Given this dependency, the expected life of batteries in station blackout conditions was an important issue. In the analyses for draft NUREG-1150, effective battery life was estimated to be 6 hours (in station blackout conditions) for the Peach Bottom plant. Based upon reviews and discussions with the Philadelphia Electric Company and accounting for operator load-shedding actions, this battery life was extended to 12 hours for the present risk analysis. A 12-hour battery life was also calculated for the Grand Gulf analysis.

Comment: Confidence limits on the success probabilities assigned to the fault trees appear to be derived from the high and low probabilities of operator action. This is not mathematically correct, and such limits should be viewed only as upper and lower bounds.

Response:

Operator actions were treated the same as all other events in the fault trees. That is, the failure probability of each action was assigned a mean value and a probability distribution. The combination of individual event probabilities was then performed using mathematical sampling techniques, ensuring the appropriate mathematical treatment.

Comment: The analysis of a rupture of a steam generator tube is incorrect because the frequency of a credible tube rupture sequence is multiplied by the probability of auxiliary feedwater system failure, thus lowering the frequency of a legitimate core melt accident sequence by three to four orders of magnitude. There is no reason to believe a value of 9E–9/reactor year when detailed plant-specific PRAs have consistently estimated the frequency in the range of 1.6E–6 to 1.0E–5/reactor year.

Response:

The analysis of steam generator tube rupture frequency and consequences has been substantially modified for the present version of NUREG–1150. Core damage frequencies obtained using the new analyses are consistent with the values cited above as results from other recent PRAs.

Comment: The models of core degradation are unrealistic. Severe fuel damage is defined to cover all cases where fuel cladding is damaged, but cladding integrity is not a measure of fuel damage and results in an overestimate of risk when it is defined as such.

Response:

The models that are used to predict fuel damage do not attempt to describe all the complex phenomena associated with severe core degradation in detail. The thermal-hydraulic model in the Source Term Code Package (STCP) (Ref. D.24) uses simplified models and assumptions for the treatment of some of the very complex steps in the core degradation process, such as fuel slumping into the lower plenum of a reactor vessel. However, the current version of NUREG–1150 did not rely heavily on the thermal-hydraulic model in the STCP for the estimation of the core meltdown process. The results of analysis with the MELCOR code (Ref. D.26), the MELPROG code (Ref. D.27), and the MAAP code (Ref. D.42) assisted project analysts and other experts in estimating the magnitude of parameters directly associated with core melt progression, such as hydrogen production and the mode of reactor vessel failure. Although the MELCOR code and the MELPROG code predict some core meltdown processes in more detail than the models in the STCP, the simplifications in the models of these codes must also be recognized.

In the current version of NUREG–1150, core damage is defined as a significant core uncovery occurrence with reflooding of the core not imminently expected. The result is a prolonged uncovery of the core that leads to damaged fuel and an expected release of fission products from the fuel.

The current version of NUREG–1150 treats the recovery of core with fuel damage differently from earlier probabilistic risk analyses. Under a broad range of conditions, given that a water supply is recovered prior to vessel failure, the likelihood of recovering a core and arresting an accident was evaluated. Based largely on the experience of the accident at Three Mile Island Unit 2, debris bed coolability analyses, and supplemental calculations of head failure, the likelihood of arresting further core damage decreased as the fraction of the fuel relocated to the bottom head at the time of coolant recovery increased.

D.6 Containment Loads and Structural Response

D.6.1 Accident Progression Event Tree: Logic

Comment: The large amount of detail in the accident progression event trees (APETs) gives the impression that more is known about containment events and phenomena than is actually known. It is difficult to believe the results of a complex tree that yields a tremendous number of pathways that are then aggregated into a dozen or so groups. Not only does a complex tree give a false impression but it limits any review of how the tree was constructed.

Response:

The rationale for developing a detailed APET is to provide an explicit treatment of all phenomena that can have a significant impact on the accident progression and the magnitude of the fission product source terms. Even if the existing capability to predict some of these phenomena is limited, it is important that the phenomena be recognized, at least for characterizing the uncertainty in the results.

The size of an APET does not affect the clarity of the results. Pathways with similar characteristics can be grouped to form simpler event trees as has been done to present results in the current version of the main report.

A detailed review of the APETs is difficult but not impossible. In the current documentation, the APETs were sufficiently described. Reviews were performed as described in Section D.3.3.

Comment: Early containment failure calculations are based on flawed accident progression event trees. Pathways with a potential for pressure reduction in the reactor coolant system are neglected.

Response:

In the draft analyses, the PWR event trees included several important depressurization mechanisms, namely, induced reactor coolant system hot leg failure, induced steam generator tube rupture, and reactor coolant pump seal failure. The BWR event trees took into account the operation of the automatic depressurization systems. In the current analyses, the PWR event trees have been revised to include a possible reactor coolant primary system and/or secondary system depressurization by the operators and by power-operated relief valves (PORVs) sticking open. The combined effect of these depressurization mechanisms was found to be important in the present risk analysis.

Comment: The risk reduction due to containment venting can be assessed only after a detailed study of venting procedures, relevant hardware, and plant response has been done. There is much disagreement as to the scenarios leading up to venting, the manner in which to vent, the vent size, re-isolating, and the effectiveness of venting. Containment venting should be included in the analyses only if procedures and equipment exist at the given plant.

Response:

The actions included in the NUREG–1150 analyses that could result in deliberate containment venting were those permitted by plant-specific operational procedures. Of the five plants studied, only two (Peach Bottom and Grand Gulf) had such procedures. For these two, the probability of successful venting was a function of the available procedures and hardware. For the Peach Bottom plant, it was found that venting with existing hardware and procedures was viable (had a high probability of success) for one type of accident, the long-term loss of decay heat removal. For other sequences, the probability of successful venting was of low probability, principally because of hardware limitations (Ref. D.43). For the Grand Gulf plant, the situation is similar. For the long-term loss of decay heat removal sequences, the procedures exist and operators can vent from the control room. Credit was not given in the most frequent accident sequences (i.e., station blackouts) because of unavailability of needed dc power.

D.6.2 Containment Loading Phenomena

Comment: Studies of severe accident phenomena are conflicting. Some studies predict global hydrogen burns while others conclude that global combustion cannot occur. Some studies show that hydrogen detonations can occur while others show that diffusive burning will occur. Some studies show early containment failure while other studies show that an early failure can occur only as an interfacing-system loss-of-coolant accident. Some studies suggest that the steel shell of a containment can be breached when contacted by molten core debris while other studies suggest that heat will be conducted away from the shell at a sufficient rate to prevent meltthrough. The conflicting studies give little confidence in the conclusions that are drawn from them.

Response:

It is agreed that the present information base on severe reactor accident phenomenology is limited and that this base sometimes contains conflicting data. The state of this information base is one reason why the NRC staff chose to characterize individual phenomenological issues and the associated risk analyses by probability distributions, rather than single-valued estimates, and make use of expert judgment, as done in other analyses of poorly understood issues, to review and interpret the available information.

Comment: NUREG–1150 states that for BWRs direct containment heating is not a prominent cause of containment failure because the core support design will allow limited portions of the core to melt and fall into the lower vessel head, causing localized vessel failure before the bulk of the core accumulates in the lower head. This is assumed; no calculations or experiments are offered in support of this hypothesis.

Response:

Many of the values used in the analyses found in draft NUREG-1150 were inadequately justified. The current analyses have been more thoroughly documented. Issues such as the loads on a containment building at the time of reactor vessel breach, including the effects of direct containment heating, were determined through expert interpretation of available calculations and experiments. Details of these analyses are discussed in References D.18 and D.39.

Comment: There are large uncertainties associated with direct containment heating (DCH). An industry group studied DCH and found it not to be a contributor to containment failure in the Sequoyah cavity design. Small-scale experiments indicate that 90 percent of the ejected melt will remain inside the cavity. It is thought that, in the PWR reactor coolant system (RCS), hot leg failure can occur prior to the bottom head failure, precluding direct containment heating because the RCS would be at low pressure at the time of vessel failure. But other experimental studies done at a national laboratory indicate that DCH can occur. Analytical studies suggest that assuming depressurization by operators will not alleviate the problem since there are some accident scenarios in which depressurization cannot be achieved because of a lack of dc power.

Response:

Since draft NUREG-1150 was published in February 1987, the information base for quantifying important issues has been expanded. Among those that received considerable attention were contributors to containment loads during high-pressure melt ejection (including direct containment heating) and the potential for depressurizing the reactor vessel (including RCS hot leg failure). However, the information base remains incomplete. In the present study, experts in severe accidents were asked to interpret the information base and to generate probability distributions required for risk analyses. The experts who participated in this assessment and the information used to quantify containment loads at vessel breach are discussed in Sections C.5 and C.6 of Appendix C.

Through expert judgment, it was concluded that the upper end of the range of potential containment loads accompanying high-pressure melt ejection reached high values (i.e., several times the containment design pressures). A containment analysis indicated, with relatively high confidence, that the Surry and the Zion containment structures could accommodate all but the highest of these loads. A similar conclusion could be drawn for the Sequoyah containment only if certain containment safety features operate (e.g., a substantial inventory of ice is in the ice condenser at the time of vessel breach). Regarding the RCS pressure at vessel breach, the current analyses indicate that in the majority of station blackout accident scenarios, at least one of several mechanisms (e.g., temperature-induced hot leg failure, reactor coolant pump seal failures) will reduce the pressure in the reactor coolant system to sufficiently low values to make high-pressure melt ejection unlikely. During other types of severe accident scenarios, manual actions, such as opening the pressurizer PORV, are likely to occur to similarly reduce the likelihood of high-pressure melt ejection.

Comment: When the necessary conditions exist, steam explosions can occur in a reactor vessel as a result of a degraded core contacting water in the lower head, and the explosion can be sufficiently energetic to cause reactor coolant system and containment building failure. Because of large uncertainties in this technical issue, it should be mathematically treated as a full issue in and of itself and not part of some other issue. The treatment should be consistent with previous studies done by the NRC and the industry.

Response:

The probability of steam explosions sufficient to fail the containment building was treated as a separate issue and a probability distribution developed for the present version of NUREG-1150. This distribution was developed by the NUREG-1150 project staff using, as an initial basis, the work of the Steam Explosion Review Group (Ref. D.44). This work was updated (incorporating the possible effect of new information) by polling the individual members of the review group.

Comment: NUREG-1150 continues to carry the steam explosion issue along even though all but a few steam explosion researchers concluded that no issue remains. Considering this issue as a mechanism for reactor vessel and containment failure is inconsistent with a previous study done by the NRC (the Steam Explosion Review Group).

Response:

As discussed in the previous response, the present NUREG-1150 analyses of the potential for containment building failure by in-vessel steam explosions are based on the work of the Steam Explosion Review Group.

Comment: In draft NUREG-1150, there is an accident sequence in which a containment failure leads to a failure of emergency core cooling, leading to core meltdown. This does not seem plausible.

Response:

This type of accident sequence was first identified in the Reactor Safety Study (Ref. D.7) in 1975 as the S2C accident sequence in the Surry plant and as the TW sequence in the Peach Bottom plant. Since that time, analyses have been performed that indicate that the S2C sequence would not result in core damage in the Surry plant. However, the TW sequence has been investigated in a number of boiling water reactor PRAs and found in some cases to have a not insignificant frequency. For the present study of the Peach Bottom plant, this accident sequence made a small contribution to the core melt frequency, principally because the progression of the accident was slow, permitting operator intervention to preclude core damage.

Comment: The major contributors to Peach Bottom containment building failure appear to be the assumed overpressure failure in the wetwell above the water line, drywell head failure, or the assumed meltthrough of the drywell shell. None of these failure modes are supported to the degree necessary to warrant the level of confidence in the central estimate.

Response:

Consideration of these failure modes in the present version of NUREG-1150 made use of a spectrum of experimental and calculated data. Because these data were often conflicting, expert judgment was used to interpret the data and to develop the probability distributions needed for the risk studies.

The results of the Peach Bottom risk analysis for the present version of NUREG-1150 indicate that meltthrough of the drywell shell is the principal cause of early containment failure in that plant. However, high-pressure melt ejection remains a significant contributor to early containment failure.

Comment: The probabilities of hydrogen detonation in the BWR Mark III containment building and hydrogen combustion-induced failure of the Sequoyah containment building are overestimated. A detonation requires a very high concentration or a geometric configuration that will produce a sufficient flame acceleration. Data from over 40 tests in the Hydrogen Control Owners' Group 1/4-Scale Test Facility and other test data support the notion of containment-wide mixing that precludes high local concentrations and thus local detonations. The effects of diffusive combustion at the suppression pool surface controls the global hydrogen concentration from 4 to 6 volume percent, precluding flame acceleration. In a study of Sequoyah by an industry group, calculations indicate that during a station blackout accident, natural circulation in the containment will permit the recombination of combustible gases as the gases pass over hot debris in a reactor cavity without placing severe loads on the containment. The large hydrogen loads that have been calculated result from inadequate credit given to the recombination of combustible gases.

Response:

Experts, listed in Section C.4, of Appendix C, were asked to interpret information on hydrogen combustion. Each expert was familiar with published data and analyses regarding hydrogen combustion phenomena and their applicability to the distribution of hydrogen in a containment building, the ignition of hydrogen, and the attendant loads. Using these data, they developed distributions characterizing their estimate of and uncertainty in selected parameters, such as ignition frequency, probability of deflagration/ detonation transition, and combustion loads. A summary of the probability distributions and their application in the Grand Gulf and the Sequoyah risk analyses is provided in Section C.4 of Appendix C.

Comment: The initiation of containment sprays in a BWR Mark III containment building should not lead to significant de-inerting. As the sprays cool the containment atmosphere, the reducing pressure will allow the suppression pool to flash as the saturation temperature is reached. The flashing will produce steam and at least partially re-inert the containment atmosphere. It is believed that a hydrogen burn after re-inerting is insufficiently energetic to cause containment building failure.

Response:

The recovery of ac power allows both the containment sprays and the residual heat removal (RHR) system to operate. Eventually, the suppression pool will be cooled by the RHR system; this precludes flashing, and, hence, the containment will de-inert. It is thought that in a containment previously inerted by steam, and after the recovery of both ac power and containment sprays, the severity of a hydrogen burn depends on the relative timing of ignition. Ignition well after recovery of ac power (when the containment atmosphere is not inert) could result in a severe hydrogen burn. However, if ignition occurs soon after ac power recovery, a slow incomplete burn that does not threaten the containment or the drywell could occur. Such incomplete burns are considered in the accident progression trees for Grand Gulf.

The spray de-inerting scenario is not as important in the current analyses as it was in the analyses of the earlier draft of NUREG-1150. Previously, the core damage frequency was dominated by long-term station blackout scenarios when the containment atmosphere is inert at the time of reactor vessel breach. Currently, it is dominated by short-term station blackouts when the containment atmosphere is not inert at the time of vessel breach.

Comment: The phenomena in core-concrete interactions are not well understood; the models used are only approximations that are inadequately validated. But even if detailed models could be formulated, it is unnecessary to be concerned with such details while neglecting to examine the location and the behavior of previously evolved fission products. Because a molten core has lost nearly all its fission product gases, the core-concrete interaction is depleted of fission products.

Response:

The major phenomena of a core-concrete interaction are reasonably well known and understood. Based on experimental studies, models have been developed that adequately represent the phenomena (Refs. D.24 and D.45 through D.47).

While a molten core has lost essentially all the volatile fission products by the time it has penetrated the reactor vessel and begun to interact with concrete, it will still retain a majority of the nonvolatile species. The subsequent evolution of these nonvolatile species can have a significant impact on the overall consequences.

D.6.3 Containment Structural Response

Comment: There is no universal definition of what constitutes a containment failure, including leak failures and penetration failures.

Response:

The accident progression event trees make a distinction between different failure locations and magnitudes of leakage. The source terms for each accident progression bin account for the effects of these differences in leakage behavior. The issue of location and mode of failure is probably treated in greater detail in this study than in any previous PRA, making use of available calculations and experimental data on containment building responses to severe accident loads.

Comment: Experimental data on the ultimate potential strength of containment buildings and their failure modes are lacking. This lack of data renders questionable the methods used in draft NUREG-1150 for assigning probabilities and locations of failures.

Response:

The present data on the potential strength of containment structures under severe accident loadings and the potential modes of failure are limited. For the current analyses, the structural engineering experts who reviewed and interpreted the available information are listed in Section C.8 of Appendix C.

Except for the Grand Gulf plant, the experts addressed the response of the containment buildings studied in NUREG-1150 to a range of quasistatic pressure loads associated with severe accident conditions. Other containment failure mechanisms, such as penetration by a missile, structural failure due to impulse loads (e.g., hydrogen detonation for Grand Gulf and Sequoyah), and meltthrough by molten material, were not addressed by structural experts directly but were addressed for some plants through other aspects of the risk analysis. For example, drywell shell meltthrough in the Peach Bottom reactor (BWR Mark I containment) was addressed by a separate expert panel.

The results of the experts' reviews and judgments are described in Appendix C of NUREG-1150. Briefly, the median failure pressure ranged from 50 and 65 psig for Grand Gulf and Sequoyah, respectively, to 125 and 130 psig for Surry and Zion, respectively. The uncertainty in these estimates spanned a range of approximately 50 to 70 psi, regardless of the absolute range of the design or failure pressure. For the two large, dry containments, Surry and Zion, the median failure pressure corresponds to approximately three times the containment design pressure. The median failure pressure of 65 psig for the ice condenser containment, Sequoyah, was substantially lower than that for the large, dry containments; however, this value corresponds to more than six times the design pressure. For the two BWR containments, the Mark I at Peach Bottom and the Mark III at Grand Gulf, the ultimate capacity of the containments was estimated to be 150 psig and 50 psig, respectively. This corresponds to approximately three times the respective design pressures. The failure pressure of the Mark I containment was judged to be extremely sensitive to the drywell atmospheric temperature. As described in Appendix C, Section C.8, the ultimate capacity of the Peach Bottom drywell shell may decrease to levels at or below the containment design pressure if the drywell temperature exceeds 1200°F.

Comment: The Electric Power Research Institute (EPRI) has been conducting experiments to confirm the hypothesis that steel-lined concrete containments will develop small leaks before experiencing gross failure when subjected to high internal pressures. Industry computer programs have been modified to represent the behavior of steel liners and concrete. Codes have been developed that have been validated against experiments and can be used to analyze actual containments. This source of information should be used in NUREG-1150.

Response:

Results of the EPRI tests were discussed in the expert elicitation process and were used in quantifying the failure pressure and modes of the concrete containments. A participant in the EPRI program served on the containment performance panel of experts.

Comment: In BWR analyses, secondary containments should be taken into account because there are divergent views on the capability of this structure to withstand a failure of the primary containment and to retain aerosols.

Response:

In the current version of NUREG-1150, the decontamination factor (DF) of the reactor building was quantified through expert interpretation of available data. The judgments on the DFs (for several release rates, steam concentrations, and flow patterns) were based on models of and calculations from mechanistic codes, personally developed models, and experiments. A DF was not applied to the reactor building of the Grand Gulf plant because the most likely failure location is at the top of the containment;

the only structure between the anticipated failure location and the environment is a corrugated metal structure that is judged to fail immediately after containment failure.

Comment: The Peach Bottom analysis was based on the concept of a freestanding structure. However, the failure pressure would be higher than considered in draft NUREG-1150 because the steel shell would get support from the concrete as it expands under pressure loading.

Response:

The performance of the Peach Bottom steel shell was reviewed by an expert panel of structural engineers. Data available to the panel members included an analysis by the Chicago Bridge and Iron Company on the structural capability of the Peach Bottom steel shell, explicitly including the effects of the concrete biological shield surrounding the shell. The results of the expert review are discussed in Section C.8 of Appendix C.

Comment: The assumption that the drywell shell fails as the molten core material contacts the shell is driven by expert judgment. The failure might be delayed or averted if the shell conducts heat away from the contact point rapidly.

Response:

The potential for drywell shell failure by direct contact of molten core material has been analyzed by a number of organizations, including Brookhaven National Laboratory, Oak Ridge National Laboratory, Sandia National Laboratories, Massachusetts Institute of Technology, University of Wisconsin, Fauske & Associates, Inc., and the Electric Power Research Institute. The results of these analyses are conflicting.

For the present version of NUREG-1150, the analyses were reviewed by experts listed in Section C.7 of Appendix C; their results are presented in the same section.

D.7 Source Terms and Consequences

D.7.1 Methods

Comment: The NUREG-1150 study and the findings are inconsistent with past research trends, which have been toward more mechanistic codes resulting in smaller uncertainties. The computer codes used in NUREG-1150 are becoming less mechanistic and the uncertainties appear to be increasing.

Response:

The NUREG-1150 study used a simplified approach for calculating radioactive releases because the large number of such estimates needed to express uncertainty could not all be made with the long-running and resource-intense codes such as the Source Term Code Package (STCP). Simple algorithms were used to make these calculations; the algorithms are collectively known as the XSOR codes. However, the bases for these algorithms were calculations with a set of more mechanistic codes, including the STCP (Ref. D.24), CONTAIN (Ref. D.25), MELCOR (Ref. D.26), and MELPROG (Ref. D.27).

In order to address the adequacy of the estimates provided by the parametric computer codes, complementary calculations were performed with the STCP to benchmark the parametric analyses. In general, the parametric calculations were found to be in reasonable agreement with the calculations from the STCP. Discrepancies in the parametric calculations, relative to the STCP calculations and to expert judgment, could be explained. Details of these comparisons are reported in Reference D.30.

Comment: The uncertainties in risk have not been properly quantified because the Source Term Code Package used in NUREG-1150 does not account for reevolution or resuspension of deposited fission products either in the reactor coolant system following vessel failure or in the containment. High temperatures would cause ruthenium, which is normally nonvolatile, to form oxides, which are volatile.

Response:

The XSOR analyses in both the draft and the present studies account for a number of processes such as revaporization of material deposited on reactor coolant system surfaces and the volatilization of iodine from water pools late in the accident. The characterization of these processes was made in terms of probability distributions from expert elicitations. The bases for the expert judgments were provided by direct experimental evidence and analyses using mechanistic computer codes such as TRENDS (Ref. D.48), which predicts iodine transport in containment, and a revaporization model developed by the Sandia National Laboratories for the SCDAP/RELAP code (Ref. D.49).

The basis for the STCP analysis of the ruthenium release is the CORSOR model, which is a semiempirical model based on a number of reasonably prototypic experiments. The distribution of release estimates that was actually used in the risk study was obtained from a panel of source term experts. The range of release for the ruthenium group is quite broad. The thermodynamics of ruthenium are considered explicitly in the VANESA model in the STCP, which predicts core-concrete release. Vapor species considered by VANESA are Ru, RuO, RuO_2, RuO_3, and RuO_4. Ruthenium oxidation was also considered in the development of source terms for direct containment heating and steam explosions.

Comment: All research efforts in the past several years have been geared toward making the source term estimates more mechanistic. The NUREG-1150 study goes against these trends by developing and using simple algorithms to estimate source terms. The algorithms do not represent the same level of understanding of source terms as do the mechanistic codes, such as the Source Term Code Package (STCP). The algorithms are merely linear combinations of aggregated variables representing many factors determining source terms, one of the most important of which, timing, is not included. At least the important source terms should be calculated with the STCP, not extrapolated with simple parametric codes. There is no justification for assuming that the variables are linearly related as was done in the simplified parametric source term codes.

Response:

With the introduction of quantitative uncertainty analyses in NUREG-1150, a large number of source term calculations became necessary. The number needed was far too many to be performed with a mechanistic code. In addition, no one code contained the "best" models for all phenomena considered potentially important to the transport analyses. As a result, parametric computer codes were developed, based on the results of detailed calculations of accidents by a number of computer codes, including the Source Term Code Package (STCP) (Ref. D.24), the CONTAIN code (Ref. D.25), and other codes.

While time is not a formal variable in the parametric codes, time dependency of fission product release is included, in that the releases are broken up into in-vessel and ex-vessel portions. Other factors include the timing of containment failure, the time periods over which the containment sprays operate, and the timing of concrete attack.

Comment: The NRC must convince the public and the nuclear community that the Source Term Code Package (STCP) is reliable. The STCP imposes choices on model selection, and no attempt has been made to determine if a different choice would give a significantly different outcome. The NRC should hold a workshop on the STCP and publish a description of the STCP. Because the STCP has not been extensively used in the nuclear community, it is important to review the code; an international consensus may be needed.

Response:

The STCP has been extensively reviewed, as discussed in Reference D.24. In that study, the NRC staff assessed the technology for estimating source terms. The study was reviewed by the American Physical Society (reported as Ref. D.36). The STCP and the results obtained with it were an integral part of a series of meetings between the NRC and an industry group (IDCOR) to exchange technical information. In addition, the STCP has also been the subject of validation and verification efforts by groups other than those involved in its development (Refs. D.50 and D.51). It has been used by numerous organizations in the United States and abroad and, through agreements with the NRC, researchers report their experience in using the code.

The default model selections in the STCP are those that are believed to be most consistent with the understanding of relevant phenomena and considering the limitations of the code. A different choice of models yields different results and, for this reason, choices other than default choices are discouraged. Because the modeling of particular scenarios may require use of alternative assumptions, the model choices are still available in the STCP.

The input decks to operate the STCP were developed using the Final Safety Analysis Report of a plant, information from the plant owners, information from the reactor manufacturers, and information obtained during visits to plants. For any particular STCP calculation, the input data were verified.

The NRC staff recognizes that the STCP has limitations, which are reported in Reference D.24. To compensate for the limitation in the NUREG-1150 study, expert judgment was used. The judgment is based on the information available at the time of the calculations, such as experimental studies and analytical studies using the STCP and other codes. The judgments were factored into the risk estimates through empirically based algorithms collectively known as the XSOR codes; the variables in these codes are more general and subjective than the variables found in the mechanistic codes such as the STCP but semiquantitatively account for phenomena for which rigorous models do not exist. Whenever possible, a mechanistic source term calculation benchmarks the XSOR estimates. A study that compares expert opinion, STCP calculations, and XSOR estimates is reported as Reference D.30.

Comment: Natural circulation is a complex phenomenon. There is no evidence in NUREG-1150 to suggest that the complexity is appreciated or that it is adequately modeled.

Response:

The effect of natural circulation in the reactor coolant system on the potential for early failure and depressurization prior to meltthrough of the vessel was an issue considered by one of the expert panels in the current version of NUREG-1150. The panel had access to the results of a number of analyses performed with industry and NRC computer codes, as well as experimental data (Ref. D.18). The likelihood of vessel meltthrough with the reactor coolant system at high pressure is low in the current analyses, in part because of this potential for early depressurization. The possibility of containment bypass resulting from temperature-induced failure of steam generator tubes is also represented in the current analyses. However, the likelihood of this bypass mechanism is assessed to be small.

Natural circulation patterns can also affect the progression of a core meltdown and the production of hydrogen in a reactor vessel. This was taken into account in the current NUREG-1150 analyses; experts interpreted the results of analyses performed with NRC-sponsored codes and also with codes sponsored by the industry.

D.7.2 Supporting Data Base and Modeling Assumptions

Comment: Evidence suggests that cesium iodide is stable but in NUREG-1150 it is modeled otherwise. No data or experience suggest that iodine will revolatilize from a basic aqueous solution that would form because of the high percentage of cesium in fission product releases. Iodine remained in solution in the Three Mile Island plant for several years after the accident.

Response:

Cesium iodide is not completely stable either in transport through the reactor coolant system or in solution. The issue is how much of the more volatile form is produced. Recent experimental evidence and analysis indicate that the production of volatile forms in the reactor coolant system is smaller than characterized in the previous draft of NUREG-1150. The late release of iodine from the suppression pool is an issue that was addressed by an expert review panel for the current version of NUREG-1150. Results of TRENDS code analyses and direct experimental data were considered by the expert panel. The projected pH of the pool was an important consideration. The extent of reevolution obtained in the current study is not as great as in the draft report. For a subcooled suppression pool, the upper bound of the distribution used is 10 percent and the median value is 0.1 percent. For a saturated or boiling suppression pool, larger releases are predicted.

Comment: Key issues that lead to high source terms, such as drywell liner meltthrough, core melt progression, and late iodine release, should be subject to further experimental evaluation.

Response:

Significant research results have been obtained in each of these areas subsequent to the release of draft NUREG–1150. At the time this response is being written, the NRC is in the process of reorganizing and reprioritizing its Severe Accident Research Program, in part to account for insights generated in this study. It is anticipated that the highest priority for research over the next few years will be given to resolving issues associated with potential threats to containment integrity, such as drywell shell meltthrough, rather than to source term phenomenological issues, such as the late release of iodine from water pools.

Comment: Severe fuel damage (SFD) scoping tests on decladded fuel collapse are inappropriate for validating the models of core melt phenomena because the conditions for the experiments and the conditions represented in the codes are different.

Response:

The performance of integral fuel damage experiments always involves substantial compromise in achieving prototypic severe accident conditions. A considerable effort has recently been initiated at the Brookhaven National Laboratory to provide a quantification of scale distortion and the effects associated with extrapolations, correlations, or models used beyond their data base to quantify code uncertainties.

Comment: Steam generator tube rupture occurring as a result of a core damage accident was found to be an important contributor to the probability of containment bypass. This assumes that fission products get into the steam generator; detailed analyses indicate that fission products will deposit in the pressurizer and pressurizer surge line, not in the steam generator. Industry studies suggest otherwise.

Response:

It is not necessary for fission products to deposit in the steam generator to obtain overheating and failure of the tubes. The Westinghouse experiments on natural circulation indicate that the convective flow path can occur to the steam generators by means of stratified flow in the hot legs. Failure of steam generator tubes prior to hot leg or surge line failure is not considered likely. In part because steam generator tubes may be degraded, some likelihood of tube failure was assessed (by an expert panel) and is included in the analyses.

Comment: Many assumptions are made in the modeling of core degradation phenomena, such as 50 percent of a core becoming molten before slumping occurs and a single well-defined melting point. These assumptions have a large effect on the predicted source terms.

Response:

There are some simplifications in the core meltdown models in the Source Term Code Package (STCP), such as the use of a single temperature for fuel melting, which can affect the magnitude of the source term. Of greater significance is the lack of models in the STCP to predict some highly uncertain processes, such as revaporization of fission products from reactor coolant system surfaces after vessel failure. It is important to understand the relationship among the STCP, the XSOR codes, the detailed mechanistic codes, and the use of expert judgment in treating uncertainties in this study. The STCP was only used as a benchmark for the XSOR codes. It played a very small role in the quantification of the accident progression event trees. Probability distributions for the most important uncertain parameters affecting the source terms were determined by expert panels, based in large part on the results of mechanistic code analyses and experimental results. The source term ranges obtained in this study are dominated by the treatment of these uncertain parameters, not by modeling approximations of core melt progression in the STCP.

Comment: Debris cooling is assigned a low likelihood of occurrence in cases where models based on experiments would predict a coolable geometry.

Response:

In the present version, debris coolability was considered for conditions involving water and debris interactions both in-vessel and ex-vessel. Based in part on the experience from the accident at Three Mile Island on March 28, 1979, it was assumed that water recovery of a damaged core in-vessel could result in arresting core degradation. The likelihood of arrest was decreased as a function of the time into the accident. A window of time for recovery was estimated that was determined by the amount of core debris estimated to be on the lower head of the vessel. For minimal core degradation, a high likelihood (0.9) of arrest is assumed if an emergency coolant supply is reestablished. Beyond a level of debris accumulation, the likelihood of arrest is assumed to be low (0.1). The likelihood of arrest decreases linearly over the time interval.

The likelihood of a coolable debris bed being established ex-vessel was assessed for a variety of different conditions that depended on whether the reactor vessel was at high or low pressure at the time of vessel failure, the size of the failure, the depth of the water in the reactor cavity, and the temperature of the debris. For the different cases, the likelihood of the debris bed being coolable ranges from 20 to 90 percent. Of course, a continuous water supply is a prerequisite to long-term coolability.

Comment: Water in the lower head or water injected into the reactor vessel can have a significant effect on accident progression. The possibility that debris can become critical when flooded is never considered.

Response:

In the PWRs, emergency coolant water is borated. The likelihood of recriticality following flooding is considered small and was not represented in this study. In some BWR sequences, a period of time exists when the control blades may have melted and relocated while the fuel pellets are essentially in their normal configuration. Under these circumstances, reflooding could result in a critical condition. In the present study, the likelihood of recriticality under these specific conditions was considered high but the possibility of an energetic excursion with the potential to fail the vessel was assessed to be small (Ref. D.18).

Comment: In the Sequoyah analysis, it is recognized that water can boil away from debris in a reactor cavity leading to a core-concrete interaction after the ice is depleted and the containment has failed. The scenario does not seem correct because the steam should condense and replenish the debris bed with coolant.

Response:

The analyses account for condensation in the containment and the replenishment of water in the reactor cavity. Dryout of the debris bed only occurs after containment failure and an extended period of steam loss from the containment.

Comment: Data from the accident at Three Mile Island do not support the core melt and fission product release models.

Response:

To the extent that TMI-2 data can be interpreted to evaluate the magnitude of fission product release and the extent of fuel damage during core uncovery, the TMI-2 data are consistent with the Source Term Code Package (STCP) analyses. Benchmark analysis with the STCP shows good agreement with the pressure history (Ref. D.52). Examination of TMI-2 core debris indicates that 70 to 80 percent of the iodine and cesium had escaped the core debris, much of which had never been completely melted (Ref. D.53). These results are consistent with the STCP analyses. Because of the subsequent flooding of the TMI-2 vessel, the examination of the reactor coolant system samples was not able to provide information on the extent of radionuclide deposition during the period of core uncovery and fission product release from the fuel.

Comment: In the risk study, an assumption was made that 5 percent of a population surrounding a nuclear power plant will not evacuate during a severe accident. No basis for this assumption was given. Actual emergency events, both with and without emergency plans, should be used to develop a well-founded value. The assumption should take into account a failure in offsite emergency response, such as may be caused by the failure of buildings, roads, and bridges during an earthquake. A realistic assumption must be used because it significantly affects the calculated consequences.

Response:

The assumed 5 percent non-evacuation of the population within the 10-mile emergency planning zone (EPZ) of a reactor that was used in the calculations for draft NUREG-1150 is conservative. In the current study, the assumption was changed to 0.5 percent, based on the following rationale. The plants that were studied in NUREG-1150 have detailed and well-maintained emergency plans, which also have provisions for evacuating from special facilities within the EPZ. Because an evacuation is preplanned, it is expected to be nearly complete. The preplanned evacuation should be distinguished from unplanned and impromptu evacuations prompted by transportation accidents involving toxic chemicals, accidents at chemical plants, or natural disasters. The specific value used (0.5 percent) was derived from an actual use of a nuclear emergency plan (for a nearby chemical accident). The current study includes displays of the offsite consequences and risk with assumptions on the alternative modes of emergency response within the EPZ, such as evacuation, early relocation, sheltering, and partial (i.e., 0 to 5 miles) evacuation/partial (i.e., 5 to 10 miles) sheltering. Sensitivity calculations of severe accident consequences during an earthquake, assuming a degraded emergency response, are reported in the supplements of NUREG-1150 (Refs. D.12 and D.13).

Comment: It is unreasonable to assume that, once an individual evacuates beyond 15 miles from a damaged reactor, no further dose is received.

Response:

In the current consequence analysis, an individual is assumed to avoid further radiation exposures after reaching a radial distance of 20 miles from a reactor. People who evacuated from the 10-mile emergency planning zone (EPZ) of the Grand Gulf and Peach Bottom plants would need about 3 to 4 hours to travel to a distance of 20 miles from these reactors; people who evacuated from the 10-mile EPZ of the Surry, Sequoyah, and Zion plants would need about 7 to 11 hours to travel to a distance of 20 miles from those reactors. It seems reasonable to assume that by then the location of the radioactive plume and the area contaminated by it would have been known and people would have been advised on how to avoid it.

Comment: The MACCS code should be thoroughly documented and benchmarked. A study should be done of how the MACCS code compares to other codes. The MACCS code should be thoroughly verified and validated to ensure the validity and accuracy of the models, data, and assumptions.

Response:

In late 1987 the NRC staff began an inhouse benchmarking activity on the version of the MACCS code (Version 1.4) used for draft NUREG-1150. This activity consisted of a comparison of MACCS 1.4 results with the results from various research groups calculating consequences using their own consequence codes. The research groups were members of the Organisation for Economic Co-operation and Development/Nuclear Energy Agency/Committee on Safety of Nuclear Installations (OECD/NEA/CSNI). The benchmarking activity revealed errors in MACCS 1.4 and these are reported in References D.54 and D.55.

A comparative review has been performed at the Institute for Energy Technology, Norway, of the chronic exposure pathways modeled in MACCS 1.5 with other consequence codes used by the OECD member countries. The findings are reported as Reference D.33.

The Idaho National Engineering Laboratory (INEL) performed the quality assurance and verification of MACCS 1.5. Sandia National Laboratories (SNL), the developers of the code, assisted INEL because the code had not been adequately documented. The quality assurance program was a line-by-line check of the Fortran coding and crosschecking with the model equations for consistency.

Corrections of most of the errors identified in the Norwegian and INEL reviews were completed in MACCS 1.5 used in this version of NUREG-1150. The residual errors in the MACCS code appear to cause an error of about a factor of two in the latent cancer fatality and population dose estimates; the errors in the code will be corrected before the code is made available to the public. The documentation (user's manual, model descriptions, and programmer's manual), the Norwegian review, and a report of the final crosschecking by INEL (Ref. D.56) are expected to be completed in the summer of 1989.

Comment: It seems to be surprising and erroneous that the bulk of the late fatalities associated with large radionuclide releases is derived from the long-term doses committed via food chain pathways of exposure at low levels of individual dose. It contradicts the results of earlier studies (such as the Reactor Safety Study (Ref. D.7), the German Risk study (Ref. D.57), and the Sizewell B calculations (Ref. D.58)), which concluded that the bulk of the late fatalities is associated with relatively high levels of individual dose attributable to relatively short-term exposure following an accident. Substantial work is required to verify this important difference.

Response:

Errors were found in the calculations of the radiological consequences from the food chain pathways of exposure. These errors were partly from errors in the input to the consequence analysis code, MACCS 1.4, and partly from incorrect modeling of the removal of cesium from the root zone. The incorrect modeling assumed little removal of cesium from the root zone because of irreversible binding of cesium to the soil or cesium percolating through the soil beyond the root zone. This caused the root uptake pathway ingestion doses to be unreasonably large.

D.7.3 Comparisons with Accident at Chernobyl

Comment: Offsite doses (versus distances) reported in NUREG-1150 for some of the source terms are too high and indicate a potential for causing prompt fatalities in the offsite population. In contrast, there were no prompt fatalities outside the plant in the Chernobyl accident.

Response:

The potential of a large radioactive release to the atmosphere to result in high doses and prompt fatalities in the public depends on the meteorological conditions during and immediately following the release and the energy content of the release.

A large radioactive release during favorable meteorological conditions may not have the potential for causing prompt fatalities in the public. The opposite is possible if releases occurred during unfavorable meteorological conditions. In a PRA framework, many alternative meteorological conditions are used (based on actual site data), some of which are favorable and some of which are unfavorable so that the effect of virtually all meteorological scenarios can be represented.

A release accompanied by a large quantity of thermal energy may result in the plume lifting off from the building wake and rising in the atmosphere while being transported by the wind, resulting in low offsite doses. This happened during the release from the Chernobyl accident and, therefore, there were no offsite prompt fatalities.

Comment: After the Chernobyl accident, it is difficult to justify a lack of accounting of doses beyond 50 miles in the risk calculations.

Response:

In draft NUREG–1150, the radiological consequence calculations were limited to 500 miles from a damaged reactor. In the analysis for the present version of the report, all radioactive material (except for the noble gases) remaining in the plume at 500 miles from the plant was deposited on the ground between 500 miles and 1,000 miles from the reactor. The contribution of all pathways of exposure between 500 and 1,000 miles are also included in the estimates of the consequences. This ensures nearly 100 percent accounting of the released radionuclides in the consequence calculations. The impact of the small quantities of noble gases leaving the 1,000-mile region is negligible.

Comment: It should be made clear that variations in the relocation/decontamination/interdiction dose criteria are included in the cost uncertainties.

Response:

Uncertainties in the offsite consequence models and the values of the input parameters to the consequence code were not treated in the previous analyses. In those analyses, the long-term relocation criterion of the Reactor Safety Study (Ref. D.7) of 25 rems in 30 years from groundshine was used. In the current analyses, the relocation criterion of 4 rems in 5 years from groundshine is used for base case calculations; this criterion is an approximation of the criterion currently proposed by the U. S. Environmental Protection Agency (Ref. D.59). The criterion found in the Reactor Safety Study is also used in the current calculations but only to show the sensitivity of the long-term health effects.

Comment: More discussion is needed on the assumptions that have been made about relocation and time scales for the decontamination of property after the 7-day emergency phase when doses could still be high.

Response:

In the current consequence calculations, decontamination of both land and buildings was assumed to reduce the levels of radioactive material by a factor of three or 15. A reduction by a factor of three was assumed to require 60 days of decontamination work; a reduction by a factor of 15 was assumed to require 120 days of decontamination work. The decontamination efforts were assumed to commence at the end of the 7-day emergency phase. The affected people were assumed to be relocated during the decontamination period.

D.8 Uses of NUREG–1150 As a Resource Document

D.8.1 Uses

Comment: The way that NUREG–1150 will be factored into the regulatory process is unclear.

Response:

NUREG–1150 is not intended to represent a quantitative and systematic evaluation of regulations. However, NUREG–1150 does provide a source of information that can support, at least in part, such an objective. The document provides an information base for analyzing plant-specific and generic safety issues. The NUREG–1150 models can be used for assessing the safety significance of operational occurrences and as a basis for evaluating alternative design changes to improve safety. A discussion of the use of NUREG–1150 in the regulatory process is provided in Chapter 13 of NUREG–1150 and in Reference D.8.

Comment: Risk assessment offers a logical framework to review regulations and examine safety issues. The decision to use risk assessment as a basis for regulatory decisionmaking is a major advancement in the regulation of nuclear power plants. But regulators need to be fully aware of the strengths and the weaknesses of their tools and to be concerned with the degree of precision needed to ensure safety.

Response:

NUREG–1150 is not intended to represent a quantitative and systematic evaluation of regulations. However, NUREG–1150 does provide an information source that can support such an objective. The results in NUREG–1150 will be used with full recognition of the uncertainties involved and the strengths and the weaknesses of the methods from which the results were derived (as described in Chapters 1 and 13).

Comment: The interpretation of expert judgments about containment response in terms of probabilities has a large effect on risk estimates. The NUREG–1150 study suggests that the major contributor to risk is early containment failure, but the large uncertainty precludes any regulatory decision on the need for risk reduction using, for example, venting strategies, refractory-lined cavities, and in-plant emergency procedures.

Response:

There are necessarily large uncertainties associated with severe accident risks. These large uncertainties are due in part to a lack of understanding associated with many of the complex phenomena in severe accidents. The uncertainty in risk does not preclude its use in decisionmaking. Decisions must be made in spite of the uncertainties, but the uncertainties may change the type of decision being made.

Comment: The application of NUREG–1150 should be discontinued until the risk estimates are improved.

Response:

Draft NUREG–1150 has not been widely used as a basis for regulatory action during the comment period or while modifications were under way. However, the interim findings of the draft report and the methods developed were not completely ignored. Rather, items identified as being potentially important in the draft report were considered in developing the listing of items for consideration in the guidance provided for the individual plant examination process. Similarly, the information on containment performance provided one input into the NRC's containment performance improvement study. The draft NUREG–1150 information provided a starting point for the development of a more focused in-depth analysis of various issues. As discussed in Chapter 13, the future uses envisaged for NUREG–1150 do not rely heavily on the quantitative results obtained for the five plants analyzed.

D.8.2 Cost/Benefit Analysis

Comment: The models used in calculating the cost of a severe accident lack many factors that should be taken into account. Many of the assumptions are questionable and unfounded. The models have not been benchmarked. Some interpretations and conclusions that were made in draft NUREG–1150 are questionable. The cost estimates need to be more thoroughly documented to understand and evaluate the calculations.

Response:

The present version of NUREG–1150 provides a limited set of risk-reduction calculations, principally related to the potential benefits of accident management strategies in reducing core damage frequency. It does not assess the costs of these or other improvements. Such analyses are more properly considered in the context of specific regulatory actions.

Comment: The averted cost (in terms of risk reduction) does not include secondary costs. The draft risk study recognizes that secondary costs may significantly increase the benefit of some safety options but no attempt was made to quantify this increase. The underestimate can be attributed to the following:

1. The cost of shutdowns of similar reactors on the same site and at other sites.

2. The possibility of a moratorium on nuclear power due to a severe accident.

3. The value of $1 million per averted acute fatality and $100,000 per latent cancer fatality may be too low.

4. The values of the interdiction dose used in the calculations may be too high.

5. No allowance has been made for the decrease in long-term value of land and buildings that have been contaminated.

6. Decontamination costs used in the calculations may be based on decontamination of test sites in deserts instead of agricultural, residential, and commercial property.

7. The radiation releases are calculated out to 50 miles; a radius as much as 500 miles may be more appropriate.

Response:

The draft NUREG-1150 cost/benefit analyses reflected the conventional NRC methods for assessing costs and benefits. Because cost/benefit analyses are more properly considered in the context of specific regulatory activities, they are not provided in this version of NUREG-1150.

D.8.3 Safety Goal Comparisons

Comment: NUREG-1150 finds that the U.S. safety goals are met; this discourages further improvements in safety.

Response:

As discussed in Chapter 13, this version of NUREG-1150 indicates that the five plants studied in NUREG-1150 are below the Commission's quantitative health objectives. The NRC staff disagrees that the findings of NUREG-1150 discourage further improvements in nuclear safety. Many improvements have been made at the five plants studied in NUREG-1150 since the draft report was first published in February 1987; some of the safety improvements arose from this study of various features of the five plants. The NRC staff believes that a comprehensive risk analysis on a plant enhances safety because it presents an overall and comprehensive view of interactions among plant systems and operator actions. Similarly, the variety of perspectives drawn in NUREG-1150, particularly in Chapters 8 and 12, provide information that other plants may consider as they perform individual plant examinations.

D.8.4 Extrapolation of Results

Comment: It is unclear how NUREG-1150 will be used in conjunction with individual plant evaluations or plant-specific PRAs as a basis for regulatory decisionmaking.

Response:

Perspectives gained from NUREG-1150, previous industry-sponsored PRAs, and analyses done by industry groups, such as IDCOR's analysis of four containment configurations, have been assembled into several NRC reports (Ref. D.60). These reports provide information to the analysts performing individual plant examinations (IPEs) concerning plant features and operator actions that are important to the evaluation of risk. Chapter 13 discusses how NUREG-1150 is used in the IPE; details of the IPE process are presented in Reference D.61.

REFERENCES FOR APPENDIX D

D.1 U. S. Nuclear Regulatory Commission (USNRC), "Draft NUREG-1150 for Public Comment: Issuance and Availability," *Federal Register*, Vol. 52, p. 7950, March 13, 1987.

D.2 USNRC, "Seminar on Methodology Used in NUREG-1150, 'Reactor Risk Reference Document'," *Federal Register*, Vol. 52, p. 8390, March 17, 1987.

D.3 ACRS, "ACRS Comments on Draft NUREG-1150, 'Reactor Risk Reference Document'," letter from W. Kerr, Chairman, Advisory Committee on Reactor Safeguards, to L. W. Zech, Chairman, USNRC, July 1987. ACRS, "Report on NUREG-1150, 'Reactor Risk Reference Document'," letter from W. Kerr, Chairman, Advisory Committee on Reactor Safeguards, to L. W. Zech, Chairman, USNRC, August 16, 1988. ACRS, "NUREG-1150: Resolution of ACRS Comments," letter from F. Remick, Chairman, Advisory Committee on Reactor Safeguards, to L. W. Zech, Chairman, USNRC, January 23, 1989.

D.4 H. J. C. Kouts et al., "Methodology for Uncertainty Estimation in NUREG-1150 (Draft): Conclusions of a Review Panel," Brookhaven National Laboratory, NUREG/CR-5000, BNL-NUREG-52119, December 1987.

D.5 W. E. Kastenberg et al., "Findings of the Peer Review Panel on the Draft Reactor Risk Reference Document, NUREG-1150," Lawrence Livermore National Laboratory, NUREG/CR-5113, UCID-21346, May 1988.

D.6 L. LeSage et al., "Initial Report of the Special Committee on Reactor Risk Reference Document (NUREG-1150)," American Nuclear Society, April 1988.

D.7 USNRC, "Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

D.8 USNRC, "Integration Plan for Closure of Severe Accident Issues," SECY-88-147, May 25, 1988.

D.9 J. S. Evans et al., "On the Propagation of Error in Air Pollution Measurements," *Environmental Monitoring and Assessment*, Vol. 4, pp. 139-53, 1984.

D.10 H. W. Lewis et al., "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission," Ad Hoc Review Group, NUREG/CR-0400, September 1978.

D.11 D. I. Chanin , H. Jow, J. A. Rollstin et al., "MELCOR Accident Consequence Code System (MACCS)," Sandia National Laboratories, NUREG/CR-4691, Vols. 1-3, SAND86-1562, February 1990.

D.12 R. J. Breeding et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, October 1990.

D.13 A. C. Payne, Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Revision 1, SAND86-1309, December 1990.

D.14 J. J. Gregory et al., "Evaluation of Severe Accident Risks: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, December 1990.

D.15 T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 6, Revision 1, SAND86-1309, December 1990

D.16 C. K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

D.17 D. B. Clauss, "Round-Robin Analysis of the Behavior of a 1:6-Scale Reinforced Concrete Containment Model Pressurized to Failure: Posttest Evaluations," Sandia National Laboratories, NUREG/CR-5341, SAND89-0349, October 1989.

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

D.18 F. T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

D.19 R. C. Bertucio and J. A. Julius, "Analysis of Core Damage Frequency: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Revision 1, SAND86-2084, April 1990.

D.20 A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

D.21 R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

D.22 M. T. Drouin et al., "Analysis of Core Damage Frequency: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 6, Revision 1, SAND86-2084, September 1989.

D.23 M. B. Sattison and K. W. Hall, "Analysis of Core Damage Frequency: Zion Unit 1," Idaho National Engineering Laboratory, NUREG/CR-4550, Vol. 7, Revision 1, EGG-2554, May 1990.

D.24 M. Silberberg et al., "Reassessment of the Technical Bases for Estimating Source Terms," USNRC Report NUREG-0956, July 1986.

D.25 K. D. Bergeron et al., "User's Manual for CONTAIN 1.0, A Computer Code for Severe Reactor Accident Containment Analysis," Sandia National Laboratories, NUREG/CR-4085, SAND84-1204, July 1985.

D.26 R. M. Summers et al., "MELCOR In-Vessel Modeling," *Proceedings of the Fifteenth Water Reactor Safety Information Meeting* (Gaithersburg, MD), NUREG/CP-0091, February 1988.

D.27 S. S. Dosanjh (Ed.), "MELPROG-PWR/MOD1: A Two-Dimensional, Mechanistic Code for Analysis of Reactor Core Melt Progression and Vessel Attack Under Severe Accident Conditions," Sandia National Laboratories, NUREG/CR-5193, SAND88-1824, May 1989.

D.28 R. L. Iman and M. J. Shortencarier, "A Fortran 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use with Computer Models," Sandia National Laboratories, NUREG/CR-3624, SAND83-2365, June 1984.

D.29 M. L. Corradini et al., "A Review of the Severe Accident Risk Reduction Program (SARRP) Containment Event Trees," University of Wisconsin, NUREG/CR-4569, May 1986.

D.30 P. Cybulskis, "Assessment of the XSOR Codes," Battelle Columbus Division, NUREG/CR-5346, BMI-2171, November 1989.

D.31 R. L. Iman et al., "PARTITION: A Program for Defining the Source Term/Consequence Analysis Interfaces in the NUREG-1150 Probabilistic Risk Assessments," Sandia National Laboratories, NUREG/CR-5253, SAND88-2940, May 1990.

D.32 E. D. Gorham-Bergeron et al., "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration, and Uncertainty Analyses," Sandia National Laboratories, NUREG/CR-4551, Vol. 1, Draft Revision 1, SAND86-1309, to be published.*

D.33 U. Tveten, "Review of the Chronic Exposure Pathway Models in MACCS and Several Other Well-Known Probabilistic Risk Assessment Models," Institutt for Energiteknikk, Norway, NUREG/CR-5377, June 1990.

D.34 Commonwealth Edison Company of Chicago, "Zion Probabilistic Safety Study," September 1981.

D.35 USNRC, "Nuclear Power Plant Severe Accident Research Plan" (G. P. Marino, Ed.), NUREG-0900, Revision 1, April 1986.

D.36 R. Wilson et al., "Report to the APS of the Study Group on Radionuclide Release from Severe Accidents at Nuclear Power Plants," *Reviews of Modern Physics*, Vol. 57, No. 3, Part II, July 1985.

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

D.37 D. L. Berry et al., "Review and Evaluation of the Zion Probabilistic Safety Study: Plant Analysis," Sandia National Laboratories, NUREG/CR-3300, Vol. 1, SAND83-1118, May 1984.

D.38 D. M. Ericson, Jr., (Ed.) et al., "Analysis of Core Damage Frequency: Internal Events Methodology," Sandia National Laboratories, NUREG/CR-4550, Vol. 1, Revision 1, SAND86-2084, January 1990.

D.39 T. A. Wheeler et al., "Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation," Sandia National Laboratories, NUREG/CR-4550, Vol. 2, SAND86-2084, April 1989.

D.40 R. M. Harrington and L. C. Fuller, "BWR-LTAS: A Boiling Water Reactor Long-Term Accident Simulation Code," Oak Ridge National Laboratory, NUREG/CR-3764, ORNL/TM-9163, February 1985.

D.41 W. J. Luckas, Jr., "A Human Reliability Analysis for the ATWS Accident Sequence with MSIV Closure at the Peach Bottom Atomic Power Station," Brookhaven National Laboratory, May 1986.

D.42 Fauske and Associates, Inc., "MAAP Modular Accident Analysis Program User's Manual," Vols. I and II, IDCOR Technical Report 16.2-3, February 1987.

D.43 D. J. Hanson et al., "Containment Venting Analysis for the Peach Bottom Atomic Power Station," Battelle Columbus Laboratories, NUREG/CR-4696, EGG-2464, February 1987.

D.44 USNRC, "A Review of the Current Understanding for the Potential for Containment Failure from In-Vessel Steam Explosions," NUREG-1116, June 1985.

D.45 D. A. Powers et al., "VANESA: A Mechanistic Model of Radionuclide Release and Aerosol Generation During Core Debris Interactions with Concrete," Sandia National Laboratories, NUREG/CR-4308, SAND85-1370, July 1986.

D.46 F. Muir et al., "CORCON-MOD1: An Improved Model for Core/Concrete Interactions," Sandia National Laboratories, NUREG/CR-2142, SAND80-2415, September 1981.

D.47 R. K. Cole, Jr., et al., "CORCON-MOD2: A Computer Program for Analysis of Molten-Core Concrete Interactions," Sandia National Laboratories, NUREG/CR-3920, SAND84-1246, October 1984.

D.48 E. C. Beahm et al., "Calculations of Iodine Source Terms in Support of NUREG-0956," Oak Ridge National Laboratory, ORNL/NRC/LTR-86/17, Technical Letter Report, July 1986.

D.49 G. A. Berna et al., "RELAP5/SCDAP/MOD0 Code Manual," EGG-RTH-7051, EG&G, September 1985.

D.50 T. S. Kress, "Review of the Status of Validation of the Computer Codes Used in the Severe Accident Source Term Reassessment Study (BMI-2104)," Oak Ridge National Laboratory, ORNL/TM-8842, April 1985.

D.51 M. Khatib-Rahbar et al., "Independent Verification of Radionuclide Release Calculations for Selected Accident Scenarios," Brookhaven National Laboratory, NUREG/CR-4629, BNL-NUREG-51998, July 1986.

D.52 R. O. Wooton, "MARCH Calculations Performed for the TMI-2 Analysis Exercise," *Nuclear Technology*, Vol. 87, August 1989.

D.53 D. W. Akers et al., "TMI-2 Core Debris Grab Samples—Examination and Analysis, Part 1," Idaho National Engineering Laboratory, GEND-INF-075, p. 110, September 1986.

D.54 S. Acharya et al., "Benchmarking of the MACCS Code," *Transactions of the American Nuclear Society* (San Diego, California), Vol. 56, pp. 353-4, June 12-16, 1988.

D.55 S. Acharya et al., "Benchmarking of the MACCS Code," *Proceedings of the Second Part of the Joint CEC/OECD/NEA Workshop on Recent Advances in Reactor Accident Consequent Assessment* (Rome, Italy), pp. 50-2, January 25-29, 1988.

D.56 C. A. Dobbe et al., "Quality Assurance and Verification of the MACCS Code, Version 1.5," Idaho National Engineering Laboratory, NUREG/CR-5376, EGG-2566, February 1990.

D.57 The Federal Minister of Research and Technology, "The German Risk Study—Summary," Gesellschaft fur Reaktorsicherheit (Germany), August 15, 1979.

D.58 M. R. Hayes et al., "The Technical Basis of 'Spectral Source Terms' for Assessing Uncertainties in Fission Product Release During Accidents in PWRs with Special Reference to Sizewell-B," United Kingdom Atomic Energy Authority, SRD-R-256, November 1982.

D.59 U.S. Environmental Protection Agency, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," Office of Radiation Programs, Draft, 1989.

D.60 Brookhaven National Laboratory, "Assessment of Severe Accident Prevention and Mitgation Features," NUREG/CR-4920, Vols. 1-5, BNL-NUREG-52070, July 1988.

D.61 NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR §50.54f," Generic Letter 88-20, dated November 23, 1988.

APPENDIX E

RESPONSE TO COMMENTS ON SECOND DRAFT OF NUREG–1150

CONTENTS

TABLES

E.1 Introduction

In June 1989, the NRC published NUREG–1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," as a second draft for peer review (Ref. E.1). At that time, the NRC also formed a peer review committee under the provisions of the Federal Advisory Committee Act to review the second draft report and answer certain questions with respect to its adequacy. This committee was chaired by Dr. Herbert J.C. Kouts; its entire membership is shown in Table E.1. In parallel, the American Nuclear Society (ANS) continued its review of the report, using a special committee of ANS members. This committee was chaired by Dr. Leo G. LeSage; its entire membership is shown in Table E.2. The comments of both committees were provided to the staff in the summer of 1990 (Refs. E.2 and E.3). This appendix summarizes the comments of the NRC-established committee (the "Kouts Committee") and the ANS committee. Summary staff responses are provided for each specific comment.

The second draft of NUREG–1150 has also been the subject of review by the NRC's Advisory Committee on Reactor Safeguards (ACRS). Its technical review was completed in October 1990; a letter providing its comments was submitted on November 15, 1990. This letter is provided as an attachment to this appendix.

Public comment was also requested on the second draft of NUREG–1150. Four comment letters were received (Refs. E.4 through E.7). These comments have also been assessed and, where appropriate, changes made in the final version of NUREG–1150.

Before discussing the comments provided by the committees on particular topics, it is worth describing the overall conclusions and findings expressed in their reports.

The overall conclusions of the Kouts committee were:

- "NUREG–1150 is a good report, and it represents a great deal of detailed high-quality work. It is commendable that an endeavor was made to consult a wider range of competence apart from that possessed by those directly engaged in producing NUREG–1150. The benefit of constructive openness to criticism is felt in the revised draft."
- "NUREG–1150 draws upon a decade and a half of practice of PSA [probabilistic safety analysis] beyond WASH–1400, mainly in the United States but also in other countries. In most respects, it represents the state of the art of this kind of analysis. It is a step forward from WASH–1400."
- "The data drawn on include many years of experience in plant operation, and a similar period of theoretical and experimental research into severe accident methodology."
- "The disciplined use of expert opinion elicitation was an important advance over previous methods of using expert opinion. It is noted that the prime motive of this technique was to assess the uncertainty in the results of the PSA."
- "The results were derived in great detail, and they are presented by methods which show well their probabilistic spread."
- "NUREG–1150 should be a valuable source of data and methodology to guide future PSAs for individual plants. Like its predecessor, WASH–1400, it should help to show the path for future PSA developments for some time to come." (Kouts 7.2)

The overall findings of the ANS committee review were:

- "NUREG–1150 is a major achievement."
- "The revised draft reports essentially a new study."
- "The revised draft provides a balanced presentation of the central tendencies and uncertainties in risk."

Table E.1 'Membership of Special Committee to Review the Severe Accident Risks Report.

Herbert J.C. Kouts	Committee Chairman, Defense Nuclear Facility Safety Board
George Apostolakis	University of California, Los Angeles
E.H. Adolf Birkhofer	Gesellschaft fur Reaktorsicherheit Forschungsgelande, Federal Republic of Germany
Lars G. Hoegberg	Swedish Nuclear Power Inspectorate
William G. Kastenberg	University of California, Los Angeles
Leo G. LeSage	Argonne National Laboratory
Norman C. Rasmussen	Massachusetts Institute of Technology
Harry J. Teague	Safety and Reliability Directorate, United Kingdom Atomic Energy Authority
John J. Taylor	Electric Power Research Institute

Table E.2 Membership of American Nuclear Society Special Committee on NUREG-1150.

Leo G. LeSage	Committee Chairman, Argonne National Laboratory
Edward A. Warman	Committee Vice Chairman, Stone & Webster Engineering Corporation
Richard C. Anoba	Carolina Power and Light Company*
Ronald K. Bayer	Virginia Power Company**
R. Allan Brown	Ontario Hydro, Canada
James C. Carter, III	Tenera Risk Management
J. Peter Hosemann	Paul Scherrer Institute, Switzerland
W. Reed Johnson	University of Virginia
Walter B. Loewenstein	Electric Power Research Institute***
Nicholas Tsoulfanidis	University of Missouri
Willem F. Vinck	Associated Consultant, Belgium****

* Currently with Science Applications International Corporation.
** Member in 1987 and 1988.
*** Member until June 1989.
**** Corresponding member.

- "The use of expert opinion in the revised study was greatly improved."
- "NUREG-1150 should supplant WASH-1400."
- "The NRC safety goals are shown to be met for all five plants studied."
- "The NUREG-1150 documentation is a useful compendium of current severe accident analysis information and data."
- "The quality of the report is substantially improved."
- "[The report] is adequate for its stated uses."

The general comments of the ACRS were:

- "We have reviewed the reports prepared by the ANS Special Committee and by the Special Committee to Review the Severe Accident Risk Report appointed by the Commission [the Kouts Committee] and found them helpful. We have no serious disagreements with either of these reviews, nor with their findings."

- "The work described in this [second] draft of NUREG-1150 is an improvement over that described in the first version entitled, 'Reactor Risk Reference Document.' Many previously identified deficiencies in the expert elicitation process have been corrected. The exposition and organization of the report have been improved. The presentation of results is clearer. There is considerable information that was not in the original version."

- "The portion that deals with accident initiation and development up to the point at which core heat removal can no longer be assured is unique, compared to other contemporary PRAs, in that a method for estimating the uncertainty in the results has been developed and applied. This method and its application are significant contributions. Although the larger contributions to uncertainty in risk come from the later parts of the accident sequences, this portion is enhanced also by an extensive identification of events that can serve as accident initiators as well as an associated set of hypothesized event trees. This information should be of considerable assistance to licensees in the performance of an Individual Plant Examination (IPE). It should also be useful to plant operators and to designers."

- "The formulation of a more detailed representation of accident progression after severe core damage begins, and an improved description of containment performance, contribute some additional information to this important area. However, understanding of many of the physical phenomena that have an important bearing on this phase of accident progression is still very sparse, and the report may give the impression that more is known about this portion of the accident sequence than is actually the case."

- "The part of the sequence that begins with the release of radioactive material outside the containment is treated by a relatively new and unevaluated code system. Furthermore, there is no estimate of the uncertainties inherent in the calculations that describe this part of the sequence. Those who use the quantitative values of reported risk must recognize that these uncertainties are not accounted for in the calculated results."

The ACRS letter contained two other comments of particular note. These were:

- "It is disappointing that the staff asserts that virtually no general conclusions can be drawn from a study that took almost five years and seventeen million dollars to complete. We recommend that the Commission encourage the staff to mine more deeply the wealth of information that has been collected in the course of this study in an effort to identify generic conclusions that might be reached."

- The NUREG-1150 "results should be used only by those who have a thorough understanding of its limitations."

These last comments are discussed in Section E.8 ("Uses of NUREG-1150"). Specific limitations noted in the ACRS letter are discussed throughout this appendix.

The remaining sections of this appendix provide itemizations of comments (including more specific findings of the ANS committee) received from the review committees and the ACRS on the second draft of NUREG-1150 and staff responses. Comments relating to two general areas, scope and documentation, are itemized first (in Sections E.2 and E.3), followed by comments on specific technical areas: use of expert judgment; accident frequency analysis; accident progression analysis; and source term and offsite consequence analysis (in Sections E.4 through E.7). Finally, Section E.8 itemizes comments on the uses of NUREG-1150.

It should be noted that all committees concluded that issuance of the final version of NUREG-1150 should not be delayed for the conduct of further research or analysis. As such, the responses to certain comments indicate that issues requiring significant effort may be the subject of future NRC work rather than included in the final version of NUREG-1150.

E.2 Scope

Chapter 1 of the second draft of NUREG-1150 described the scope of the risk analyses and identified certain limitations of these analyses. The review committees also noted these limitations, as well as others. Some more general comments by the committees with respect to scope included:

- "The second draft of NUREG-1150 addressed many of the shortcomings identified in the first draft and it provided a more comprehensive and incisive view of risk from the existing light water reactors than did WASH-1400." (Kouts 4.1)
- "In general, NUREG-1150 represents state-of-the-art methodology in PSA and associated uncertainty analysis. However, comparison of resulting risk figures between individual plants and with quantitative safety goals must be made with caution, taking into account questions as to the completeness of the analysis and uncertainties in methods and data." (Kouts 4.12) (Such reservations are itemized in the comments below.)
- "Many of the limitations and uncertainties mentioned above [in Section 4.12 of the Kouts report] may be reduced by improved PSA methodology and by improved experimental and empirical data. Such improvements should be made part of the IPE [Individual Plant Examination] program, but not delay it. We note that many such improvements in methods and data have become available since the closure date for the NUREG-1150 analysis." (Kouts 4.12)

The review committees also provided a number of more specific comments. These are itemized below and staff responses provided.

Comment: The list of initiating events was extensive, and, in most respects, state of the art, but it was not complete. Initiating events not considered included:

- Human errors of commission;
- Incidents starting from low power and shutdown conditions;
- Leaks or breaks of PWR steamlines; and
- Sabotage (understandable in view of methodological and other difficulties involved) (Kouts 3.2.1.1; ACRS).

The effects of aging were not included in the analysis (Kouts 4.12, 7.2).

Response:

The staff acknowledges that human errors of commission have not been included. The treatment of such errors has been the subject of considerable research for several years, but had not sufficiently evolved to permit its use when the NUREG-1150 risk analyses were initiated in 1985. The NRC is currently studying ways in which errors of commission can be practically included in future PRAs (Ref. E.8).

The staff acknowledges that accidents initiated during low power and shutdown operations have not been included in the NUREG-1150 analyses. Recent PRA studies and events in the United States and Europe indicate that the core damage frequency from accidents initiated in such plant operational modes may be significant. The NRC has initiated studies of low power and shutdown accident frequencies and risks for two of the NUREG-1150 plants, Surry and Grand Gulf. Interim, scoping results of these studies are expected in mid-1991. In addition, the NRC has initiated a more general review of non-full-power operational modes to identify the need for additional regulatory requirements. This review is scheduled to be completed in 1991.

Sabotage risks have not been included in the NUREG-1150 risk studies. While the effects of sabotage actions may be similar to that of accidents included in the risk studies, the estimation of the frequencies of such actions is highly uncertain and requires a detailed analysis of the spectrum of threats. Because this threat may be highly variable with time, the staff does not consider it meaningful to attempt to include sabotage risks in PRAs.

The potential for PWR steamline breaks to lead to core damage was assessed (using conservative screening analyses) and determined to be of little significance in the NUREG-1150 PWRs. For some break locations, a steamline break can be similar to a loss of power conversion system transient event and thus can be subsumed into that event. For other break locations, steamline breaks can be recovered through any one of several methods (e.g., feed and bleed cooling, or use of crossties of auxiliary feedwater or emergency core cooling injection from a second unit, if such crossties exist). Using such logic, the core damage potential resulting from such events was judged to be of sufficiently low frequency that it could be screened out early in the analysis. It should be noted, however, that steamline breaks could be important in other PWRs with different plant layouts and system redundancy.

Aging effects were not explicitly included in the analyses. Some consideration of such effects occurs indirectly, however, in that the data base of component and other failures includes failures resulting from aging. The NRC has an extensive program to investigate the impact of aging on plant equipment and to develop and test methods for more explicitly including aging effects in PRA. This work is described in Reference E.9.

Chapter 1 of NUREG-1150 has been updated to better reflect these comments on limitations of the risk analyses.

Comment: The Kouts committee had reservations with respect to the completeness of modeling of interdependencies of technical systems, including detailed modeling of auxiliary systems, formally regarded as not safety-related (Kouts 4.12).

Response:

A major portion of the analysis was devoted to accurately modeling the important auxiliary/support systems, such as component cooling water, and normal and emergency service water. Dependency matrices were developed to identify the dependence of each frontline system on such systems. Connections between safety and nonsafety systems, such as connections to electric power buses, were explicitly considered. Failures of the support systems were also explicitly considered as initiating events. Although most of these events could be ruled out (in initial screening analyses) as initiating events because of train separation, the use of alternative systems, and operator recovery actions, failures of some support systems did contribute to the estimated core damage frequencies (e.g., the component cooling water system failure at Zion and some electric power bus failures).

Comment: The Kouts committee had reservations with respect to uncertainties associated with probabilities mainly based on expert judgment, especially where considerable divergence of opinion existed (Kouts 4.12).

Response:

This comment is discussed in Section E.4.

Comment: The Kouts committee had reservations with respect to the impact of "safety culture" and the fact that the potential effects of management quality are not included (Kouts 4.12).

Response:

This comment is discussed in Section E.5.1.

Comment: Users of the report should be aware of assumptions made in the screening process in which low frequency accident sequences were eliminated from further consideration and that it may not be appropriate to screen out potential sequences in other plants based on the NUREG-1150 studies (ACRS).

Response:

The staff agrees with this comment.

Comment: The frequency of disruptive failure of the reactor pressure vessel was estimated to be between 1E-7 and 1E-6 per reactor year, yet the event was not treated in the analysis. Reviews published in recent years indicate failure probabilities typically in the range of 1E-6 to 1E-9 per reactor year based mainly on probabilistic fracture mechanics considerations. These considerations show a significant influence of plant-specific parameters such as material properties and aging, positioning of welds, and inspection programs. Thus, a somewhat more extensive discussion might have been warranted in NUREG-1150 (Kouts 3.2.1.7).

Response:

A limited screening analysis was performed for NUREG–1150 which indicated that the relative contribution of vessel rupture to core damage frequency would be negligible. For this reason, this issue was not pursued further.

One issue that could have a significant effect on the estimated core damage frequencies of PWRs due to pressure vessel rupture is pressurized thermal shock (PTS). In 1985, the NRC issued new regulations (Ref. E.10), and defined a screening criterion, to limit the potential impact of PTS. Estimates have been made as to when each licensed PWR would reach this screening criterion (Ref. E.11); none of the three PWRs studied in NUREG–1150 is close to reaching this criterion.

Comment: The lack of analysis of external events for three of the plants studied is a deficiency (Kouts 7.2).

The fire analysis in NUREG–1150 was limited to Surry and Peach Bottom. It was generally state of the art but should have been extended to all five plants (Kouts 4.3.3, 7.2).

Response:

The original intent of (what was to become) the NUREG–1150 risk analyses was to provide perspectives on the mid-1980's revisions to source term technology and thus early analyses did not include accidents initiated by external events. In response to comments on the first draft report, the risk analyses of two plants were extended to include external-event analyses. All five plants were not subjected to external-event analyses because of time and budget constraints. The staff concurs, however, with the basic point made that modern PRAs should include consideration of externally initiated accidents.

Comment: Although the two seismic PRAs in NUREG–1150 have been carried through Level 3, these results have not been reported. We believe that these results might provide valuable insights about seismic vulnerabilities of containment systems (ACRS).

Response:

As discussed in Chapter 1, the seismic risk calculations are not described in NUREG–1150 because of certain issues relating to the nonradiological consequences of large earthquakes. While some data are provided in NUREG–1150 with respect to containment performance during seismic events, detailed information is provided in supporting contractor reports (Refs. E.12 and E.13).

Comment: The methods and data used [in the fire analysis] were probably the best available at the time the work was performed. However, certain issues identified more recently may result in increased fire risk estimates (ACRS).

Response:

The staff agrees that the more recently identified issues could be significant. The staff is currently investigating these issues further with respect to their importance to plant safety. As the results of these investigations become clear, the staff will reassess the adequacy of current PRA methods and, if appropriate, initiate work to improve the methods.

Comment: It is not clear as to why loss of instrument air was judged not to be important (Kouts 3.2.1.1).

Response:

The loss of instrument air was examined as a potential initiating event. The plants were examined to determine: if the loss of instrument air resulted in a plant trip and the need for decay heat removal; and the effects of loss of instrument air on accident prevention and mitigation systems. For the plants considered, this event was examined and determined to be of minimal importance. Reasons for this conclusion included plant-specific design features such as separation of air supplies, coupled with the availability of backup systems, and/or that loss of instrument air resulted in plant conditions similar to those of other initiating event groups of higher frequencies, such as a transient with the loss of the power conversion system.

Comment: Recognizing and supporting NRC's desire to publish a final NUREG-1150, we recommend that the report indicate the likely impact of Commonwealth Edison Company's committed modifications on the Zion plant results (Kouts 4.2.2).

Response:

The NRC staff has identified the specific modifications that have now been made to the Zion plant (Ref. E.14). Using this information, sensitivity studies have been performed to assess a revised mean core damage frequency and risk for the Zion plant. Chapters 7, 8, and 12 have been revised to indicate the impact of the modifications made at Zion. More detailed documentation of the sensitivity studies performed is provided in Section 15 of Appendix C.

E.3 Documentation and Display of Results

As discussed in Appendix D, the display of results in the first draft of NUREG-1150 was the subject of considerable controversy. Because of the displays used (and other reasons), the first draft was considered inscrutable. In response, the second draft of NUREG-1150 made significant changes to the displays. Some general comments made by the two review committees on this subject included:

- "[With respect to display of results,] the second draft, reviewed by this [Kouts] committee, followed a more conventional course, showing the probability distributions and the major parameters. This choice responds well to the criticisms of both WASH-1400 and the first draft of NUREG-1150, and the present Committee endorses the decision." (Kouts 4.11.1)

- "The current version does a much better job of presenting the results. A particularly helpful form of the results are the matrix-like figures in which mean values of accident progression bins are combined with mean plant damage states and their frequencies. Pie charts are used effectively to display qualitatively the contributions of various initiating events and accident progression scenarios." (ANS 2.a.12.c)

A related question to the choice of display techniques is the appropriateness of citing and using mean values (vs. median values) to describe uncertain parameters. The NRC-sponsored committee addressed this question and noted the following:

- "There has been much discussion over the matter of preference between use of the mean and the median as a point indicator in such cases. Which is the one that most accurately represents the full distribution? We leap forward to the answer: the preference depends on the precise question being asked. In some applications the mean would be preferred; in others it might be the median. There may be instances in which neither would suffice." (Kouts 4.11.3)

Some other general issues related to documentation were also addressed by the committees. These were:

- The (ANS) committee agrees with the decision not to include the radiological consequences of seismic events (ANS 2.a.9.b).

- The ANS committee agrees with the deletion of the analyses of accident prevention and mitigation features (ANS 2.a.10).

- The Kouts committee notes that the staff presentation of the Peach Bottom ATWS sequence demonstrated good traceability of the methods and data used in the analysis, as did the detailed documentation of the Grand Gulf case (Kouts 4.8.4).

The review committees also provided a number of more specific comments. These are itemized below and staff responses provided.

Comment: Experience shows that neglecting sequences with a frequency about two orders of magnitude below the calculated mean core damage frequency does not noticeably change the overall core damage frequency. Thus, for plants that have a mean core damage frequency of 1E-5 per year, a cutoff frequency of 1E-7 per year seems reasonable (Kouts 4.10.2).

It is reasonable to neglect individual risks that are about one order of magnitude or more below the value associated with the U.S. safety goals. A *de minimis* threshold of 1E-7 per year would appropriately represent this reasoning (Kouts 4.10.3).

Taking into account remaining uncertainties in the PRA methodology, e.g., with respect to completeness in the treatment of human factors and external events, estimated core damage frequencies much below 1E-5 per reactor year should be regarded with some caution (Kouts 4.12).

Response:

The staff basically agrees with the frequency cutoff suggested above. In general, accident sequences identified in NUREG-1150 as having frequencies roughly two orders of magnitude or lower below the accident sequence with the highest mean frequency were eliminated.

The staff also basically agrees with the suggestion of neglecting individual risks at levels one order of magnitude or more below the NRC safety goals. In some circumstances, however, values below such levels have been included in NUREG-1150 to permit comparisons with such risk measures as the frequency of a "large release" goal (see Section 13.2).

Chapter 1 has been modified to discuss the cautionary statements on interpretation of PRA results. Throughout the report, figures and tables have also been modified to indicate these cautions.

Comment: The last six chapters of the second draft of NUREG-1150 are the least effective and most difficult to follow portions of the report. Certain of the material is very worthwhile but much of the discussion seems forced, and the observations range from the obvious to those for which the analysis provides no apparent basis (ANS 2.a.12.d).

Response:

These chapters have been reviewed by the staff and its contractors and updated as appropriate. In addition, Appendix C has been expanded to provide additional discussion of issues important to the results and perspectives provided in Chapters 8 through 13.

Comment: Appendix B provides a valuable example of an accident sequence carried through from accident initiation to offsite consequence estimates. However, the example provided did not include early containment failure; hence many of the more interesting issues that are important to risk are not included in the discussion (ANS 5.e.2).

Response:

An example containing early containment failure was originally considered for Appendix B. The early containment failure example was considered interesting but, however, not typical. That is, a more typical sequence was chosen to avoid giving the wrong impression about the importance of early containment failure to risk at Surry. More detailed discussion of specific risk-important issues is provided separately in Appendix C.

Comment: The purpose of Appendix C was to provide some insight to the resolution of key issues. These discussions are sketchy and the information and reasoning that led to the expert judgments generally not provided. There seems to have been no concerted effort to provide a discussion of those issues that were most important to risk (ANS 5.e.1).

Response:

Appendix C has been reviewed and expanded to address other important issues. However, the information provided is still at a somewhat summary level. The reader seeking more detailed information than that in Appendix C should turn to the extensive issue discussions provided in References E.15 and E.16.

Comment: Recovery actions should be discussed in Chapter 2 and their impact quantified in Chapters 3 to 7 (Kouts 7.3).

Response:

Appendix A has been modified to clarify how recovery actions were treated in the risk studies. Important operator actions (including recovery actions) are addressed in qualitative terms in Chapters 3 through 7 along with other types of failures. The large number of events involved makes it impractical to provide discussion in the summary report. However, more detailed information, including sensitivity studies and importance calculations, is provided in Appendix C and in the plant-specific accident frequency analysis reports (Refs. E.17 through E.21).

Comment: To facilitate a comparison between estimates of offsite consequences in WASH-1400 and NUREG-1150, it is suggested that the final version of NUREG-1150 include comparisons of estimated probabilities of exceeding whole-body or thyroid doses as a function of distance from the site. These data are available from calculations already completed, so no delay in issuance of the report should be caused by incorporating such comparisons (Kouts 5.5; ANS 2.b.3, 2.b.10).

Response:

Although the consequence model used in NUREG-1150, MACCS 1.5 (Ref. E.22), can calculate center-line whole-body and thyroid doses as a function of distance from the site, neither of these specific results was generated and saved in the NUREG-1150 analyses. Thus, this information is not now available for generating dose versus distance plots. Because of the time required to develop such information and transform it into a form directly comparable with the Reactor Safety Study (Ref. E.23), it has not been included in the final version of NUREG-1150 but may be appropriate for study and publication in other forums.

Comment: The contributions of the unavailabilities of safety systems to the total core damage frequency should be displayed (Kouts 7.3).

Response:

The calculation and display of system unavailabilities is most appropriately performed on an accident sequence basis and should account for the operability states of support systems (e.g., the unavailability of the auxiliary feedwater system is different if ac power is or is not available). The staff believes that tabulating a single unavailability contribution (e.g., to core damage frequency) could thus be somewhat misleading and has chosen not to include such information in the final version of NUREG-1150. More detailed tabulations of system unavailabilities, accounting for support system availability, etc., could not be generated in a time period consistent with completion of the final report and thus have also not been included.

Comment: Since the supporting documentation upon which NUREG-1150 depends could be helpful to those performing an individual plant examination (IPE), these reports should be published as soon as feasible (ANS 2.b, ACRS).

Response:

Roughly 80 percent of the contractor reports supporting NUREG-1150 (including methods descriptions, computer code descriptions, and documentation of data and results) have now been published. The present staff and contractor schedules indicate publication of all reports by the end of March 1991.

Comment: In the plant-specific chapters, the substantial differences in the methods used for the Zion plant analysis are not highlighted (ANS 2.a.12.b).

Response:

Chapter 7 has been modified to highlight the differences in methods for the Zion accident frequency analysis.

Comment: The final version of NUREG-1150 should clearly state that it should be viewed as a new study and as a replacement for the first draft (ANS 2.b.6).

Response:

Chapter 1 has been modified to clearly state that the final version of NUREG-1150 is so different from the first draft that the latter should no longer be used.

Comment: The first draft of NUREG-1150 was better in one respect, in that it provided a schematic drawing of the containment and reactor coolant system in each plant-specific section (ANS 5.a).

Response:

Plant schematic diagrams have been added to each of the plant-specific chapters (i.e., Chapters 3 through 7).

Comment: Some presentations of results are so small, or so little contrast provided, that the results are unreadable (ANS 5.b).

Response:

Presentations of results throughout the report have been reviewed and improved where needed.

E.4 Use of Expert Judgment

The use of expert judgment is another issue that was the subject of considerable controversy during the review of the first draft of NUREG-1150. Serious criticisms of the methods used in the first draft to obtain these judgments led the staff and its contractors to implement more formal and rigorous methods. The committees reviewing the second draft had a number of general comments on the use of expert judgment. These included:

- "The formal methods that NUREG-1150 employed for such elicitation and the extensive debates that have ensued constitute a significant advance in PSA methodology, since they force visibility on the use of 'engineering judgment,' which is abundant, yet often hidden, in safety studies. The critical element of the whole process, e.g., the selection of the experts, is now widely recognized and appreciated." (Kouts 4.7)
- "Expert opinion elicitation is technically less satisfying than the use of detailed, validated analytical procedures, or experimental data. Considering the lack of understanding of some phenomena, the uncertainties in the scenarios, and the state of development of many of the analytical procedures, some form of expert opinion was unavoidable, however." (Kouts 4.4) (The committee then continued with a set of more specific comments, some of which are appropriate for staff response. These are discussed below.)
- "It can be hoped that, in the long term, the accumulation of experience will help to narrow the distributions in many inputs and outputs of risk assessments. This is, however, unlikely for many of the important ones, because the objective of safety is specifically to avoid just those events that would generate the data useful for risk analysis." (Kouts 4.11.2)
- "There is a general agreement that the techniques used for eliciting expert opinion in preparation of the second draft were significantly better than those used for the first draft. However, with insufficient information there can be no experts. Thus, use of the term 'expert opinion' in a description of some of the Level 2 work may be misleading. We applaud efforts to improve on the Level 2 treatment of previous PRAs. We nevertheless believe that the results from Level 2 presented in this latest [second] draft must be regarded as having major uncertainties in both calculated mean values and in estimated uncertainties." (ACRS)

More specific comments by the review committees are itemized below and staff responses provided.

Comment: Formal, professionally structured expert opinion is preferable to the current alternative, according to which the individual PRA analysts make informal judgments that are not always well documented. However, it is not as technically defensible as analysis using detailed, validated codes. The reproducibility of expert opinion results is a concern (Kouts 4.4).

Response:

The staff agrees that a PRA will be improved by having as many robust calculations as possible. However, it should be noted that it will also never be possible to remove expert judgment from a PRA. A PRA is a procedure for assembling information from many sources, including experimental data, theoretical calculations, and mechanistic code calculations, some of which are conflicting and incomplete. The process of obtaining expert opinions such as used in NUREG-1150 provides a way to review this information and put it in a form that is suitable for use in a PRA. The outcome of this process will always be improved by better information, including calculations by detailed, validated codes. However, some type of expert judgment is always associated with the use of code calculations for several reasons. First, a code calculation is performed for a very specific accident, but the results of this calculation are used in a PRA for groups of "similar" accidents. This type of aggregation requires judgment since the performance of a calculation for every possible accident is not feasible. Second, it is not possible to fully "validate" the mechanistic codes that are used in reactor accident calculations. Thus, there is always a judgment that must be made with respect to the acceptability of a code calculation for a specific application. Third, judgments with respect to model formulation and model parameters must be made to use a code. Thus, the opinion of this "expert" will always enter into the calculations and results.

In the NUREG-1150 uses of expert judgments, two factors acted to reduce the potential impact of this concern: the information being obtained from experts was in the form of probability distributions rather than single or best estimates; and, for key issues, a diversity of judgments was sought. Nonetheless, the staff agrees that the reproducibility of expert judgments can be of concern and expects to support research in this area in the future.

Comment: There is always a question as to "who is an expert on a given issue." The membership of expert panels for the second draft of NUREG-1150 seemed to be better than for the first draft. Yet it still seemed to be unbalanced in that panels still contained more analysts and fewer persons with practical engineering experience who might have expertise on the phenomena; the panels included more users and fewer generators of data than is preferable (Kouts 4.4, 7.2; ACRS).

Response:

The method used to select the members of the expert panels for the NUREG-1150 risk analyses is discussed in Reference E.24. As described there, one goal was to select experts with a diversity of backgrounds. However, experts familiar with reactor safety were usually selected for practical purposes. That is, the project schedule did not permit the time, in general, to educate experts in very specialized areas in the more general area of reactor safety. Two experts on specific phenomena with no familiarity with reactor safety analysis were selected: one on the source term panel and one on the containment loadings panel. One of the experts felt uncomfortable extrapolating his knowledge to reactor accident sequences and declined to continue participation. The second expert went through the effort to educate himself on reactor risk and provided valuable input.

Comment: Expert opinion may have been relied upon too heavily in some instances. An important example is the treatment of core cooling after containment failure. In this case, expert opinion was used to argue that equipment would fail 70-80 percent of the time if environmental temperatures exceeded equipment qualification limits. No explicit analysis was performed to determine the impact of local environmental conditions on equipment heatup and the potential for subsequent failure. It may have been thought that the analysis would have been too time-consuming. It would have been appropriate if possible to have developed these analyses and then to have subjected them to critical review to which expert opinion could have been directed (Kouts 4.4).

Response:

The staff and its contractors did obtain additional information and perform extensive analyses to eliminate the need for or support expert judgments and to supplement the information available in the literature. For the specific issue cited, the experts did receive, for example, information on equipment tolerances and lubricant breakdown temperatures. More generally, many calculations were commissioned specifically for the NUREG-1150 study and presented to the expert panels for review. Some examples of code calculations commissioned include those performed with CONTAIN, CORCON, the Source Term Code

Package, MELCOR, and MELPROG. Such calculations were performed for specific issues and are described in Reference E.16.

Comment: There are some subjects for which the expert opinions were either incomplete or were not targeted on the correct issue because definition of the issue evolved subsequent to the elicitation process and resources were lacking to update it. In these cases, the Sandia staff modified the expert opinion in order to treat the redefined issue. Unfortunately, these new calculations were not reviewed with the expert panel and are not reported in the NUREG-1150 main report or other documentation available to the Kouts committee (Kouts 4.4).

Response:

There were issues in which the responses of the experts were used in a slightly different context than was originally intended. There were two reasons for this:

- The experts had different perceptions of the question asked of them; thus, the information was received from the individual experts in different formats. To aggregate these issues, it was necessary to extrapolate and interpolate some of the expert responses.

- The definition of the issue sometimes evolved subsequent to the elicitation process. In some cases, the issue was much more complex than was anticipated at the time of the elicitation; an example is the treatment of multiple containment failure modes during fast pressure rises. In these cases, the information from the expert panels was reformatted or extrapolated in order to aggregate the response.

In all cases, the original elicitation notes for the accident progression issues and the source term issues have been documented (after review by the experts) in Reference E.16. Any manipulations that were performed on the expert elicitation are described in a section that preceded the individual expert issue documentation, entitled "Method of Aggregation." In virtually all cases, the manipulations were discussed with the experts prior to its use to ensure that the information was not misused.

Comment: The study assigned equal weight factors to the opinions of all experts. Other methods that can develop unequal weight factors were not used (Kouts 4.4).

Response:

The staff and its contractors considered a variety of methods of combining expert judgments, including methods using unequal weighting factors. As noted in Appendix A, the method of equal weighting was chosen because this simple method has been found in many studies (e.g., Ref. E.23) to perform the best.

Comment: The ACRS was told that the budget for the study provided only enough funding to support the participation of about 20 percent of the experts who served on the panels. The remainder were drawn from the NRC staff or from organizations with contractual relationships to the NRC. This biased the selection toward people whose organizations depend upon the NRC for support (ACRS).

Response:

Roughly 30 percent of the experts were funded directly by the NUREG-1150 study. However, the remainder of the experts were supported by two groups: the NRC and the nuclear industry (e.g., EPRI). Overall, approximately 30 percent of the experts were supported directly by the NUREG-1150 study, 45 percent by other NRC projects, and 25 percent by the nuclear industry.

Comment: The expert opinion procedure is complex, time-consuming, and expensive. Therefore, the full scope of the methodology may have very limited future application. It is unlikely that an expert opinion procedure of this magnitude will be repeated for several years, although expert elicitation on single or narrow issues may be practical. However, it should be remembered that throughout the study analysts had to decide how to use technical information of all kinds; this "expert judgment" is necessary in all PRAs (Kouts 4.4; ACRS).

Response:

The staff agrees that the expert judgment methods used in NUREG–1150 may have limited utility in future work because of the time and cost involved. The staff intends to pursue research in this area with the intent of making the formal uses of expert judgments and the performance of quantitative uncertainty analyses more practical.

Comment: The discussion of issue quantification could be substantially improved, with much clearer indication of what probability distributions were developed by the staff and which specific issues were quantified by the expert review panels (Kouts 7.3; ANS 2.a.8.a).

Response:

The staff agrees with this comment. A table indicating what variables were included in the uncertainty study for the Surry plant (Ref. E.12), and how they were quantified (by expert panels, by NUREG–1150 staff, or by user function), is provided as an example in Section C.1 of Appendix C. Similar tables for the other four plants are provided in References E.13 and E.26 through E.28.

E.5 Accident Frequency Analysis

The review committees and the ACRS had a number of general and specific comments on the accident frequency analysis performed in the NUREG–1150 project. These comments are itemized below, beginning with the subject receiving the most comment, human reliability analysis, discussed in Section E.5.1. Section E.5.2 then provides a discussion of comments on external-event analysis, and Section E.5.3 provides a discussion of other comments on the accident frequency analysis.

E.5.1 Human Reliability Analysis

The Kouts committee provided considerable comment on the subject of human reliability analysis (HRA). As a general comment, the committee noted that:

- "Given the current state of the art in HRA, it would be unreasonable to expect NUREG–1150 to resolve all the outstanding issues including use of a universally accepted model." (Kouts 4.8.2)

The ACRS also provided a general comment on this subject:

- "As other reviewers have reported, there are recognized deficiencies in the state-of-the-art treatments of human performance; and this report is not free of these deficiencies."

In addition, a number of specific comments were provided. These are itemized below along with staff responses.

Comment: NUREG–1150 pioneered the explicit treatment of model uncertainties and the use of expert panels to weigh the relative merits of alternative methods of analysis, but such an approach was not been employed for human actions such as errors of commission and complex situations in control rooms such as in the early phases of an BWR ATWS accident (Kouts 4.8.2).

Response:

The staff agrees that the human reliability analysis should have been performed in a manner more consistent with the remainder of the risk analyses.

Human reliability analysis has been the subject of extensive research in the past few years and has led to the development and initial application of techniques to deal with such issues as human errors of commission. NRC continues to perform a substantial amount of research in HRA, as described in Reference E.8. The demonstration and more widespread use of improved HRA methods in PRA is planned to be the subject of future work by NRC.

Comment: It would have been valuable if the theoretical HRAs of the ATWS sequences had been tested against analysis of real events as a basis for an in-depth analysis of uncertainties in HRA. This could be done as part of expert opinion input on the merits of different HRA models. Such an approach to the

ATWS HRA appears more appropriate and consistent with the use of expert panels in the remainder of NUREG-1150 (Kouts 4.8.4, 7.2).

Response:

The validation of human reliability models (by comparisons with actual events, simulator exercises, etc.) is an integral part of the present NRC program in HRA (Ref. E.8). Future NRC PRA work will make use of such models and thus should provide a better assessment of human performance and its importance to risk.

Comment: For NUREG-1150, the argument was advanced that the conservative screening procedures that have been employed and the wide uncertainty ranges that have been assigned to human error rates have the effect of including the results that other models would have generated. However, such an approach goes against the presumed goal of a PRA, namely, the realistic estimation of risks. Furthermore, the use of an error factor does not necessarily cover the possibility that the models systematically overestimate or underestimate the human error rates (Kouts 4.8.2).

Response:

Conservative screening values were used in the initial quantification of human error probabilities. However, for those events that were potentially significant contributors to core damage frequency, more detailed analyses were performed (this approach being designed to expend significant resources only on those events that are most important). Different types of probability distributions, such as maximum entropy or lognormal, were assigned as appropriate. It is possible that the mean values produced in the analyses could be systematically high or low because of various types of systematic errors. However, the uncertainty analysis did account for these errors in the sense that many of the human error uncertainty distributions were correlated. That is, when a value near the high end of the distribution was chosen for one variable, then a value near the high end of the distribution was chosen for all similar human errors. Thus, the variability did account somewhat for systematic errors. This approach, coupled with the fact that very wide uncertainty distributions were applied to these variables, leads the staff to believe that the treatment of human error uncertainties was adequate for the types of actions included within the scope of the study, recognizing the state of technology of HRA at the time when the work was performed. As noted above, the NRC is currently funding considerable research in the area of HRA (Ref. E.8).

Comment: Considering the different Grand Gulf and Peach Bottom analyses of operator failure to initiate the standby liquid control system during an ATWS event, it is unclear to what extent the differences in estimated probabilities is due to the different methods employed and to the different groups of analysts that have implemented them. It may be questioned if the relatively simple methods used are the most appropriate for very complex, high-stress situations (Kouts 4.8.4).

Response:

The HRA methods used for the Grand Gulf and Peach Bottom ATWS analyses included a detailed task analysis, using the THERP method (Ref. E.29) for Grand Gulf and the SLIM-MAUD method (Ref. E.30) for Peach Bottom. The staff acknowledges that use of different methods and analysts can have an impact on the results obtained and that the impact on the two plant ATWS studies of these differences cannot be easily estimated.

While the use of different analysts can influence the results, it should be recognized that plant design differences were found to be important in NUREG-1150. With respect to ATWS accident sequences in Grand Gulf and Peach Bottom, several such important design differences exist. For example, the standby liquid control system in Grand Gulf is designed to inject boron via the high-pressure core spray sparger, while in Peach Bottom boron is injected into the bottom of the reactor vessel. This difference leads to differences in timing of ATWS events and the procedures established by the plants (operator actions to lower and raise water levels required at Peach Bottom are not needed in Grand Gulf).

Comment: It is beyond the capabilities of present PRA models to account for the influence of management quality on risk; thus it is understandable that NUREG-1150 does not address these issues. While management quality may not be quantifiable in PRA in the near future, its impact on safety is currently being addressed through other NRC and INPO [Institute of Nuclear Power Operations] work. It is important to bear in mind that management quality is not reflected in the risk information as results and perspectives are used (Kouts 4.9).

Response:

Such influences have not been included in NUREG-1150 (or in any other PRA). The present NRC human factors research program (Ref. E.8) includes the study of organizational and management influences on plant safety, including consideration of how such influences can be accounted for in risk studies such as NUREG-1150. Completion of this research should provide some perspective on the degree to which these influences can be incorporated.

Comment: The inclusion of some recovery actions was state of the art. However, the assumptions behind actual recovery curves are not always clear (Kouts 3.2.1.7).

Response:

The recovery analysis included an evaluation of both the time available for recovery and the probability of the operator correctly performing the task. For some faults, actual historical data exist. For example, data exist for all electrical-type faults (i.e., offsite power and diesel generator faults) and faults associated with the power conversion system. For other type faults, historical data did not exist. This recovery information is documented in Reference E.31. For these situations, an HRA or recovery analysis was performed to determine the probability of failure to recover. These recovery curves and "generic" human behavior curves are obtained directly from use of the THERP method (Ref. E.29).

Comment: Innovative recovery actions not covered by operating or emergency procedures should not be included in the baseline analysis, but should be reserved for potential reductions in risk (Kouts 3.2.1.7).

Response:

For some of the accidents analyzed in NUREG-1150, several hours pass before the onset of core damage. In severe accidents of such time duration, an emergency response team would be involved to support the operating crew. It would, therefore, be unrealistic not to allow any innovative recovery actions, considering that such options would be under active investigation and consideration. For these reasons, and recognizing the goal of performing realistic analyses in PRA, credit for innovative recovery in such accidents was permitted in the NUREG-1150 analyses.

It should be noted that, while permitted, very few innovative actions were ultimately incorporated into the analyses. Although several innovative recovery actions were proposed, some of these were incorporated into plant procedures (by the licensee), while others were found to be unnecessary for further analysis because of the already low estimated frequency of the associated accident sequences or the low probability of success.

Comment: Special attention should be given to further development of human reliability analysis, and to proper calibration of the procedures used for it, to enable comparisons to be made between plants and quantitative safety goals (Kouts 4.12, 7.3).

Response:

As discussed in the responses to a number of the previous comments, the NRC has a significant research program under way in the area of human reliability analysis (Ref. E.8).

E.5.2 External-Event Analysis

Specific Comments

Comment: A simplified approach was taken in NUREG-1150 in defining seismic initiators, which leads to failure from all resulting transients, small or large (Kouts 4.3.2).

Response:

All seismically induced transients were not assumed to result in "failure." It is assumed in the analysis that an earthquake will lead to at least one initiator that will require the plant to shut down (either automatically or as a result of operator action).

The occurrence of any of these initiating events, however, does not necessarily imply that core damage will occur. Given that such an initiating event occurs, the same (event tree and fault tree) process is used to assess the conditional probability of core damage as was performed for the internal-event analyses. System failure probabilities may be higher because of earthquake-induced damage, but they are not assumed to be of unity probability.

Comment: Although plant experience was used to establish fire initiation frequencies, judgmental factors were used to determine whether a fire, once started, would persist and cause damage in spite of fire mitigation systems and actions. It would seem that the same data base that was used for fire initiation could and should have been used to give a more realistic value for fire persistence (ANS 3.f.3.a).

Response:

Credit was taken for fire mitigation systems, both manual and automatic, in each fire scenario where applicable. In the case of manual suppression, the same fire data base used to develop the fire initiation frequencies was also used to develop a probability of suppression in any given time frame. For automatic suppression systems, several other studies were used to determine reliability values, as these could not be determined directly from the fire occurrence data base. The data indicated that, for fires in critical locations, the fire was always eventually suppressed (either automatically or manually) but seldom before damage to critical equipment would be predicted to occur (using the COMPBRN model of fire propagation (Ref. E.32)).

Comment: Research in seismic modeling is warranted with the objective of improving the basic model for prediction of attenuation and ground motion and for developing a consensus of the use of one model or model set based as much as practicable on region-specific spectral shapes. Effort should also be made to improve the basic model to reflect greater source depths and regional variations with the appropriate reflections of substrata waves (Kouts 7.3).

Response:

NRC and others continue to sponsor research to improve the general understanding of seismic hazard, including the areas noted above. Such work is described in Reference E.33.

E.5.3 Other Accident Frequency Comments

In addition to the comments itemized above on human reliability and external-event analyses, the committees had a number of other comments on the NUREG-1150 accident frequency analysis. Some general comments provided by the committees included:

- "[The plant damage state analysis] was more detailed than the corresponding analysis in other recent PSA's. It provided an efficient interface with the detailed and complex accident progression and containment loads analysis, and constitutes an advance in PSA methodology." (Kouts 3.2.1.8)
- "In respect to including the modes of containment failure, and in the level of detail, the [accident sequence event tree] analysis was advanced other than typically seen in Level 1 PSA's performed at the time of the NUREG-1150 analysis." (Kouts 3.2.1.2)
- "Although NUREG-1150 is described as being 'a set of modern PRAs, having the limitations of all such studies,' the level of modeling in the accident frequency analysis is not as detailed in some areas as that found in other current PRAs." (ANS 2.a.8.b)
- "A rigorous analysis would always combine the generic and plant-specific [failure data] information. In fact, this is often done using Bayes' theorem. However, we note that in general the numerical differences between the approximate methods of NUREG-1150 and the rigorous approach are insignificant." (Kouts 3.2.1.6)

More specific comments made by the committees are itemized below and staff responses provided.

Comment: Since the first draft was issued, considerable effort was devoted to making the accident frequency analysis more robust. However, the NRC staff recognizes that the state of the art with respect to common-cause failures and human reliability analysis is imperfect and that further improvements can be made in these crucial areas. These areas have not been treated as top-level issues in the expert elicitation process (Kouts 6.3).

Response:

Common-cause failures and selected human error probabilities were offered to the accident frequency analysis panel as issues. The panel concluded that the approach being taken by the analysis team for common-cause failures was appropriate and that expert judgment would not significantly improve the process. Human errors could not be readily considered as a single issue because each action being considered was unique, requiring a separate analysis. The panel did consider several specific human error issues considered to be particularly important. In addition, sensitivity studies on the importance of human error were performed, as discussed in Appendix C.

Comment: The consideration of operating experience in the so-called subtle interactions, represents a good attempt to ensure completeness of failure modes. The method of treatment of dependent failures was state of the art in most respects. The documentation of common-cause failure analysis is difficult to follow. For example, in some instances references were made to EPRI common-cause methods and data, but it appears that in reality a modified beta-factor method was used, which was itself state of the art. The probability of failure of all station batteries is critical to the final results and therefore necessitates better substantiation. Recovery from common-cause failure was restricted to selected electrical equipment (Kouts 3.2.1.4).

Response:

Common-cause failures are discussed in Appendix C and in Reference E.31. The common-cause analysis used in the NUREG-1150 analyses was based primarily on EPRI methods and data. EPRI generic component beta factors were used in the calculation of the common-cause failure (CCF) rates. The CCF rates were calculated as follows:

$$CCF = Q * \beta_n$$

where

Q = Total failure rate

β_n = Beta factor for n components.

For some components, there was not a generic component beta factor for the number of components modeled. In these cases, the EPRI beta factor was modified. In addition, for some components (e.g., batteries, air-operated valves), there were no EPRI generic component beta factors. For these components, other sources or methods were used to calculate the beta factor.

Common-cause failures of the batteries were analyzed in detail in other studies (Ref. E.34) and were used in the NUREG-1150 analysis. Recovery credit for common-cause failures was included where data existed.

Comment: In the analysis of loss of feedwater initiating events, it was assumed that condensate would also be lost, thereby eliminating a potential source of injection capability. For such an initiating event, the recovery potential may be underestimated because of this assumption (Kouts 3.2.1.1).

Response:

The loss of feedwater (LOFW) was treated on a plant-specific basis. For Grand Gulf, upon examination of LOFW, it was determined that condensate would not be lost. For Peach Bottom, it was assumed that condensate was lost with LOFW; however, credit was given for the recovery of the power conversion system, which included recovery of condensate. For PWRs, loss of condensate was included as one of the contributors to LOFW. However, because the LOFW initiating event was not an important contributor to the estimated core damage frequency, no credit for recovery of condensate was considered necessary nor given.

Comment: In general, it appears that very little plant-specific thermal-hydraulic analysis was conducted. Instead, the analysts relied on the results of generic analyses and made judgments as to the degree of applicability in many scenarios (Kouts 3.2.1.2).

Response:

When necessary, plant-specific thermal-hydraulic calculations were performed (e.g., BWR ATWS sequences, ice condenser containment spray actuation timing, and boiloff calculations). Additional thermal-hydraulic calculations were not deemed necessary because a large library of calculations already existed, including those from NRC research programs, vendor analyses, and other industry programs. In addition, actual plant experience was used. For example, the thermal-hydraulic response to a steam generator tube rupture was based in part upon the data from the North Anna tube rupture incident.

Comment: Some success criteria may be too conservative, e.g., both PORVs are assumed to be required for feed and bleed in PWRs (Kouts 3.2.1.2).

Response:

As much as possible, success criteria were developed to be realistic, as opposed to conservative. For example, low-pressure systems were allowed to lead to success in BWR ATWS sequences, including loss of the standby liquid control system, whereas previous studies might not have considered that possibility. In some cases, the success of a particular system was questionable based on information available in the time frame of the study. In these cases, conservative choices were made. Plant procedures (e.g., those that called for both PORVs to be opened in the case of feed and bleed cooling) were also influential in the decisions made in such cases.

Comment: The Grand Gulf ATWS analysis included the two event tree branches of early and late closure of the main steam isolation valves. In the Peach Bottom ATWS analysis, it was, probably conservatively, assumed that the main steam isolation valves (MSIVs) closed for all scenarios. We have found no justification for this difference based on design data or plant operating experience (Kouts 3.2.1.3).

Response:

A plant-specific analysis of MSIV response during ATWS was performed for both Grand Gulf and Peach Bottom. It was not assumed in the Peach Bottom ATWS analysis that all scenarios resulted in MSIV closure. Based on a detailed analysis, it was concluded that all ATWS sequences (with MSIVs open) would lead to isolation signals to the MSIVs.

Comment: Electrical control and actuation circuits were not included in the common-cause failure analysis (Kouts 3.2.1.4).

Response:

Electrical control and actuation circuits faults were included as part of the component random failure rate. The same applies for the common-cause failures. The faults comprising the common-cause failures for components (i.e., valves, pumps, diesels, etc.) were dominated by electrical control and actuation circuit faults.

Comment: Expert judgments assign large uncertainty to the issue of reactor coolant pump seal failure, which is actually susceptible to experimental determination. It is not readily apparent how the bimodal distribution of NUREG–1150 would be affected by the revised estimates of leakage rates and times for initiation of leakage (Kouts 4.6.2).

More recent information and the development of some new reactor coolant pump seal designs since the NUREG–1150 risk studies were completed would lead to a prediction of risk less than that reported (ACRS).

Response:

The expert judgment process was intended to characterize the current understanding of the issue rather than provide resolution. The information base used by the experts included data from experimental programs by Westinghouse, by NRC, and in France. (Appendix C now includes a section of how this issue was addressed; a detailed description is provided in Ref. E.15.)

It should also be noted that the expert judgment process considered the potential importance of the new Westinghouse seal design (not yet in place in the plants analyzed). The experts concluded that seal failure with the new seals would be very unlikely. This would have two effects on the NUREG-1150 analyses. First, the core damage frequency would be reduced because more recovery time would be available prior to core damage. However, for those accident sequences that continue on to core damage, the core damage may occur with the reactor coolant system at high pressure, leading to high containment building loadings at time of reactor vessel breach.

Comment: It is likely that the performance of relief valves, which must function if the feed and bleed operation is to be successful, is not well represented by the data for valve performance used in the NUREG-1150 calculations (ACRS).

Response:

The staff agrees that there is now operating experience data that suggest that the PORV failure rates are optimistic. However, since failure of the feed and bleed function is assessed to be dominated by human errors (to actuate the system), it does not appear that increased failure rates for the PORVs would significantly affect the likelihood of failure to feed and bleed.

Comment: There is now a significant body of evidence to indicate that the failure probability used to describe the operation of certain key motor-operated valves is too low. This may have an important bearing on the outcome of several accident sequences described in the report (ACRS).

Response:

The staff agrees that there is now evidence that motor-operated valve failure rates are, under some conditions, higher than those used in NUREG-1150. The NUREG-1150 analyses have not been reevaluated in detail to assess the potential impact of the newer failure rates. It is the staff's judgment that, while the impact would be noticeable, it would not be dominating.

Comment: Plant-specific information is becoming increasingly important in PRA; such information should be collected and placed on file for future use (Kouts 7.3).

Response:

The NRC has developed a data base for the accident frequency analysis models developed in NUREG-1150 (and for other PRAs as well). This data base can be accessed via two computer codes, SARA and IRRAS (Refs. E.35 and E.36), which permit the manipulation of the data for sensitivity analyses, etc. These codes and the data base have been installed and are seeing use in several locations at NRC (and its contractors).

In 1990, the NRC initiated work to assess the feasibility of developing a similar data base and acquisition/analysis system for the accident progression, source term, and risk analysis models of NUREG-1150. This system would make use of data generated with the detailed NUREG-1150 codes, such as EVNTRE (Ref. E.37) and PRAMIS (Ref. E.38).

Comment: The NUREG-1150 documentation does not allow a reviewer to determine how particular events contributed to the frequency of loss of offsite power and subsequent recovery (Kouts 3.2.1.1).

Response:

As noted in the report, NUREG-1150 provides a summary of the methods and results of the five PRAs performed. More detailed information is contained in the underlying contractor reports (Refs. E.15 through E.21, E.12, E.13, and E.26 through E.28). Even these, however, do not contain some of the raw data used to develop and quantify the risk models. Such data are retained in the project files. Included in these files are the data on specific losses of offsite power and its recovery. These data included all events at U.S. nuclear power plants through 1987. These included plant-centered, grid, and weather-related faults. Particular events that could not occur at a particular site were eliminated from the data base for that plant. Further, the analysis considered the operating history at each plant. Plant-specific recovery curves were then generated based on an aggregate of all loss of offsite power events, as opposed to separate recovery curves for each type of failure event.

E.6 Accident Progression Analysis

The review committees had a number of specific comments on the NUREG–1150 accident progression analysis, the most important of which appears to relate to the level of detail in the analysis, compared with the detailed accident phenomenogical computer codes and with the present level of understanding of accident phenomenology. These specific comments are itemized below and staff responses provided. Comments dealing with the closely related subject of accident source term methods are discussed in Section E.7.

Comment: The level of detail in the accident progression analysis appears to have exceeded the understanding of the phenomena involved. It implied greater insight into the processes assumed to be taking place than was justified (Kouts 3.2.2.1, 7.2; ACRS).

If phenomenological models are not provided and directly used, the dependence of the results of the accident progression analysis on governing physical phenomena is hidden (Kouts 3.2.2.1, 7.2).

The generality of the structure of trees and the flexibility to use different levels of modeling capability and details to answer the questions at branch points make the method very powerful, but concern can arise about the meaningfulness of computed results if little information is available about the issues. The possibility of introducing high-level issues makes the method efficient, but this feature should be used with caution if applied to issues with a weak information basis (Kouts 3.2.2.1, 7.2).

We note that in the back end subjective distributions are given for high-level parameters ("issues") that describe the outcomes of complex physical or chemical processes whose basic uncertainties are at lower levels. Mechanistic computational models that would relate these lower-level parameters to the higher-level issues are not employed (for example, the amount of core debris involved in ex-vessel steam explosion is an issue, and its dependence on such lower-level parameters as heat generation rates and chemical reaction rates is not modeled explicitly). Developing subjective probability distributions for high-level parameters may not always be the best approach since the physics of the underlying processes does not get the attention that would be desirable (Kouts 4.7).

Response:

The comments first question whether the detail exceeds the state of knowledge. The staff does not believe so. The intended use of a study to some extent defines the appropriate level of detail. The level of detail was chosen to pass the appropriate information on to the source term analysis and to allow the variation of parameters in the integrated uncertainty analysis. In order to meet these two objectives, it was necessary to form the probabilistic models with high-level issues. Uncertain responses to the high-level issues resulted in wide uncertainty distributions. The use of wide uncertainty distributions to characterize processes that are not well understood should not imply greater insight into the process than is justified but should highlight the uncertainty of that process.

The information presented in NUREG–1150 provides insight into the importance of the high-level parameters and not the governing physical phenomena (e.g., chemical reaction rates). Also, the accident progression event trees used to model the accident progression are based on these high-level parameters. To evaluate the branch point probabilities, however, the high-level parameters are decomposed to the level of the governing physical phenomena (as documented in Ref. E.16).

Because of the complexity of the accident progression, it would have been computationally impossible to model the accident framework for each accident sequence at the physical process level (heat transfer correlations, oxidation rates, etc.). To obtain the insights necessary on the underlying physical processes, it is necessary to establish what high-level parameters are important and then refer to the copious documentation provided on parameter distribution development in Reference E.16.

The staff agrees with the comment that the user should interpret the results of the study carefully when there is a weak information base associated with high-level issues. The NUREG-1150 approach was to include these issues in our models and apply appropriate uncertainty bounds to the parameter distributions.

Comment: There is inconsistency in the detail of the accident progression analysis. This is in part because the state of knowledge with respect to severe accident phenomenology in BWRs versus PWRs is different, the use of expert elicitation for severe accident issues was not the same for all plants, and there was a large uncertainty in operator behavior with respect to post-core-damage recovery actions (Kouts 6.3).

Response:

The general process for performing the accident progression analysis was consistent between PWRs and BWRs. The BWR accident progression event trees (APETs) tended to be larger and more complex because there are more interactions between the containment and the reactor coolant system in BWRs and because the failure location in BWR containments can have a large impact on release fractions. The quality of the information available for input parameter distribution varied for the different issues because of the different amount of experimental and analytical studies performed, but was not clearly superior for either BWRs or PWRs.

There are some issues that have been studied more extensively for PWRs (in-vessel melt progression, direct containment heating). This may have resulted in some inconsistency in the quality of the response on some issues, but the selection criteria for the expert elicitation issues were applied consistently to all plants analyzed.

Comment: The bin "no vessel breach" has a relatively high conditional probability for all plant damage states of PWRs. Yet, the capability to model the issue of core degradation before vessel breach is rather poor. We are unable at present to judge the validity of the conditional probabilities associated with this accident progression bin (Kouts 4.6.1).

Response:

The staff agrees that there are considerable uncertainties associated with this issue. However, it is felt that the approaches used in this study adequately represent the knowledge base as it pertains to this issue. The approach used in the NUREG-1150 analyses is described in Part 6 of Reference E.16.

Comment: Only one of the three experts whose opinions were elicited provided a distribution function for temperature-induced hot leg failure. The other two made the statements "...if necessary conditions for high temperature were met, the leg would always fail...," and "...if high temperatures lasted long enough hot leg would always fail. For shorter time at high temperature hot leg would sometimes fail..."

Since the crucial point in the analysis is the estimation of the hot leg temperature, we cannot see how the two cited statements were incorporated into the aggregated probability distribution presented in NUREG-1150. Therefore, we are unable to judge the validity of the result (Kouts 4.6.3).

Response:

The three experts that considered the temperature-induced hot leg failure all addressed the estimation of the hot leg temperature in their assessments. Two of the experts' decompositions of the issue established continuous distributions for failure probability, the other decomposition provided a point estimate. Each decomposition of the issue was different, yet all addressed hot leg temperatures.

There were many cases in which the distributions (and the associated rationales) provided by experts on the same issue differed significantly. For example, one expert might have felt that the uncertainty in an issue was primarily stochastic in nature while another expert might have felt that the uncertainty was entirely the result of the lack of understanding of the physical process. The method of aggregating distributions (described in Ref. E.16) accommodated different perceptions of the results. Further information on the specific expert analysis of temperature-induced hot leg failure may be found in Part 1 of Reference E.16.

Comment: The treatment of the pressure rise at vessel breach as a single issue by the expert panel obscured a more complete understanding of how the various components contributed to the reduced probability of early containment failure (Kouts 4.6.4).

Response:

The containment loads expert panel felt that tightly coupled phenomena were responsible for the loads that accompany vessel failure. Furthermore, the experts felt that there were synergistic relationships among the various phenomena. Thus, because a simple relationship that ties together the various phenomena involved did not exist, the expert panel did not believe that these phenomena could be isolated without sacrificing the credibility of the final distribution (i.e., the load experienced by the containment). It was their opinion that artificially breaking apart the loads would not provide a realistic picture of the events that are taking place.

The phenomena that contribute to the loads at vessel breach and the importance of the various phenomena for a given distribution are discussed in Reference E.16. From the descriptions of the experts' rationales, the importance of various events to the loads at vessel breach can be obtained. Discussion of the reasons for which these loads are less important now than in the first draft of NUREG–1150 is provided in Section C.5 of Appendix C.

Comment: We note that the concrete erosion progresses faster and with greater intensity than is estimated in NUREG–1150, with a corresponding influence on hydrogen production. However, we agree with the assessment in NUREG–1150 that the meltthrough per se introduces no important influence on health risk (Kouts 4.6.5).

For reasons explained in the section on basemat meltthrough, we believe that this process (MCCI) [molten core-concrete interactions] is modeled incorrectly, with the consequence that the hydrogen generation rate in the ex-vessel phase of accidents in PWRs would be underestimated (Kouts 4.6.6).

Response:

The hydrogen generation rate during core-concrete interactions was based on calculations with the CORCON computer code, as discussed in Reference E.16. The amount of hydrogen produced in the core-concrete interaction phase is dependent on how much unoxidized metal is available, which in turn is dependent on how much has been oxidized in prior phases. From the CORCON calculations, it appears that most of the unoxidized metals remaining in the debris as core-concrete interactions begin are oxidized rapidly. The staff therefore believes that most release rates predicted by current techniques and considering experimental evidence are bounded by the range of release rates in NUREG–1150, when considering in-vessel hydrogen production rate, the at-vessel-breach hydrogen release rate, and the early core-concrete interaction hydrogen production rate.

Because early containment failures and containment bypass accidents tended to dominate the risk, and because there was already so much hydrogen in the containment at the beginning of core-concrete interaction, the amount of hydrogen produced during core-concrete interaction was not considered to be highly risk significant and thus was not varied in the overall uncertainty analysis. As such, while an estimate of hydrogen production based on CORCON calculations may not agree closely with all estimates from experiments such as those performed in the BETA facility, such differences are not believed to be important to the overall risk estimates.

Comment: A separate accident progression bin should be used for basemat meltthrough because knowledge of the consequences of this form of release, though not important from the standpoint of damage to the public, is useful for other purposes (Kouts 3.2.2.3).

Response:

The accident progression results that are shown in NUREG–1150 are summary accident progression bins, grouped together for presentation purposes only. It is possible to separate the basemat meltthrough event from the long-term containment failure event when presenting results of the detailed accident progression analysis. However, in the source term analysis, the two events were binned together. Thus it is not possible to extract separate source term and consequence results for the basemat meltthrough and late containment failure events without performing new calculations. This was not done in the NUREG-1150 analyses because of the low estimated risk significance of both of these failure mechanisms.

Comment: The lack of information about many of the physical phenomena that determine the performance of a containment system in a severe accident situation is such that only educated guesses can be made for some sequences that might make significant contributions to risk. Some important phenomenological issues (e.g., direct containment heating, Mark I shell meltthrough) were characterized quite differently in the first and second drafts even though there was not a major change in the information base. Further, no consideration was found on the impact of ex-vessel steam explosions on early containment failure. There is little unambiguous guidance here for a licensee performing an IPE (ACRS).

Response:

While the staff believes that significant progress has been made in the understanding of severe accident phenomenology, it also agrees that there is a need for more information on a number of specific issues, such as those highlighted by the ACRS. The staff recognized this in developing the guidance for licensee individual plant examinations (IPEs). Appendix 1 to the IPE generic letter (Ref. E.39) provides guidance on how licensees should deal with this lack of information.

It is correct that a number of phenomenological issues were characterized quite differently in the first and second draft versions of NUREG-1150. This reflects a greater information base on a number of important issues such as direct containment heating. The technical bases used by expert panels to assess such issues are discussed in considerable detail in Reference E.16.

The consideration of ex-vessel steam explosions is discussed in Section C.9 of NUREG-1150. This phenomenon was assessed to be of relatively minor importance in the five plants studied, in part because of the greater impacts of such issues as hydrogen combustion loads, etc. Its most prominent impact was in the Grand Gulf plant; Section 6.3 describes its importance relative to other phenomena.

Comment: The aggregate distribution for the probability of drywell shell meltthrough depends critically on the composition of the expert panel. Since this issue combines severe offsite consequences with very large uncertainties, a better resolution of the issues involved is clearly demanded (Kouts 4.6.7).

Response:

The staff agrees with the comment that a better resolution to the drywell shell meltthrough issue is advisable. This issue has been the subject of continuing research by NRC, as discussed in Reference E.40.

Comment: Large uncertainty contributions associated with some phenomena indicate the need for further research. These include the thermal-hydraulic phenomena associated with reactor coolant system (RCS) depressurization (as an accident management strategy), the ways in which the RCS may fail during high-pressure accident sequences in PWRs, and the assessment of threats to (and means to ensure the integrity of) the containment structure in case of a core meltdown resulting from pressure vessel failure (Kouts 7.3).

Response:

The staff agrees that the wide uncertainty distributions associated with specific phenomena provide one indication of where further research is desirable. Other considerations include the importance of the phenomena in question to risk (some wide uncertainty distributions may be acceptable if the contribution to risk is negligible) and the feasibility that further research will reduce the uncertainty bounds. All of these considerations are included by the staff when identifying and prioritizing future research.

Comment: Containment failure from seismic events was based on broad assumptions rather than structural analyses (Kouts 4.3.2).

Response:

Past work has shown that gross structural failure of a typical reinforced concrete containment due to earthquake motion is highly unlikely (Ref. E.41). Rather, it is the pipe penetrations that are most likely to fail because of the loads put on the penetrations by motion of the pipes passing through the penetrations. The loads most likely to cause penetration failure would arise from large motion or support failures of steam generators (in PWRs) or the reactor vessel (in BWRs). Hence, in the NUREG-1150 seismic analysis, containment failure was based on failure of the penetrations resulting from the failure of supports of the major reactor coolant system components. Since the vessel failure and the large LOCA initiating events included in the seismic analysis are based on support failures, it was assumed that some failure of the containment would occur, given either of these initiating events.

This assumption is based on a review of typical containment penetration configurations and discussion with structural experts and is based on the assumption that support failure would result in piping displacements of 1 to 2 feet, and that this would provide a sufficient load to fail the penetration. There are currently no data on the failure capacity of penetrations, given such loads. Hence, this assumption is based on engineering judgment.

In addition, estimates were needed on the size of the leak, given the failures described above. Again, based on typical penetration configurations, it was judged that the most likely crack size would be approximately 1/2 inch by 18 inches, similar to the small leak definition used in the rest of NUREG-1150. It was then assumed that a small leak would occur with a conditional probability of 0.9 and that a larger leak would occur with a conditional probability of 0.1. This assumption was based on the fact that piping supports inside containment would absorb a significant portion of the displacement-induced load and thus limit the leak size. Again, however, there are no data or calculations to substantiate this assumption.

E.7 Source Terms and Consequences

E.7.1 Source Terms

The Kouts committee had two general comments in the area of source term analysis. These were:

- "The overall strategy for generating the uncertainty values in Level 2, including the use of the XSOR codes, appears reasonable, since the tests that were made indicated that the uncertainties introduced by the codes are small compared to the overall Level 2 uncertainties." (Kouts 4.5)
- "Considerable caution is recommended in the use of the results obtained with the approximate XSOR codes without confirmation by more detailed codes." (Kouts 7.3)

The ANS committee had the following general comment:

- "The source terms reported in NUREG-1150 and the resultant offsite consequences should be considered as approximations, due to the reliance on the simplified mass balance XSOR models used to produce large numbers of source terms." (ANS 3.d.3)

In addition to these general comments, the review committees had a number of more specific comments. These are itemized below and staff responses provided.

Comment: The readers of NUREG-1150 should be aware that, of the thousands of source terms results presented, only a few were obtained using the detailed state-of-the-art calculational methods. The remainder were calculated using the parametric XSOR codes. This was a tradeoff to meet the need to generate many results in order to evaluate the uncertainties. The XSOR codes should be used with caution without confirmation by more detailed calculations (Kouts 7.3; ANS 2.a.8.d).

Response:

The XSOR codes were used for two reasons: (1) to generate source terms for the large number of accident progression bins identified in the accident progression analysis, and (2) to provide a means of incorporating the uncertainty in important analysis parameters into the integrated plant studies. Even if uncertainties were not being incorporated into the plant studies, it would be a very demanding undertaking to perform a mechanistic source term calculation for every accident progression bin. The alternative choices are assigning all accident progression bins to the results of a limited number of mechanistic calculations or attempting to modify the results of these calculations to more appropriately match the conditions associated with individual accident progression bins. The latter approach was chosen for the NUREG-1150 analyses.

The XSOR codes actually consist of three parts: a data base developed in the expert elicitation process, a mapping between the accident progression bins and this data base, and an algorithm for constructing source terms on the basis of individual accident progression bins and their associated data. In developing the data base, an attempt was made to use all available sources of information, including mechanistic code calculations, analytic solutions, and experimental data. Thus, the results of mechanistic calculations, as interpreted in the expert review process, are incorporated into the source terms generated by the XSOR codes.

Calculations were performed in which the SOR codes were benchmarked against a Source Term Code Package calculation for a specific scenario. The SOR code was then used to estimate the source terms for a similar scenario. The results compared favorably to a Source Term Code Package calculation made specifically for the second scenario (Ref. E.42).

Chapter 2 and Appendix A to NUREG-1150 have been modified to clarify the role of the XSOR codes.

Comment: Because of the approximate nature of the XSOR codes, the final version of NUREG-1150 should note the need for more exacting analysis of risk-significant accident sequences. The more detailed analysis should be performed and published in a supplement to NUREG-1150. This analysis should concentrate on best-estimate modeling and should be compared with the source terms in the final version of the report (Kouts 7.3; ANS 2.a.8.d).

Response:

The staff agrees with the comment. The staff intends to investigate the practicality of linking risk analysis calculations more closely to accident analysis codes such as MELCOR (Ref. E.43), potentially reducing the dependence on the XSOR codes. As noted below, the staff intends to initiate more detailed studies of the bypass accident sequences.

Comment: With respect to the containment bypass source term, it would be helpful to cite recent work (by EPRI) to help guide the reader to detailed assessments of some of the most important accidents identified in NUREG-1150. Citing more recent studies should help guide the users of NUREG-1150 to existing analyses that provide detailed assessments of some of the most important accident sequences identified in NUREG-1150 (Kouts 4.2.1).

The source terms for containment bypass accident sequences, including interfacing-system LOCAs and steam generator tube ruptures, were not the subject of detailed analyses and may be characterized as conservative approximations (ANS 2.a.8.d).

Response:

A number of Source Term Code Package computer analyses were performed to estimate the source terms for bypass accidents (Ref. E.44). Model development would be required, however, to more realistically treat certain aspects of such accident sequences as deposition in steam generators in a steam generator tube rupture-initiated core damage accident. The staff intends to perform more detailed studies of bypass sequences in followup work to NUREG-1150 and to compare the results of the new studies with those of NUREG-1150. More recent work by EPRI and others will be reflected in such followup comparisons.

Comment: A time cutoff of 24 hours after the onset of core degradation for the release of radionuclides was used throughout NUREG-1150, although no mention of this fact is contained in the report (ANS 2.a.8.d).

Response:

A time cutoff of 24 hours after the onset of core degradation was only used when considering the issue of late revolatilization from the reactor coolant system. Some of the members of the source term expert panel were concerned that the majority of the releases were going to occur extremely late in the accident (much later than 24 hours after the beginning of core damage). The project staff instructed the panel to consider late releases only up to 24 hours after core degradation. The reason for this was that some operator action to cool the reactor coolant system would be expected by that time (e.g., using external cooling by the containment sprays). The time cutoff was not an issue for the other source term processes that were considered because the majority of radionuclides were released well before 24 hours.

Appendix A has been modified to acknowledge this assumption.

Comment: The source terms and consequences of two classes of accidents, containment bypass and early containment failure, should be reported separately as well as the combined data presently displayed (ANS 2.b.9).

Response:

The plant-specific risk reports (Refs. E.12, E.13, and E.26 through E.28) present exceedance frequency curves for the source terms associated with different types of accidents, including containment bypass and early containment failure (e.g., see Figs. 3.3-4 and 3.3-9 in Ref. E.12). Equivalent information for consequences was not generated. However, the individual plant studies do present detailed information on the contribution of different accident types to risk.

Comment: It is not clear how credit is taken for radionuclide retention in the auxiliary building for PWR containment bypass accidents and the reactor building for BWR containment failures (ANS 5.e.4).

Response:

Two types of bypass accidents are considered in the PWR analyses: steam generator tube ruptures (SGTRs) and interfacing-system LOCAs (Event V). During an SGTR accident, the radionuclides are released directly to the environment; therefore, no radionuclide retention in the auxiliary building is considered. For the Event V accident, two methods for retention of radionuclides in the auxiliary (or safeguards) building are considered: retention associated with the building itself and retention from either water pools or water sprays. (At Surry, retention in the relatively small safeguards building is limited; however, there is the potential that the release will occur under a pool of water. At Sequoyah, the release could be mitigated by the fire spray system in the auxiliary building.)

Radionuclide retention in the Peach Bottom reactor building was considered, but none was considered for the Grand Gulf analysis. That portion of the reactor building surrounding the Grand Gulf containment is a relatively weak structure (compared with possible severe accident loadings), and it was judged to have little retention value. The decontamination factors applied in all these plants were provided by the source term expert panel and are documented in Reference E.16.

Comment: At this time, only the MELCOR code is available to the staff for source term calculation. Although it appears to be an improvement over the Source Term Code Package, it is not yet fully developed, nor is it generally available in its current form. Some method for calculating a source term will be needed by the staff and its contractors for performing or reviewing PRAs as well as other tasks (ACRS).

Response:

The MELCOR code is intended to be the staff's principal analytical model for the accident progression portions of its risk analyses. It has been used in the NUREG-1150 work (e.g., Ref. E.45) and is now being used to support other staff risk analysis work. The staff's planning for further MELCOR development, etc., is described in Reference E.40. As noted above, the staff also plans to investigate the practicality of more closely linking risk analysis calculations to codes such as MELCOR, reducing dependency on parametric models such as the XSOR codes.

E.7.2 Offsite Consequences

The review committees had a number of specific comments in the area of offsite consequence analysis. These are itemized below and staff responses provided.

Comment: The uncertainties in offsite consequences were not included in the NUREG–1150 risk uncertainty estimates (Kouts 7.2; ACRS)

Response:

As indicated in the report, it was not possible because of time constraints to include offsite consequence uncertainties in NUREG–1150. The development of needed probability distributions for parameters included in offsite consequence assessments and the incorporation of these distributions into risk uncertainty assessments is planned to be initiated in 1991.

Comment: There are also a number of uncertainties in the modeling of consequences due to decisions that would be made only during or after a severe accident. These decisions, of a sociopolitical nature, include such things as evacuation, interdiction of land and foodstuff, and the value of real property. These uncertainties have not been included in NUREG–1150, although they have been discussed elsewhere. Recent experience suggests that much lower interdiction levels than those used in NUREG–1150 are sometimes used, which would have the effect on NUREG–1150 results of increasing economic impacts and decreasing health impacts (Kouts 3.2.4, 4.12).

Response:

The staff agrees that issues such as interdiction levels actually used in the event of a reactor accident may be quite different than those used in NUREG–1150. As discussed in Chapter 2 and Appendix A, the evacuation and interdiction assumptions in NUREG–1150 were based on Environmental Protection Agency and Food and Drug Administration guidelines, respectively (Refs. E.46 and E.47). The results of sensitivity studies on these assumptions are provided in Chapters 11 and 12 of the summary report.

Comment: The MACCS code used in NUREG–1150 for offsite consequence analysis is a relatively new code, still under development. It has been neither benchmarked nor validated. Additional uncertainties are introduced by the use of such a new and relatively untested code (ACRS).

Response:

The staff agrees that the use of relatively new computer codes introduces additional uncertainty. Two efforts were undertaken as part of the NUREG–1150 project to improve the reliability of the MACCS code. These were an independent review of the chronic exposure pathway model in the code (Ref. E.48) and an independent line-by-line review of the code (Ref. E.49).

Benchmarking of the MACCS code is now under way under the auspices of an international project sponsored by the Committee on the Safety of Nuclear Installations and the Commission of the European Communities.

Comment: Important information on the offsite consequence calculations is not provided, such as the fact that inhalation doses reflect lifetime dose commitments (ANS 2.a.8.e).

Response:

In its role as a summary document, NUREG–1150 can only give a relatively brief description of the individual models used in the analysis. Detailed descriptions of the individual models are given elsewhere. For the MACCS program used to calculate offsite consequences, detailed descriptions of both the models and the computer program are given in Reference E.22. Further, the data used in the NUREG–1150 consequence calculations are described in Part 7 of Reference E.16.

E.8 Uses of NUREG–1150

The review committees had a number of specific comments in the area of the uses of NUREG–1150. These are itemized below and staff responses provided.

Comment: NUREG–1150, along with other PRAs and recent work in severe accident analysis, should be used to close out as many open issues as can reasonably be achieved and help prioritize limited research resources on the remaining safety issues. A definitive program for the use of NUREG–1150 and its supporting documents should be developed and implemented (Kouts 7.3).

The information presented in NUREG-1150 must be carefully examined in the context of the plant being studied to determine the priority ranking of safety issues, and we caution against broad generalities (ANS 6).

Use of NUREG-1150 to assist in prioritization and resolution of safety issues should be considered a priority application and a principal benefit of the substantial resources expended on this multiyear study (ANS 2.a.13.e).

Response:

As discussed in Chapter 13, the risk analyses of NUREG-1150 are intended to be used as one tool in the prioritization of research and safety issues, as well as in a number of other ways by the staff. Some applications of NUREG-1150 methods and results have already been made, such as in supporting the development of guidance for individual plant examinations (Refs. E.50 and E.51). (Chapter 13 has been updated to reflect some of the more recent uses.) As appropriately noted by the ANS comment, the plant-specific nature of the NUREG-1150 analyses should be and has been kept in mind in such applications.

Following publication of the final version of NUREG-1150, the staff intends to provide additional guidance to potential users of the report within NRC as to its strengths and weaknesses, etc.

Comment: The results of NUREG-1150 should be used only by those who have a thorough understanding of its limitations (ACRS).

Response:

The staff agrees with this comment. As noted in Section E.5.3, the staff has developed a data base and computer codes that permit the staff to modify the NUREG-1150 (and other PRA) accident frequency analyses and plans to develop similar data bases and codes for the remainder of the risk analyses. The staff intends to develop quality assurance procedures as part of this effort to minimize the potential for inappropriate calculations.

Chapter 1 has been modified to note this caution.

Comment: It is disappointing that the staff asserts that virtually no general conclusions can be drawn from a study that took almost 5 years and 17 million dollars to complete. We recommend that the Commission encourage the staff to mine more deeply the wealth of information that has been collected in the course of this study in an effort to identify generic conclusions that might be reached (ACRS).

Response:

The staff agrees that NUREG-1150 provides a substantial body of information, much of which has not yet been "mined" for use in other staff work. It is expected that this body of information will see its principal use by the staff to support the resolution of specific issues, such as study of alternative safety goals, generic issue resolution, PRA reviews, etc. The staff also intends to commit resources to the study of more general issues (e.g., the extrapolation of results for five plants to other plants).

Comment: It is recommended that the NRC issue additional guidance on the treatment of external events in the individual plant examination (IPE) process (Kouts 7.3).

Response:

Such guidance was issued in draft form (for public comment) in July 1990 (Ref. E.51).

Comment: The NUREG-1150 methodology is of special value with respect to guiding risk-reduction and risk-management actions because it makes possible a more sophisticated approach to risk management, addressing not only major contributors to risk, taken as point values, but also contributors associated with large uncertainty bands (Kouts 4.13).

Taken together with the individual plant examinations, NUREG-1150 should help guide evaluation of accident management from a risk-reduction perspective. However, such uses of NUREG-1150 would seem to be limited due to the parametric nature of the study (ANS 6).

Response:

NUREG-1150 information is being used in the development of general accident management guidance (Ref. E.52). As with the individual plant examination process, the NRC in ensuring that each licensee has developed an adequate accident management program. Such a program will be prepared by the licensee reflecting plant-specific information from a plant's individual plant examination as well as from more generic information such as NUREG-1150.

Comment: In many European countries, safety goals and objectives are related to a low risk of releases with disruptive effects on society, typically meaning releases with a potential for long-term restrictions on land usage over large areas. The summary presentations of the results in the main report do not facilitate comparisons with such alternative safety goals. An addition of such comparisons or later documentation might enhance the value of the report, especially outside the United States, since many of these may not be calculable with data in the report (Kouts 4.14).

Response:

The staff agrees that a comparison of the spectrum of national safety goals using the NUREG-1150 plant models would be of considerable interest. Such a comparison could not be accomplished in time for inclusion in NUREG-1150 but is being considered by the staff for future study.

Comment: The limited information presented in NUREG-1150 with respect to the NRC staff's proposed large-release goal would not be particularly useful in the evaluation of implementation strategies (ANS 2.a.13.d).

Response:

The staff agrees that NUREG-1150 provides very limited information on possible large-release goals and implementation strategies. The discussion provided in Chapter 13 of the report was intended as a demonstration of how NUREG-1150 risk models could be used in assessing alternative goals and applying the then-recommended definition of large release, rather than providing a definitive study of a complex technical issue. Since that time, the Commission has provided the staff with additional guidance on safety goal implementation (Ref. E.53) and possible definitions of large releases. It is expected that the NUREG-1150 models will be used by the staff as part of the further consideration of large-release definitions.

REFERENCES FOR APPENDIX E

E.1 U.S. Nuclear Regulatory Commission (USNRC), "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, Second Draft for Peer Review, June 1989.

E.2 H. J. C. Kouts et al., "Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)," NUREG-1420, August 1990.

E.3 L. LeSage et al., "Report of the Special Committee on NUREG-1150, The NRC's Study of Severe Accident Risks," American Nuclear Society, June 1990.

E.4 Walston Chubb letter to Denwood F. Ross, Jr., USNRC, August 14, 1989.

E.5 Steven C. Sholly, MHB Technical Associates, letter to Denwood F. Ross, Jr., USNRC, August 7, 1989.

E.6 Roy R. Wight, State of Illinois Department of Nuclear Safety, letter to Chief, Rules and Procedures Branch, USNRC, December 15, 1989.

E.7 G. A. Hunger, Jr., Philadelphia Electric Company, letter to Chief, Rules and Procedures Branch, USNRC, January 11, 1990.

E.8 F. Coffman et al., "Human Factors Regulatory Research Program Plan," NUREG-1384, Vol. 1, October 1989.

E.9 USNRC, "Nuclear Plant Aging Research Program Plan," NUREG-1144, Revision 1, September 1987.

E.10 USNRC, "Analysis of Potential Pressurized Thermal Shock Events," *Federal Register*, Vol. 50, p. 29937, July 23, 1985.

E.11 USNRC, "Regulatory Analysis for Proposed Amendment of the PTS Rule, 10 CFR 50.61, 'Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock'," cited in "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events," *Federal Register*, Vol. 54, p. 52946, December 26, 1989.

E.12 R. J. Breeding et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, October 1990.

E.13 A. C. Payne, Jr., et al., "Evaluation of Severe Accident Risks: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Revision 1, SAND86-1309, December 1990.

E.14 R. A. Chrzanowski, "March 13, 1989, letter from Cordell Reed to T. E. Murley," NRC Docket Nos. 50-295 and 50-304, August 24, 1990.

E.15 T. A. Wheeler et al., "Analysis of Core Damage Frequency from Internal Events: Expert Judgment Elicitation," Sandia National Laboratories, NUREG/CR-4550, Vol. 2, SAND86-2084, April 1989.

E.16 F. T. Harper et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," Sandia National Laboratories, NUREG/CR-4551, Vol. 2, Revision 1, SAND86-1309, December 1990.

E.17 R. C. Bertucio and J. A. Julius, "Analysis of Core Damage Frequency: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Revision 1, SAND86-2084, April 1990.

E.18 A. M. Kolaczkowski et al., "Analysis of Core Damage Frequency: Peach Bottom Unit 2," Sandia National Laboratories, NUREG/CR-4550, Vol. 4, Revision 1, SAND86-2084, August 1989.

E.19 R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, Revision 1, SAND86-2084, April 1990.

E.20 M. T. Drouin et al., "Analysis of Core Damage Frequency: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 6, Revision 1, SAND86-2084, September 1989.

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

E.22 D. I. Chanin, H-N Jow, J. A. Rollstin et al., "MELCOR Accident Consequence Code System (MACCS)," Sandia National Laboratories, NUREG/CR-4691, Vols. 1-3, SAND86-1562, February 1990.

E.23 USNRC, "Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

E.24 E. D. Gorham-Bergeron et al., "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration, and Uncertainty Analyses," Sandia National Laboratories, NUREG/CR-4551, Vol. 1, Draft Revision 1, SAND86-1309, to be published.*

E.25 H. F. Martz et al., "Eliciting and Aggregating Subjective Judgments—Some Experimental Issues," *Proceedings of the 1984 Statistical Symposium on National Energy Issues* (Seattle, WA), NUREG/CP-0063, July 1985.

E.26 J. J. Gregory et al., "Evaluation of Severe Accident Risks: Sequoyah Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, December 1990.

E.27 T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 6, Revision 1, SAND86-1309, December 1990.

E.28 C. K. Park et al., "Evaluation of Severe Accident Risks: Zion Unit 1," Brookhaven National Laboratory, NUREG/CR-4551, Vol. 7, Draft Revision 1, BNL-NUREG-52029, to be published.*

E.29 A. D. Swain and H. E. Guttman, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," Sandia National Laboratories, NUREG/CR-1278, SAND80-0200, August 1983.

E.30 E. A. Rosa et al., "Application of SLIM-MAUD: A Test of an Interactive Computer-Based Method of Organizing Expert Assessment of Human Performance and Reliability: Main Report," Brookhaven National Laboratory, NUREG/CR-4016, Vol. 1, BNL-NUREG-51828, September 1985.

E.31 D. M. Ericson, Jr., (Ed.) et al., "Analysis of Core Damage Frequency: Internal Events Methodology," Sandia National Laboratories, NUREG/CR-4550, Vol. 1, Revision 1, SAND86-2084, January 1990.

E.32 V. Ho et al., "COMPBRN III—A Computer Code for Modeling Compartment Fires," University of California at Los Angeles, UCLA-ENG-8524, November 1985.

E.33 USNRC, "Seismic Safety Research Program Plan," NUREG-1147, Revision 1, May 1987.

E.34 P. W. Baranowsky et al., "A Probabilistic Safety Analysis of DC Power Requirements for Nuclear Power Plants," NRC report NUREG-0666, April 1981.

E.35 H. D. Stewart et al., "System Analysis and Risk Assessment System (SARA), Version 4.0, Volume 1—Reference Manual," Idaho National Engineering Laboratory, NUREG/CR-5022, EGG-2522, to be published.*

E.36 K. D. Russell et al., "Integrated Reliability and Risk Analysis System (IRRAS) Version 2.0 User's Guide," Idaho National Engineering Laboratory, NUREG/CR-5111, EGG-2535, June 1990.

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

E.37 J. M. Griesmeyer and L. N. Smith, "A Reference Manual for the Event Progression Analysis Code (EVNTRE)," Sandia National Laboratories, NUREG/CR-5174, SAND88-1607, September 1989.

E.38 R. L. Iman et al., "PRAMIS: Probabilistic Risk Assessment Model Integration System," Sandia National Laboratories, NUREG/CR-5262, SAND88-3093, May 1990.

E.39 USNRC, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR § 50.54(f)," Generic Letter 88-20, November 23, 1988.

E.40 USNRC, "Revised Severe Accident Research Program Plan: FY 1990-1992," NUREG-1365, August 1989.

E.41 M. Amin et al., "An Analytical Study of Seismic Threat to Containment Integrity," Sandia National Laboratories, NUREG/CR-5098, SAND88-7018, July 1989.

E.42 P. Cybulskis, "Assessment of the XSOR Codes," Battelle Columbus Division, NUREG/CR-5346, BMI-2171, November 1989.

E.43 R. M. Summers et al., "MELCOR In-Vessel Modeling," *Proceedings of the Fifteenth Water Reactor Safety Information Meeting* (Gaithersburg, MD), NUREG/CP-0091, February 1988.

E.44 R. S. Denning et al., "Report on Radionuclide Release Calculations for Selected Severe Accident Scenarios: Supplemental Calculations," Battelle Columbus Division, NUREG/CR-4624, Vol. 6, BMI-2139, August 1990.

E.45 S. E. Dingman et al., "MELCOR Analyses for Accident Progression Issues," Sandia National Laboratories, NUREG/CR-5331, SAND89-0072, to be published.*

E.46 U. S. Environmental Protection Agency, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," Office of Radiation Programs, Draft, 1989.

E.47 U.S. Department of Health and Human Services/Food and Drug Administration, "Accidental Radioactive Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies," *Federal Register*, Vol. 47, No. 205, pp. 47073-47083, October 22, 1982.

E.48 U. Tveten, "Review of the Chronic Exposure Pathway Models in MACCS and Several Other Well-Known Probabilistic Risk Assessment Models," Institutt for Energiteknikk, Norway, NUREG/CR-5377, June 1990.

E.49 C. A. Dobbe et al., "Quality Assurance and Verification of the MACCS Code, Version 1.5," Idaho National Engineering Laboratory, NUREG/CR-5376, EGG-2566, February 1990.

E.50 USNRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August 1989.

E.51 USNRC, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, Draft Report for Comment, July 1990.

E.52 W. J. Luckas et al., "Assessment of Candidate Accident Management Strategies," Brookhaven National Laboratory, NUREG/CR-5474, BNL-NUREG-52221, March 1990.

E.53 Samuel J. Chilk, "SECY-89-102—Implementation of the Safety Goals," Memorandum to James M. Taylor, dated June 15, 1990.

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

ATTACHMENT TO

APPENDIX E

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 15, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: REVIEW OF NUREG-1150, "SEVERE ACCIDENT RISKS: AN ASSESSMENT FOR FIVE U.S. NUCLEAR POWER PLANTS"

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we discussed the second draft of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." The Committee had previously discussed this matter with the staff and its consultants and with Dr. Herbert Kouts, Chairman of the Special Committee to Review the Severe Accident Risk Report. Our Subcommittees on Severe Accidents and Probabilistic Risk Assessment discussed this report during a number of joint meetings with members of the staff, Sandia National Laboratories (SNL) and the American Nuclear Society (ANS) Special Committee (Dr. Leo LeSage, Chairman). We also had the benefit of the documents referenced.

1. <u>INTRODUCTION</u>

In this report, we first offer some general comments. We then offer recommendations concerning the publication of NUREG-1150 and provide comments and cautions concerning interpretation or use of some of the components of this document. And finally, we provide more detailed comments on some key parts.

We have reviewed the reports prepared by the ANS Special Committee and by the Special Committee to Review the Severe Accident Risk Report appointed by the Commission and found them helpful. We have no serious disagreements with either of these reviews, nor with their findings.

2. <u>GENERAL COMMENTS</u>

The work described in this draft of NUREG-1150 is an improvement over that described in the first version entitled, "Reactor Risk Reference Document." Many previously identified deficiencies in the expert elicitation process have been corrected. The exposition and organization of the report have been improved. The presentation of results is clearer. There is considerable information that was not in the original version.

The portion that deals with accident initiation and development up to the point at which core heat removal can no longer be assured is unique, compared to other contemporary PRAs, in that a method for estimating the uncertainty in the results has been developed and applied. This method and its application are significant contributions. Although the larger contributions to uncertainty in risk come from the later parts of the accident sequences, this portion is enhanced also by an extensive identification of events that can serve as accident initiators as well as an associated set of hypothesized event trees. This information should be of considerable assistance to licensees in the performance of an Individual Plant Examination (IPE). It should also be useful to plant operators and to designers.

The formulation of a more detailed representation of accident progression after severe core damage begins, and an improved description of containment performance, contribute some additional information to this important area. However, understanding of many of the physical phenomena that have an important bearing on this phase of accident progression is still very sparse, and the report may give the impression that more is known about this portion of the accident sequence than is actually the case.

The part of the sequence that begins with the release of radioactive material outside the containment is treated by a relatively new and unevaluated code system. Furthermore, there is no estimate of the uncertainties inherent in the calculations that describe this part of the sequence. Those who use the quantitative values of reported risk must recognize that these uncertainties are not accounted for in the calculated results.

3. RECOMMENDATIONS

We recommend that the current version of NUREG-1150, with the corrections suggested by several of those who have already reviewed it in detail, be published. However, its results should be used only by those who have a thorough understanding of its limitations. Some of these limitations are discussed in subsequent sections of our report.

Since the supporting documents upon which NUREG-1150 depends could be helpful to those who perform an IPE, we recommend that these also be published as soon as feasible.

Both the Commission and the ACRS have raised questions about generic conclusions that might result from a careful examination of the results of this study. It is disappointing that the staff asserts that virtually no general conclusions can be drawn from a study that took almost five years and seventeen million dollars to complete. We recommend that the Commission encourage the staff to mine more deeply the wealth of information that has been collected in the course of this study in an effort to identify generic conclusions that might be reached (see Section 5.5 of this letter).

4. COMMENTS AND CAUTIONS CONCERNING USES OF THE MATERIAL IN NUREG-1150

We discuss below certain areas in which the methods or results should be used with caution.

4.1 Differences Among Levels of the PRA

The phenomena which contribute to sequence progression in Level 1 are generally well understood. Power plant or other related experience with system and component performance has provided sufficient data to permit predictions of sequence progression with considerably greater confidence than for those parts of the sequence described in Levels 2 and 3. NUREG-1150 is unique in the amount of effort that went into estimating uncertainties in the calculated Level 1 results. It is our view that the results of Level 1 can be used with more confidence than those of Levels 2 and 3. However, as other reviewers have reported, there are recognized deficiencies in the state-of-the-art treatments of human performance; and this report is not free of those deficiencies. In addition, some possibly important initiators, e.g., those at low power operation or at shutdown, and sequences initiated by fire, are either treated superficially or are neglected altogether.

The Level 2 analyses in NUREG-1150 include more detailed containment event trees than those found in any previous PRA. However, we have some concern that the amount of detail may lead to a conclusion that much more is known about the phenomena in this area than is actually the case.

Since there is a dearth of information concerning many of the phenomena that determine severe accident progression, expert elicitation was used most extensively in the Level 2 portion of the PRAs. There is general agreement that the techniques used for eliciting expert opinion in preparation of the second draft were significantly better than those used for the first draft. However, with insufficient information there can be no experts. Thus, use of the term "expert opinion" in a description of some of the Level 2 work may be misleading. (Further comments about the expert elicitation process are given in Section 5.3). We applaud efforts to improve on the Level 2 treatment of previous PRAs. We nevertheless believe that the results from Level 2 presented in this latest draft must be regarded as having major uncertainties in both calculated mean values and in estimated uncertainties.

The Honorable Kenneth M. Carr 4 November 15, 1990

The MELCOR Accident Consequence Code System (MACCS) was used for the consequence calculations of Level 3. Use of MACCS is a departure from many existing PRAs that use the Calculation of Reactor Accident Consequences (CRAC) series of codes. MACCS is a relatively new code, still under development. It has been neither benchmarked nor validated. Thus, in addition to the uncertainties inherent in the physical phenomena that enter into consequence modeling, additional uncertainties are introduced by the use of a new and relatively untested code.

No effort was made to estimate the uncertainties in the Level 3 calculations. Thus, the estimates of uncertainties in risk that are given in the report are only those arising from the uncertainties calculated for Levels 1 and 2. It is our judgment that the uncertainties in modeling the consequences of a release can be at least as large as those estimated for Level 2. For example, the health effects, especially for low dose exposures, are subject to large uncertainty, and the exposures themselves depend on actions (e.g., evacuation, sheltering, interdiction of land and crops) for which the uncertainty in prediction is largely unknown.

4.2 Assumptions Made in Screening

Users of the report should be aware of the assumptions made in the screening process for low-probability, high-consequence events. For example, the analysts assumed that the probability of total loss of DC power was less than 1×10^{-7} per year and thus could be neglected. The same assumption was made for loss of all service water. Thus, those who use the results in IPE work should recognize that these assumptions may not be valid for all operating plants.

4.3 Credit for Decay Heat Removal by Feed and Bleed

The success of the feed and bleed operation is highly dependent on human performance. Everyone seems to agree that there are large uncertainties in its treatment in this report. In addition, it is likely that the performance of valves, which must function if this maneuver is to be successful, are not well represented by the data for valve performance used in the calculations.

4.4 Performance of Motor-Operated Valves

There is now a significant body of evidence which indicates that the failure probability used to describe the operation of certain key motor-operated valves is too low. This may have an important bearing on the outcome of several accident sequences described in the report.

4.5 Contribution of Pump-Seal Failure to the Risk of Small Break LOCAs

We believe that more recent information and some new seal designs developed since the study was made would lead to a prediction of risk less than that reported.

4.6 Containment Performance

The lack of information about many of the physical phenomena that determine the performance of a containment system in a severe accident situation is such that only educated guesses can be made for some sequences that might make significant contributions to risk. Although the large number of event trees developed in the containment analyses is indicative of what was hypothesized by the analysts, the amount and quality of information concerning a number of key phenomena that determine behavior at branch points are low. The difficulty of arriving at a result with significant confidence is illustrated by two examples. In the analysis of the performance of the Mark I containment used in early BWRs, the experts in the original study predicted a large conditional probability of early failure. In the second study a different group of experts produced a bimodal distribution because part of the panel concluded that the probability of early failure was high, and part considered it low. A second example is the calculation of risk produced by postulated direct containment heating (DCH). In the first study, the calculated risk due to DCH for PWRs with large dry containments was a major contributor to the total risk. In the second version, its contribution was significantly less. In neither case had there been a major change in the information about relevant physical phenomena available at the time of the first study. Further, we find no consideration of the impact of ex-vessel steam explosions on early containment failure. There is little unambiguous guidance here for a licensee performing an IPE.

5. AREAS FOR SPECIAL COMMENT

In this section, we provide more detailed comments on some areas that appear to us to deserve special attention.

5.1 Fire Risk

The fire contribution to core-damage probability was estimated for two plants using insights gained during previous fire PRAs and studies, the latest methods and data bases developed under NRC sponsorship, and the benefits of extensive plant walkdowns. The methods and data used were probably the best available at the time the reported work was performed. Nevertheless we conclude, on the basis of later information, that the results should be viewed as being incomplete. The models used were not able to take full account of several issues identified by SNL in a scoping study of fire risks that was completed more recently. These are issues that have not been adequately considered in past fire risk studies and may increase the risk. Of particular concern are seismic-fire interactions, adequacy of fire barriers, equipment survival in the environment generated by the fire, and control systems interactions. The PRA for the LaSalle nuclear plant, which is nearing completion, may provide insights concerning the risk importance of these issues.

5.2 Seismic Risk

The seismic PRAs for the Surry and Peach Bottom nuclear plants were performed using two quite different representations of the seismic hazards. The results however, at least for sequences leading to core damage, were similar in terms of which accident initiators and sequences were important. This tends to support the acceptability of using the seismic margin approach rather than a PRA in the search for plant-specific seismic vulnerabilities in the IPE-External Events (IPEEE) program. However, the success of either approach in finding vulnerabilities depends strongly on walkdowns to identify those systems and components to be evaluated. Knowledge of what to look for is derived chiefly from PRAs done on other plants, and these have tended to focus primarily on core damage rather than releases of radioactive material to the environment. Although containments are usually quite rugged seismically, this is not necessarily true for containment cooling systems, containment isolation systems, etc.

Although the two seismic PRAs in NUREG-1150 have been carried through Level 3, these results have not been reported. We believe that these results might provide valuable insights about seismic vulnerabilities of containment systems.

5.3 The Expert Elicitation Process

There is general agreement that the use of expert elicitation in the preparation of the results in this draft of the report is improved compared to that used for the first version. However, we have reservations about some parts of the application of the process. For example, during our discussions of the choice of the participating experts we got the impression that an effort was made to choose participants in such a way that a wide spectrum of viewpoints would be represented. This was defended as proper, based on the assumption that unless this wide spectrum of opinion was represented, the uncertainty in expert opinion would not be appropriately accounted for. We found this argument unconvincing, and would have preferred to see individuals chosen primarily on the basis of their knowledge and understanding of the phenomena being considered. Furthermore, we were told that the budget for the study provided only enough funding to support the participation of about 20 percent of the experts who served on the panels. The remainder were drawn from the NRC staff or from organizations with contractual relationships to the NRC. This biased the selection toward people whose organizations depend upon the NRC for support. We also observe that the membership of the panels seems to have been dominated by analysts in contrast to those who have done significant research on phenomena of importance to the accident sequences being described.

5.4 Source Term Description

The staff, or at least that part of it closely associated with this study, has discarded for future use the Source Term Code Package (STCP) that was one of the resources used by the expert panels in the preparation of NUREG-1150. The expert elicitation method is too resource intensive to be used generally. At this time, only the MELCOR code is available to the staff for source term calculation. Although it appears to be an improvement over the STCP, it is not yet fully developed, nor is it generally available in its current form. Some method for calculating a source term will be needed by the staff and its contractors for performing or reviewing PRAs, as well as for other tasks, such as a revision of the siting rule.

5.5 Lack of General Conclusions

We have asked the staff whether the results reported in NUREG-1150 shed any light on the risk expected due to operation of the population of plants now licensed. With few exceptions, it is the staff's view that one can tell little or nothing about the expected risk of plants not studied from the results of the study of these five plants in NUREG-1150. In spite of these statements, however, those who prepared the report propose that applications will include evaluation and resolution of generic issues and prioritization of future research and prioritization of inspection activities. If, as we were told, the results from the analyses of these plants have little or no generic significance, application of these results must be made with considerable caution.

We believe that the large amount of information collected as input to the calculations made during this study, and the results of the large number of analyses undertaken, must surely permit some more general conclusions to be drawn than we find in this report. For example, the risk calculated for each of the five plants analyzed (although calculated only for internal initiators) falls within the Quantitative Health Objectives (QHOs) set forth in the Safety Goal Policy Statement. Each was designed and constructed and is operating within the rules and regulations promulgated by the Commission. There must be some significance in the fact that plants supplied by a number of different vendors, constructed at different locations, under supervision of different organizations, over a period of more than a decade, with rather different balance of plant configurations, and different containments, nevertheless fall within the QHOs. Is application of the NRC's regulations achieving the objectives of the NRC Safety Goal Policy?

Another area of interest is the risk reduction achieved by some recently promulgated rules. The report indicates that station blackout is a significant risk contributor for three of the plants studied. Answers to questions we asked during our meetings with the staff indicated that some of the plants analyzed had implemented most of the requirements of the Station Blackout Rule, while others had only just begun the process. Could one draw any conclusions from the plants studied as to the risk reduction to be expected from implementation of the Station Blackout Rule? Or could one estimate the risk reduction for some "average" plant? This would be interesting, since in the typical cost benefit analysis associated with backfit it is assumed that some such conclusion can be drawn about plants generally. It would be useful to see what an examination of these five plants would indicate.

The five nuclear power plants chosen for the study were selected partly on the basis of the different types of containment represented. We find little or no discussion of relative containment performance or identification of containment designs that might be expected to have superior mitigation capabilities. For example, in light of the containment being proposed for the Advanced Boiling Water Reactor (ABWR), it would be helpful to have any information or conclusions that were developed during the course of the study as to relative efficacy of the containment being proposed for that design as compared to the Mark I or the Mark III containments. Or, for large dry containments, does the subatmospheric operation of the Surry system provide a substantial decrease in risk (because, for example, of its continuous indication of leak tightness) as compared to a large dry containment operated at atmospheric pressure?

Although it may not be feasible to make major changes in containments of reactors now in operation, it is possible to choose containments with superior mitigation characteristics for nuclear plants not yet constructed. It might even be feasible, as a result of the study, to recommend a containment design that combines the best features of several of the existing systems. If in the course of this study information has been developed that could be used to reduce the conditional failure probability of containment, given severe core damage, the risk uncertainty in new designs might be reduced without requiring any additional studies of core damage progression.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Volumes 1 and 2 (Second Draft for Peer Review), dated June 1989.
2. American Nuclear Society, "Report of the Special Committee on NUREG-1150, The NRC's Study of Severe Accident Risks," L. LeSage (Chairman), dated August 1990.
3. U.S. Nuclear Regulatory Commission, NUREG-1420, "Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)," H. Kouts (Chairman), dated August 1990.
4. U.S. Nuclear Regulatory Commission, NUREG-1150, "Reactor Risk Reference Document," Volumes 1, 2, and 3, Draft issued for comment, dated February 1987.

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1150
Vol. 3

2. TITLE AND SUBTITLE

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

Appendices D and E
Final Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	1991

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Final Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

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 which provide the detailed discussion of the methods used and results obtained in these risk studies.

Volume 3 of this report contains two appendices. Appendix D summarizes comments received and staff responses on the first (February 1987) draft of NUREG-1150. Appendix E provides a similar summary of comments and responses, but for the second (June 1989) version of the report.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

severe accidents
risk analysis
probabilistic risk analysis

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE

THIS DOCUMENT WAS PRINTED USING RECYCLED PAPER.