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Vol. 2

PRA PROCEDURES GUIDE

A Guide to the Performance of Probabilistic
Risk Assessments for Nuclear Power Plants

Final Report
Vol. 1 - Chapters 1-8
Vol. 2 - Chapters 9-13 and Appendices A-G

Prepared under the auspices of
The American Nuclear Society and
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Foreword

The development of safety design requirements for nuclear power plants in the last 20 to 25 years took place in a subjective, deterministic framework. Little use was made of the techniques of quantitative probabilistic risk assessment (PRA), largely because these techniques were not fully developed for analyzing nuclear power plants. It was F. R. Farmer who introduced the idea of reactor safety based on the reliability of consequence-limiting equipment in the early 1960s. The first major application of PRA techniques was the Reactor Safety Study (WASH-1400), which demonstrated that a nuclear power plant could be analyzed in a systematic fashion by PRA techniques. Since the completion of the Study in 1975, the Nuclear Regulatory Commission (NRC) has been exploring ways of systematically applying probabilistic analysis to nuclear power plants, and the use of PRA techniques has been rapidly becoming more widespread in the nuclear community.

Contributing to these developments has been a growing appreciation of the wisdom of the strong recommendations made by the Lewis Committee to use PRA techniques for reexamining the fabric of NRC's regulatory processes to make them more rational.* After the accident at Three Mile Island, these recommendations were reinforced by the Kemeny† and Rogovin reports,‡ which also encouraged the use of these techniques. As Lewis stated in his March 1981 Scientific American article,§ "the Three Mile Island incident illustrates graphically how important it is to quantify both the probability and the consequences of an accident, and to generate some public awareness of these issues.... This is an issue that goes to the heart of many regulatory and safety decisions, where one must have some measure of the risks one is willing to accept on as quantitative a basis as the expert community can provide."

The NRC has recently raised questions about potential accident risks for nuclear plants near high population concentrations. To answer these questions, the industry has performed PRAs for the Indian Point, Limerick, and Zion plants. Moreover, the utilities themselves are showing considerable interest in taking advantage of the safety and availability insights afforded by risk assessments. As a result of these forces, an increasing number of PRAs are either under way or being planned. Finally, the NRC is contemplating a future program (National Reliability Evaluation Program, NREP) in which many licensed nuclear power plants will be required to perform a probabilistic risk assessment.

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

†J. G. Kemeny et al., Report of the President's Commission on the Accident at Three Mile Island, Pergamon Press, 1979.

‡M. Rogovin, Three Mile Island: A Report to the Commissioners and to the Public, USNRC Report NUREG/CR-1250 (Vol. 1), 1979.

§H. W. Lewis, "The Safety of Fission Reactors," Scientific American, March 1981.

Because of this increasing application of PRA techniques within the industry and the regulatory process, there is a need for technical guidance on methods and procedures. It was this need that led to the creation of the PRA Procedures Guide project and ultimately to this document.

The objective of this project was to compile a procedures guide describing the principal methods now used in PRAs. To accomplish these objectives, a Steering Committee and a Technical Writing Group were formed. Funding has been provided by the NRC, the Department of Energy (DOE), and the Electric Power Research Institute (EPRI), and expertise was contributed by the nuclear industry.

The group responsible for the document is the Steering Committee. The Committee includes representatives from the American Nuclear Society, the Institute of Electrical and Electronics Engineers, the NRC, the DOE, the Atomic Industrial Forum, EPRI, and utilities (see Chapter 1 and Appendix B for the membership list). The Technical Writing Group, whose members were selected by the Steering Committee (see Appendix B), consists of technical specialists experienced in the application of probabilistic and reliability techniques to the analysis of nuclear power plants.

To obtain the wide peer review desired for the Procedures Guide, the Steering Committee decided on two mechanisms: criticism by a carefully selected peer review group and open review in two conferences. The objective in establishing the peer review group was to bring additional technical expertise and, in some instances, alternative viewpoints to the project. An effort was also made to include experts who are not members of the nuclear community. Candidates for the peer group were proposed by the Steering Committee and members of the Technical Writing Group; those who were finally selected are listed in Appendix B.

The first of the two conferences, held on October 26-28, 1981, included a series of workshops in risk assessment. It was sponsored by the Institute of Electrical and Electronics Engineers. The second was held on April 4-7, 1982, by the American Nuclear Society. These meetings have allowed the Steering Committee to obtain comments from a large number of experts in disciplines related to probabilistic risk assessment as well as potential users of the Procedures Guide. The disposition of these comments, like those of the peer review group, has been resolved by the Technical Writing Group under the guidance of the Steering Committee.

Actual writing of the Procedures Guide by the Technical Writing Group began only in April 1981, and by July a working draft was produced for review by the Steering Committee. It was followed by a review draft that was distributed for peer review and discussion at the October 1981 conference. The October 1981 conference was heavily attended, and many comments were submitted to the Steering Committee. A major revision of the Procedures Guide resulted in a second draft, published in April 1982 for the attendees of the ANS Executive Conference, which reflected many, but not all, of the comments.

After the ANS Executive Conference, a final revision was made, and this document resulted. Thus, the methods described herein have received broad review from both PRA practitioners and potential users of PRA techniques.

Upon completion of the PRA Procedures Guide project, the Steering Committee, which has guided the project, was disbanded. Future questions or comments on the Guide should be directed to Robert M. Bernero, Division of Risk Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

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VOLUME 2

Chapter 9

Environmental Transport and Consequence Analysis

9.1 INTRODUCTION

9.1.1 OBJECTIVE AND SCOPE

This chapter describes how to calculate the consequences of radionuclide releases into the environment and how to interpret the results of such calculations. It is primarily intended for the would-be user of consequence models, someone who has perhaps obtained a consequence-modeling computer package off the shelf and wishes to know what to do with it--the information that is required as input, the kind of output that might result and how it is to be interpreted, the pitfalls associated with the use of the code, and the uncertainties inherent in the data and the modeling. It is expected, however, that this chapter will also appeal to a wider audience. The layman should be able to find enough qualitative material to give him a good idea of what consequence modeling is about; and the expert, it is hoped, will benefit from the discussion of various topics that are still subject to debate and controversy in the consequence-modeling and scientific communities.

The remainder of this section contains a brief description of the scope of this chapter. The overview in Section 9.2 delineates the major tasks of a consequence analysis, explains why each task is done, what information results, and how it is to be used.

Section 9.3 discusses the various elements of a consequence analysis: (1) transport and diffusion in the atmosphere and/or water; (2) deposition processes; (3) processes that lead to the accumulation of radiation doses; (4) protective measures, such as evacuation, that can reduce radiation doses; (5) the effects of radiation doses on the human body; and (6) economic impacts. Some topics are subject to argument and continuing development since consequence modeling is not a precise science but contains large uncertainties and gaps in knowledge. Where an understanding of the current debate is deemed necessary for a sensible interpretation of the results, a discussion of this debate is included, wind-shift models being a case in point (see Appendix D4). Other areas that are described in some depth are those in which the user's choice of input data can significantly affect the output. A particularly important example is that of evacuation and sheltering (see Appendix E).

Section 9.4 is devoted to the collection and processing of the input data. Section 9.5 is a step-by-step set of procedures for a consequence analysis, starting with the information requirements and ending with the final product, including examples of the results and how they can be displayed. This section also discusses the probabilistic aspects of a consequence analysis in the context of a complete probabilistic risk assessment.

Section 9.6 covers assumptions, sensitivities, and uncertainties. Sections 9.7 and 9.8 describe the methods of documentation and provisions for the assurance of technical quality, respectively. Supporting material is presented in Appendix D, which covers issues in dispersion modeling; Appendix E, which describes evacuation and sheltering; and Appendix F, which discusses liquid pathways.

9.1.2 PURPOSE AND SCOPE OF CONSEQUENCE MODELING

The complete range of calculations carried out in the course of a PRA bridges the gap between the engineering and operations associated with the reactor and the potential risks that the reactor poses to the public. Consequence analysis provides the final link in this chain of calculations and is intended to assess the effect of accidental releases of radionuclides on the surrounding population and the environment.

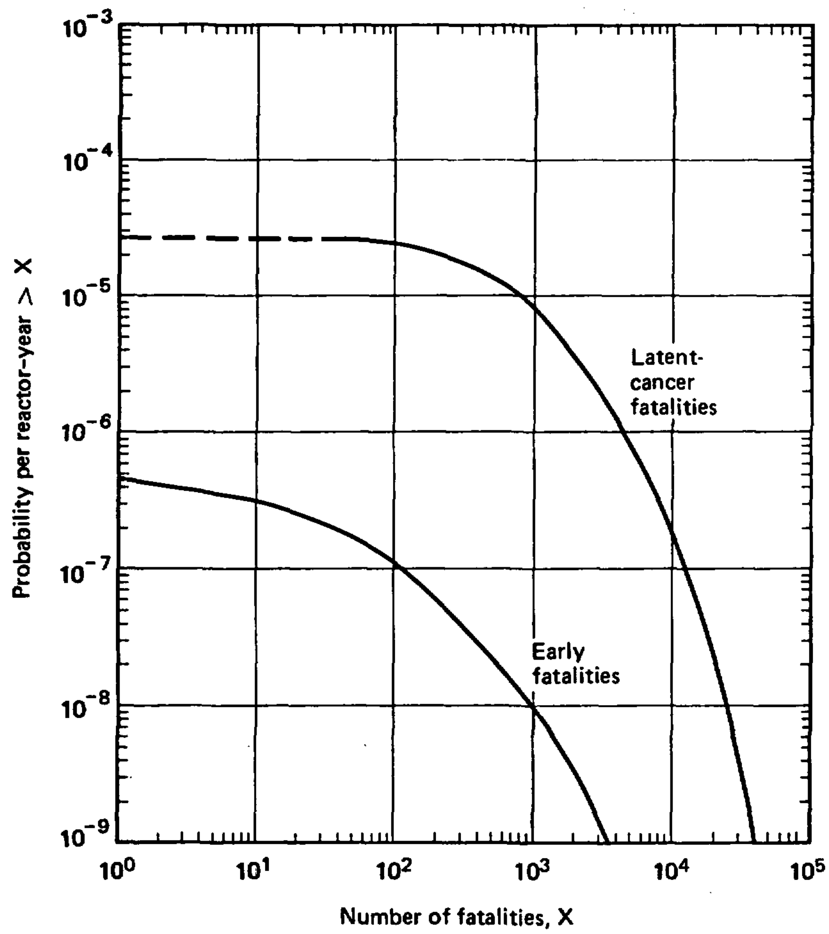
Consequence modeling can therefore be defined as a set of realistic calculations of the ranges (probabilities of occurrence and magnitudes) of adverse impacts that would follow from an accidental release of radionuclides. These adverse impacts, commonly referred to as "public risks," include (1) early and long-term deaths; (2) early and long-term injuries; (3) genetic damage; (4) the contamination of property, land, and water; and (5) economic impacts. These outputs are discussed in more detail in Section 9.5. Consequence modeling provides the means for relating these risks, at both the individual and the societal level, to the characteristics of the radioactive release.

Consequence modeling has many actual or potential applications, including the following examples:

1. Risk evaluation--generic or site specific, societal or individual.
2. Evaluation of alternative design features.
3. Environmental impact assessment.
4. Rulemaking and regulatory procedures.
5. Emergency planning and response.
6. The development of criteria for the acceptability of risk.
7. The provision of focus for research needs.
8. Accident liability.
9. Instrumentation needs and dose assessment.

In the short space allotted to this introduction, it is not possible to describe the application of consequence analysis to each of the topics listed above. The reader may find the examples that follow instructive, however.

<u>Risk Evaluations--Generic or Site Specific</u>. It is usual to present risks as "complementary cumulative distribution functions" (CCDFs), and two examples are given in Figure 9-1, which shows the predicted probability per reactor-year (frequency) with which an accident might occur and cause the deaths of as many as, or more than, the corresponding number of people. These CCDFs, which are probably the most natural form of the output of a



Note: Approximate uncertainties are estimated to be represented by factors of 1/4 and 4 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

Figure 9-1. Frequency distribution for early fatalities and latent-cancer fatalities. From the Reactor Safety Study (USNRC, 1975).

consequence analysis, are taken from the most celebrated of all risk assessments performed to date, the Reactor Safety Study (RSS--USNRC, 1975). The CCDFs themselves can be used as a measure of public or societal risk. Some authors take the integrals under the CCDFs, which are generally approximately equal to the expected (in the statistical sense) number of early or latent fatalities per year, and use these figures as a measure of public risk. The CCDFs and/or the expected values can then be compared with similar quantities for other industrial activities, in order to put them in perspective.

The Reactor Safety Study was a generic study. Numerous site-specific studies are also under way or have recently been completed, such as that for the Limerick site (Philadelphia Electric Company, 1981), the Zion PRA (Commonwealth Edison Company, 1981), and the German Risk Study (Gesellschaft fuer Reaktorsicherheit, 1980; EPRI, 1981).

Rulemaking and Regulatory Procedures. A good example of this use of consequence modeling is the current Sandia Nuclear Power Plant Siting Study (Strip et al., 1981), which is intended to assist the U.S. Nuclear Regulatory Commission (NRC). The NRC is currently revising its regulations on the siting of nuclear power plants and has asked Sandia National Laboratories to provide technical guidance for establishing (1) numerical criteria for the population density and distribution around the sites of nuclear power plants and (2) standoff distances for offsite hazards. In order to provide this guidance, calculations have been performed to address the following questions:

1. What range of risk is associated with currently existing sites?

2. What characteristics of the surrounding population (distance, distribution) influence risk?

3. What impacts do other site and design characteristics have on risk?

4. What is the influence of emergency response on risk?

These calculations included population data for 91 sites at which reactors are operating or under construction and sensitivity analyses to examine the influence on risk of emergency-response alternatives, weather conditions at the site, release characteristics (including release fractions), reactor power level, and reactor design.

A report describing the Sandia Nuclear Power Plant Siting Study in detail should be available soon. It is a good example of a study of its kind because it combines several of the applications of consequence analysis and also contains a sensitivity study.

Further examples of the use of consequence analyses in the regulatory context can be found in the recent series of supplements to the environmental reports for several reactor sites (see, for example, USNRC, 1981). These supplements have been produced by the NRC in order to fulfill its interpretation of the requirements of the National Environmental Policy Act, as put forward in the Commission's Statement of Interim Policy:

> ...Environmental Impact Statements shall include consideration of the site-specific environmental impacts attributable to accident sequences that lead to releases of radiation and/or radioactive materials, including sequences that can result in inadequate cooling of the reactor fuel and to melting of the reactor core. In this regard, attention shall be given both to the probability of occurrence of such releases and to the environmental consequences of such releases.

To implement this policy, the NRC has carried out site-specific probabilistic consequence analyses of Class 9 accidents.

The Sandia Nuclear Power Plant Siting Study and the NRC's applications of consequence modeling in environmental impact statements are examples of uses of consequence analyses that do not fit into the PRA categories defined in Chapter 2. In each of these studies, a generic source term* was used. Thus, there are uses in which a consequence analysis can be, and has been, carried out outside the context of a PRA.

Emergency Planning and Response. Several recent studies have used the RSS consequence model, CRAC, for guidance in emergency planning and response. One such study (Aldrich, McGrath, and Rasmussen, 1978; Aldrich et al., 1978) examined the relative merits of evacuation and sheltering followed by population relocation as protective measures for core-melt accidents, the distances to which (or areas within which) they might be needed, and the time available for their implementation. Partly on the basis of this analysis, the NRC has required the implementation of emergency-planning zones for the plume-exposure pathway, with a radius of approximately 10 miles, for all operating plants in the United States (Collins et al., 1978).

Another study has been performed to provide guidance to policy makers concerning (1) the effectiveness of potassium iodide as a blocking agent in potential reactor-accident situations, (2) the distance to which (or the area within which) it should be distributed, and (3) its relative effectiveness in comparison with other available protective measures (Aldrich and Blond, 1980, 1981). Again, the analysis was performed with the RSS consequence model. The conclusion was that potassium iodide does not appear to be a cost-effective protective measure.

Evaluation of Alternative Design Features. Carlson and Hickman (1978) considered a number of design alternatives for light-water reactors (LWRs): (1) stronger containment, (2) shallow underground siting, (3) deep underground siting, (4) increased containment volume, (5) filtered atmospheric venting, (6) compartment venting, (7) thinned basemat, (8) evacuated containment, and (9) double containment. For each of these alternatives, they carried out a consequence analysis and calculated the integrals under the CCDFs for early fatalities, latent-cancer fatalities, and property damage. These results were then used as a basis for estimating the cost effectiveness of each design alternative.

The list of the uses of consequence analysis given above, together with the examples that follow it, should give the reader a good idea of the range of applications of a consequence analysis.

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*In the Sandia study, there are five source terms ranging from a gap-activity release to a core melt with a large radionuclide release directly to the atmosphere (Aldrich et al., 1981a). These source terms were developed by the NRC specifically for the siting study. In the consequence analyses for the environmental impact statements, the NRC used the "rebaselined" pressurized-water reactor (PWR) or boiling-water reactor (BWR). These are essentially representations of the Surry PWR or the Peach Bottom BWR, which were the reactors analyzed in the Reactor Safety Study, with some modifications to account for calculations and reanalyses carried out since the report of the Study was written. Each environmental impact statement contains an appendix describing the appropriate rebaselined reactor.

9.2 OVERVIEW

There are a number of tasks involved in a consequence analysis, as outlined below.

1. Acquiring background. A beginner must first acquaint himself with what is typically done in a consequence analysis and with the various codes that are available. Once this has been done, he can make an intelligent choice of code for his own use.

2. Deciding on the purpose of the analysis. This decision is important. It influences the choice of code, the requirements for input data, and the choice of output.

3. Choosing a computer code for consequence modeling.

4. Computer-code debugging and/or modification. The purpose of the calculations may require modifications to the code.

5. Collecting input data. For the user of consequence-modeling codes, this is the most important task in the analysis. It offers him the chance to make a significant impact on the results of the calculations.

6. Exercising the code. In principle, this is straightforward, assuming that the ground has been well prepared by the conscientious performance of the earlier tasks. This task also includes any sensitivity studies that may be carried out as part of an uncertainty analysis.

7. Report writing and interpretation of results.

The experience of the consequence modeler will, of course, determine which of these tasks he needs to do. A complete beginner would start with task 1. An experienced member of a PRA team would need to carry out task 2, deciding on the purpose of a consequence analysis, but could then begin with task 5, the collection of input data.

9.2.1 TASK 1: BACKGROUND STUDY

As an introduction to the subject, Appendix VI of the Reactor Safety Study contains a comprehensive survey of all of the essential elements. In order to understand the meteorological modeling, Meteorology and Atomic Energy--1968 (Slade, 1968) is a thorough review that will be shortly updated and retitled Atmospheric Science and Power Production. As examples of the use of existing codes in recent risk assessments, it is instructive to review the Limerick study (Philadelphia Electric Company, 1981), the German Risk Study (Gesellschaft fuer Reaktorsicherheit, 1980; an English translation has been prepared by EPRI, 1981), and the Zion study (Commonwealth Edison Company, 1981).

Figure 9-2 gives a schematic outline of the arrangement of the computational elements or submodels of a typical consequence-modeling code. Most codes are made up of similar elements. Each of the submodels is discussed briefly below, drawing heavily on material contained in the Overview of the Reactor Safety Study Consequence Model (Wall et al., 1977).

9.2.1.1 Description of Radionuclide Release

The calculation begins with a description of the characteristics of the radionuclide release, including the quantity of each radionuclide released to the environment, the amount of energy associated with the release, the duration of the release, the time of the release after accident initiation, the warning time for evacuation, and the frequency of occurrence predicted for the accident. An example of this kind of input data, generated by the engineering analysis of the PWR and BWR reactors examined in the Reactor Safety Study, appears in Table 9-1. This input is discussed more fully in Section 9.4.2.

9.2.1.2 Atmospheric Dispersion and Weather Data

Most consequence-modeling codes simulate the atmospheric dispersion of the released radioactive material by using a Gaussian dispersion model to calculate ground-level instantaneous and time-integrated airborne concentrations and deposited levels of radioactivity. This is done as a function of time and of distance from the reactor. In the consequence-modeling codes available in the United States, the Gaussian model is generally used in such

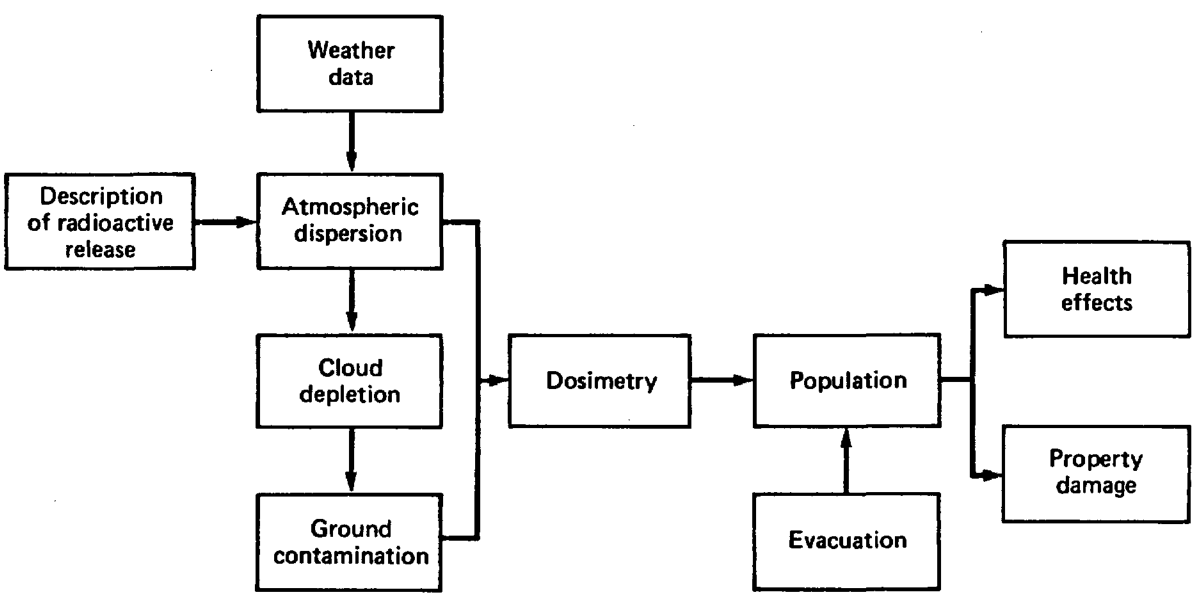

Figure 9-2. Schematic outline of a typical consequence model. From Wall et al. (1977).

Table 9-1. Summary of RSS release categories for hypothetical accidents[a]

Release category[d]	Probability per reactor-yr	Time of release (hr)	Duration of release (hr)	Warning time for evacuation (hr)	Elevation of release[b] (meters)	Energy of release (10^6 Btu/hr)	Fraction of core inventory released[c]						
							Xe-Kr	I[e]	Cs-Rb	Te-Sb	Ba-Sr	Ru[f]	La[g]
PWR-1	9×10^{-7}	2.5	0.5	1.0	25	20 and 520[h]	0.9	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR-2	8×10^{-6}	2.5	0.5	1.0	0	170	0.9	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR-3	4×10^{-6}	5.0	1.5	2.0	0	6	0.8	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR-4	5×10^{-7}	2.0	3.0	2.0	0	1	0.6	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR-5	7×10^{-7}	2.0	4.0	1.0	0	0.3	0.3	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR-6	6×10^{-6}	12.0	10.0	1.0	0	NA	0.3	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR-7	4×10^{-5}	10.0	10.0	1.0	0	NA	6×10^{-3}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR-8	4×10^{-5}	0.5	0.5	NA[i]	0	NA	2×10^{-3}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR-9	4×10^{-4}	0.5	0.5	NA	0	NA	3×10^{-6}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR-1	1×10^{-6}	2.0	0.5	1.5	25	130	1.0	0.40	0.40	0.70	0.5	0.5	5×10^{-3}
BWR-2	6×10^{-6}	30.0	3.0	2.0	0	30	1.0	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR-3	2×10^{-5}	30.0	3.0	2.0	25	20	1.0	0.10	0.10	0.03	0.01	0.02	4×10^{-3}
BWR-4	2×10^{-6}	5.0	2.0	2.0	25	NA	0.6	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1 x 10-4
BWR-5	1×10^{-4}	3.5	5.0	NA	150	NA	5×10^{-4}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0

[a]From Wall et al. (1977).

[b]A 10-m elevation is used in place of zero representing the midpoint of a potential containment break. Any impact on the results would be slight and conservative.

[c]Background on the isotope groups and release mechanisms is presented in the Reactor Safety Study, Appendix VII (USNRC, 1975).

[d]The definition of release categories is discussed in Section 9.4.2.9.

[e]Organic iodine is combined with elemental iodine in the consequence calculations. Any error is negligible since the release fraction of organic iodine is relatively small for all large release categories.

[f]Includes Ru, Rh, Co, Mo, Tc.

[g]Includes Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm.

[h]Accident sequences within the PWR-1 category have two distinct energy releases that affect consequences. The PWR-1 category is subdivided into PWR-1A, with a probability of 4×10^{-7} per reactor-year and an energy of release of 20×10^6 Btu/hr; and PWR-1B, with a probability of 5×10^{-7} per reactor-year and an energy of release of 520×10^6 Btu/hr.

[i]Not applicable.

a way as to allow changes in atmospheric stability, wind speed, and precipitation for each successive hour of travel time. Some codes also allow the wind direction to change. The hourly weather data required as input are usually generated by processing data collected at the reactor site itself or at nearby weather stations.

In general, consequence-modeling codes simulate the behavior of the radioactive plume as it travels tens or hundreds of kilometers downwind--that is, for many hours during which the weather conditions may change. In principle, there may be a different sequence of hourly weather changes for each of the 8760 hours during a full year at which the accident might take place. In practice, it is usually prohibitively expensive to run each of these sequences in turn, and some method must be devised for selecting a sample. In some codes this can be done randomly, in others by selecting starting times that are equally spaced throughout the year. Another possibility is to first combine the weather sequences into groups in which the pattern of hourly weather changes is similar and then to ensure that the sampling process covers all of the groups. This question of how best to sample weather data is important and is discussed more fully in Appendix D4.1.2. Some other important issues, such as the differences between codes that do or do not allow changes in wind direction as the plume travels downwind, are also addressed in Appendix D4.

The basic Gaussian model is modified to take into account a number of phenomena. Among them are radioactive decay and daughter buildup, which are treated in ways that can be found in any standard textbook. Allowance is usually made for the mixing of the radioactive plume as it emerges into the turbulent wake of the reactor building. The atmospheric boundary layer, which is the layer of turbulent air adjacent to the surface of the earth, is almost always capped by an overhead inversion, which is a layer of very stable air that acts as an effective barrier to the upward dispersion of the plume. The height of the base of this layer, often termed "the inversion lid," depends on several phenomena, including the intensity of turbulence in the layer of air beneath it, which in turn depends on the time of day and the wind speed. Methods of treating the inversion lid as a function of time can become quite sophisticated. (See Appendix D4 for a further discussion.)

If the plume is buoyant, it is allowed to rise according to standard procedures available in the literature. Some codes allow the plume to penetrate the inversion lid.

9.2.1.3 Deposition--Ground Contamination

As the plume of radioactive material travels outward from the reactor, various mechanisms remove the airborne material. In addition to radioactive decay, the radioactive material is removed by such deposition processes as impaction on obstacles (dry deposition) and by precipitation scavenging (wet deposition).

These deposition mechanisms cannot be specified precisely. There are significant dependences of removal rates on, among other things, the type and rate of precipitation, particle density and size distribution, the surface characteristics of the ground, and weather conditions. For simplicity, the dry-deposition velocity (ratio of the deposition flux to the air concentration at a particular distance from the surface) is assumed to be constant for particulate matter. When it rains or snows, wet deposition occurs simultaneously with dry deposition. Wet deposition is modeled by a simple exponential removal rate, which should be dependent on the rate of rainfall. When the occurrence of precipitation is specified by the weather data, it is assumed to occur uniformly within time and throughout the spatial interval in which the plume is located. The removal rate is a function of the thermal stability. The noble gases are assumed to be insoluble and nonreactive, and therefore are not removed by either dry or wet deposition.

The ground concentration is calculated from the air concentration and the deposition rate. The material deposited on the ground is subtracted from the airborne material.

Both dry deposition and wet deposition are still matters of considerable discussion among consequence modelers. Dry deposition is discussed more fully in Appendix D3.

9.2.1.4 Processes That Lead to the Accumulation of Radiation Doses (Dosimetry)

Using the procedures described above, for each selected accident starting time, spatial distributions of instantaneous and time-integrated airborne concentrations and deposited levels of radioactive material are estimated. These quantities are then used to calculate the potential radiation doses that would be received by individuals and populations--doses that could be accumulated in a number of ways. Figure 9-3 shows some of these possible pathways by which radioactivity could reach people. (This figure is not intended to be a comprehensive summary.) It is convenient to classify the exposure pathways as those associated with the passing cloud and those associated with ground contamination.

The airborne radioactive material leads to radiation doses caused by external radiation from the plume ("cloudshine") and radiation from inhaled radionuclides. To receive the external radiation, a person must be either immersed in the plume or in its general vicinity. The consequence-modeling code relates the concentration of radioactive material in the air to an external dose delivered to various body organs (e.g., bone marrow, gastrointestinal tract).

The radiation dose from inhaled radioactive material is proportional to the exposure to the airborne concentration of radionuclides at roughly 2 meters above the ground and to the individual's breathing rate. The dosimetric model used to derive the dose-conversion factors describes the time-dependent movement of the radioactive material within the body. An important element of the model from which the data used in most consequence-analysis codes are derived is the well-known lung model of the International Commission for Radiological Protection (ICRP, 1966).

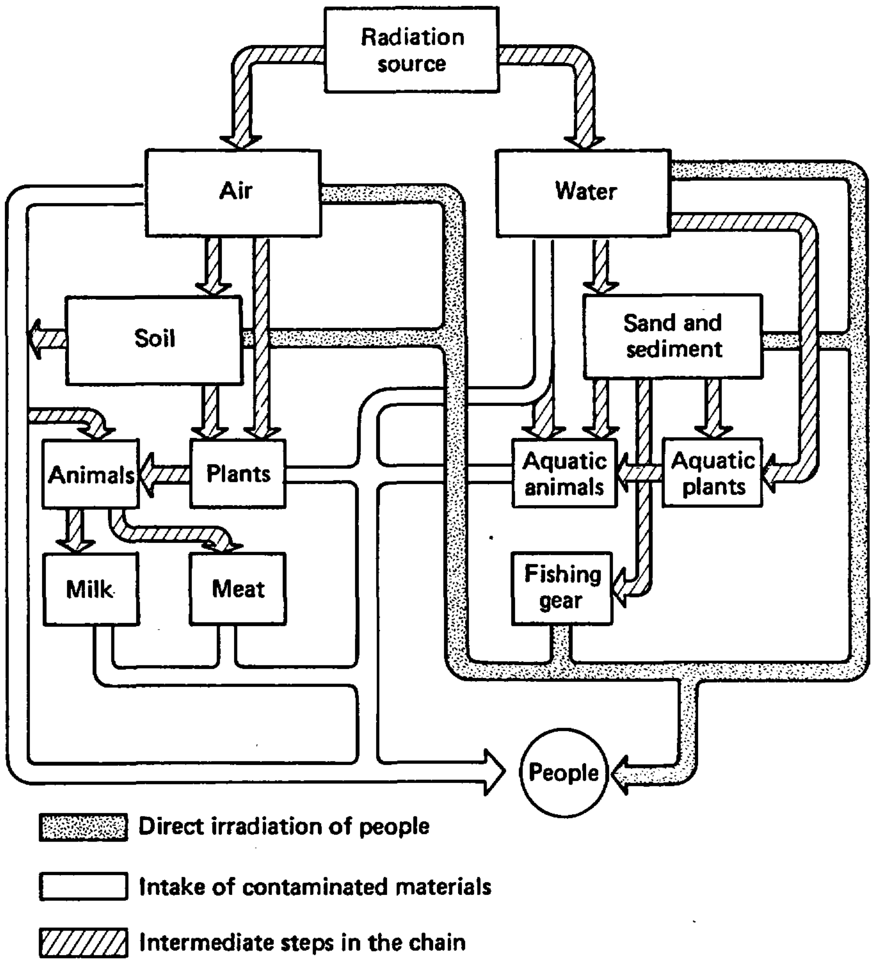

Figure 9-3. Examples of radiation pathways. From Safety and Nuclear Power, United Kingdom Atomic Energy Authority, London, England.

In essence, the radioactive material deposited on the ground delivers radiation doses through three pathways: external irradiation due to gamma rays emitted by deposited material ("groundshine"), the inhalation of resuspended radioactive material, and the ingestion of contaminated food and water. The ingestion of the radioactive material may result from direct deposition onto vegetation, which is consumed by people or by animals furnishing food for people, or from the more indirect pathways involving the uptake of ground-deposited radioactive material through the roots of plants.

9.2.1.5 Population Distribution

Once the radiation doses delivered to individuals have been calculated, they must be combined with the population distribution. In general, consequence models assign the population to a grid consisting, first, of a number of sectors. Within each sector, radial intervals are defined. The population within a sector and between two radial intervals is effectively assumed to be uniformly distributed. Population data are usually obtained by processing U.S. Census data, carrying out house counts, and examining aerial photographs. Some users extrapolate the census data to plant midlife.

9.2.1.6 Evacuation and Other Measures That Reduce Radiation Doses

Evacuation is the expeditious movement of people to avoid or reduce immediate exposure to the passing cloud. It is in the choice of such parameters as the delay time (the period between the declaration of a general emergency by the plant emergency director and the time at which evacuation actually begins) and the effective evacuation speed that the user can profoundly influence the results of his calculations. This is particularly true of the predicted numbers of early fatalities and early injuries, which are very sensitive to the radiation dose accumulated through exposure to gamma rays emitted by deposited fission products during the first few hours after the accidental release of radioactivity has taken place.

It is very important that the evacuation model be sensibly handled by the user of the code. For this reason, Appendix E presents a thorough description of some evacuation models and an in-depth discussion of the input-data requirements.

It is assumed in consequence modeling that people will take advantage of structures in the neighborhood of reactors in order to shelter from external irradiation by gamma rays. Gamma rays emitted by the passing cloud (cloudshine) are attenuated by, for example, the walls of buildings; gamma rays emitted by deposited radioactive material (groundshine) are attenuated both by buildings and by surface rugosities. Consequence-modeling codes require shielding factors for both cloudshine and groundshine for people assumed to be using shelters. Also required may be shielding factors for people waiting to evacuate, people evacuating, and people behaving normally. Appendix E explains how to calculate such shielding factors.

Another measure that can be used to reduce radiation doses is relocation. This is the permanent or long-term removal of people from a contaminated area in order to reduce the radiation dose accumulated by long-term exposure to the deposited radioactive material.

The countermeasures treated in many consequence-modeling codes also include interdiction and decontamination. The radioactive contamination of a large area may result in the contamination of milk produced by cattle grazing on contaminated pastures, in the external contamination of crops, and/or in excessive radiation doses to people. In such events, the milk and crops may be impounded and/or the people relocated for a period of time. All of these actions are called "interdiction."

The interdiction model is based on the concept of maximum acceptable doses. The dose criteria used in the Reactor Safety Study (USNRC, 1975) were based on the recommendations of the U.S. Federal Radiation Council (FRC, 1965) and the British Medical Research Council (MRC, 1975).

The dose criteria are translated into corresponding contamination levels (curies per square meter) of different radionuclides by dosimetric models like those described in the Reactor Safety Study and an environmental model that incorporates the grass-cow-man or soil-root-crop-man pathways. Since the milk interdiction level is the most restrictive, the area over which milk would be impounded would be the largest. Conversely, the interdiction level for human occupancy is the least restrictive, and therefore the area from which people would be relocated would be the smallest. The "weathering" of deposited radionuclides is also incorporated so that the interdiction distance slowly moves toward the reactor.

Decontamination is defined as the cleanup and removal of radionuclides (see Section 9.3.4.4). A measure of the effectiveness of decontamination operations is the decontamination factor--that is, the original concentration of the contaminant (in curies per square meter) divided by its concentration after decontamination. The decontamination model is illustrated in Figure 9-4. Without decontamination, the interdiction criterion translates to a distance R_1. With a maximum decontamination factor of 20, the land area between R_1 and R_2 will become available for reoccupation. Subsequent weathering of the radionuclides will reopen the land area between R_2 and R_3.

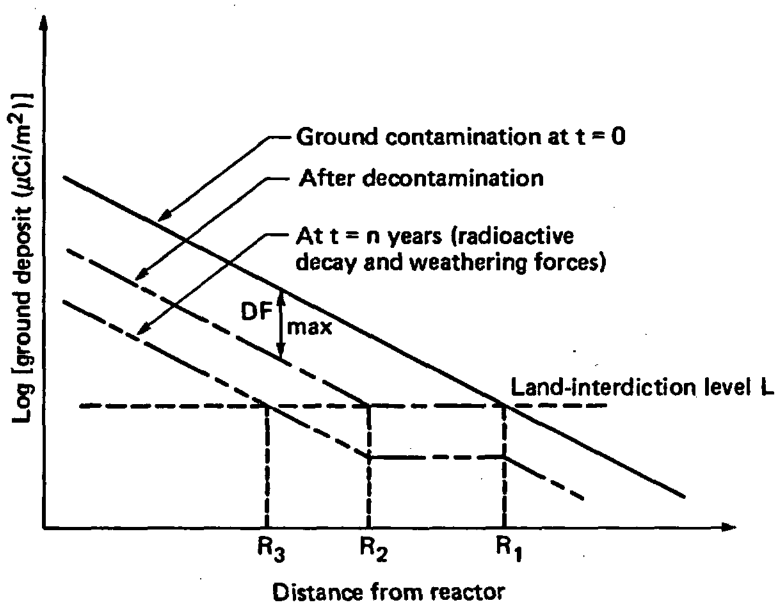

Figure 9-4. Illustrative decontamination model for ground-level releases. From Wall et al. (1977).

9.2.1.7 The Effect of Radiation on the Human Body (Health Effects)

Three categories of potential health effects may be calculated: early and continuing somatic effects, late somatic effects (cancers), and genetic effects. Early and continuing somatic effects manifest themselves within days up to a year after exposure. By contrast, latent cancers would probably be observed from at least 2 to 40 years after exposure and genetic effects in succeeding generations.

Early and continuing fatalities may result from radiation damage to the bone marrow, the lung, or the gastrointestinal tract. Past studies indicate that radiation damage to the bone marrow is the most important contributor, λ, given the inventory of radionuclides likely to be released to the atmosphere in the event of an accident in a light-water reactor. The relationships between the radiation dose to these various organs and the probability of death (dose-risk or dose-response relationships) are discussed in Section 9.3.5 and reviewed in depth in Appendix VI of the Reactor Safety Study. The RSS also estimated the number of prenatal deaths and of early injuries, including hypothyroidism, temporary sterility, congenital malformations, growth retardations, cataracts, and prodromal vomiting. In general, it is not necessary to consider early effects in such detail. The predicted numbers of early fatalities and injuries are usually sufficient to give an adequate notion of the public risks associated with early effects.

Late somatic effects consist of latent-cancer fatalities, nonfatal cancers, and benign and cancerous thyroid nodules. After the irradiation of a large number of people, there is generally a latent period during which no increase in cancer incidence is detectable. After this period, the radiation-induced cancers tend to appear at an approximately uniform rate for a period of years, which is termed the "plateau." The plateau period could in some cases extend over the lifetime of the individual. The dose-response relationships for cancer induction are discussed in Section 9.3.5 and in Appendix VI of the Reactor Safety Study.

9.2.1.8 Economic Costs (Property Damage)

Property damage after a postulated reactor accident is not of the same nature as that resulting from most other potential catastrophic events (i.e., there is no physical damage to offsite property). The damage arises from contamination with radioactive material and the possible radiation dose that could be received if the property were used in its intended manner. The restriction in the use of the property results in economic loss.

The components of property damage, as assumed and modeled in the consequence model, are evacuation costs, loss of agricultural products, decontamination costs, and population-relocation costs. The main problem is to ensure that realistic figures are used for the various elements of these costs (see Section 9.4).

9.2.2 TASK 2: DECIDING ON THE PURPOSE OF THE CONSEQUENCE CALCULATIONS

Before embarking on the choice of a consequence-modeling code and its use, it is necessary to take some time to think of the output that is required and the purpose for which it is to be used. This influences, for example, which code is to be used and what input data are required. Some examples are as follows:

1. In the recent Zion study (Commonwealth Edison Company, 1981), it was deemed sufficient to calculate CCDFs for early fatalities and

injuries, latent-cancer fatalities, and the population dose (man-rem). This removed the requirement for the collection of data pertaining to economic costs.

2. A study like Sandia's Nuclear Power Plant Siting Study (Strip et al., 1981) would require the handling of large amounts of meteorological and population data from a considerable number of sites.

3. The NRC has begun to publish supplements to the environmental impact statements for various reactors in response to the requirements of the National Environmental Policy Act (NEPA). These supplements contain fairly stylized calculations. Any utility that wishes to respond to the requirements of NEPA would presumably deem it sufficient to do similar calculations.

4. Organizations wishing to carry out a detailed analysis of evacuation procedures, taking into account existing road networks, might consider looking at a code that is capable of mapping the road network.

The above examples demonstrate the importance of having a clear idea of the purpose and the required output of a consequence analysis at a very early stage.

9.2.3 TASK 3: CHOICE OF CODE FOR CONSEQUENCE MODELING

The potential user of consequence models may wish to be told categorically that code X is manifestly the best that is available and should be used in preference to all others. Unfortunately, consequence modeling is as much an art as a science. There are large gaps in knowledge that can be filled only by the judgment of the modeler. The area is still being developed, with new and promising changes to codes. Often it is how intelligently the code is used, rather than which particular one, that determines whether the results are meaningful or not.

In the United States, there are four codes that can be used for a complete consequence analysis. The reader may be inclined to object that there are many more than four such codes. To be precise, there are four codes that both contain all of the necessary elements of a consequence model and perform the probabilistic manipulations that are necessary for the calculation of CCDFs. Other codes may contain many excellent and sophisticated features, but they are not fully developed consequence-modeling codes. The four in question are CRAC, the code used during the RSS, and three offshoots, CRAC2, CRACIT, and NUCRAC.

CRAC. The code CRAC (Calculation of Reactor Accident Consequences) was developed for the Reactor Safety Study (USNRC, 1975, Appendix VI). It was the first code to integrate all of the elements of a consequence model into a package capable of generating CCDFs and contains what were, at the time, innovative features (in the context of consequence modeling), such as the treatment of changing weather conditions and the incorporation of chronic pathways.

CRAC2. The CRAC2 code is a revision of CRAC (Ritchie et al., 1981a). Recently issued by Sandia National Laboratories, it incorporates significant improvements in the area of weather-sequence sampling (Ritchie et al., 1981b) and emergency response (Aldrich et al., 1978; Aldrich, Ritchie, and Sprung, 1979).

Weather data are normally collected at reactor sites at hourly intervals. Weather sequence sampling is the selection of a limited number of starting times for accident sequences from the 8760 that are possible in a full year, in order to reduce computing time. CRAC employs a stratified sampling technique whereby weather sequences are selected every 4 days plus 13 hours to cover diurnal, seasonal, and 4-day weather cycles. In this manner, 91 sequences are chosen to represent a year of data. Sensitivity studies performed with CRAC indicate considerable uncertainty in the predicted results, attributable to sampling by this method.

CRAC2 uses a new weather-sequence sampling method that greatly reduces the uncertainty attributable to sampling. Before sampling sequences, the entire year of data is sorted into 29 weather categories, or bins. Categories include sequences in which either rainfall or wind-speed slowdowns occur within specified distance intervals from the plant. Atmospheric-stability and wind-speed categories are also considered. The probability of each weather category is estimated from the number of sequences in the category. Sequences are then sampled from each of the 29 categories (and weighted with appropriate probabilities) for use in risk calculations, thus ensuring that low-probability adverse weather conditions (e.g., rainfall, wind-speed slowdowns) are adequately included.

The emergency-response model in CRAC2 is considerably more realistic than that in CRAC. In CRAC, evacuation was assumed to commence immediately upon warning and to proceed at a very slow speed. Any person overtaken by the plume was assumed to be exposed to the full extent of the plume and to receive a 4-hour ground dose. In contrast, the CRAC2 model includes a delay time between warning and the start of evacuation, more reasonable evacuation speeds, and an explicit calculation of the time during which people are exposed to airborne and deposited radionuclides. The revised model also allows the user to consider population sheltering.

A number of refinements in the calculation of plume rise, washout, and atmospheric dispersion were also incorporated into CRAC2.

CRACIT. Developed by Pickard, Lowe and Garrick, Inc., CRACIT (CRAC Including Trajectories) incorporates major modifications in the atmospheric-dispersion and evacuation models that permit some unique features of a site to be considered (Woodard and Potter, 1979; Commonwealth Edison Company, 1981). The atmospheric-dispersion model in CRACIT used the "modified potential flow" (MPF) method developed by Lantz and Coats (1971). The MPF method incorporates the effect of site-specific topographic features by using digitized terrain data to calculate a temporally and spatially dependent wind field. Using the calculated wind field, CRACIT solves the set of transport and diffusion equations by numerical methods and is thus more realistic than the Gaussian plume model. In CRACIT, the numerical solution is used only to a maximum distance of 14.5 km; the model used beyond this distance is a segmented Gaussian-plume model that incorporates changes in wind direction by changing the trajectory of the plume.

The evacuation model in CRACIT takes into consideration the likely evacuation routes at a site (the CRAC and CRAC2 models assume evacuees move radially away from the reactor) as well as traffic jams that may occur during an evacuation. CRACIT also contains a number of additional refinements in the calculation of atmospheric dispersion, plume rise, washout, and interactions between the plume and the inversion layer.

Because it calculates a three-dimensional wind field, performs numerical dispersion calculations, and incorporates an actual road network into the evacuation model, CRACIT requires considerably more input data and computation time than does CRAC.

NUCRAC. NUCRAC, developed by Science Applications, Inc., incorporates major modifications in two areas: plume depletion by dry deposition and chronic-exposure pathways (Kaul et al., 1980; Kaul, 1981a). The model allows for a distribution of particle sizes in the material released from the containment. Dry deposition is modeled with Overcamp's (1976) surface-depletion method, which takes into account the gravitational settling of particles on the basis of particle size. NUCRAC, however, does not currently consider plume depletion by wet deposition. The improved model of chronic-exposure pathways in NUCRAC treats a larger number of radionuclides and better reflects the site-specific details of agricultural production.

Other Codes. A document that will discuss the full range of consequence models available worldwide and their capabilities is the forthcoming report of the International Benchmark Comparison of Reactor Accident Consequence Models,* henceforth referred to as the "Benchmark." This document, when it becomes available, should be made required reading for all would-be users of consequence models, particularly those who wish to interpret the results generated by consequence-modeling codes and to assess the impact of uncertainties. Examples of other consequence-modeling codes are TIRION, developed by the United Kingdom Atomic Energy Authority (Kaiser, 1976; Fryer and Kaiser, 1979), ALICE, developed by the French Commissariat à l'Energie Atomique (Maire et al., 1981) and the Finnish code ARANO (Nordlund et al., 1979).

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*The Benchmark exercise is being carried out under the aegis of the Committee for the Safety of Nuclear Installations (CSNI), Nuclear Energy Agency, Organization for Economic Cooperation and Development. The exercise has consisted of the definition of a number of standard problems and their solution by some 20 participants from Europe, Japan, and the United States, using their own consequence models. The results are to be presented in a forthcoming report together with interpretation by various problem coordinators. Preliminary presentations on the activities of the Benchmark are given by Blond et al. (1981) and Aldrich et al. (1981b). Detailed specifications of the Benchmark problems, including site and release characteristics, are available on request from Division 9415, Sandia National Laboratories, Albuquerque, N.M. 87185.

The user should be aware that the state of the art of consequence modeling is one of change. He should always be on the lookout for improvements because models are continually being updated.

9.2.4 TASK 4: CODE DEBUGGING AND MODIFICATION

Once the code has been obtained, the tedious but necessary process of debugging it and making sure that it can run on the user's machine must be undertaken. At the same time, any necessary modifications to the code should be carried out--modifications designed to produce additional output data, for example.

9.2.5 TASK 5: COLLECTION OF INPUT DATA

Consequence-modeling codes are elaborate and require what seem to be endless amounts of input data. This is where the user can have a considerable impact on the results, and his choice of certain inputs will determine whether the results are meaningful or not. An example that has already been mentioned is the choice of delay time for evacuation.

In general, it is the user's responsibility to collect and process data in some or all of the following areas: (1) input from the calculations of radionuclide release and transport (e.g., magnitude, duration and rate of release, energy of release, frequencies); (2) population and meteorological data; (3) economic data; (4) health-physics data; (5) emergency-response information; and (6) criteria for interdiction and decontamination. Section 9.4 describes where to obtain such data and how to process it. It is to be emphasized that data collection is a time-consuming procedure and must be started at an early stage in a consequence analysis, considerably in advance of running the code.

9.2.6 TASK 6: EXERCISING THE CODE

This is usually the most straightforward part of the consequence-analysis calculation. It is necessary to carry out runs of the code for the various cases needed to generate the required CCDFs or other results and to repeat some of the runs for changes in some of the parameters (e.g., evacuation speed, deposition velocity) for which sensitivity studies are thought to be desirable. These sensitivity studies may be used as the basis for an uncertainty analysis.

9.2.7 TASK 7: REPORT WRITING AND INTERPRETATION OF RESULTS

Once the results have been completed, it is necessary to describe what has been done, perhaps in the form of a report like that outlined in Section 9.7. Included in the report will be the display and interpretation of the results, as described in Section 9.6.

9.3 METHODS

This section describes some of the more common methods used in consequence modeling.

9.3.1 RADIONUCLIDE TRANSPORT AND DIFFUSION

According to Figure 9-2, the first step in the chain of calculations is the release of radioactivity into the atmosphere or into water. The atmospheric pathway is generally the most important in the case of nuclear reactors, but there are postulated circumstances--for example, core melt-through into an aquifer, or the release of sumpwater--in which the water pathway should be considered. In general, the water pathway is not treated on the same footing as the atmospheric pathway in consequence modeling, because it can be shown to be a less significant contributor to the magnitude of the predicted consequences. Liquid-pathway modeling has been examined in two recent reports (USNRC, 1978; Niemczyk et al., 1981). The state of the art of water-pathway modeling is summarized in Appendix F, but the subject is not treated further in this section.

9.3.1.1 The Gaussian Plume Model

The most commonly used model of atmospheric dispersion in consequence-modeling codes is the Gaussian one. Appendix D shows that this popularity arises for the following reasons:

1. Economical use of computer time.

2. General lack of availability of the meteorological parameters necessary for input to more complicated models.

3. Evidence that in some circumstances, such as dispersion over flat terrain, the results do not differ sufficiently from those of more-complicated models to make the use of the latter worthwhile in analyses that require repeated use of the meteorological model.

The conventional Gaussian formula for the time-integrated concentration χ at the point (x,y,z) is

$$\chi(x,y,z) = \frac{Q \exp\left[-y^2/2\sigma_y^2(x)\right]}{2\pi\sigma_z(x)\ \sigma_y(x)\bar{u}} \left\{ \exp\left[-\frac{(z+h)^2}{2\sigma_z^2(x)}\right] + \exp\left[-\frac{(z-h)^2}{2\sigma_z^2(x)}\right] \right\} \quad \text{Ci-sec/m}^3 \qquad (9\text{-}1)$$

where the pair of exponentials summed inside the braces expresses the fact that total reflection at the ground has been assumed. The symbols are defined as follows:

Q = the total amount of effluent emitted (curies).

h = the height of the source (meters).

$\sigma_z(x)$, $\sigma_y(x)$ = the vertical and horizontal standard deviations (meters), respectively.

x = the distance downwind (meters).

y = the distance across wind (meters).

z = the height above the ground (meters).

$\bar{u}$ = the mean wind speed (m/sec).

Since the wind speed varies with height, it is not possible to define $\bar{u}$ unambiguously. In many experiments, $\bar{u}$ is the wind speed at the height of the source or at the height of a nearby tower. Smith and Singer (1965) show that a reasonable estimate of $\bar{u}$ is obtained by calculating the wind speed at a height $0.62\sigma_z(x)$. This conclusion, however, is model dependent. Unless otherwise stated, it is assumed in this chapter that $\bar{u}$ is the wind speed measured at a height of 10 meters, that is, $\bar{u} = \bar{u}(z = 10) = \bar{u}(10)$. Clarke et al. (1979) say that the product $\sigma_y\bar{u}$ tends to remain constant with in creasing height, so that it is acceptable to use $\bar{u}(10)$, provided that appropriately measured σ_y values are also taken. Some computer codes (e.g., CRACIT) allow $\bar{u}$ to change with height.

The use of the Gaussian model can be justified in a qualitative way by appealing to the random properties of atmospheric turbulence. A small particle of radioactive material, while being carried downwind at the mean wind speed, is also thrown about at random by the turbulent forces acting upon it; that is, it can be regarded as taking a random walk. As is well known, the distribution of a large number of such particles, each of which has taken a large number of random steps, can be described by the Gaussian formula.

Some authors approximate the lateral spreading of diffusing plumes by a "top-hat" distribution. In the Reactor Safety Study (USNRC, 1975, Appendix VI), the quantity

$$\frac{1}{\sqrt{(2\pi)}\ \sigma_y} \exp\left(-\frac{y^2}{2\sigma_y^2}\right) \qquad (9\text{-}2)$$

in Equation 9-1 is replaced by

$$(3\sigma_y)^{-1} \quad \text{for} \quad -1.5\sigma_y \le y \le 1.5\sigma_y \qquad (9\text{-}3)$$

The method has been refined by the authors of the German Risk Study, who used a cross-plume profile with four distinct steps (Aldrich, Bayer, and Schueckler, 1979), which is therefore more nearly akin to the true Gaussian shape.

9.3.1.2 The Dispersion Parameters $\sigma_z(x)$ and $\sigma_y(x)$: Stability Categories

The quantities $\sigma_z(x)$ and $\sigma_y(x)$ have been adjusted by various authors to make the Gaussian distribution fit the measured data. The many possible parametrizations have been reviewed by Gifford (1976). In general, there are two important considerations: the dependence of σ_z and σ_y on the degree of atmospheric instability (or equivalently the intensity of turbulence in the boundary layer) and whether the sigmas are best described as functions of travel time t or of travel distance x. Since most consequence-modeling codes require the user to input stability categories that he has to define, it is worth going over their definition in some detail.

A widely used system for turbulent-diffusion typing was originally proposed by Pasquill (1961), who presented information on the lateral spreading ϕ and the vertical spreading H of diffusing plumes. The latter was shown as a graph and the former as a table. Both are functions of six atmospheric-stability classes, A through F, varying from the "extremely unstable" category A--that is, rapid diffusion--to the "stable" category F, with relatively slow diffusion. The stability category is chosen by reference to a table (see Table 9-2) that defines these categories in terms of the observed wind speed, cloud cover, and insolation conditions--quantities that are widely and routinely observed throughout the world. The values of H and ϕ can be converted into families of curves of the plume standard deviations σ_z and σ_y (Gifford, 1961).

Pasquill's stability categories were chosen subjectively (Gifford, 1976); however, they are approximately linearly related to the intensity of turbulence (Luna and Church, 1974). Ideally, the definition of stability categories should be based on quantities directly related to

Table 9-2. Meteorological conditions defining Pasquill turbulence types

Surface wind speed (m/sec)	Daytime insolation			Nighttime cloudiness[a]	
	Strong	Moderate	Slight	$\geq 4/8$	$\leq 3/8$
<2	A	A-B	B	--	--
2	A-B	B	C	E	F
4	B	B-C	C	D	E
6	C	C-D	D	D	D
6	C	D	D	D	D

[a]The fraction of the sky covered by clouds.

turbulence intensity. Such a quantity is σ_θ, standard deviation of a horizontal wind-direction trace (Singer and Smith, 1966). Indeed, the NRC, in Regulatory Guide 1.23 (USAEC, 1972), has recommended the use of bands of σ_θ for defining stability categories, and these are displayed in Table 9-3. However, a workshop held by the American Meteorological Society (AMS, 1977) did not recommend the use of σ_θ as the basis for determining the vertical standard deviation σ_z. It appears that the method needs further refinement before it can be easily applied (Sedefian and Bennett, 1980).

Another typing scheme recommended by the NRC in Regulatory Guide 1.23 is the ΔT method, which was also used in the Reactor Safety Study. It directly relates the stability category to the value of the atmospheric temperature gradient dT/dz, as shown in Table 9-3. This is an attractive scheme from the user's point of view because, in general, the values of dT/dz can easily be estimated from measurements of the temperature difference ΔT between two points on the meteorological tower. The reliability of the ΔT method has been questioned by several authors (Weber et al., 1977; Sedefian and Bennett, 1980). Vogt et al. (1978) have proposed a method for determining turbulence regimes on the basis of both wind speed and ΔT, parameters that are usually available at reactor sites. Table 9-4 gives an example of such a scheme, developed at Sandia National Laboratories for use in consequence models. Appendix D1 gives reasons why the scheme in Table 9-4 is recommended for consequence modelers. The paper by Gifford (1976) is recommended as a comprehensive review of turbulent-diffusion typing schemes.

9.3.1.3 Parametrizations of σ_z and σ_y

In the Pasquill-Gifford scheme, the parameters σ_z and σ_y are generally presented as functions of travel distance x. Turner (1969) and Doury (1972, 1976), however, use parametrizations that depend on travel time t. There is little to be said here other than that the existence of widely accepted schemes like those of Pasquill, Turner, and Doury, which use such

Table 9-3. Ranges of values of σ_θ and ΔT corresponding to the Pasquill-Gifford stability categories

Stability category	σ_θ (10 m) (degrees)	ΔT (K/100 m)
A	>22.5	<-1.9
B	17.5 to 22.5	-1.9 to -1.7
C	12.5 to 17.5	-1.7 to -1.5
D	7.5 to 12.5	-1.5 to -0.5
E	3.75 to 7.5	-0.5 to 1.5
F	2.0 to 3.75	1.5 to 4.0
G	<2.0	>4.0

Table 9-4. Stability categories defined by reference to both temperature difference and wind speed[a,b]

Wind speed (m/sec)	$\Delta T/\Delta Z$ (°C/100 m) <-1.9	-1.9 to -1.7	-1.7 to -1.5	-1.5 to -0.5	-0.5 to 1.5	1.5 to 4
	Stability as defined by NRC Regulatory Guide 1.23					
	A	B	C	D	E	F
<2	A	B	B	B	E	F
2 to 3	A	B	C	C	E	F
3 to 5	B	B	C	D	E	F
5 to 6	C	C	C	D	D	D
>6	C	C	C	D	D	D

[a]Scheme based on the typing scheme recommended by Pasquill and the scheme presented in NRC Regulatory Guide 1.23 (USAEC, 1972).

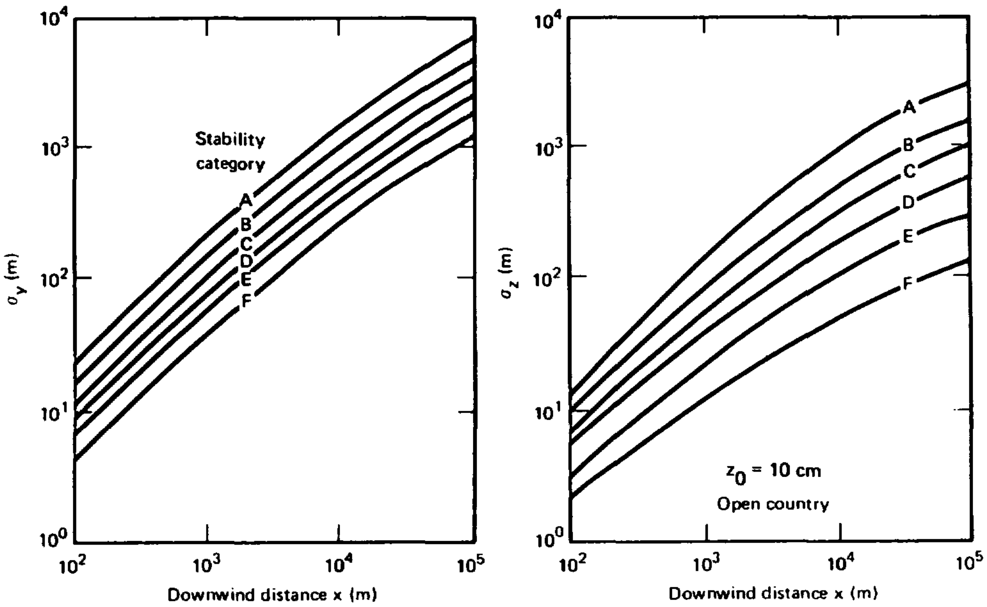
[b]From D. J. Alpert and D. C. Aldrich, Sandia National Laboratories, "Note on Turbulence-Typing Schemes for Use in Reactor Accident Consequence Models," to be published.

different parametrizations of σ_z and σ_y, "probably fairly well reflects the uncertainty of the data" (Gifford, 1976).

In general, consequence modelers use schemes with six or seven stability categories. An example of a widely used scheme appears in Figure 9-5, and a parametrization suitable for use on a computer has been given by Hosker (1974). This model has the advantage that dependence on surface roughness is incorporated. In the Reactor Safety Study, the parametrizations of σ_y and σ_z are a version of the Pasquill-Gifford scheme attributed to Martin and Tikvaart (1968) and described by Eimutis and Koricek (1972). Vogt et al. (1978) have developed a set of parametrizations resulting from tracer measurements over a terrain of major surface roughness. Again, the paper by Gifford (1976) is excellent background reading in this area.

Several of the available parametrizations of σ_y and σ_z were compared in the Benchmark exercise; included were those used in the Reactor Safety Study and the German Risk Study, the schemes by Hosker and Doury alluded to above, and others used in the United States, Europe, and Japan. It appears that, for releases near ground level, the predicted values of χ for a release lasting 1 hour can vary by an order of magnitude simply through the choice of σ_y and σ_z. The reasons for these differences are to be fully discussed in the Benchmark document (see footnote on page 9-17).

For releases of short duration, the predicted time-integrated concentration in the plume is likely to be within a factor of 3 of the actual concentration if measured values are used for all parameters and the correct stability has been assigned. The values of the parameters in the models are most reliable for dispersion over distances of up to a few tens of kilometers; when considering dispersion over distances approaching 100 km, predictions are likely to be increasingly less accurate (Clarke



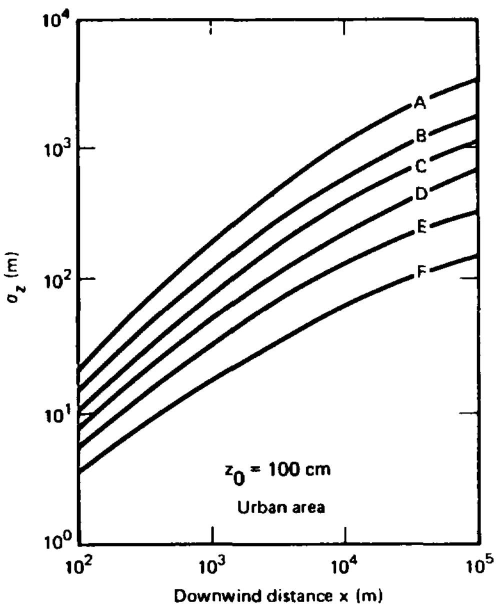


Figure 9-5. Typical variation of lateral and vertical standard deviations with stability category and distance. The meteorological roughness length z_0 is defined in Equation D-2. From Hosker (1974).

et al., 1979). The validity of the Gaussian approach is discussed more fully in Appendix D1.2.

9.3.1.4 Very Low Wind Speeds

Equation 9-1 clearly breaks down as the wind speed tends to zero, and the user of consequence models needs to know what to do when his meteorological data indicate a calm. In CRAC and CRAC2, the code automatically assigns a wind speed of 0.5 m/sec if the true wind speed is lower. In this case, the stability category remains as indicated. Some experiments have been carried out at low wind speeds (Sagendorf, 1974; Gifford, 1976). In all cases the observed diffusion was very much greater than that predicted by the application of the Gaussian model with the appropriate stability category. One possible objection to the neglect or approximation of low-wind-speed conditions--that ground-level concentrations may be very much greater than observed even under category F or G conditions and that the worst case is being ignored--can therefore be discounted.

9.3.1.5 Specific Effects

The basic Gaussian model must be modified to take account of various effects that cannot be neglected. The most important of these are dry and wet deposition, which have been allocated a section of their own (Section 9.3.2). The specific effects covered in this section are radioactive decay, duration of release, building wakes, inversion lid, and plume rise.

Radioactive Decay

As is well known, a single radionuclide will decay, so that, if there was a quantity Q present at t = 0, a quantity Q $\exp(-\lambda t)$ will remain at time t. The time t can be equated to $x/\bar{u}$ if a constant wind speed is assumed. If there is a chain of radionuclides, the buildup and decay of daughters can be treated by standard methods; a convenient reference is a report by Kaiser (1976).

Duration of Release

If a release is of prolonged duration T_r and the wind direction is nominally unchanging during that period, the action of large-scale eddies will cause the time-averaged plume to be wider than it would be for a release of shorter duration. Naden and Leeds (1972) have described how in principle plume models can be modified to account for long averaging times, but their methods are too cumbersome for general use, and simplifying assumptions are needed. A comprehensive review of the T_r dependence of $\sigma_y(x)$ has been given by Griffiths (1977). A report by Clarke et al. (1979) is a useful reference.

One of the simplest methods is to make the substitution

$$\sigma_y \rightarrow \sigma_y \left(\frac{T_r}{T_E}\right)^p \qquad (9\text{-}4)$$

where T_E is the duration of release in the experiments from which the values of σ_y were derived. In the Reactor Safety Study, p was taken to be 1/3 and T_E to be 1/2 hour. In CRAC2, p is taken to be 0.2 for 3 minutes $\leq T_r \leq$ 1 hour and 0.25 for 1 hour $< T_r$. A practical upper limit on T_r is 10 hours; T_E is taken to be 3 minutes. This CRAC2 scheme is in accord with the recommendations of the American Meteorological Society workshop (AMS, 1977) and should therefore be preferred. Another method is to break the release into puffs of short duration and to superpose the time-integrated concentrations from each puff. UFOMOD, the code used for the

German Risk Study, incorporates this option (Schueckler and Vogt, 1981). In CRACIT the option of a multiphased release is also available.

Building Wakes

It is very likely that in an accident the radioactive effluent will be emitted into the turbulent reactor-building wake. Unfortunately, as Gifford (1976) has remarked, little is known about the properties of diffusion in the wakes that exist in the atmosphere downwind of the structure, and therefore arbitrary assumptions about the effect of such a wake are required. For example, the quantity $[\pi \bar{u}\ \sigma_y(x)\ \sigma_z(x)]^{-1}$ in Equation 9-1 can be replaced by

$$[\pi \sigma_y(x)\ \sigma_z(x) + cA]^{-1}\ \bar{u}^{-1}$$

where A is the area of the building projected onto a plane perpendicular to the wind direction and c is a constant with a value $\leq$ 0.5 (Gifford, 1976). Equation 9-5 is a little difficult to manage in the sense that some assumption about concentration profiles within the building wake is also needed, and this can lead to difficulties with the conservation of mass or of released activity.

In CRAC and CRAC2 it is assumed that the concentration profiles are Gaussian both laterally and vertically, with boundaries at the width W or height H of area A. As a result, the airborne time-integrated concentration within the wake is

$$\chi(x,y,z) = \frac{Q\ \exp[(-z^2/2\sigma_H^2) - (y^2/2\sigma_W^2)]}{\pi \sigma_H \sigma_W \bar{u}} \qquad (9\text{-}6)$$

where $H = 2.14\sigma_H$ and $W = 3\sigma_W$. Equation 9-6 is roughly equivalent to Equation 9-5 with c = 0.4.

As a crude approximation, the wake is supposed to persist for a certain number of building heights N (say five); the subsequent atmospheric dispersion is calculated by assuming that there is an area source at this time-integrated distance downwind, in which case it is easy to show that the airborne concentration beyond the end of the wake is given by

$$\chi(x,y,z) = \frac{Q\ \exp\{[-z^2/2\sigma'^2_z(x)] - [y^2/2\sigma'^2_y(x)]\}}{\pi\ \bar{u}\sigma'_z(x)\ \sigma'_y(x)} \qquad (9\text{-}7)$$

where

$$\sigma'^2_y(x) = \sigma^2_y(x - NH_W) + \sigma^2_W \qquad (9\text{-}8)$$

and

$$\sigma'^2_z(x) = \sigma^2_z(x - NH_W) + \sigma^2_H \qquad (9\text{-}9)$$

Equation 9-6 cannot be expected to reproduce the concentration profiles within the wake, but it has the merit of being a simple approximation. In practice it is convenient to take N = 0. This does not affect the airborne concentrations at points far enough downwind to be of interest in reactor safety studies. Other simple approximations can be found in the literature (see, for example, Slade, 1968).

The transition to a point outside the wake is very easily accomplished by Equation 9-7 without the need for time-consuming integrations. Furthermore, the calculations of radiation doses from the passing cloud are also relatively easy. The method described here can therefore be justified as a simple and economical way of taking into account the initial dilution of the plume by the building wake. An example of recent advances in the prediction of pollution concentrations near buildings is given by Britter et al. (1976). The range of existing models will be reviewed in the expected report of the international Benchmark exercise. An experimental investigation of plumes emitted within a reactor complex has been reported by Start et al. (1977).

Inversion Lid

Equation 9-1 has been written so as to make it plain that reflection at the ground has been assumed. In practice, there is also a limit to the vertical spread of the plume because the atmospheric boundary layer is capped by a very stable layer with a strongly positive temperature gradient. The turbulence intensity is much reduced within such a layer, and indeed the base of such a layer forms an effective barrier to the upward dispersion of a plume. This is known as the inversion lid. Holzworth (1964, 1972) has given estimates of the height ℓ of this lid for the United States. In general, consequence-modeling codes assume values of ℓ that are typical of the weather being considered. These values can be found from reviews like that of Holzworth.

One simple way of taking the lid into account is to assume multiple reflections at the ground and the lid. If this is done, Equation 9-1 is replaced by

$$\chi(x,y,z) = \frac{Q \exp(-y^2/2\sigma_y^2)}{2\pi\bar{u}\sigma_z\sigma_y} S(h,z,\ell) \tag{9-10}$$

where

$$S(h,z,\ell) = \exp\left[-\frac{(z-h)^2}{2\sigma_z^2}\right] + \exp\left[-\frac{(z+h)^2}{2\sigma_z^2}\right] + \exp\left[-\frac{(2\ell+z+h)^2}{2\sigma_z^2}\right]$$

$$+ \exp\left[-\frac{(2\ell+z-h)^2}{2\sigma_z^2}\right] + \ldots \tag{9-11}$$

Ultimately, the plume becomes spread uniformly between the ground and the lid, in which case

$$\chi(x,y,z) = \frac{Q \exp\left(-y^2/2\sigma_y^2\right)}{\sqrt{(2\pi)}\ \sigma_y \ell \bar{u}} \qquad (9\text{-}12)$$

The transition between Equations 9-10 and 9-12 can be conveniently made when $\sigma_z = \ell$, when the two expressions for χ differ by at most a few percent.

Since the treatment of the inversion lid is somewhat arbitrary, other methods are equally acceptable. For example, in the Reactor Safety Study Equation 9-1 is used at all times: σ_z is allowed to grow until it equals 0.8ℓ.

Recent studies indicate that CCDFs are not sensitive to values of ℓ (Sprung and Church, 1977a). On the other hand, certain quantities, such as the final height of plume rise, can be extremely dependent on ℓ (Kaiser, 1981). An instructive treatment of inversion lids that vary with time and of the special case of lids at coastal sites appears in the discussion of the CRACIT code in the Zion study (Commonwealth Edison Company, 1981).

Plume Rise

The treatment of plume rise is a source of uncertainty in modeling the consequences of large hot releases of radioactive material. At first sight, this may seem surprising. After all, more than 100 plume-rise models have been described in the literature, and there have been extensive reviews, such as those by Briggs (1969, 1975).* Nonetheless, these reviews, comprehensive as they are, do not encompass all of the elements necessary in a plume-rise model for radioactive plumes:

1. Definition of the mode of release.

2. Liftoff--the behavior of a buoyant plume in the turbulent wake of a building.

3. Plume trajectory.

4. Ground-level concentrations under a rising plume.

5. Termination of plume rise.

6. Passive dispersion.

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*G. A. Briggs also discusses this topic in "Plume Rise and Buoyancy Effects," a draft chapter (1979) for Atmospheric Science and Power Production, the projected replacement for Meteorology and Atomic Energy--1968.

These elements are to be thoroughly discussed in the forthcoming report of the international Benchmark exercise. They are also described by Kaiser (1977, 1981) and Fryer and Kaiser (1979, 1980); a more-detailed discussion appears in Appendix D2.

9.3.2 DEPOSITION PROCESSES

9.3.2.1 Dry Deposition

The standard way of dealing with deposition is to assume that, if $\chi(x,y,0)$ is the ground-level time-integrated concentration in curies per second per cubic meter, the deposited activity is given by

$$\chi_D(x,y) = v_d \chi(x,y,0) \quad \text{Ci/m}^2 \qquad (9\text{-}13)$$

where v_d is the velocity of deposition, which can occur by a number of processes, including gravitational settling, turbulent and molecular diffusion, and inertial impaction (Horst, 1977). Sehmel (1980) has tabulated some 80 factors that influence dry-deposition rates! The concept of deposition velocity is introduced here to help the reader understand the microphysical processes involved in dry removal.

Particulate Matter

For particles, v_d depends on a variety of parameters: the chemical properties of the material being deposited, the size and shape of the particles, the surface-roughness length z_0, the nature of the vegetation, the atmospheric stability category, and so on. As a result, a survey of published data on the value of v_d produces figures varying between 0.0001 and 20 cm/sec (Hosker, 1974). Since this remains an area of great uncertainty, it is discussed in some depth in Appendix D, where it is shown that, for particulate matter emitted in the aftermath of a reactor accident, it is reasonable to expect v_d to be in the range 0.1 to 10 cm/sec. Hence, the value of 1 cm/sec chosen for use in the Reactor Safety Study seems as good as any other. The large range of possible values for v_d has prompted speculation that v_d should be treated probabilistically. It is pertinent to remark in this context that Beyea (1978a,b) incorporates v_d into his models as an uncertain parameter that varies between 0.1 and 10.0 cm/sec for stability classes A through D, 0.1 to 3.0 cm/sec for stability class E, and 0.1 to 1.0 cm/sec for stability class F.

The deposition velocity v_d is one of the parameters to which many of the results of consequence calculations are sensitive since, as can be seen from Figure 9-3, deposition on the ground is the starting point for most of the pathways to people. (See also Section 9.6.4.1.)

Vapors

The important fission products that have been considered to be gases or vapors in past consequence analyses are the noble gases, elemental iodine, and iodine as methyl iodide. Their deposition velocities are discussed in Appendix D, and the following conclusions can be drawn:

1. The deposition velocities for the noble gases should be taken to be zero since the noble gases are almost totally unreactive.

2. There is no need to consider iodine separately from particulate matter. Experimental evidence shows that it is unlikely that iodine will be released from the fuel in its elemental form; it will probably be in the form of a metallic iodide, most probably cesium iodide (Campbell et al., 1981), which is far less volatile than elemental iodine. Hence v_d should be the same as for particulate matter.

3. Methyl iodide can be neglected in consequence calculations because it makes only a small contribution to public risk. This is a lesson that has emerged from experience gained during the Reactor Safety Study.

9.3.2.2 Modification of the Gaussian Formula

The modification of Equation 9-1 to take into account deposition is achieved by replacing Q (the total emitted activity) by Q(x), the activity remaining at a distance x downwind, where

$$\frac{Q(x)}{Q} = \exp\left\{-\left(\frac{2}{\pi}\right)^{1/2} \frac{v_d}{\bar{u}} \int_0^x \frac{dx'}{\sigma_z(x')} \exp\left[-\frac{h^2}{2\sigma_z^2(x')}\right]\right\} \qquad (9\text{-}14)$$

The proof of this result can be found in the article by Van der Hoven (1968) in Meteorology and Atomic Energy. Since, at distances of many tens of kilometers in the more stable weather categories, Q(x) may be less than a tenth of Q, the modification contained in Equation 9-14 must be included in Equation 9-1.

Appendix D discusses the validity of Equation 9-14 and shows that it is adequate in almost all circumstances that are likely to be considered in a typical consequence analysis. Modifications that may be needed in the future to account for gravitational settling are also discussed.

9.3.2.3 Wet Deposition

If a plume of radioactive material encounters rain as it travels downwind, aerosols will be deposited onto the ground. This wet-deposition process, also known as precipitation scavenging, occurs in one or both of two ways. The first is in-cloud scavenging, which takes place because the radioactive aerosol is a source of condensation nuclei that act as centers for the formation of water droplets. This form of precipitation scavenging is known as rainout. The second occurs through rainfall from clouds above the plume. The falling water droplets collide with and collect the particles that make up the radioactive plume; this is called washout. The fraction of material removed from a plume per unit time is known as the washout coefficient and can be defined as

$$\Lambda = -\frac{1}{\chi}\frac{d\chi}{dt} \tag{9-15}$$

The theoretical calculation of Λ is not easy, because a great number of physical processes are involved, examples being thermophoresis, diffusiophoresis, turbulence, raindrop evaporation, and condensation. It is therefore necessary to make simplifying assumptions; for example, the analysis of experimental data suggests that (Ritchie et al., 1976)

$$\Lambda = CR^{\alpha} \tag{9-16}$$

where R is the rainfall rate (mm/hr); α is a constant, taken to be unity; and C is a constant that can be on the order of 10^{-4} sec^{-1}/mm-hr for stable and neutral atmospheric conditions or 10^{-3} sec^{-1}/mm-hr for unstable atmospheric conditions (i.e., a convective storm) (Ritchie et al., 1981b). The quantity Λ can vary from 10^{-5} to 10^{-2} sec^{-1}.

For the simple case in which it rains everywhere at a constant rate, Equation 9-1 becomes

$$\chi(x,y,z) = \frac{Q \exp\left[-y^2/2\sigma_y^2(x)\right]}{2\pi\ \sigma_z(x)\ \sigma_y(x)\ \bar{u}}\ S(h,x,z)\ \exp\left(-\frac{\Lambda x}{\bar{u}}\right) \tag{9-17}$$

where

$$S(h,x,z) = \exp\left[-\frac{(z+h)^2}{2\sigma_z^2(x)}\right] + \exp\left[-\frac{(z-h)^2}{2\sigma_z^2(x)}\right]$$

and Λ is chosen by the user. The quantity of material deposited on a unit area of the ground at the point (x,y,0) is

$$\chi_D(x,y) = \chi(x,y,0)\left\{v_d + \Lambda\ \sqrt{(2\pi)}\ \sigma_z(x)\ \exp\left[+\frac{h^2}{2\sigma_z^2(x)}\right]\right\} \tag{9-18}$$

where v_d is the dry-deposition velocity.

In general, R and Λ depend on time and position since a typical rainstorm moves and is highly structured. Ritchie et al. (1976), for example, describe a simplified rainstorm that covers several tens of thousands of square kilometers and may persist for up to a few days. This area, known as the synoptic region, contains regions known as large mesoscale areas (LMSAs), which cover about 4000 km^2 each and take up about one-third of the total storm area. These areas are not fixed but undergo continuous periods of growth or decay with a lifetime of 12 hours or so; the typical rainfall rate inside them is twice the rate outside them. Within each LMSA there are five or six small mesoscale areas, covering typically 250 km^2, in which the storm lasts for about an hour with a rainfall rate four times that in the synoptic region. Finally, each of these small mesoscale areas contains "cells," 10 km^2 or so in area, in which the storm persists for short periods and in which the rainfall rate can be 25 times that in the synoptic region. The model of Ritchie et al. also allows for the phenomenon of runoff, that is, the washing of deposited radioactive material into rivers and lakes by rain. This complex effect is not, to the authors' knowledge, incorporated into any currently available, fully probabilistic consequence-modeling code, however.

In the context of the discussion of rain, it is pertinent to note that, in CRAC2, for example, it is conventional to assume that, once the plume has passed beyond the farthest point of the computational grid, it is assumed to be completely deposited on the ground by the action of rain, in an interval such as that between 500 and 2000 miles. This artificial procedure is implemented in order to avoid a well-known difficulty in consequence modeling, the nonconvergence of the total population dose (in man-rem). In brief, most dispersion models would predict radiation doses decreasing like r^{-a} at large distances r from a reactor, with $a < 2$. Assuming that the plume is confined to a sector of angular width θ, with a uniform population distribution, the whole-body population dose is proportional to

$$\int_0^\infty \frac{\theta r \, dr}{r^a} \propto r^{2-a}$$

which does not converge. This is an unrealistic result, because various depletion processes will act on the plume as it moves to very large distances. The washout of the plume described above is an artificial, but reasonable, means of avoiding this difficulty.

An alternative would be to truncate the above integral when the radiation doses become negligible--some small fraction of those delivered by the natural background, such as 10 mrem. Such truncations are always controversial, however, and it is preferable to use the plume-washout method.

9.3.2.4 Changing Weather Conditions

It is clear from the foregoing that a realistic treatment of the effects of rain can be achieved only within a scheme that treats changes in weather conditions over time. The most significant difference between the predictions of workers who use a statistical model and those who rely on methods that do not incorporate changing weather conditions is to be found in the quantity and the position of deposited gamma emitters. If a radioactive plume encounters a region of heavy rain as it passes over a city some distance downwind, a relatively large fraction of the material within it could be deposited in a densely populated area. This means that the predicted dose rates due to irradiation by deposited gamma emitters could be much higher than would ever be predicted by a code like TIRION (Kaiser, 1976), in which the weather conditions are constant. As a result of rainout, people living in the city could rapidly accumulate a dose to the bone marrow that would exceed the thresholds for early death or morbidity. The number of early deaths predicted for an airborne release of radionuclides is extremely sensitive to assumptions about the dose delivered by deposited gamma emitters.

Codes like CRAC incorporate provisions for changing the weather conditions as the plume travels downwind. It is possible to distinguish a hierarchy of codes of increasing levels of sophistication:

1. Constant-weather codes like TIRION (Kaiser, 1976).

2. Codes with changing weather conditions but an unchanging wind direction, an example being CRAC2 (Ritchie et al., 1981a).

3a. Codes with changing weather conditions and wind directions, and single-station meteorological data, an example being UFOMOD (Schueckler and Vogt, 1981).

3b. Codes with changing weather conditions and wind directions, multiple-station meteorological data, and the effect of topographical features; an example is CRACIT (Commonwealth Edison Company, 1981).

The use of codes belonging to one or another of the stages in this hierarchy is an extremely important element in the current debate among consequence modelers about how best to handle changing weather conditions. This extremely important aspect of consequence modeling is discussed in depth in Appendix D.

9.3.3 PROCESSES THAT LEAD TO THE ACCUMULATION OF RADIATION DOSES

There are five processes that account for most of the ways in which people can accumulate a radiation dose after an accidental release of radioactive material to the atmosphere:

1. Inhalation.

2. Exposure to external irradiation from the passing cloud (cloudshine).

3. Exposure to external irradiation from deposited radionuclides (groundshine).

4. Ingestion, including contaminated vegetation, milk, milk products, and crops contaminated by root uptake.

5. Inhalation of resuspended radionuclides.

For estimating early effects, the most important of these pathways are (1) inhalation from the cloud, (2) cloudshine, and (3) short-term exposure from contaminated ground (hours to days). For estimating latent health effects, the important pathways include (1) external exposure from contaminated ground (both short and long term), (2) inhalation exposure from the passing cloud and from the subsequent resuspension of radionuclides, and (3) the ingestion of contaminated foods.

9.3.3.1 Inhalation

The preceding sections have discussed the methods required to calculate the time-integrated concentration $\chi_n^i(x,y,z)$ of the i^{th} member of the n^{th} chain of radionuclides. The total inhaled activity of this nuclide is $I_n^i(x,y)$, given by

$$I_n^i(x,y) = b_r \; \chi_n^i(x,y,z = 0) \quad \text{Ci} \tag{9-19}$$

where b_r is the breathing rate, a parameter that depends on the age of the person involved and on his being engaged (or not) in vigorous activity. The breathing rate commonly assumed for adults (ICRP, 1975) is

$$b_r = 2.66 \times 10^{-4} \; m^3/sec \tag{9-20}$$

This is the breathing rate averaged over the entire day--that is, 16 hours of light activity and 8 hours of resting:

Activity level	Breathing rate (m^3/sec)
Light (16 hr/day)	3.33×10^{-4}
Resting (8 hr/day)	9.03×10^{-5}
Daily average	2.66×10^{-4}

The breathing rate will clearly vary during different phases of evacuation or sheltering. For example, people preparing to evacuate may well be highly active. People traveling in cars will be somewhere between resting and light activity. People who have retired to their basements to shelter will also most likely be in a light or lesser state of activity. However, these activity levels and associated breathing rates may not account for possible effects of anxiety. In principle, different breathing rates during different phases of the emergency-response procedure should be taken into account.

The calculation of the radiation doses delivered as a result of the inhalation of radioactive material is extensively reviewed in the Reactor Safety Study. The model used there incorporates the ICRP lung model (ICRP, 1966), with a separate treatment for gaseous radionuclides (Bernard and Snyder, 1975). This allows the calculation of a quantity $F_{n,k}^i(t)$, the dose in rem to organ k at time t after the inhalation of 1 Ci of the i^{th}

nuclide of the n^{th} chain, hereafter referred to as nuclide (n,i), at t = 0. Thus the total dose to organ k integrated to time t is

$$D_k(x,y,t) = \sum_n \sum_i F^i_{n,k}(t)\ I^i_n(x,y) \tag{9-21}$$

The quantities $F^i_{n,k}(t)$ are known as inhalation-dose-conversion factors.

The library of dose-conversion factors compiled for the Reactor Safety Study was calculated with the code TIMED (Watson et al., 1976). Many of the consequence-modeling codes available in the United States still have the same library. At present, considerable effort is being devoted to the updating of inhalation-dose-conversion factors. Other codes, such as INREM II (Killough et al., 1978a; Dunning et al., 1979; 1981), have been developed. The International Commission on Radiological Protection (ICRP) is making use of a revised version of TIMED (Watson and Ford, 1980) in a systematic update of these dose-conversion factors. Revised guidance was recently published by the ICRP in Publication 30 (ICRP, 1979, 1980). Dose-conversion factors have also been published by the British National Radiological Protection Board (Kelly et al., 1977; Adams et al., 1978; Hunt et al., 1979).

It appears that, in general, these revised dose-conversion factors do not make a significant difference to the results of consequence analyses for LWR plants. This should not be interpreted to mean that the revised factors are numerically similar to those in the Reactor Safety Study. On the contrary, there are some significant differences, particularly among the actinides. However, for the typical inventory of radionuclides that is predicted to escape into the atmosphere in the event of an LWR accident, these differences do not propagate significantly into the results.

One of the questions most frequently asked about dose-conversion factors is whether the age distribution of the population has been properly accounted for since the dose-conversion factors depend on the age of the exposed person. In the Reactor Safety Study, the dose-conversion factors were developed strictly for the adult male.* The assumption of an adult dosimetry model is a convenient simplification. More detailed studies have revealed that the effect of the closer proximity of the organs to each other in an infant or child is approximately offset by lower intake and higher metabolism (Snyder, 1975). Infants compose only about 2 percent of the population. Thus, even if the dose factors for children were fivefold greater than adult factors, the error in the collective dose would be only about 10 percent, well within the overall uncertainty. Therefore, the convenient approach of using adult parameters for dose calculations does not in general cause significant errors for LWR-accident consequence calculations involving the whole population.

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*The same remark applies to the dose-conversion factors for ingestion, cloudshine, and groundshine, which are discussed in Sections 9.3.3.2 and 9.3.3.3.

The NRC is funding the development of a comprehensive library of dose-conversion factors, which is to be suitable for easy use by consequence modelers and is to be published in 1983 by Sandia National Laboratories.

9.3.3.2 External Irradiation

External Irradiation from the Passing Plume (Cloudshine)

For estimating external cloudshine exposure, let the time-integrated airborne concentration of nuclide (n,i) be $\chi_n^i(x,y,z)$ and let the radionuclide deliver a radiation dose from exposure to cloudshine, $D_{cn}^i(x,y,z)$, to a mathematical element of tissue at point (x,y,z). If the cloud is assumed to be infinite in extent and of uniform concentration, then

$$D_{cn}^{i}(x,y,z) = C_n^i \chi_n(x,y,z) \qquad (9\text{-}22)$$

where C_n^i is known as the cloud-dose-conversion factor and Equation 9-22 is an expression of the well-known semiinfinite-cloud approximation. In general, Equation 9-22 is evaluated with z set equal to 1 m, that is, for a person standing at ground level. The quantity C_n^i does not take into account self-shielding of the body, a subject that is discussed below. For a mixture of radionuclides, the total radiation dose is obtained by summing Equation 9-22 with respect to n and i.

If the cloud is finite, the semiinfinite approximation is not applicable (Van der Hoven and Gammill, 1969), and C_n^i must be multiplied by the quantity $CF(\sigma_z,z/\sigma_z)$, which is a correction factor to be applied when the cloud has its center at a height z above the ground and a vertical standard deviation σ_z. Table VI 8-1 of the Reactor Safety Study contains a compilation of these cloud-dose-correction factors as a function of σ_z and z/σ_z, taken from Meteorology and Atomic Energy--1968 (Slade, 1968). The finite-cloud dose is calculated as follows: the dose is calculated as if the person were located in a semiinfinite cloud with a uniform concentration equal to that at the centerline of the cloud. The correction factor $CF(\sigma_z,z/\sigma_z)$ accounts for the finite extent of the cloud and the vertical displacement (z) between the cloud centerline and ground level.

It is important to remember that the product $CF(\sigma_z,z/\sigma_z)C_n^i$ is an approximation to a three-dimensional integral over the plume. This integral must in principle be evaluated numerically for each gamma ray emitted by each radionuclide in the atmospheric release of radioactivity. This can be extremely time consuming, and it is often the most costly calculation in a consequence-analysis code. It is therefore highly desirable to approximate the integral by an expression not requiring an integration; hence the need for approximations like $CF(\sigma_z,z/\sigma_z)$. It is recommended that the user of consequence-modeling codes use these time-saving approximations as much as possible.

Thus CRAC2 contains an array of values of $CF(\sigma_z,z/\sigma_z)$ for selected values of σ_z and z/σ_z, between which interpolation is carried out for other values of these parameters. It also contains a library of quantities $C_{n,k}^{i}$ that are related to C_{n}^{i} as follows: the dose delivered to a particular body organ k through external irradiation by gamma rays does not necessarily equal C_{n}^{i} since the organ may be shielded by the rest of the body. Hence, CRAC2 contains values of C_{n}^{i} modified for each body organ k. The modifications were calculated with the code EXREM III (Trubey and Kaye, 1973).

External Exposure from Gamma Radiation Emitted by Deposited Radionuclides (Groundshine)

The deposited activity of nuclide (n,i) per square meter is given by $\chi_{Dn}^{i}(x,y)$. At time t after the accident, this quantity will have changed because of the action of two mechanisms. The first is radioactive decay, which can be treated in a standard manner and changes the term $\chi_{Dn}^{i}(x,y)$ to $\chi_{Dn}^{i}(x,y,t) = \chi_{Dn}^{i}(x,y)\ RD_{n}^{i}(t)$, where $RD_{n}^{i}(t)$ accounts for radioactive decay and daughter buildup over time t. The second is weathering, which reduces the gamma dose observed above a contaminated surface by a variety of mechanisms, including the removal of dust by the wind, the carrying away of material dissolved in water, the penetration of radionuclides into the soil, and uptake by vegetation. Therefore, the concentration of each nuclide should be modified by a weathering factor $f_{n}^{i}(t)$ that in principle should be different for each radionuclide. In practice, the only nuclide for which much information is available is Cs-137 (Gale et al., 1964). It has been shown experimentally that, if the dose rate above land contaminated by Cs-137 is $\dot{D}_g(t = 0)$ immediately after the contamination has occurred, the dose rate t_y years later is

$$\dot{D}_g(t_y) = \dot{D}_g(t = 0)\ \exp(-0.023t_y)\ \left[0.63\ \exp(-1.13t_y) + 0.37\ \exp(-0.0075t_y)\right] \qquad (9\text{-}23)$$

The single exponential $\exp(-0.023t_y)$ gives the rate of radioactive decay for Cs-137. The term in brackets is the weathering factor for this nuclide.

The weathering of other nuclides is discussed in Appendix VI of the Reactor Safety Study (USNRC, 1975). It is concluded that so little is known about this subject that it is as well to assume that all nuclides behave like cesium, apart from the different radioactive-decay constants.

At time t after the initial deposition has taken place, the rate at which nuclide (n,i) delivers a radiation dose through groundshine is

$$\dot{D}_{gn}^{i}(x,y,t) = \chi_{Dn}^{i}(x,y)\ RD_{n}^{i}(t)\ f_{n}^{i}(t)\ \dot{G}_{n}^{i} \qquad (9\text{-}24)$$

where $\dot{G}_{n}^{i}$ is the dose rate at a reference height Z_r (usually 1 m) above a surface uniformly contaminated by 1 Ci/m^2 of nuclide (n,i). Methods for calculating $\dot{G}_{n}^{i}$ have been described (Slade, 1968). The values of $\dot{G}_{n}^{i}$ are used as approximations to a two-dimensional integral over the contaminated area. As with the cloudshine, they have been introduced in order to eliminate the need for time-consuming numerical integration.

A code like CRAC2 contains a compilation of dose-conversion factors for groundshine, $G_n^i(t)$, calculated by integrating $\dot{G}_n^i$ for two time intervals--8 hours and 7 days. The CRAC2 data bank contains these quantities $G_n^i(t)$ for each of several organs k (see Section 9.4.8.2).

9.3.3.3 Ingestion

A thorough discussion of the ingestion-exposure pathways is presented in Appendix VI (Chapter 8 and Appendix E) of the Reactor Safety Study (USNRC, 1975). The paragraphs that follow lean heavily on that discussion.

There are two distinct periods of ingestion hazard. Immediately after deposition a significant portion of the radioactive material could be deposited on vegetation that is consumed by people or by animals furnishing food for people. Only a single crop would be affected by direct deposition, so that the potential for exposure would exist for less than a year. (This is the only significant mechanism for ingesting the short-half-life radionuclides like I-131.) The level of contamination on the vegetation would decrease with time because of the influence of weather; for example, wind and rain would remove deposited material from vegetation.

The radioactive material deposited on the soil would be available for incorporation into vegetation by uptake through the roots. This is a long-term exposure mechanism and is relatively unimportant in comparison with the others discussed above. The radioactive material contaminating the soil would be available for plant uptake over a period of several years, but generally only a few percent, at most, would be taken up by plants in one growing season. With time, the material may become unavailable for uptake by plants by migrating below the root zone, for example.

The metabolic characteristics of the radionuclides in people and animals determine which of them would contribute significantly to the "internal" dose. These radionuclides have been identified in extensive experimental studies of fallout from nuclear weapons. The radionuclides selected in the Reactor Safety Study were I-131, I-133, Sr-89, Sr-90, Cs-134, Cs-136, and Cs-137. The radioiodines were considered only for the ingestion of milk because of their short half-lives. It should be noted, however, that chronic exposure may be highly dependent on agricultural practices and food-consumption patterns, and the relative importance of certain radionuclides may be changed. These practices and patterns should be considered carefully in any site-specific application of a consequence-modeling code before assuming that the treatment contained in the Reactor Safety Study is applicable.

Direct Contamination of Vegetation

The calculation of contamination levels on vegetation involves a large number of parameters, many of which are poorly known or extremely variable. There can be large variations in local conditions that directly affect the level of contamination ingested, but since the areas affected are large, this variability is expected to average out in such a way that the effects of local "hot spots" would be offset by a person's consuming food from a wide area.

For the specific reactor site and date of accident, a test should be made to determine whether the accident occurs during the growing season for crops or forage. If not, then the direct contamination of vegetation is not considered to be a feasible mode of radiation exposure.

The major factors considered in calculating the ingestion of radionuclides deposited on vegetation are (1) the fraction of deposited material initially retained on vegetation, (2) its behavior on vegetation as a function of time, and (3) the possible mechanisms that would lead to eventual ingestion by people. The explicit models and data are described in Appendix E to Appendix VI of the Reactor Safety Study, and only a brief discussion is given here.

The fraction of deposited material initially retained on vegetation is taken to be 0.5. Weathering effects would reduce the amount of material remaining on vegetation. The fraction remaining t days after deposition is described by the empirical function

$$f_w(t) = 0.85 \exp\left(- \frac{0.693t}{14.0}\right) + 0.15 \qquad (9\text{-}25)$$

In addition to weathering, radioactive decay would also reduce the amount of radioactivity remaining on vegetation, and this can be treated in a standard way.

The above factors are then used to determine the time required for vegetation-contamination levels to fall to an acceptable level. In the Reactor Safety Study, the criteria by which the acceptability of the levels of contamination can be determined were adapted from recommendations by the British Medical Research Council (MRC, 1975) and the U.S. Federal Radiation Council (FRC, 1964, 1965). For example, the limits set for the milk pathway were 3.3 rem to the bone marrow in the first year from strontium, 3.3 rem to the whole body from cesium, and 10.0 rem to the thyroid from iodine. These radiation doses were related to levels of contamination on vegetation by a model that includes--

1. The initial daily intake of a given radionuclide by an average cow.

2. The decay and weathering processes discussed above.

3. The fraction A' of the activity ingested by the cow that is transferred to the milk. This fraction depends on many factors, including the breed of cow, milk yield, and season.

4. Radioactive decay between the production and the consumption of milk (an average delay of 3 days is assumed).

5. The amount of milk consumed by a person each day, a typical value being 0.7 liter.

6. The ingestion factor, which is the radiation dose delivered to a given organ after the ingestion of 1 Ci of a given radionuclide. Thus, the ingestion factors are similar to the inhalation factors discussed earlier. Examples have been given by Adams et al. (1978).

Factors 1 through 5 above are multiplicative and lead to a concentration factor, which relates deposited radioactivity (Ci/m^2) to the curies ingested by an individual. Consequence-modeling codes generally contain estimates of these concentration factors in a data bank. It should be apparent that these simple factors incorporate many assumptions and complex calculations, and indeed their values are uncertain. Examples of both concentration factors and ingestion factors are given in Table 9-5.

Table 9-5. Examples of parameters used in calculating the dose commitment from ingesting contaminated milk

Nuclide	Concentration factor ($Ci/Ci\text{-}m^{-2}$)	Ingestion factor (rem/Ci ingested)[a]		
		Thyroid	Whole body	Bone marrow
I-131	0.692	1.68×10^6	8.79×10^2	2.87×10^2
I-133	0.0042	3.21×10^5	2.70×10^2	1.48×10^2
Sr-89	0.402	5.81×10^2	1.91×10^3	5.26×10^3
Sr-90	0.588	3.18×10^3	5.52×10^4	2.08×10^5
Cs-134	4.22	7.33×10^4	7.14×10^4	7.34×10^4
Cs-136	1.42	9.23×10^3	8.96×10^3	9.29×10^3
Cs-137	4.22	5.55×10^4	5.49×10^4	5.61×10^4

[a]These parameters are based on the "reference man" (ICRP, 1966).

From Table 9-5 it is apparent that the deposited activity of a given nuclide and the subsequent predicted radiation dose accumulated in a given organ are related by the product of a dose-conversion factor for an ingestion concentration factor. Hence a limit like 10.0 rem to the thyroid can readily be translated into an acceptable deposited level of activity for I-131. This level can then be used as the basis for computing areas within which interdictive measures, such as the destruction of crops, are required and the time for which they are needed. The vegetation-contamination models in CRAC consider total radionuclide ingestion from milk, milk products, meat, vegetables, and other foods.

Incorporation of Contaminants from Soil into Vegetation

It is not necessary to calculate acceptable soil-contamination levels for the growing of crops. The uptake of radionuclides by plant roots would be an inefficient mechanism of radiation exposure. At most, a few percent of the deposited radionuclides would be taken up by plants in one growing season. Furthermore, the fraction of material taken up declines rather rapidly in subsequent growing seasons. An area that had enough soil contamination to produce unacceptable doses by plant uptake and the ingestion of food plants would most likely be already forbidden by other restrictions (e.g., external irradiation).

The dose commitment for this mode is calculated as for the direct contamination of vegetation--that is, by making use of concentration factors and dose-conversion factors for ingestion. Rather than the initial retention by vegetation and subsequent weathering, the important factors are the rate of uptake by plants and the rate at which availability to plants decreases (e.g., by leaching to below root zones). Apart from these differences, the two methods are conceptually the same.

9.3.3.4 Resuspension

Radionuclides deposited on the ground will be resuspended by the action of the wind. It is conventional to define a resuspension constant $K(t_y)$ (m^{-1}) such that, if the initial deposited concentration of a radionuclide is 1 Ci/m^2, the concentration in the air just above the ground after t_y years is $K(t_y)$ Ci/m^3. Experimental data on the behavior of $K(t_y)$ as a function of time are meager. A suggested form for $K(t_y)$ is (USNRC, 1975)

$$K(t_y) = 10^{-5} \exp(-0.67t_y) + 10^{-9} \quad m^{-1} \qquad (9\text{-}27)$$

Experience shows that, for typical predicted releases of radioactivity from light-water reactors, the inhalation of resuspended radioactive material is relatively unimportant (see Tables 9-6 and 9-7). This conclusion would not necessarily be true if actinide releases from a reprocessing plant were being considered, however. An alternative expression for $K(t_y)$ has been given by Anspaugh et al. (1974). A useful review has been published by Linsley (1979). Lassey (1979) discusses a modification of Equation 9-27 that is appropriate for nonarid climates.

9.3.3.5 Discussion

After the preceding, somewhat lengthy, discussion, it is convenient to put the various pathways into perspective by asking, Which are the most important? There is no single answer to this question, since the importance of each pathway varies with, for example, the composition of the radionuclide release, the weather conditions at the time of the accident, and the consequences considered. However, the discussion below leads to some conclusions that are generally applicable to accidental releases from LWR plants.

Figure 9-6 shows the relative doses delivered to the bone marrow at 0.5 mile from the reactor in a BWR-1 release (one of the release categories in the Reactor Safety Study) in neutral weather conditions without rain; it is this dose that has been found to be the most important cause of early fatalities. As can be seen, the predicted radiation doses from inhalation,

Table 9-6. Contribution of different exposure pathways to latent-cancer fatalities for the PWR-1 release category[a,b]

	Percentage contribution							
Pathway	Leukemia	Lung	Breast	Bone	GI tract[c]	All others[d]	Total	Whole body[e]
External irradiation from cloud	0.1	0.1	0.3	0.1	0.1	0.1	1	1
Inhalation from cloud	0.3	22	0.4	0.1	0.5	0.2	24	5
External ground								
<7 days	2	2	5	0.9	0.7	2	13	18
>7 days	8	5	19	2	2	7	43	64
Inhalation of resuspended contamination	0.2	13	0.2	0.3	0.2	0.2	14	4
Ingestion of contaminated foods	1	0.6	2	0.6	0.5	0.7	5	8
Total	12	43	27	4	4	10	100	100

[a]Data from Wall et al. (1977).

[b]This table does not include latent fatalities from thyroid cancer, which are calculated separately, as discussed in Appendix VI of the Reactor Safety Study (USNRC, 1975).

[c]The gastrointestinal tract includes the stomach and the rest of the alimentary canal.

[d]"All others" denotes all cancers except those specified in the table.

[e]Whole-body values are proportional to the 50-year whole-body population dose commitment (man-rem).

Table 9-7. Contribution of different exposure pathways to latent-cancer fatalities for the PWR-2 release category[a,b]

Pathway	Percentage contribution							
	Leukemia	Lung	Breast	Bone	GI tract[c]	All others[d]	Total	Whole body[e]
External irradiation from cloud	0.2	0.1	0.5	0.1	0.1	0.1	1	1
Inhalation from cloud	0.5	4	0.7	0.2	0.4	0.2	6	3
External ground								
<7 days	3	2	7	0.7	0.9	3	16	16
>7 days	12	8	28	3	4	11	66	68
Inhalation of resuspended contamination	0.2	1	0.2	0.4	0.2	0.1	3	2
Ingestion of contaminated foods	2	1	3	1	1	1	9	10
Total	18	16	39	5	6	16	100	100

[a]Data from Wall et al. (1977).

[b]This table does not include latent fatalities from thyroid cancer, which are calculated separately, as discussed in Appendix VI of the Reactor Safety Study (USNRC, 1975).

[c]The gastrointestinal tract includes the stomach and the rest of the alimentary canal.

[d]"All others" denotes all cancers except those specified in the table.

[e]Whole-body values are proportional to the 50-year whole-body population dose commitment (man-rem).

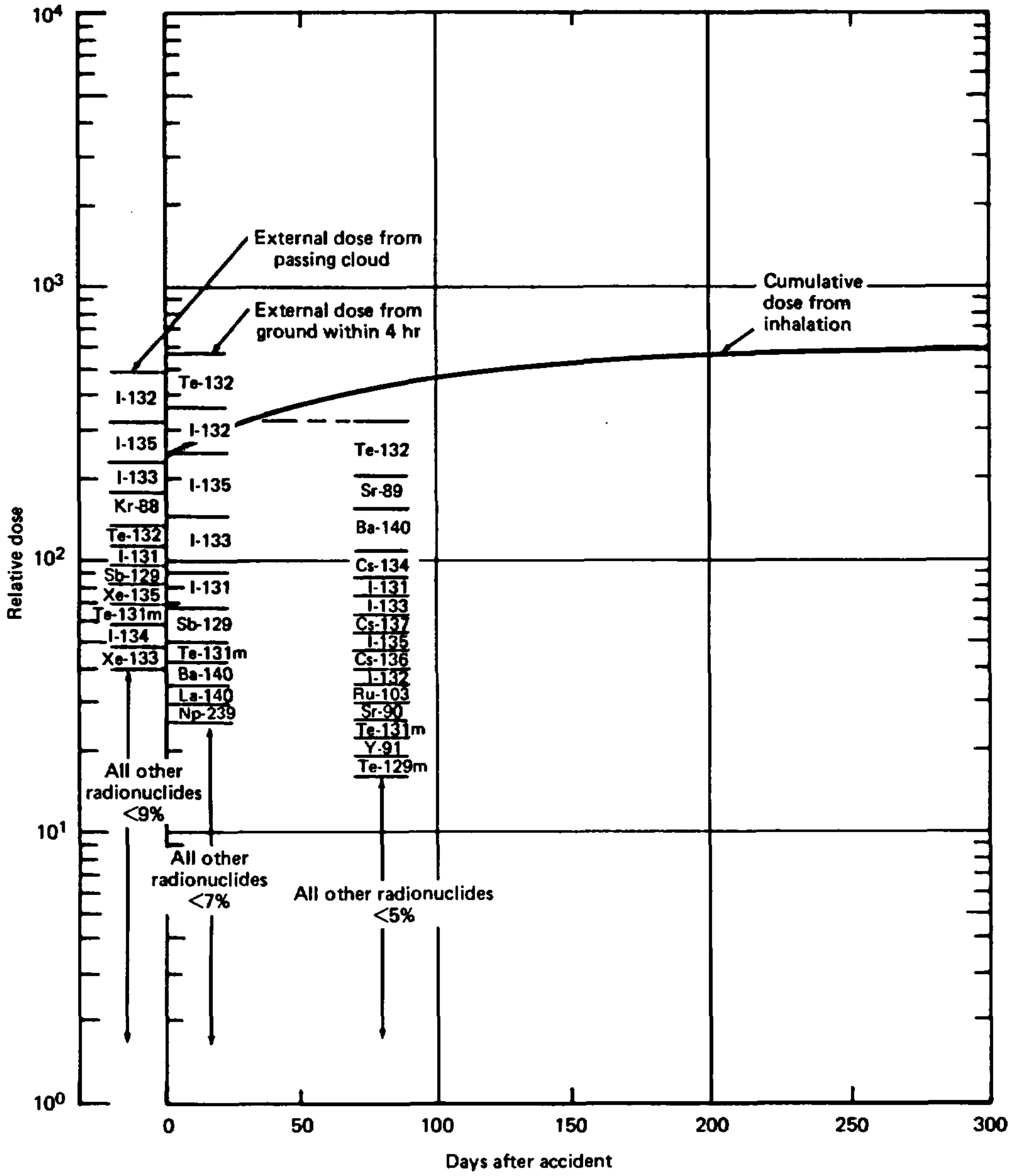

Figure 9-6. Relative doses delivered to the bone marrow at 0.5 mile from reactor. From Wall et al. (1977).

cloudshine, and a 4-hour exposure to groundshine are all comparable, and all of these pathways are important in the calculation of early effects. If the groundshine were extended to 1 day, the relative dose accumulated via this pathway would increase from 500 to about 2000 and would clearly be the dominant contributor (see Figure VI 13-1 of the Reactor Safety Study). This illustrates how important it is to consider evacuation and sheltering strategies that would minimize the effect of this pathway (see Appendix E).

The contribution of different exposure pathways to latent-cancer fatalities is shown in Tables 9-6 and 9-7 for RSS release categories PWR-1 and PWR-2. In both cases, the groundshine radiation dose accumulated over the long term is the dominant contributor, with a further substantial contribution from the external gamma-ray dose delivered in the first 7 days. The long-term external dose from groundshine is dominated by Cs-137 and its daughter Ba-137m. For the PWR-1 release, the inhalation of the cloud and of resuspended nuclides is also important because this release has a high proportion of insoluble, long-lived Ru-106, which preferentially resides in the lung once inhaled. In general, a PWR-2 release is more typical of those expected from an accidental escape of radioactivity from a light-water reactor, since PWR-1 is characteristic of containment failure from a steam explosion, an event that is now thought to be much less likely than was assumed in the Reactor Safety Study.

In calculating the long-term contamination of the ground, leading to the need for relocating people or expensive decontamination, the external dose delivered by gamma rays emitted by deposited Cs-137 is found to be dominant. For estimating the areas within which agricultural produce must be destroyed, the results from the milk pathway are by far the most important.

In summary, the following are generally the most important pathways in calculating the consequences of LWR accidents:

1. Inhalation from the cloud (particularly for early effects).

2. Cloudshine (early effects).

3. Groundshine in the first few hours or days (early effects, latent effects).

4. Groundshine in the long term (latent effects, interdiction of land).

5. Milk ingestion (interdiction of crops).

9.3.4 MEASURES THAT CAN REDUCE PREDICTED RADIATION DOSES

Various protective measures can be envisaged whereby the accumulation of radiation doses by individuals can be much reduced or eliminated. These include evacuation, sheltering, interdiction, and decontamination.

9.3.4.1 Evacuation

It is in the choice of parameters for an evacuation model that the user of consequence-modeling codes can make a highly significant impact on the calculated results. This is illustrated by the CCDF for early fatalities in Figure 9-7, which is taken from Aldrich, Ritchie, and Sprung (1979). The various evacuation schemes used are explained on the figure. It is apparent that the choice of delay time can make a difference of orders of magnitude to the mean public risk.

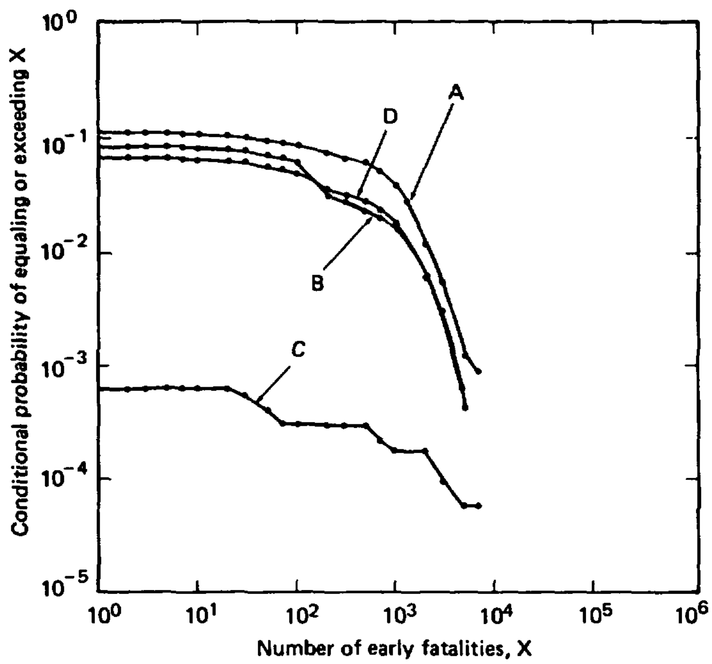

Figure 9-7. Conditional probability versus early fatalities, calculated with the CRAC2 evacuation model. CCDFs are conditional on RSS release categories PWR-1 through PWR-4. Evacuation within 25 miles at a speed of 10 mph. Curves A, B, and C are for 5-, 3-, and 1-hour delays, respectively. Curve D is the weighted sum (5-hour delay, 30%; 3-hour delay, 40%; 1-hour delay, 30%). From Aldrich, Ritchie, and Sprung (1979).

Because of the importance of this topic to the user of consequence-modeling codes, Appendix E has been set aside for a relatively thorough review of evacuation.

The most useful references for background reading are (1) the Reactor Safety Study (USNRC, 1975); (2) the German Risk Study (Gesellschaft fuer Reaktorsicherheit, 1980; EPRI, 1981); (3) a review paper (Aldrich et al., 1978); (4) two reports from Sandia National Laboratories (Aldrich, Blond, and Jones, 1978; Aldrich, McGrath, and Rasmussen, 1978), which describe the updated evacuation model contained in CRAC2; and (5) Section 6 of the Zion PRA (Commonwealth Edison Company, 1981), which describes the evacuation model used in CRACIT and shows how to account for the interaction between an existing road network and wind-shift consequence models.

9.3.4.2 Sheltering

The attenuation of gamma rays by buildings and by surface rugosities has already been mentioned in Section 9.2.1.6. For further information, the reader is referred to Appendix E, which gives guidance about the choice of shielding factors for a typical consequence-modeling code.

9.3.4.3 Interdiction

The process of interdiction would involve the denial of land and its improvements for normal intended use. For example, if the land were contaminated to such an extent that a specified radiation dose would be exceeded over a period of time, use of the land could be prohibited until such time as the radiation dose that an individual would receive over the succeeding period has decreased (through radioactive decay and weathering forces) below the specified criterion. In a decreasing order of impact, interdiction could fall into any of the following categories:

1. Total land and asset interdiction for long periods (more than 10 years).

2. Limited land interdiction (restrictions imposed for a few years).

3. Interdiction of crops.

4. Interdiction of milk consumption.

The criteria for establishing any of these categories of interdiction are based on projected doses to the population. Examples have already been given for the milk pathway in Section 9.3.3.3. Other examples can be found in Table VI 11-6 of the Reactor Safety Study and include, for instance, 25 rem to the whole body from external radiation delivered over a period of 30 years to people living in an urban area and 10 rem delivered over the same period to people living in a lightly populated rural area.* Consequence-modeling codes generally establish the areas within which the given acceptable levels would be exceeded. By assuming that people within those areas would be relocated or that crops would be destroyed, the predicted population dose is reduced and hence the number of predicted health effects is also reduced.

As explained above, the area of interdicted land would decrease with time as the level of contamination decreases. However, decontamination would make it possible to recover some of this land immediately.

9.3.4.4 Decontamination

Decontamination, in the broad sense of the word, is the cleanup and removal of radionuclides. The possible modes of decontamination include the physical removal of the radionuclides, stabilization of the radionuclides in place, and management of the environment. The particular procedure used in a given case would depend on many factors, including (1) the type of surface contaminated, (2) the external environment to which the surface is exposed, (3) the possible hazards to people, (4) the costs, (5) the degree of decontamination that is required, and (6) the consequences of the decontamination operation.

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*In practice, CRAC and CRAC2 make use of the criterion of 25 rem in 30 years for both urban and rural areas.

There is a large body of experimental data on the decontamination of structures, pavements, and land. Most of these data were generated for planning reclamation in the event of a nuclear war. Because of differences in the sizes of contaminant particles and in decontamination criteria, some of these experimental data are not directly applicable to reactor accidents. These problems are discussed more fully in Appendix K to Appendix IV of the Reactor Safety Study (USNRC, 1975).

A measure of the effectiveness of decontamination operations is the decontamination factor (DF), which is defined as the contaminant concentration (in curies per square meter) before decontamination divided by the contaminant concentration after decontamination. Therefore, the larger the DF, the better the decontamination method. For example, a 90-percent removal of contaminants from a surface gives a DF of 10; a 99-percent removal gives a DF of 100.

Typical procedures that can be followed to remove radioactivity are as follows:

1. Hard surfaces (roofs, walls, pavements, etc.)
   a. Replacement of roofing material.
   b. Sandblasting of walls and pavements.
   c. Resurfacing of pavements.

2. Land areas (soil, vegetation, etc.)
   a. Vegetation removal and disposal.
   b. Surface soil removal and burial.
   c. Deep plowing.

The maximum decontamination factor that was considered practical on the basis of the review carried out for the Reactor Safety Study, averaged over large areas, is 20. This limitation is based on the practicality of large-scale decontamination operations, the costs, and the consequences of decontamination operations. The German Risk Study suggests a factor of 10; clearly this is another area of uncertainty.

In CRAC and CRAC2, land that is contaminated to between 1 and 20 times the acceptable level is assumed to be decontaminated just sufficiently to bring it down to that level~~;~~ land that is more severely contaminated is assumed to have the benefit of a full DF of 20. This then reduces the interdiction time required to allow the weathering and decay processes to decrease the contamination to acceptable levels.

9.3.4.5 Miscellaneous

There are some countermeasures that can in principle be incorporated into consequence models, although this is not always done.

Thyroid Blocking

Potassium iodide or iodate, if ingested in time, reduces the amount of radioiodine that can be taken up by the thyroid. The distribution of potassium iodide or iodate tablets, immediately before or after an accidental release of radioactivity, to the population at hazard would reduce thyroid doses. This possibility is not usually considered in consequence models, because the planning procedures, whereby the prompt distribution of a blocking agent would be possible in the event of an accident, have not been implemented for U.S. reactors. A comprehensive examination of this subject has been made by Aldrich and Blond (1980, 1981), who conclude that "although the effective use of KI could significantly reduce the number of thyroid nodules resulting from a serious accident, it would have no, or only minor, impact on other accident consequences, including immediate deaths or injuries, delayed cancer deaths, and long-term land contamination. Therefore, the availability of KI would provide only a supplemental strategy to be considered along with other protective measures."

Ventilation

If people are assumed to be sheltered from external irradiation by, for example, taking refuge in a basement, then it is conceivable that a significant quantity of radioactive material can be excluded from a structure, either by natural effects or by certain ventilation strategies. This could reduce the amount of inhaled radionuclides (Aldrich and Ericson, 1977) and might reduce the inhalation dose by a factor of 2 (Cohen et al., 1979). The Limerick study (Philadelphia Electric Company, 1981) is an example of a recent risk assessment that takes this effect into account.

Medical Treatment

The effectiveness of medical treatment can readily be incorporated into the dose-response relationship that is used to relate the radiation dose to the probability of some health effect. For example, the Reactor Safety Study (USNRC, 1975) proposes three dose-response relationships for early fatalities, assuming minimal, supportive, and heroic medical treatment. Similarly, it is usually assumed that only 5 to 10 percent or so of thyroid cancers are fatal, and this is easily incorporated into a dose-response relationship.

Respiratory Protection

Recently, a study was carried out to determine what benefit, if any, would result if people covered their faces with sheets, towels, or other crude forms of mask while inhaling air contaminated with radioactive material (Cooper et al., 1981). The study consisted of a series of experiments with various fabrics, aerosols, and vapors, followed by an estimation of the likely efficacy of these fabrics as face masks. A summary of the results appears in Table 9-8. A glance at this table reveals a considerable sensitivity to aerosol-particle diameter, which, as is discussed in Appendix E, is a poorly known quantity for accidental atmospheric releases of radioactivity. Nonetheless, it appears that, of the materials likely to be available in an ordinary house, a wet towel folded into four layers could be quite effective, even for small aerosol-particle diameters, with reduction factors of 5 to more than 100 being feasible. The study does not, however, estimate the radiation dose delivered by gamma rays emitted by fission products trapped in the face mask.

Table 9-8. Estimated penetration through expedient respiratory-protection materials[a,b]

Material	Number of layers	Particle diameter (μm) 0.5	1	5	Molecular iodine (I_2)[c]	Methyl iodide[c]
		DRY MATERIALS				
3M respirator[d] No. 8710	2	0.03	0.004	<0.01		
Sheet	20	0.66	0.64	0.020	1.0	0.6[e]
Shirt	15	0.54	0.59	0.070		
Lower-quality towel	20	0.53	0.41	0.015		
Higher-quality towel	6	0.24	0.13	0.01		0.6[e]
Handkerchief	14	0.61	0.54	0.32		
		WET MATERIALS				
Sheet	6	0.91	0.88	0.22	0.45	0.8[e]
Shirt	6	1.0	0.51	<0.02	0.15[f]	1.0[f]
Higher-quality towel	4	0.20	<0.01	<0.01	0.21	1.0
Handkerchief	2	0.98	0.95	0.37	0.10[f]	

[a]Data from Cooper et al. (1981).

[b]Tests at a pressure drop of 50 Pa (0.2 in. H_2O) and a face velocity of 1.5 cm/sec.

[c]Taken from tests at 1.0 cm/sec, assuming penetration is the product of single-layer penetrations.

[d]Available commercially in single-layer thickness.

[e]Not shown to be statistically different from 1.00.

[f]Wetted with 5-wt% baking-soda solution.

It is not yet common practice to incorporate methods of respiratory protection into emergency plans. However, the distribution of respirators to persons within 10 miles or so of a reactor would be easy, as would special sheltering plans that would instruct people to make emergency respirators out of materials they have in their houses. It follows that, in the future, it may be necessary to consider the effects of respiratory protection in carrying out a consequence analysis.

9.3.5 THE EFFECT OF RADIATION DOSES ON THE HUMAN BODY

After calculating the radiation-dose commitments, it is necessary to consider the adverse health effects that may result in the exposed population. Three kinds of health effects from exposure to accidental releases of radioactive material from a reactor are considered: early and continuing somatic effects, late somatic effects, and genetic effects. Early and continuing somatic effects consist of the early injuries and fatalities that are usually observed after high acute doses ($\geq$50 rads to the whole body) and can occur within days to weeks after exposure. Since these effects are observed only at high doses and high dose rates, they are generally, but not always, limited to persons living in the immediate vicinity of the reactor.

Late somatic effects consist of cancer fatalities and illnesses; they are observed in populations several years to decades after exposure. Finally, changes in the genetic coding of chromosomes can affect the well-being and stability of future populations. Unlike the early and late somatic effects, genetic effects manifest themselves not in the irradiated individuals but in their descendants. Consequence-modeling codes usually have the capability of calculating genetic risk. Whether this is done or not depends on the intended application of the consequence model.

Appendix VI of the RSS (USNRC, 1975) provides the most extensive and complete discussion of early and continuing somatic effects available at present. Section 9.3.5.1 discusses the concepts presented in Appendix VI and some of the uncertainty associated with the estimates of early somatic effects.

Late somatic and genetic effects have been studied by many groups, including the Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR) of the National Academy of Sciences, the United Nations Scientific Committee on Radiological Protection (UNSCEAR), the National Council on Radiation Protection (NCRP), the International Commission on Radiological Protection (ICRP), and the RSS Advisory Group on Health Effects. These groups have made estimates of the risk of adverse health effects from the effects observed in exposed populations. Many of these estimates are based on effects from high doses and high dose rates, with extrapolation to the low dose regions. At low doses and low dose rates, it is difficult to distinguish between health effects that result from radiation exposure and those that result from other causes. Many uncertainties are associated with these risk estimates, and some of these are addressed in Section 9.3.5.1.

The health-effects model described in Appendix VI of the RSS is the most commonly used model in codes like CRAC, CRAC2, and CRACIT. It is currently being reviewed by a group of scientists at Harvard University's School of Public Health, under the sponsorship of Sandia National Laboratories. A report by D. W. Cooper and co-workers, detailing the strengths and weaknesses of the model and areas needing further research, is expected to be published in 1983.

9.3.5.1 Early and Continuing Somatic Effects

After a reactor accident, the large doses required to produce early effects could result from several pathways: external irradiation from the passing cloud, external irradiation from ground contamination, and internal irradiation from inhaled radionuclides (see Section 9.3.3.5).

A set of criteria (see Section 9.4.8.3) for relating the dose received by individuals to the early fatalities and injuries that may arise within a year after an accident are detailed in RSS Appendix VI. Early fatalities were estimated by considering damage to the bone marrow, the lung, and the gastrointestinal (GI) tract. Early injuries include respiratory impairment, temporary changes in the GI tract, hypothyroidism, thyroiditis, temporary sterility, congenital malformations and growth retardations, cataracts, and prodromal vomiting. Early injuries are defined by the RSS as the responses that require medical attention. A convenient measure of early injuries is sometimes taken to be the number of people who receive more than 200 rem to the bone marrow or the whole body. These people would require hospital treatment. The consequence modeler should be aware that codes like CRAC2 base most of their illness estimates on impairments requiring medical attention. However, CRAC2 has an option available for considering numerous injuries in detail, if the user finds this type of analysis necessary.

Fatalities

Radiation doses to the bone marrow, the lung, and the GI tract would be the major contributors to the risk of early fatalities. These three organs should be treated on a conditional risk basis, to prevent overestimation. The probability of death from bone-marrow irradiation usually dominates the corresponding probabilities for the lung and the GI tract for LWR consequence calculations.

If an accident were to occur, it is presumed that the emergency response would include medical treatment to mitigate the adverse consequences that may result from high-dose exposures. The RSS established 60-day median lethal doses, $LD_{50/60}$ (i.e., the doses that would be lethal to 50 percent of the exposed population within 60 days), for varying degrees of medical treatment (minimal, supportive, and heroic) for bone-marrow exposures. An $LD_{50/60}$ of 340 rads was recommended by the RSS Advisory Group on Health Effects as the value to use in estimating health effects if only minimal medical treatment is available. With heroic treatment (e.g., bone-marrow transplants) the $LD_{50/60}$ value may be significantly increased, but these medical procedures may have adverse side effects that could decrease the survival rate. For supportive medical treatment, the RSS used an $LD_{50/60}$ of 510 rads (curve B of Figure VI 9-1 of the RSS). The consequence modeler should be aware that the fatality-risk estimates are extremely sensitive to the $LD_{50/60}$ value.

Considering the current medical expertise and future advances, it is reasonable to assume that supportive treatment would be available to persons exposed to high doses after a reactor accident. However, the availability of supportive treatment would depend on the number of people needing hospital treatment for high radiation exposure (more than about 200 rads to the bone marrow). The RSS estimated that, in the United States, 2500 to 5000 people could be given supportive treatment (based on 1975 medical facilities). The consequence analyst should be aware that if the number of individuals receiving acute bone-marrow doses (>200 rads) exceeds 5000, it is likely that some individuals would receive less than supportive treatment (e.g., minimal treatment).

Estimates of lung damage from inhaled beta-emitting radionuclides were adequately covered in the RSS. Other useful references for fatalities from inhaled radionuclides are reports by Filipy et al. (1980) and Hahn (1979).

In the RSS, the dose-response relationship for early fatalities is applied to a radiation dose that is the sum of the following:

1. External dose from the passing cloud.

2. External dose from contaminated ground (the duration of exposure to gamma rays emitted by deposited fission products, together with the degree of shielding, depends on the assumed emergency-response strategy).

3. Internal dose received during the first 7 days from inhaled radionuclides.

4. For bone-marrow exposure only, half of the internal dose from inhaled radionuclides received from day 8 through day 30.

5. For lung exposure only, the internal dose from inhaled radionuclides received from day 8 through day 365.

It can be seen that this is a specifically defined dose commitment. The consequence modeler should be cautioned that redefining the dosimetry assumptions used in the analysis would require redefinition of the dose-response relationships for early fatalities. Repair mechanisms may modify the effects of radiation exposure if the exposure is received over an extended period of time.

Injuries

The various types of impairment listed at the beginning of this section are detailed in Appendix VI of the RSS. A sublethal dose, defined as the dose expected to cause a clinical response in 10, 50, or 90 percent of the exposed population, was estimated for the various morbidities. These responses are not as easily determined as fatalities; thus the estimates have some subjectivity and increased uncertainty.

9.3.5.2 Late Somatic Effects

Late somatic effects consist of latent-cancer fatalities, nonfatal cancers, illnesses, and benign thyroid nodules. The RSS model included a latent period during which there was no increase in cancers and a plateau period with a uniform cancer rate for each cancer type.

The estimates of latent cancer calculated by the CRAC code are based on the BEIR I report (NAS-NRC, 1972), with the leukemia and bone-cancer values modified to reflect new data that became available between 1972 and 1975. The RSS developed three estimates of risk. The upper-bound estimate used the linear, no-threshold estimators from the BEIR I report (1972). The central estimate (see Section 9.4.8.4) incorporated a dose-effectiveness factor for exposures delivered at low dose rates. The lower-bound estimate took into account the large uncertainty in estimating effects from low doses and low dose rates and assumed a threshold of 10 or 25 rem for latent-cancer fatalities. The central-estimate approach is consistent with the BEIR III report (NAS-NRC, 1980), which used a linear-quadratic model to calculate risk estimators for latent-cancer fatalities. In addition, the BEIR III report published ranges that indicate some of the uncertainty associated with these factors. The upper and the lower bounds of the ranges were obtained with the linear model and the pure quadratic model, respectively. The risk estimates, based on the linear-quadratic model, of BEIR III (1980) are approximately 2 times lower than the BEIR I (1972) estimates based on the linear model.

Recently, Loewe and Mendelsohn (1980) conducted some preliminary reassessments of the dose data for people exposed by the atomic bombs at Nagasaki and Hiroshima. Since the BEIR estimates were calculated from these Japanese data, these reassessments could have some impact on the final estimates of latent-cancer risk. The Los Alamos National Laboratory is attempting to redefine the source term from the two bombs. In conjunction with this effort, the Oak Ridge National Laboratory is recalculating dose estimates. Final resolution of the health-effects estimate will likely follow these efforts. It is important that the consequence modeler be aware of these developments.

Except for leukemia, the latent-cancer fatalities presented in Table VI 9-4 of the RSS were calculated for a 30-year plateau period, whereas the BEIR I report (1972) used the remaining lifetime as the plateau period for "solid tumors." A comparison of the values obtained by assuming lifetime and 30-year plateaus is given in Table 9-9. (The lifetime plateau is implemented in the CRAC2 code.)

Table 9-9. Expected latent-cancer deaths per 10^6 man-rem of external exposure

	Expected deaths per 10^6 man-rem	
Type of cancer	CRAC health-effects model[a]	CRAC2 health-effects model[b]
Leukemia	28.4	28.4
Lung	22.2	27.5
Stomach	10.2	12.7
Alimentary canal	3.4	4.2
Pancreas	3.4	4.2
Breast	25.6	31.7
Bone	6.9	10.1
Other	21.6	28.0

[a]Based on a 30-year plateau period for all cancers except leukemia.

[b]Based on a lifetime plateau period for all cancers except leukemia.

In calculating the incidence of thyroid nodules, both benign and cancerous, the RSS assumed the following dose-response relationship, with allowance for the age distribution of the population:

Dose range (rem)	Expected thyroid nodules per 10^6 man-rem	
	Benign	Cancerous
<1500	200	134
1500-3000	100	67
>5000	0	0

It is assumed that the thyroid gland is ablated after doses higher than 5000 rem, requiring its surgical removal and the daily use of substitute hormone pills. Aldrich and Blond (1980), in a study of the effectiveness of potassium iodide as an emergency protective measure in the aftermath of nuclear reactor accidents, used a simplified model: the probability of a thyroid nodule is 3.34×10^{-4} per rem for doses not exceeding 3000 rem; above 3000 rem, the thyroid is assumed to be ablated.

In the RSS, the radiation dose to which the dose-response relationship is applied is given by the sum of (1) the external thyroid dose from the passing cloud, (2) the external thyroid dose from contaminated ground, (3) the internal dose delivered during the first 30 days by all inhaled radionuclides except iodine-131, and (4) one-tenth of the internal dose delivered during the first 30 days by iodine-131. The RSS offers clinical evidence for the assumption that iodine-131 irradiation causes a lower incidence of thyroid nodules than do external gamma rays. In addition, from a purely radiobiological standpoint, it is thought that the more uniform distribution of dose within the thyroid from external irradiation might increase the induction of clinical hypothyroidism.

The dose-effectiveness factor of 0.1 for iodine-131 was disputed by the American Physical Society Study Group on Light-Water Reactor Safety (APS, 1975), which assumed a range of factors from 0.3 to 1.0. This issue remains unresolved (Aldrich and Blond, 1980); however, the reader should be aware that, in the case of CRAC and CRAC2, the assumption of one-tenth effectiveness for iodine-131 is built into the code.

Finally, most thyroid cancers are well differentiated, slow growing, and treatable. The mortality rate for thyroid cancers is therefore much lower than that for other forms of cancer. The 10-percent mortality rate assumed in the RSS for malignant thyroid cancers is probably an overestimate.

Genetic Effects

The RSS used the BEIR I report (NAS-NRC, 1972) to prepare tables for potential genetic disorders per 10^6 man-rem of external and internal radiation exposure. These genetic effects include autosomal dominant disorders, multifactorial disorders, disorders due to chromosomal aberrations, and spontaneous abortions. The estimates of genetic effects in the BEIR III report (NAS-NRC, 1980) are based on new data that have become available since 1972, but they are not significantly different from the 1972 estimates. Therefore, it is reasonable to continue using the genetic estimators that are implemented in codes like CRAC. In most consequence analyses, genetic disorders are not part of the final output. The latent cancers tend to dominate the risk estimates of latent effects. However, when executing codes like CRAC, the user does have the option to provide estimates of genetic risk.

9.3.5.3 Discussion

As this chapter was being written and reviewed, it became apparent that the topics of dosimetry and dose-response relationships generate considerable scientific controversy. After detailed discussion, the authors have decided to make the following recommendations. First, the state of the art has not yet "solidified" to the extent that it is possible to recommend unequivocally a replacement for the RSS methods. Hence, the RSS remains the best comprehensive treatment of dosimetry and dose-response relationships _in the context of consequence modeling_, and its methods remain acceptable. Second, because considerable work has been done since the publication of the RSS, those who wish to try to update the methods are encouraged to do so. However, those who vary from the RSS values should use sources that have been subjected to a peer review, such as the BEIR III report (NAS-NRC, 1980), the UNSCEAR (1977) report, and ICRP Publication 26 (ICRP, 1977). Finally, as already mentioned, studies intended to update the RSS methods are in progress: the NRC is funding work on age- and sex-specific dose-conversion factors at the Oak Ridge National Laboratory, and work on health-effects modeling is under way at Harvard University's School of Public Health. When their results have been published, a comprehensive updating of the RSS methods in codes like CRAC2 will be in order.

9.3.6 ECONOMIC IMPACTS

The economic models of consequence-modeling codes require several cost elements. Thus, for example, the cost of evacuating a person is assigned a given value, and the total cost of evacuation is merely that figure multiplied by the number of people evacuated. The quantity of each agricultural product condemned is calculated and multiplied by the value of a unit quantity of that product. The economic model is thus a simple counting and adding routine in which the key factors are the unit costs of each countermeasure; these are required as input to the code and are discussed further in Section 9.4. An illustrative list of required cost inputs is as follows:

1. Evacuation cost per person.

2. Value of residential, business, and public areas per person.

3. Relocation cost per person.

4. Decontamination cost per acre for farm areas.

5. Decontamination cost per person for residential, business, and public areas.

6. Compensation rate per year for residential, business, and public areas (i.e., fraction of value).

7. Average value of farmland per acre for state, county, or smaller areas.

8. Average annual value of farm sales per acre for state, county, or smaller areas.

9. Miscellaneous information, such as seeding and harvesting month, fraction of land devoted to farming, fraction of farm sales due to dairy production.

Hence, the economic model essentially adds the costs incurred for evacuation, relocation, the interdiction of land use, and the destruction of crops. Note that it does not consider the cost of health effects (this would involve assigning a monetary value to life, which would be extremely controversial). It does not consider any economic multiplier effects (e.g., the effect on employment in one area if a factory in another has to close). There is no attempt to incorporate plant costs, although in the prediction of economic risk the in-plant property damage far exceeds the offsite property damage. As demonstrated by the accident at Three Mile Island, the costs of decontaminating and reconstructing the plant and of replacing power may be several billion dollars even though the offsite consequences were very small indeed.

9.4 INPUT-DATA REQUIREMENTS

As already mentioned, it is in the choice of input data that the user can most influence the outcome of the consequence analysis.

9.4.1 BASIC RADIONUCLIDE DATA

The user of a code like CRAC2 is generally expected to provide the inventory of radionuclides present in the reactor core at the time the accident sequence starts. The inventory used in the RSS was that appropriate for a 3200-MWt Westinghouse PWR. It was calculated for an end-of-cycle equilibrium core with the ORIGEN code (Bell, 1973) and has been used to represent both PWR and BWR cores. Differences in reactor size (i.e., thermal power level) are usually accommodated by a linear scaling of the radionuclide inventories.

Since the writing of the RSS, ORIGEN has been updated (Croff, 1980). Moreover, Sandia National Laboratories, using its own version of ORIGEN,*

*The Sandia version of ORIGEN will be described by D. E. Bennett in a forthcoming report entitled Radionuclide Core Inventories for Standard PWR and BWR Fuel Management Plans.

has calculated equilibrium inventories for a 3412-MWt Westinghouse PWR, a 3578-MWt General Electric BWR, and a 1518-MWt Westinghouse PWR. Inventories of selected radionuclides are shown in Table 9-10 (Ostmeyer, 1981) as multiples of the inventory for the 3412-MWt PWR.

The significant differences between the long-lived-radionuclide inventories for the RSS 3200-MWt PWR and the 3412-MWt PWR are due to the assumption of a 25 percent greater fuel burnup for the latter (26,400 MWd/tonne in the RSS, subsequently increased to 33,000 MWd/tonne). In general, the inventories of the short-lived radionuclides are proportional to the power level. The most significant impact of the greater inventory of long-lived radionuclides, especially Cs-137, is to increase the predicted number of latent-cancer fatalities roughly in proportion to the increase in inventory. For the 3412-MWt PWR, the land-interdiction and decontamination results are approximately 30 percent larger than those of the RSS, while those for the 3578-MWt BWR are approximately 50 percent larger again. Differences for other consequences, such as early fatalities, which are more directly dependent on the inventory of short-lived radionuclides, are smaller.

Inventories most representative of the reactor being studied should be employed for reactor-accident consequence calculations. This would include using BWR inventories, if available, for BWR consequence calculations and using radionuclide inventories representative of the reactor-power level under consideration.

Once an inventory is available, it is necessary to reduce the number of radionuclides to manageable proportions, since ORIGEN considers 246 activation products, 461 fission products, and 82 transuranics. Many of these nuclides are not radioactive, but even so, the total number of radionuclides runs to several hundred. Many of them can be eliminated without significantly affecting the results of the radiation-dose calculations. The elimination process in the RSS was based on a number of factors, such as (1) quantity (curies); (2) release fraction; (3) emitted-radiation type and energy; (4) chemical characteristics; and (5) half-life. It is possible to eliminate many short-lived radionuclides because the shortest interval between the termination of the chain reaction (accident-sequence initiation) and the beginning of the escape of radionuclides to the atmosphere is 30 minutes (for the accident sequences presented in the RSS).

In the RSS, the list of nuclides considered was thus reduced to the 54 shown in Table 9-11. Recent studies have shown that, at least for LWRs, this list is more than adequate. Table 9-11 also contains some examples of nuclides that are important in some of the radiation pathways discussed in Section 9.3, for releases from LWRs only. These are taken from Chapter 13 of Appendix VI of the RSS, which presents many more details.

Finally, the list of radionuclides should be accompanied by standard information on parents and/or daughters in a radioactive-decay chain and the radioactive half-life. Also required in principle is the spectrum of the gamma rays emitted by the decaying radionuclides, in order to calculate cloudshine and groundshine (see Section 9.4.8.2) and the deposition velocity and washout coefficient (Section 9.4.5).

Table 9-10. Inventory of selected radionuclides for various reactors[a]

Nuclide	Half-life (days)	Inventory (Ci) End of cycle 3412-MWt PWR	End of cycle 3200-MWt PWR[b]	End of cycle 3578-MWt BWR	End of cycle 1518-MWt PWR	1/3 cycle 3412-MWt PWR	2/3 cycle 3412-MWt PWR
Kr-85	0.117	6.64×10^5	1.03	1.36	0.44	0.68	0.84
Mo-99	2.8	1.66×10^8	0.94	1.05	0.45	1.02	1.01
Tc-99m	0.25	1.43×10^8	1.00	1.05	0.45	1.03	1.01
Ru-103	39.5	1.25×10^8	0.85	1.06	0.44	0.87	0.96
Ru-105	0.185	8.22×10^7	0.88	1.07	0.43	0.86	0.94
Ru-106	366	2.90×10^7	0.86	1.24	0.42	0.66	0.83
Te-129m	0.34	6.70×10^6	0.79	1.06	0.44	0.88	0.96
Te-131m	1.25	1.28×10^7	1.00	1.07	0.44	0.97	0.98
Te-132	3.25	1.27×10^8	0.92	1.06	0.45	1.00	1.00
Sb-129	0.179	2.72×10^7	1.22	1.06	0.44	0.93	0.97
I-131	8.05	8.74×10^7	0.98	1.06	0.45	0.99	1.00
I-132	0.096	1.29×10^8	0.92	1.05	0.44	0.99	1.00
I-133	0.875	1.84×10^8	0.94	1.05	0.45	1.02	1.01
I-134	0.037	2.02×10^8	0.95	1.05	0.45	1.02	1.01
I-135	0.28	1.73×10^8	0.88	1.06	0.45	1.02	1.01
Cs-134	750	1.26×17^7	0.60	1.20	0.38	0.55	0.76
Cs-136	13.0	3.91×10^6	0.77	1.04	0.41	0.67	0.84
Cs-137	11,000	6.54×10^6	0.72	1.39	0.44	0.67	0.83
Ba-140	12.8	1.68×10^8	0.94	1.05	0.45	1.02	1.01
Ce-144	284	9.15×10^7	0.92	1.14	0.45	0.77	0.90

[a]From Ostmeyer (1981).
[b]The reference PWR for the Reactor Safety Study.

Table 9-11. Radionuclides considered in the Reactor Safety Study consequence analysis[a]

Element	Radionuclide	Element	Radionuclide
Cobalt	Co-58,* Co-60*	Iodine	I-131,[c,g,h,i] I-132,[b,g,h] I-133,[b,g,i] I-134, I-135[b,g,h]
Krypton	Kr-85,* Kr-85m,* Kr-87,* Kr-88[b]		
Rubidium	Rb-86*	Xenon	Xe-133, Xe-135
Strontium	Sr-89,[c] Sr-90,[d,e] Sr-91	Cesium	Cs-134,[c] Cs-136, Cs-137[j]
Yttrium	Y-90,* Y-91	Barium	Ba-140[c]
Zirconium	Zr-95, Zr-97	Lanthanum	La-140
Niobium	Nb-95*	Cerium	Ce-141, Ce-143,* Ce-144[f]
Molybdenum	Mo-99	Praseodymium	Pr-143*
Technetium	Tc-99m*	Neodymium	Nd-147*
Ruthenium	Ru-103, Ru-105,* Ru-106[f]	Neptunium	Np-239
Rhodium	Rh-105*	Plutonium	Pu-238,[e] Pu-239, Pu-240, Pu-241[e]
Tellurium	Te-127,* Te-127m, Te-129,* Te-131m, Te-132[b,c,g]	Americium	Am-241*
		Curium	Cm-242, Cm-244
Antimony	Sb-127, Sb-129		

[a]Applicable to releases from LWRs only. The radionuclides marked with an asterisk are negligible contributors to health effects. The most significant contributors are signaled with superscript letters for the modes or effects listed below.

[b]Cloudshine.
[c]Inhalation (early effects).
[d]Leukemia (inhalation dose).
[e]Bone cancer (inhalation dose).
[f]Lung cancer (inhalation dose).
[g]Groundshine (early effects).
[h]Thyroid dose.
[i]Milk-ingestion pathway.
[j]Long-term groundshine.

9.4.2 SPECIFICATION OF THE SOURCE TERM

It is necessary to obtain data that will have been calculated by the methods described in Chapters 7 and 8. For simplicity, this will be referred to as the specification of the source term. An example of this sort of information, taken from the RSS, appears in Table 9-1, the various entries of which are discussed below.

Some of the discussion that follows goes into greater depth than is required for the straightforward use of codes like CRAC2, but, because of the current rapid developments in this field, it is necessary to be able to talk intelligently about the likely future impact on consequence modeling.

9.4.2.1 Magnitude of Radionuclide Releases to the Atmosphere

The activity of each radionuclide escaping to the atmosphere is calculated by the methods described in Chapter 8. In the RSS, the various radionuclides were classified into eight groups (see Table 9-1), mainly on the basis of their volatility. It is still common practice to classify radionuclides in this way in order to reduce the burden of calculation with codes like CORRAL.

The rate of radionuclide release to the atmosphere is, in fact, time dependent, as described in Chapter 8. Current consequence-modeling codes are generally not able to handle this, but since the next generation of radionuclide-transport codes are likely to predict varying rates of release, it is possible that in the future updated versions of consequence models that can accept this input will be required.

9.4.2.2 Timing

Various times are important input to consequence calculations: the time of release, the duration of release, and the warning time.

The time of release, which is a parameter calculated in the modeling of physical processes, is the interval between the start of the accident and the predicted start of the release of radionuclides to the atmosphere. In some consequence-modeling codes, this interval is used to attenuate the source term by the process of radioactive decay. In general, the interface between radionuclide-transport codes and consequence-modeling codes allows for what is rather an artificial means of accounting for this radioactive decay. Future generations of radionuclide-transport codes may include a better treatment of this effect and require a more sophisticated way of handling input on the part of consequence-modeling codes.

The time of release can generally be obtained from the output of codes like MARCH and CORRAL. For accident sequences in which core melt occurs before containment failure, the time of containment failure is also the time of release. For sequences in which the containment fails first, the time of release can be taken as the time at which the gap release or the melt release from fuel would be predicted to occur.

The duration of release, which is generally calculated from the output of a radionuclide-transport code like CORRAL, could be used to allow for (1) radioactive decay and (2) the broadening of the plume by the action of large-scale turbulent eddies in the atmosphere. In principle, changes in wind direction during this period could also be taken into account (see Appendix D4.2) and used as input to an evacuation model for an area affected by a complicated plume like that shown in Figure D-12.

The warning time is the period between the awareness of an impending core melt and the release of radioactivity. This parameter is important in evacuation models. It should in principle be obtained by comparing the emergency-plan implementing procedures (EPIPs) with the output of a code like MARCH. The EPIPs give criteria for the declaration of a general emergency, on the basis of such quantities as the water level in the core and radiation or pressure readings in the containment. From the MARCH output one can usually deduce when these readings will be reached.

9.4.2.3 The Elevation of Release and the Dimensions of the Release

Data on these parameters are obtained from the modeling of physical processes. The simplest models of the atmospheric dispersion of radionuclides assume a point-source release from a known elevation. If the source is assumed to be of this nature, the elevation of release should be provided.

If the release does not emerge from a well-defined source, or if that source is on the face or the roof of a reactor complex, it is assumed that mixing will occur throughout the reactor-building wake. In this case, the dimensions of the reactor complex are needed (i.e., the height and width of the projection perpendicular to the wind direction).

9.4.2.4 Buoyancy

Plume rise can be important in reducing the predicted values of the mean public risk, particularly for early effects. The minimum of information required is the rate of energy release that accompanies the escaping radionuclides, and this should emerge from the modeling of physical processes as described in Chapter 7. The MARCH code contains an output variable, "QSENS," that, as is explained in an associated comment card, is intended for use in the CRAC plume-rise model. Examples are given in Table 9-1.

Ideally, momentum effects should also be considered. In principle, the orientation, size, and momentum of an escaping jet should influence plume rise. In practice, predictions of these effects are not likely to be feasible in the near future, and, in any case, most consequence-modeling codes cannot handle such information.

9.4.2.5 Particle-Size Distribution

This section has more to do with what may be possible in the future than with what is feasible now. The particle-size distribution can affect, among other factors, the dry-deposition velocity v_d (see Appendix D3.1), the washout coefficient, and the health-physics calculations (e.g., the dose-conversion factors for the inhalation pathway; see Section 9.4.8.1).

Some particle-size distribution must be assumed, either explicitly or implicitly. For example, the RSS compilations of inhalation-dose-conversion factors are for an aerosol with an activity median aerodynamic diameter (AMAD) of 1 μm, distributed lognormally. The range of particle diameters that are implicitly assumed to be compatible with the RSS assumption of a dry-deposition velocity of 1 cm/sec is discussed in Appendix D3.

It is very likely that the considerable amount of research that is under way on radionuclide source terms will produce a mass of information on the size distribution of particles emitted to the atmosphere in a severe reactor accident. It might also provide information on particle composition, since such particles will be agglomerates made up of fuel, structural materials, and fission products. It might show that different nuclides may be preferentially associated with different particle sizes. As discussed in Appendix D3.1, such considerations would imply a need for considerable changes in the deposition model normally used in consequence-modeling codes and would also require considerable extra sophistication in the input to such codes.

In practice, the user of a consequence-modeling code is likely to find that he has little freedom of choice. The health-physics parameters that are generally built into a code, where they reside in a data bank, will have been calculated on the assumption of a given particle-size distribution, and this distribution usually has an AMAD of 1 μm. The deposition velocity and washout coefficients are also input without much consideration of particle diameter. Hence, the user will generally not be required to make an explicit decision about particle sizes.

9.4.2.6 Chemical Properties

The chemical properties of the released radionuclides will influence the subsequent health-physics calculations--for example, the classification of elements into the three inhalation classes D, W, and Y, which represent respiratory-clearance half-times on the order of days, weeks, and years, respectively. It is therefore necessary to obtain either a direct classification of elements as D, W, or Y or enough information about their chemical properties for a classification to be made. Again, in practice, the user may well find that his consequence-modeling code contains a data bank of inhalation factors that are calculated on the basis of assumed inhalation classes for radionuclides, classes that are thought to be characteristic of releases from LWRs.

9.4.2.7 Moisture

The present state of the art does not permit the realistic modeling of the behavior of wet plumes like those that might be emitted from an LWR during a severe accident. The reason given for this judgment is that the present understanding of radionuclide source terms does not include the ability to predict the state of any moisture that may be carried away from the reactor by a buoyant plume, whether entirely as vapor or as a fine aerosol. The greatest potential impacts on consequence modeling may well be due to a possible decrease in the predicted final height to which a hot plume rises. This decrease would be attributable to the loss of buoyancy after the evaporation of liquid droplets. Alternatively, the rainout of radionuclides after the condensation of water vapor on particles within the plume could result in the deposition of most of the radioactive material within a short distance of the reactor.

Current or planned research into radionuclide source terms will certainly include efforts to improve predictions of the spatial and temporal variations in the state of the water in the reactor system, and this should lead to an improved understanding of the quantity and the state of the water emerging into the atmosphere.

9.4.2.8 Release Categories and Their Frequencies

In the course of a PRA, it becomes necessary to choose a limited number of source terms for analysis. It is not unusual for a PRA to identify hundreds of accident sequences, and it is out of the question to examine all of these with codes like MARCH, CORRAL, and CRAC2. It is therefore essential to sort the sequences into a small number of groups for which the source terms are expected to be similar.

This grouping of accident sequences is a somewhat subjective exercise and is more properly the province of Chapters 7 and 8. However, the following guidelines are appropriate at this point:

1. Source terms of similar magnitude but with very different times of release or warning times should not be grouped together, as the effect of the evacuation procedure on the CCDF for early fatalities or injuries will be very different.

2. It is often possible to use arguments based on the relative probability of accident sequences to show that the public-risk contribution of one is necessarily much smaller than the contribution of the other.

The frequency of occurrence that is predicted for each radionuclide release category is the sum of the frequencies of the accident sequences in that category. The frequencies of the individual accident sequences are obtained from the quantification of event trees and from the subsequent containment analyses.

9.4.3 METEOROLOGICAL DATA

A code like CRAC2 requires a file of hourly meteorological data: wind speed, wind direction, stability category, and intensity of precipitation. The most obvious place to look for these data is at the reactor site, since each reactor site in the United States has a program of meteorological measurements, as required by the NRC's Regulatory Guide 1.23. For example, at one U.S. site the program of measurements includes (1) wind speed at 300 and 30 feet; (2) wind direction at 300 and 30 feet; (3) ambient temperature and dewpoint temperature at 30 feet; (4) temperature difference between the 300- and 30-foot and the 100- and 30-foot levels, measured with matched sensors; and (5) precipitation intensity.

After these data have been obtained, they must be processed into a form compatible with the consequence-modeling code that is being used. As mentioned above, CRAC2 requires a data file containing hourly values of wind direction, wind speed, stability category, and precipitation intensity. These hourly values are required for at least a year; ideally, more than one year of data should be used to increase the likelihood that all possible weather conditions are covered. Usually, the wind speed would be chosen for a height of 30 feet. As discussed in Appendix D1, a scheme based on two parameters, such as temperature difference and wind speed, should be used to determine the stability category, if possible. Finally, the precipitation intensity can obviously be determined from the precipitation measurements.

It is usually necessary to take the advice of a trained meteorologist while compiling the meteorological data file. Sometimes there are sequences of hours or even days when data are incomplete or inconsistent and expert judgment is required to fill in the gaps. Sometimes the site data are so bad that alternative sources must be found. Here again, a trained meteorologist should be consulted, and he will probably suggest the use of data from a nearby airport or from the National Weather Service. In looking for substitute meteorological data, it is clearly important to ensure that the meteorological characteristics of the substitute site are as close as possible to those of the reactor site. If there are terrain features that greatly affect the wind rose or considerable differences in precipitation patterns, the substitute may not be acceptable. If care is taken in choosing such data, however, experience in the use of codes like CRAC2 suggests that the errors introduced into the results of the consequence modeling are not great.

Other codes may require data in a different form. For example, CRACIT uses the wind speed and direction at more than one height. It can also use meteorological data from a number of sites in the neighborhood of a reactor. Each code manual should contain instructions detailing the data that are needed.

The user new to consequence modeling should note that the processing of the meteorological data looks deceptively simple. Problems inevitably arise, however, usually in connection with the accessibility, quality, and completeness of the data.

9.4.4 POPULATION DATA

Codes like CRAC invariably require that the population be assigned to the elements of a grid like that shown in Figure 9-8. Needless to say, the population grid should be taken consistent with the available meteorological data. Particular care should be taken to ensure that, when the meteorological data show the wind blowing toward, say, the north, it is the population north of the site and not south of the site that is assumed to be affected. This is a trap into which users of CRAC and CRAC2 often fall.

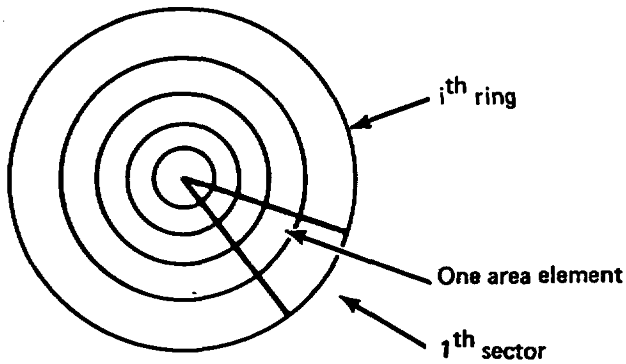

Figure 9-8. Population grid in CRAC.

The available data base is the U.S. census, 1970 being that which is currently available. (The 1980 data should be released soon.) These data give the number of people residing within a census enumeration district (CED). Where the CED lies entirely within an area element, it is simple to assign all of the population to that element. Where two or more elements lie within a CED, there are generally two alternative procedures. One is to assume that the population is uniformly spread over the CED. Another is to assign the population to some "center of gravity" of the CED. Each of these methods can assign people to places where there are none or underestimate the number of people in a given area. The problem becomes severe within 5 to 10 miles, where CEDs may overlap many grid elements. Within this region, it is advisable to carry out house counts and make use of aerial photographs.

Often, the data are required for years ahead, the predicted midlife of the plant being a typical example. This requires extrapolation, using known growth rates. These growth rates are usually available at the county level.

9.4.4.1 Transient Populations

In many areas, the population varies considerably with the time of year. Holiday resorts are an obvious example. In principle, this variation can be accommodated by collecting seasonal population distributions and causing the computer code to call the appropriate one for each weather sequence.

9.4.4.2 Diurnal Variations

There are large daily variations in population as people go to and return from work. The authors are not aware of any risk assessment in which this effect has been taken into account. In principle, information could be collected on places of work, schools, and the like. If a weather sequence (called by the CRAC2 sampling routine, for example) falls during working hours, the appropriate population distribution could be used. This variation in population could also affect evacuation and sheltering strategies. For example, people at work are more likely to be in or near large buildings with good sheltering characteristics than they are when at home (see Appendix E).

9.4.4.3 Computational Grid

In CRAC2, the population grid defines the computational grid since, within each element of the grid, such quantities as the plume-centerline airborne concentration are taken to be equal to the value calculated at the center of the grid. Other codes, such as TIRION (Kaiser, 1976), have computational grids that are not identical with the population grid. In general, consequence-modeling codes all introduce some form of computational grid within the framework of which certain approximations are made. An example is an approximation of the deposition calculation that is discussed in Section D3.2.1. In general, care should be taken to ensure that the calculational grid is not too coarse. If there is any doubt, calculations should be repeated with a finer grid. An instructive discussion of grids appears in the description of CRACIT, which uses a finer grid for dose calculations (Commonwealth Edison Company, 1981).

9.4.5 DEPOSITION DATA

The user of consequence models will probably find that his chosen code has available, in a data bank or in a standard input set, values of the dry-deposition velocity v_d and of the washout coefficient Λ for each nuclide.

As explained in Appendix D3, there are great uncertainties in the choice of dry-deposition velocities. The chosen code is very likely to assign v_d values of 1 or 0.3 cm/sec for particulate matter, and there is no reason for changing these values unless a sensitivity study is envisaged. In that case, v_d could be varied between 0.1 and 10 cm/sec. At the higher values of v_d, however, the models in some codes may become invalid.

In CRAC, Λ is assigned a value of 10^{-4} sec^{-1} for rainfall in neutral and stable weather conditions and 10^{-3} sec^{-1} in convective conditions. Sensitivity studies should allow Λ to vary from 10^{-5} to 10^{-2} sec^{-1}. If the user of consequence models wishes to justify different values of Λ, he will need a good understanding of what determines the value of Λ (e.g., particle size and rainfall characteristics); he will need to undertake a fairly ambitious review of work like that of Slinn (1977, 1978); and he will need to "benchmark" the changes to his code.

9.4.6 EVACUATION AND SHELTERING DATA

As has been explained, it is in the specification of input parameters for the evacuation model that the user can most influence the outcome of a consequence calculation. Accordingly, Appendix E is devoted to a thorough discussion of evacuation models, including the required input. The omission of a detailed discussion of evacuation and sheltering at this point is a deliberate decision on the part of the authors, who have provided Appendix E as a thorough, self-contained discussion of the subject.

9.4.7 ECONOMIC DATA

Perhaps the most useful thing that can be done here is to list as an example the data that CRAC2 requires as input.

9.4.7.1 Evacuation Cost

The modeler needs to provide the cost of evacuating a person and providing food and shelter for a few days, say, a week. These data can be obtained from such sources as an EPA study of 64 evacuations after disasters in the United States (Hans and Sell, 1974). The figure used in the RSS is $95 per person; this would have to be updated to allow for inflation, as is true for all economic costs. According to the 1980 issue of the Statistical Abstract of the United States (published by the U.S. Department of Commerce), the Consumer Price Index rose from about 150 to about 225 between 1974 and 1980. An approximate 1980 estimate of the evacuation cost would thus be about $158 per person. The contribution of evacuation costs to the total cost is simply the number of people evacuated times the individual evacuation cost.

9.4.7.2 Relocation Cost

The relocation cost essentially consists of an allowance for loss of income while an individual moves and finds a new job or while the corporation for which he works moves. It also includes a per capita allowance for household and business moving expenses. Chapter 12 of Appendix VI of the RSS (USNRC, 1975) gives details of how these components were shown to add up to $2900 in 1975, and in CRAC2 the figure (updated to 1980) is $4344. Again, CRAC2 simply multiplies this figure by the number of people who are relocated.

9.4.7.3 Value of Developed Property and Farm Property

In the RSS, the value of a property is assumed to be its market value. The CRAC2 model requires a per capita estimate of the value of depreciated residential, business, and public property over the interdiction period. The RSS figure for 1974 of $17,000 per head was obtained from the National Bureau of Economic Research; the authors of CRAC2 estimate $31,527 per head as a nationwide average. Again, the contribution to the total cost is the product of the number of people who are relocated and this per capita figure.

For farms, an estimate of the farm value per acre depreciated over the interdiction period is required. These figures vary from state to state and from county to county and can be found in such publications as Agricultural Statistics, published by the U.S. Department of Agriculture, for state averages and the County and City Data Book, published annually by the U.S. Department of Commerce, for county averages. Typical 1980 values for states vary from $100 per acre in New Mexico to $2222 per acre in New Jersey. Also required are the fractions of the area of each state that are devoted to farming. These can be found in the same Data Books.

In the RSS, the value of farmland was input state by state. In order to make use of this information, it is also necessary to identify the state to which each element of the grid of Figure 9-8 belongs. For site-specific studies, this may not be good enough. For example, the County and City Data Book for 1977 shows that, in the State of Arizona, the average value of farmland was $111 per acre. County by county, however, the values varied from $41 to $877 per acre.

If the reactor in question is situated in one of the wealthiest or poorest counties of its state, using the average for the state could give misleading answers. Hence, it may be necessary to identify elements of the population grid by county. Even within a county, the value of farmland may vary greatly; for example, farmland in the valley of a river flowing through a desert would be at a premium. Breaking the farm values down into sub-county areas requires the expenditure of considerable extra resources in the collection of local data, however.

9.4.7.4 Depreciation

If a property has an initial value made up of V_L, the value of the land, and V_I, the value of improvements, these improvements will deteriorate through the lack of maintenance at an annual rate of depreciation d_p. Assuming that the land itself retains its value in real terms, the value of the property after t_y years will be V_T, where

$$V_T = V_L + V_I \exp(-t_y d_p) \tag{9-30}$$

CRAC2 requires input for d_p, which is judged to be about 20 percent for interdicted land (USNRC, 1975, Appendix VI, Section 12.4.2.1). For residential, business, and public property, V_I is usually valued at about 70 percent of $V_L + V_I$. For farm property, the corresponding figure is 25 percent. Essentially, CRAC2 calculates the difference between V_{T_I} and

$V_L + V_I$, where T_I is the interdiction period, and uses this as the basis for estimating the cost of interdicting land.

9.4.7.5 Crop Loss

As explained in Section 9.3.3.3, the CRAC2 code considers the ingestion of milk and crops, and works on the assumption that these will be destroyed if their consumption would deliver unacceptable radiation doses to various organs. For a given element on the CRAC2 spatial mesh, the annual sales value of farm produce per acre is required, and from this it is simple to calculate the total cost of destroying the crops. It is also necessary to know the fraction of farmland devoted to dairy products; from this and the average value of sales per acre the cost of destroying milk alone can be estimated. Additional input that is required consists of the seeding month and the harvesting month. If the deposition of radionuclides occurs outside the growing season, it is assumed that the crops are not to be destroyed. All this information is fed into CRAC2 state by state or county by county.

9.4.7.6 Fraction of Habitable Land

It is necessary to know the fraction of habitable land for each element of the population grid. It is essentially the fraction of the area of that element that is fit for human habitation and excludes mountains, lakes, rivers, and oceans. It is sufficient to estimate these figures crudely from a map.

9.4.7.7 Decontamination Costs

CRAC2 requires as input decontamination costs for farmland and developed land. The calculation of these costs is discussed in the RSS (USNRC, 1975, Appendix VI, Chapter 12).

The methods that would be considered for decontaminating farmland are to scrape the surface and dispose of it, to bury surface soil in place by grading, or to bury surface soil in place by deep plowing. The RSS estimates of costs for these activities were based on data from the Robert Snow Means Company (1974), Mohon (1974), and the U.S. Department of Agriculture (1974). A typical cost in the CRAC2 manual (Ritchie et al., 1981a) is $499 per acre (1980 dollars).

The methods that would be considered for the decontamination of developed property range from the firehosing of roofs and paving, and the replacement of lawns (which would give a decontamination factor of about 2) up to the replacement of roofs and paving (for a decontamination factor of about 20). A 1979 estimate of the cost of achieving a decontamination factor of 20 is $3349 per acre, taken from the CRAC2 manual.

9.4.7.8 Discussion

The foregoing inputs for economic costs serve to illustrate the kind of data that may be required if the chosen consequence-modeling code contains an economic impact routine. Table 9-12 summarizes some of the important inputs to the economic subgroup of a consequence model. These figures are examples only and are not necessarily valid for all applications of CRAC2.

Table 9-12. Examples of important input to the economic subgroup of CRAC2[a,b]

Evacuation cost per person	\$158
Relocation cost per person	\$4344
Value of developed property, per person	\$31,527
Decontamination cost for developed property (DF 20), per person	\$3349
Decontamination cost for farmland (DF 20), per acre	\$499
Depreciation rate per year for developed property (fraction of value)	0.2
Value of farm property (state averages), per acre	From \$100 (New Mexico) to \$2222 (New Jersey)
Value of annual farm sales (state averages), per acre	From \$15 (Wyoming) to \$500 (Delaware)
Fraction of sales--dairy products (state averages)	From 0.024 (Wyoming) to 0.791 (Vermont)
Fraction of land devoted to farming (state averages)	From 0.077 (Maine) to 0.795 (Illinois)

[a]From the CRAC2 user's manual (Ritchie et al., 1981a).
[b]All figures are in 1980 dollars.

9.4.8 HEALTH PHYSICS

The health-physics calculations carried out by a code like CRAC2 require vast quantities of information (e.g., the inhalation-dose-conversion factors described in Section 9.3.3.1). A typical run of CRAC2 will make use of up to 54 radionuclides and 13 body organs, with the inhalation factors calculated for seven time periods--a total of some 5000 numbers. Most consequence-modeling codes have a data bank containing these quantities. The drawback to this is that the modeler has then imposed on the user his own views about the chemical form, the particle-size distribution, and other properties of the released radionuclides.

It is instructive to review the information required in a code like CRAC2, should the user wish to input his own information on various parameters associated with the health-physics calculations.

9.4.8.1 Inhalation Factors

Among the data files in CRAC2 is one containing dose-conversion factors for the inhalation pathway, $F^i_{n,k}(t)$. Here (n,i) identifies the radionuclide in question (i.e., the i^{th} daughter of the n^{th} radionuclide-decay chain). As already mentioned, there are 54 such nuclides. The subscript k identifies the organ, of which there are 13: lung, bone marrow, skeletal bone, endosteal cells, stomach wall, small intestine, upper large intestine and lower large intestine, thyroid, whole body, testes, ovaries, and "other" tissues.

The variable t identifies the time periods, of which there are seven: (1) period for acute exposure (1 year for the lung; 7 days for the marrow, skeletal bone, endosteal cells, stomach wall, small intestine, upper large intestine and lower large intestine; 2 days for the thyroid, whole body, testes, ovaries, and other tissues); (2) 1 year; (3) 1 to 10 years; (4) 10 to 20 years; (5) 20 to 30 years; (6) 30 to 40 years; and (7) 40 to 50 years.

All of the above quantities are required in various dose-response relationships. (See Section 9.4.2.5 for a discussion of inhalation factors and particle sizes.)

9.4.8.2 Dose-Conversion Factors: External Irradiation

CRAC2 contains the quantity $G^i_{n,k}(t)$, which is an array containing the dose-conversion factors (rem/Ci-m^2) for exposure to contaminated ground. As before, (n,i) identifies the radionuclide and k the organ; t is a variable specifying (1) the 8-hour integrated dose delivered to organ k by isotope (n,i); and (2) the 7-day integrated dose. Also contained in this array are dose-rate-conversion factors (rem$\cdot m^2$/Ci$\cdot$yr). These factors are used for calculating chronic groundshine doses. The doses are obtained by multiplying the initial deposited activity of each radionuclide by the corresponding element of $G^i_{n,k}(t)$.*

CRAC2 also contains a quantity, $C^i_{n,k}$, giving the radiation dose accumulated by organ k as a result of exposure to 1 Ci-sec/m^3 of radionuclide (n,i), that is, the dose-conversion factor for cloudshine.

Note that the elements of $G^i_{n,k}(t)$ are calculated with an infinite-plane approximation and the elements of $C^i_{n,k}$ are calculated with a semiinfinite-cloud approximation. As explained in Section 9.3.3.2, cloud-shape correction factors must subsequently be applied.

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*Care must be taken to treat daughter buildup correctly. If, as stated above, the doses are obtained by multiplying the initial deposited activities by $G^i_{n,k}(t)$, then radioactive decay is not explicitly calculated. The quantity $G^i_{n,k}(t)$ should implicitly take account of daughter buildup over the time t. If this is not done, some radiation doses can be significantly underestimated.

If the code in question makes direct use of an integral over the cloud or ground to calculate the above quantities, information must be supplied on the exposure-buildup factor and on the gamma-ray spectrum of each radionuclide. Kaiser (1976) shows how this can be done.

9.4.8.3 Computation of Early Health Effects

Figure 9-9 illustrates the information required for each early health effect that is considered. The quantity D_{jk} contains four dose-limit values (J = 1-4), with k specifying the organ in question. The quantity D_{1k} is the threshold below which the probability of the effect is zero; D_{4k} is the dose value above which the probability of the effect is 1; D_{2k} and D_{3k} are intermediate values corresponding to the probabilities given by P_{1k} and P_{2k}, respectively. The quantities D_{jk} and P_{jk} thus specify the probability of the effect over the entire dose range. The model assumes that the points described by D_{jk} and P_{jk} are connected by straight lines. The quantity P_{jk} contains two probabilities at which the slope of the dose-response relationship changes.

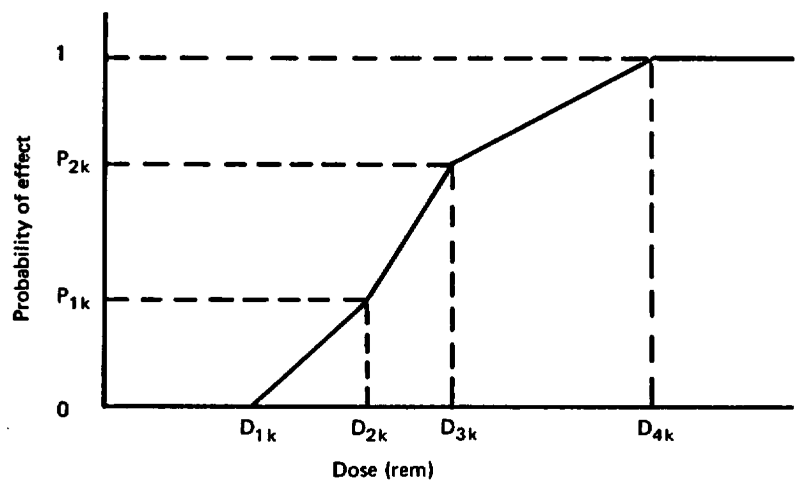

Figure 9-9. Dose-response model.

To give examples, the greatest cause of early fatalities is generally the accumulation of radiation dose in the bone marrow, where the radiation dose is specified by 0.5 (7-day + 30-day doses from inhalation) + dose delivered by the passing cloud + dose delivered within 7 days by deposited gamma emitters. The dose-limit values are 320, 400, 510, and 615 rem. The two probabilities contained in P_{jk} are .03 and .5. This is a simulation of the dose-response relationship with "supportive medical treatment" described in Appendix VI of the RSS.

For acute injuries, irradiation of the whole body is generally the most important indicator of whether hospitalization is required. The whole-body radiation dose, including the inhalation plus external-irradiation components, is used in a dose-risk relationship for which the dose limits are 55, 150, 280, and 370 rem. The two associated probability limits are .05 and 1.

Some modelers introduce more complicated dose-risk relationships with more than four dose limits or different shapes. Fryer and Kaiser (1979) give examples.

9.4.8.4 Computation of Latent Effects from Early Exposure

The key to the computation of latent-cancer effects is a population-dose conversion factor. The product of this factor and the radiation dose accumulated in a given organ gives the probability that cancer will develop in that organ. Some codes apply a single conversion factor to a single dose accumulated over a 50-year period. Typical values are 2×10^{-5} per rem delivered to the lung for lung-cancer induction and 10^{-5} per rem delivered to the bone marrow for leukemia (see Fryer and Kaiser, 1979). These figures, taken from the publications of the United Kingdom's National Radiological Protection Board, are consistent with the deliberations of the International Commission on Radiological Protection.

CRAC2 is somewhat more sophisticated in that it requires conversion factors for radiation doses delivered over different periods of time. An example for lung cancer is as follows:

1. 2.2×10^{-5} for radiation doses delivered during the first year.
2. 2.2×10^{-5} for years 1 to 10.
3. 2.2×10^{-5} for years 11 to 20.
4. 1.5×10^{-5} for years 21 to 30.
5. 8.1×10^{-6} for years 31 to 40.
6. 4.0×10^{-6} for years 41 to 50.
7. 1.5×10^{-6} for years 51 to 60.
8. 2.2×10^{-7} for years 61 to 70.

These figures reflect the fact that radiation doses delivered 50 to 60 years after the accident would manifest themselves as cancers, if at all, at least 2 to 40 years later, by which time the individual in question would almost certainly have died for other reasons; that is, these figures reflect the changing age distribution of the population that was initially exposed to the radioactive plume.

CRAC2 also allows a variation of the linear hypothesis, the central estimate discussed in Section 9.3.5. Typically, the dose-effectiveness factors would be reduced by a factor of 5 for radiation doses not exceeding 30 rem and by a factor of 2.5 for doses between 30 and 300 rem. As explained in Section 9.3.5, this is not yet a completely proved procedure. The reason it was used in the RSS is as follows: the risk estimates based on the linear extrapolation are taken from the BEIR I report (NAS-NRC, 1972). In 1975, however, the National Council on Radiation Protection and Measurements (1975) issued a report warning that the BEIR I estimates, which were derived from large radiation doses at high dose rates, are very likely to overestimate the risk from low radiation doses delivered at low dose rates. The use of the central estimate was intended as a more realistic estimate of risk, and as has been discussed in Section 9.3.5.2, the central estimate is consistent with the linear-quadratic model of the

BEIR III report (NAS-NRC, 1980), which is suitable for estimating the somatic effects of radiation with a low linear energy transfer.

9.4.8.5 Chronic Effects

As for the data needed for calculating chronic effects, perhaps the most helpful discussion for the potential user of consequence models is a brief summary of the requirements of CRAC2, which considers (1) inhalation of resuspended particles, (2) ingestion of exposed crops, (3) ingestion of milk products, (4) ingestion of milk, (5) ingestion of crops contaminated by root uptake, and (6) exposure to contaminated ground. For each of these pathways, CRAC2 requires a list of the nuclides considered important; for example, for milk ingestion these are generally taken to be I-131 and I-133. The radiation doses delivered by pathways 1 and 6 are then calculated by the methods described in Section 9.3.3.4 (resuspension) and Section 9.3.3.2 (radiation dose from deposited gamma emitters).

For the ingestion pathways, further information is required in the form of concentration factors (see Section 9.3.3.3) relating the activity ingested by a typical individual to the initial deposited activity, for each of the radionuclides identified as important for the ingestion pathway in question. Also required, for each of the nuclides being considered in all of the ingestion pathways and each of the 13 organs identified in Section 9.4.8.1, are the ingestion factors (see Section 9.3.3.3) for six time periods, measured from the time at which the ingestion took place: 0 to 10, 11 to 20, 21 to 30, 31 to 40, 41 to 50, and more than 50 years.

For each exposure pathway, the weathering half-life is required, if relevant; also required is the allowable limit of dose accumulation, together with the period of time over which the dose is accumulated. These limits are obtained from the publications of the U.S. Federal Research Council (FRC, 1964, 1965) and the British Medical Research Council (MRC, 1975), as explained in Section 9.3.3.3. Examples are as follows:

1. Inhalation of resuspended radionuclides--15 rem delivered to the lung over 70 years.

2. Ingestion of exposed crops--3.3 rem delivered to the whole body over 1 year.

3. Ingestion of milk products--3.3 rem delivered to the bone marrow over 1 year.

4. Ingestion of milk--10 rem delivered to the thyroid over 1 year.*

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*In NUREG-0396 (Collins et al., 1978), in the section discussing the emergency-planning zone for ingestion, the size of the zone is based on an expected revision of milk-pathway Protective Action Guidelines by the Food and Drug Administration's Bureau of Radiological Health. The recommended guidelines were supposed to be as low as 1.5 rem to the infant thyroid, but, to the authors' knowledge, these guidelines have never been officially published.

5. Ingestion of crops contaminated by root uptake--5 rem delivered to the bone marrow over 10 years.

6. Exposure to deposited gamma emitters--25 rem delivered to the whole body over 30 years.

In general, for accident sequences in LWR plants, it is predicted that milk ingestion and external exposure are the most important of the chronic exposure pathways, at least in the United States. However, chronic exposure may be highly dependent on agricultural practices and the patterns of food consumption, which may change the relative importance of certain radionuclides. If a probabilistic risk assessment is being carried out for a reactor site in another country, different pathways may need to be considered. In countries with a predominantly Chinese population, for example, the ingestion of milk is negligible, and it is conceivable that other pathways of chronic exposure, such as the contamination of fish farms or duck farms, would be important.

9.4.8.6 Discussion

It is apparent that the user of a consequence-modeling code like CRAC2 has a time-consuming task on his hands if he wishes to input new health-physics data. In general, the health physics of the Reactor Safety Study is still considered to be acceptable by the consequence-modeling community. However, the Nuclear Regulatory Commission is at present considering the possibility of setting up a health-physics data bank that would be freely accessible to all who wish to use it. Such a data bank is needed because the typical user of consequence models will have neither the time nor the expertise to make significant changes to the health-physics data contained in a code like CRAC. Meanwhile, a fresh compilation of inhalation factors (the same goes for ingestion factors and concentration factors) would require a fairly thorough survey of the available literature on health physics (see Sections 9.3.3 and 9.3.5).

9.4.9 DISCUSSION OF DATA REQUIREMENTS

The above review of inputs to consequence models is not comprehensive since different codes have different data requirements that cannot all be discussed here. The potential user should realize by now, however, that he is faced with a considerable amount of work and must devote thought to a variety of topics before he can begin to run his code. The amount of data required is so great, and the purposes for which it is needed cover such a range of topics, that it would be foolish to try to use the code as a "black box." For a sensible preparation of the input, a good background knowledge is required.

9.5 PROCEDURES AND FINAL RESULTS

9.5.1 PROCEDURES

The procedures presented here are aimed at a user of consequence models who has his code up and satisfactorily running on a computer.

9.5.1.1 Deciding on the Purpose of the Consequence Analysis

This involves selecting from among the list of applications given in Section 9.1.2 or devising another application not mentioned there. By doing this, the user will determine (1) which of the input options available in his code he needs to use, (2) which data sets he needs to collect, and (3) which output options he needs to exercise.

9.5.1.2 Collection of Data

As has been explained, this is a major undertaking and involves data collection in many areas.

Basic Radionuclide Data. A reactor-core inventory should be compiled for a selected list of radionuclides, together with a list of associated data, as described in Section 9.4.1.

Source-Term Data. The analyst should consult with the workers on other tasks (quantification, physical processes of core-melt accidents, radionuclide release and transport) in order to compile a table of the properties of radionuclide source terms, such as appears in Table 9-1 and is discussed in Section 9.4.2.

Meteorological Data. A tape containing hourly meteorological data for one or preferably more years should be obtained from the reactor site. In consultation with a meteorologist, the code user should decide whether the quality of the data is adequate. If not, substitute data from a nearby site should be obtained. The data set should then be processed into the form required for the computer code (see Section 9.4.3). If meteorological data from multiple stations are being used, the exercise should be repeated for each meteorological station.

Population Data. After the year for which population data are needed (e.g., plant midlife) and the radii of the elements of the population grid (see Figure 9-8) are selected, data from the U.S. census and/or the FSAR should be processed to assign the population to these elements. For some U.S. reactors, it may be necessary to obtain data from Canada or Mexico. Experience indicates that the processing of population data can be one of the more time-consuming and costly elements of a consequence analysis.

Deposition Data. Dry-deposition velocities and washout coefficients should be assigned to each radionuclide (see Section 9.4.5).

Evacuation and Sheltering Strategies. The choice of data for the evacuation model is comprehensively discussed in Appendix E. Table E-3 summarizes the kind of input that is required. In order to perform this task, site emergency plans should be consulted, together with any associated studies that give estimates of such quantities as evacuation time and effective evacuation speed, sheltering factors, and ventilation.

Economic Data (usually optional). The collection of economic data is discussed in Section 9.4.7, and examples of important input parameters are given in Table 9-12. It is important to note that it may be necessary to collect data on farming input variables at the county or smaller level. The statewide averages of the value of farmland per acre given in the CRAC2 user's manual, for example, may differ greatly from county averages.

Health-Physics Data. As indicated in Section 9.4.8, these data are usually available in a data bank associated with the code or in a standard input set. It is unlikely that a user would wish to undertake the laborious task of replacing the data bank with his own figures.

9.5.1.3 Exercising the Code

In principle, exercising the code should be straightforward. The bulk of the work should have been done in the collection and the processing of data. The code should be run for the base case and for other cases that may have been devised to test the sensitivity of the results to variations in input data.

9.5.1.4 Interpreting the Output and Writing the Report

The output is fully described in Section 9.5.2. Advice on report writing is given in Section 9.7.

9.5.2 FINAL RESULTS

The final results of a consequence analysis are also the final results of the complete probabilistic risk assessment. Hence, the results should be presented in probabilistic form, and this is why the characteristic output of a consequence analysis is in the form of a CCDF, examples of which appear in Figure 9-1.

The CCDF is a compilation of the results of many separate calculations. A code like CRAC2 essentially repeats the same calculation many times, changing each of the following variables over its full range of values:

1. The accident sequence or category (e.g., PWR categories 1 through 9 in Table 9-1).

2. The weather sequence (each set of hourly values of stability category, wind speed, and precipitation intensity).

3. The wind direction.

For each category, weather sequence, and wind direction, a code like CRAC2 will calculate the predicted number of early fatalities in the sector toward which the wind is blowing (and, of course, early injuries, latent-cancer fatalities, etc.). With this predicted number can be associated a frequency, which is the product of the frequency of occurrence of the accident sequence or category and the joint probability of occurrence of the weather sequence and the wind direction.

In principle, the calculation can be broken down still further. For example, CRAC2 allows the user to implement up to six evacuation schemes, each with an associated probability. CRACIT allows an effectively unlimited number of evacuation schemes. It is possible to envision calculations in which the deposition velocity is a variable with an associated probability distribution (Beyea, 1978a,b), or in which there could be a choice of dose-response relationships, each with an associated probability.

Such calculations yield pairs of numbers (e.g., the predicted number of early fatalities given an accident category, weather sequence, wind direction, evacuation scheme) together with the frequency with which that combination of variables is predicted to occur. Taking all such pairs of numbers for all possible combinations of the variables gives a frequency distribution that can be readily cumulated to give a CCDF.

As can be seen from Figure 9-1, CCDFs for early fatalities and latent-cancer fatalities are among the possible output of a consequence analysis; indeed, these are among the most frequently used. Other possible CCDFs include the following, with examples taken from the RSS:

1. Early illness, which is essentially defined by reference to a whole-body dose large enough to cause hospitalization (Figure 9-10).

2. Genetic effects (Figure 9-11).

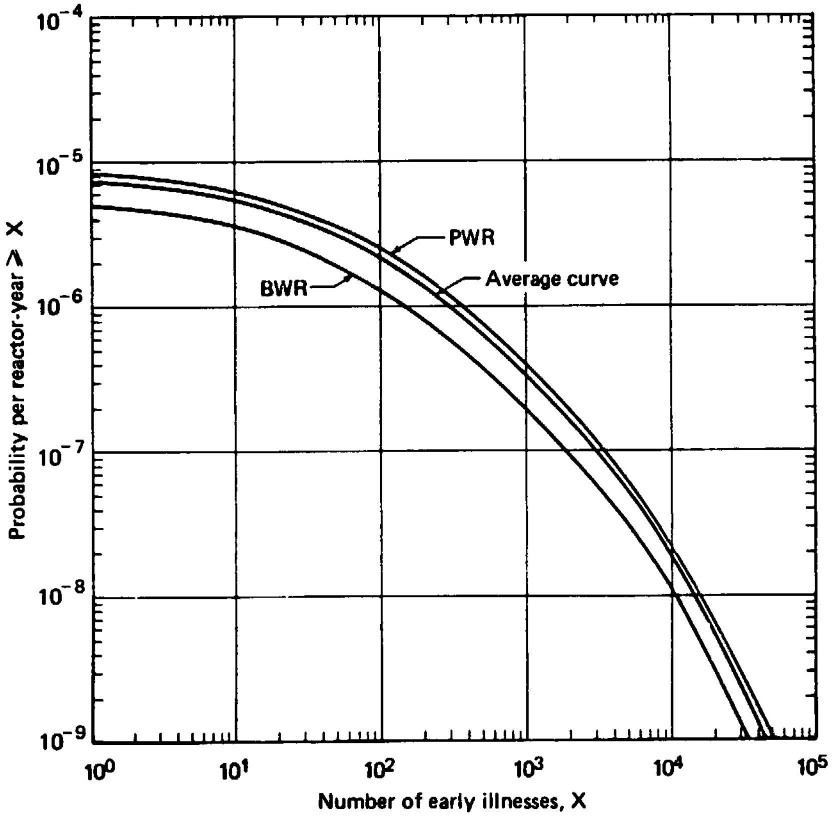
3. Areas requiring decontamination or relocation (Figure 9-12).

4. Property damage (Figure 9-13).

The CRAC2 user's manual lists a great variety of possible outputs from a consequence analysis, all of which can be expressed in the form of CCDFs. Examples are as follows:

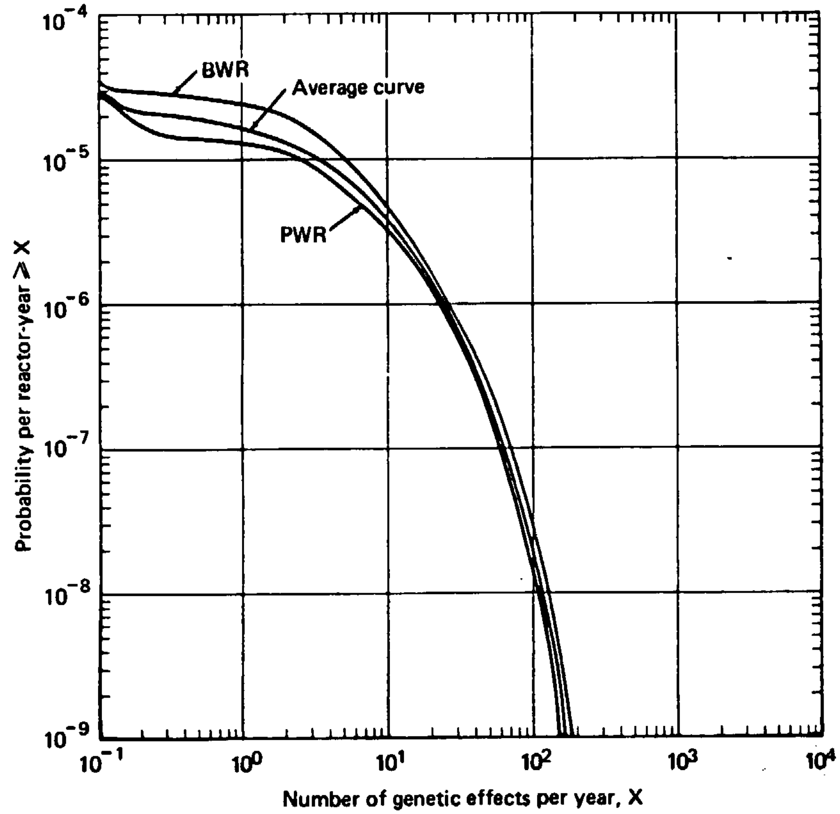
1. Number of people with an acute bone-marrow dose exceeding 200 rem (this number is of interest because it indicates how many people would require hospital treatment).

2. Risk of early fatality at the midpoint of each of several specified radial intervals on the computational or population grid.



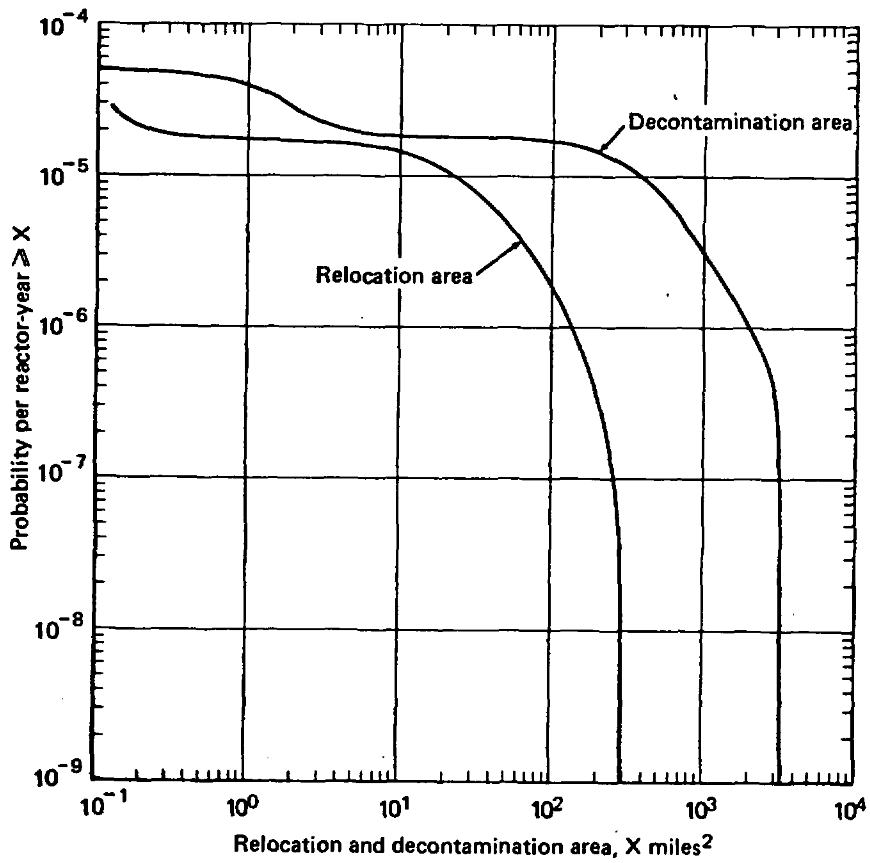
Note: Approximate uncertainties are estimated to be represented by factors of 1/4 and 4 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

Figure 9-10. Complementary cumulative distribution function for early illnesses. From the Reactor Safety Study (USNRC, 1975).



Note: Approximate uncertainties are estimated to be represented by factors of 1/3 and 6 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

Figure 9-11. Complementary cumulative distribution function for genetic effects per year. From the Reactor Safety Study (USNRC, 1975).



Note: Approximate uncertainties are estimated to be represented by factors of 1/5 and 2 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

Figure 9-12. Complementary cumulative distribution function for relocation and decontamination area. From the Reactor Safety Study (USNRC, 1975).

3. Greatest distance from the reactor at which acute fatalities or injuries are predicted to occur.

4. Number of people residing in the area that would need to be permanently interdicted (for more than 30 years).

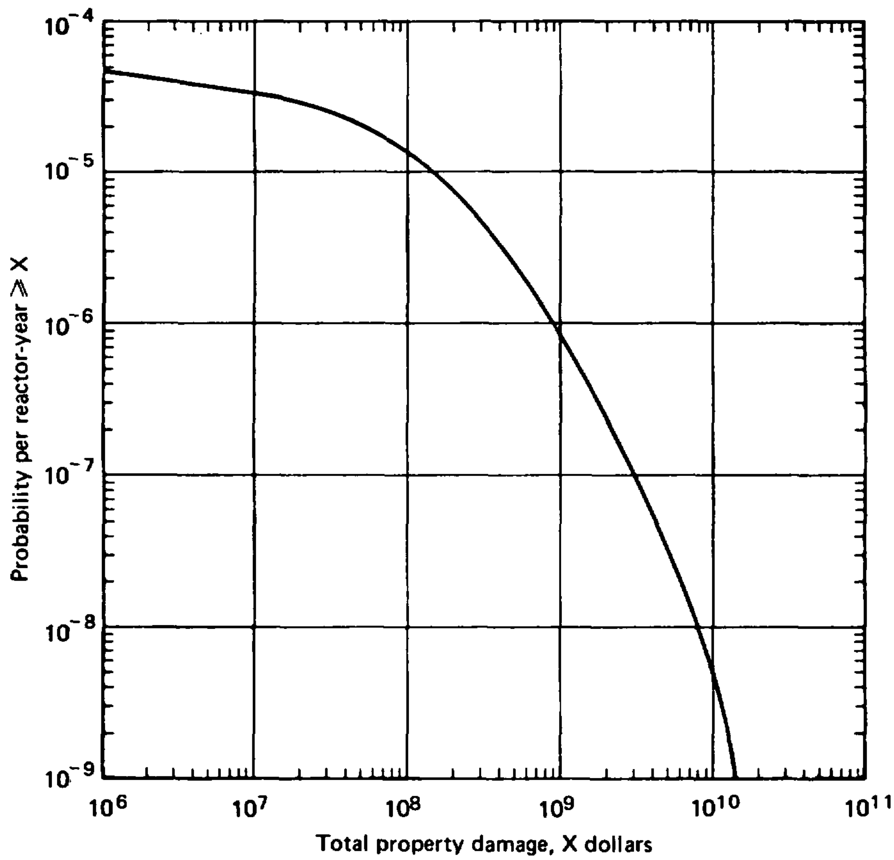
5. Cost of permanent land interdiction.

6. Total land area permanently interdicted.

7. Cost of contaminated-milk disposal.

In all, CRAC2 allows the user to choose 84 output options for display as CCDFs; the above list is a sample of the possibilities.

Consequence models are also able to output the individual risk of early fatality or latent-cancer fatality as a function of distance. Examples are



Note: Approximate uncertainties are estimated to be represented by factors of 1/5 and 2 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

Figure 9-13. Complementary cumulative distribution function for total property damage. From the Reactor Safety Study (USNRC, 1975).

given in Figure 9-14. These individual risks have recently received renewed attention because the NRC has published draft safety goals that contain target values for individual risk, an example being 5 x 10^{-7} per year for early fatality, averaged out to 1 mile (USNRC, 1982a). Since the implementation of these safety goals had not been finalized at the time of writing, however, it is premature to describe how individual risks should be calculated in the context of safety goals. It is important to be aware that there are many uncertainties; for example, the individual risk of early fatality is extremely sensitive to assumptions about evacuation and plume rise.

Consequence models can also be used to obtain output for given weather conditions, such as the radiation dose received by a given organ as a function of position. The user who so desires may, if he uses all the options available in his user's manual, produce an overwhelming stack of computer output. It is desirable to exercise a certain degree of restraint in the choice of a sensible number of output options.

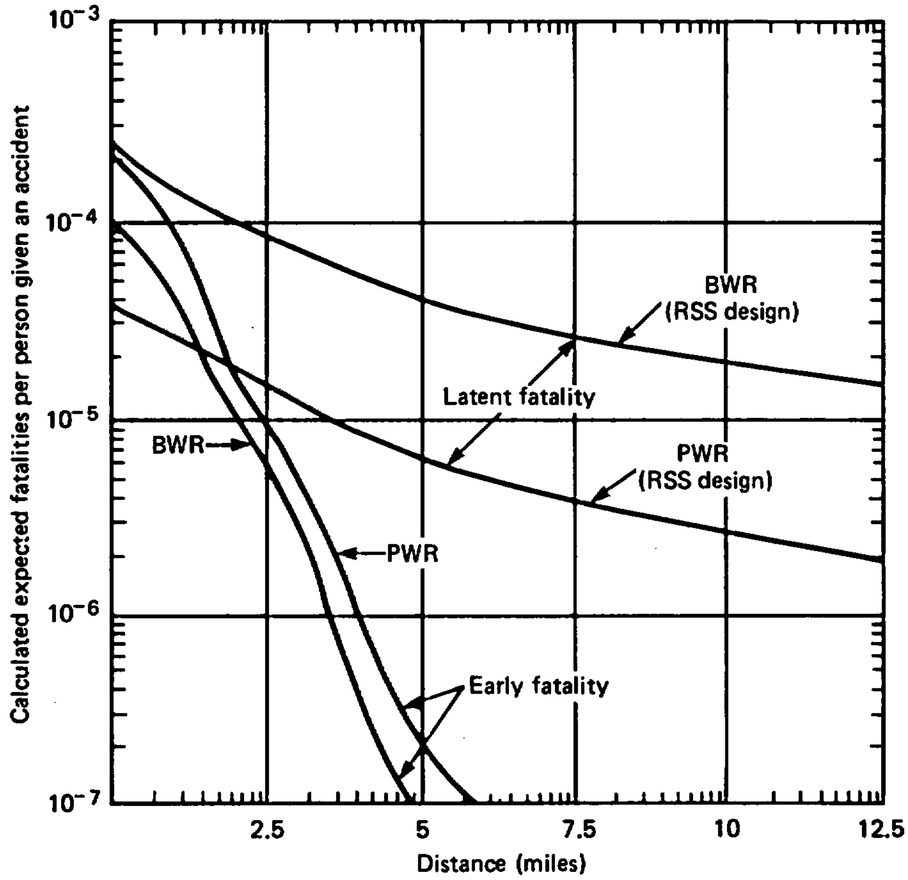


Figure 9-14. Calculated risk to an individual of early and latent fatality as a function of distance from the reactor for the accidents described in the Reactor Safety Study (including non-core-melt accidents). The latent fatalities are those attributed to immediate exposures; no chronic effects from long-term groundshine or long-term pathways are included. From NUREG-0739 (USNRC, 1980).

9.6 ASSUMPTIONS, SENSITIVITIES, AND UNCERTAINTIES

A necessary element in the interpretation of the results of a consequence analysis is an estimate of the uncertainties associated with the results.

More recent studies of uncertainties in PRA tend to produce CCDFs that look like that shown on Figure 9-15. Ideally, the upper and lower bounds are expressed in statistical terms as confidence limits, perhaps at the 5- and 95-percent levels. The median is the curve above or below which the true CCDF is equally likely to lie. The mean is shown as lying close to the 95-percent limit. This is characteristic of the highly skewed probability distributions (e.g, the lognormal) that are currently being derived to express uncertainties in PRAs. As noted by the Risk Assessment Review Group

(Lewis et al., 1978), uncertainties are larger than those given in the RSS. Typical ranges of uncertainty span two orders of magnitude or more (Philadelphia Electric Company, 1981; Commonwealth Edison Company, 1981).

It is convenient to divide the factors contributing to these uncertainties into two parts. The first consists of the factors deriving from other parts of the PRA exercise; these are discussed in the appropriate chapters and in Chapter 12. The second consists of those uncertainties that are peculiar to consequence analysis.

The parameters and modeling assumptions used at various stages in a consequence model are reviewed below in the context of their contribution to uncertainties in CCDFs. If a parameter or modeling assumption is said to be a major contributor to uncertainty, this means that, when the parameter or modeling assumption is varied over its plausible range, there is a broad band within which the corresponding CCDF may lie. A major contributor to uncertainty is one for which this band is as much as a factor of 10 in breadth (perhaps even more).

The decision as to whether a particular modeling assumption or parameter makes a major, moderate, or small contribution to uncertainty is the subjective judgment of the authors. Where possible, this judgment is based on sensitivity studies, that is, studies in which the modeling assumption or parameter value is changed and the CCDF recalculated in order to see how it varies. Hence, the uncertainties discussed in this section are not quantitative uncertainties in the sense that the bounds on CCDFs are expressed as confidence limits as shown in Figure 9-15, but are quantities that themselves have meaning in a subjective sense only.

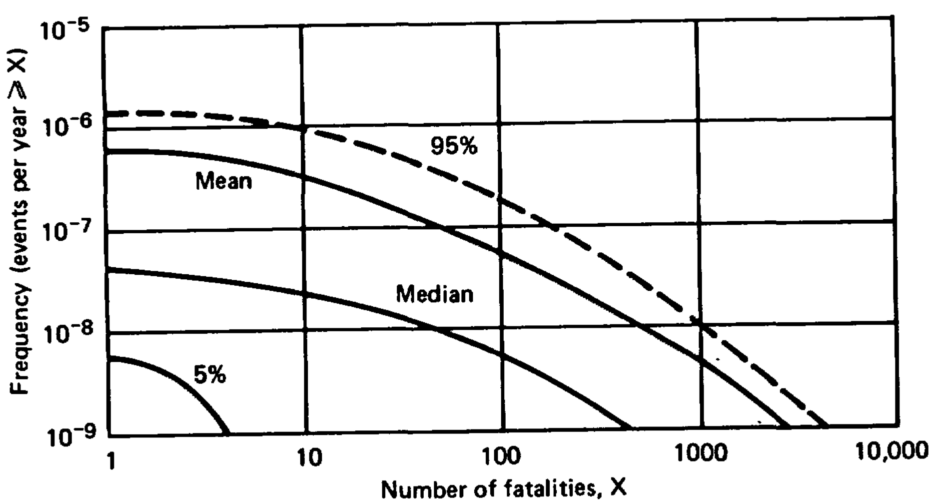


Figure 9-15. Typical uncertainty bounds on a CCDF for early fatalities.

9.6.1 INVENTORY OF RADIOACTIVE MATERIAL

The available methods for calculating the inventory of radioactive material in the reactor core at the time that the chain reaction ceases are well established. The ORIGEN code (Bell, 1973; Croff, 1980) is widely used and takes account of all the important processes--fission, radionuclide decay and daughter buildup, and neutron activation. The factors that have an important influence on the outcome of a consequence analysis are summarized in Table 9-13.

It is apparent that the user should be as realistic as possible in his choice of power level and burnup; he should also distinguish between PWRs and BWRs. Once these choices have been made, the residual uncertainties in the radioactive inventory are small. Hence the radionuclide inventory is a small contributor to the uncertainties in CCDFs.

9.6.2 SOURCE TERMS

The methods for calculating source terms are described in Chapters 7 and 8 of this guide, and a summary of important uncertainties appears in Table 9-14. It is in this area that many of the greatest uncertainties in consequence modeling arise, and it is worth devoting some attention to the factors designated as having a major influence on uncertainty.

9.6.2.1 Magnitude of the Source Term

Figure 9-16 shows an example of the effect predicted for the early-fatality CCDF if the source terms used in the RSS are reduced by factors of 5 or 10. Table 9-15 shows the impact of the same reductions on early injuries, latent-cancer fatalities, and areas interdicted for 10 years or more. Some authors have argued strongly that similar or even larger reductions in source terms are justifiable on the basis of existing evidence (Levenson and Rahn, 1981; Morewitz, 1981). Others argue that such reductions are not proved (USNRC, 1981b; Passadeg et al., 1981; Levine et al., 1982). All authors agree that there are large uncertainties in the magnitude of the source term, however.

The authors of the Limerick PRA study (Philadelphia Electric Company, 1981) identified a lack of knowledge about the mechanisms of source-term attenuation within the reactor-coolant system and the containment as the most important uncertainty in predicting the magnitude of the consequences. The authors of the Zion study (Commonwealth Edison Company, 1981) attempted to model the source-term uncertainty by assigning a probability distribution to the magnitude of release for each accident category. This was done by a process of subjective judgment, however. It is clear that the question of uncertainties in this important area remains to be settled and that considerable future research is required in order to quantify and reduce the uncertainties.

Table 9-13. Radionuclide inventory: sensitivities and uncertainties

Parameter or modeling assumption	Quantity most directly sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs: Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage	Sensitivity studies	Relevant section of this chapter
PWR vs. BWR (modeling)	Inventory of long-lived radionuclides	Low	Low	Low	Ostmeyer (1981)	9.4.1
Power level	Inventory of radio-nuclides	Low	Low	Low	Ostmeyer (1981)	9.4.1
Burnup	Inventory of long-lived radionuclides	Low	Low	Low	Ostmeyer (1981)	9.4.1

Table 9-14. Source terms: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs			Sensitivity studies	Relevant section of this report
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage		
Magnitude of source term (modeling and parameter)	Airborne and deposited levels of radioactive material	Major	Major	Major	RSS (USNRC, 1975) Wall et al. (1977) Zion study[a]	9.4.2.2, Chapter 7, Chapter 8
Frequency of occurrence of each sequence/category (modeling and parameter)	Frequency of CCDFs	Major	Major	Major	Limerick study[b] Zion study[a] Erdmann et al. (1981)	9.4.2.1
Time of release (parameter)	Time available for evacuation	Major (except peaks)	Low	Low		9.4.2.3
Duration of release (parameter)	Plume width Possibility of wind shift or weather change during release	Major (especially peaks)	Low	Low	Griffiths (1977) Benchmark exercise Zion study[a]	9.4.2.3
Warning time (parameter)	Time available for evacuation	Major (except peaks)	Low	Low		9.4.2.3
Building wake or dimensions of release (parameter)	Airborne concentration near reactor	Low	Low	Low		9.4.2.4
Rate of energy release (parameter and modeling)	Height of plume rise	Moderate to major for some sequences (low for peaks)	Low	Low	Russo (1976) Russo et al. (1977) Kaiser (1977, 1981)	9.3.1.5
Particle-size distribution (parameter and modeling)	Deposition velocity Washout coefficients Dose-conversion factors	Moderate	Moderate	Major	Kaul (1981b) Benchmark exercise Hunt et al. (1979)	Appendix D3, 9.4.2.6
Chemical form (parameter)	Dose-conversion factors Deposition velocity	Moderate	Moderate	Moderate	Hunt et al. (1979)	9.4.2.7

[a]Commonwealth Edison Company (1981).
[b]Philadelphia Electric Company (1981).

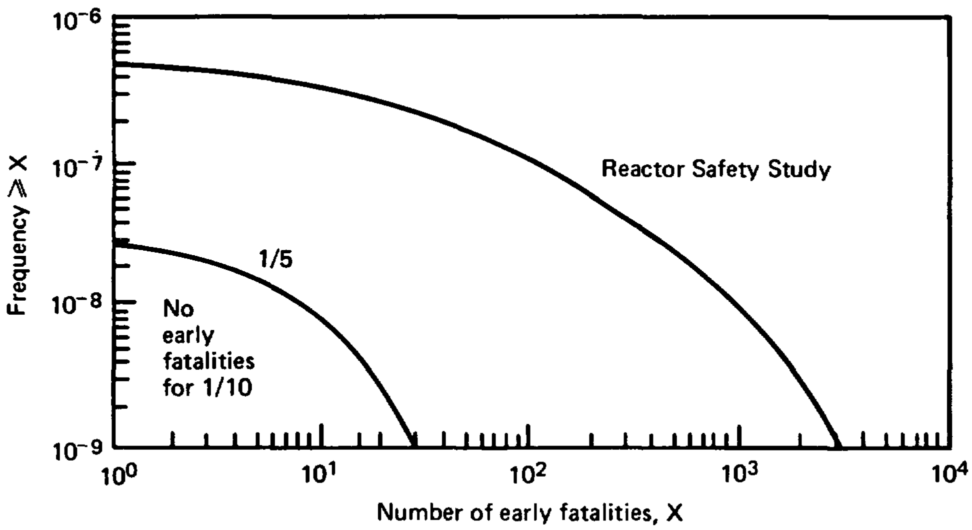


Figure 9-16. Perspective on the risk predicted by the Reactor Safety Study: iodine and particulate releases to the atmosphere reduced by factors of 5 and 10. Same modeling as was used in the Reactor Safety Study. Note that the reductions shown can be sensitive to details of modeling, such as evacuation. From C. Starr, M. Levenson, and I. B. Wall, "Realistic Estimates of the Consequences of Reactor Accidents," USNRC briefing, 1980.

Table 9-15. Impact of decreasing the magnitude of the release[a]

	Assumed reduction in quantity of iodine and particulates released to the atmosphere		
Consequence	RSS	1/5	1/10
Early injuries	1	0.032	0.0020[b]
Latent-cancer fatalities	1	0.35	0.22[c]
Area interdicted 10 years	1	0.11	0.037[c]

[a]From Starr et al. (1980). Reference for comparison is the Reactor Safety Study (USNRC, 1975).
[b]Nonlinear because of thresholds in early effects.
[c]Nonlinear because both depend on interdiction and decontamination measures, which are effectively threshold effects.

9.6.2.2 Frequency of Occurrence of Each Category

Uncertainties in the frequencies of occurrence predicted for accident categories or sequences propagate directly into uncertainties of comparable magnitude on the frequency axis of CCDFs. The source of these uncertainties is to be found in the chapters on the quantification of event trees and has nothing to do with consequence analysis per se; however, this uncertainty is the single greatest contributor to uncertainties on the frequency axis of CCDFs.

9.6.2.3 Duration of Release

In the context of uncertainty, the most important effect of an increased duration of release is to allow the possibility of a change in wind direction and/or weather conditions while the release of radioactivity is taking place. Figure D-12 and the accompanying text show the possible effect of wind shift on the dispersion of the plume. This behavior is to be contrasted with that of the single puff described in Figure D-10. A comparison of these two figures shows that the wind shift causes the plume of longer duration to spread out over a much wider area than that covered by the puff. In principle, this could cause the radiation dose delivered by the extended plume to fall below thresholds for early effects. That is, the long duration of release may introduce considerable conservative bias into the CCDFs for early effects. This bias has never been quantified, however. It is an example of uncertainties that arise because of modeling assumptions rather than uncertainties that arise because of poorly known data.

9.6.2.4 Warning Time

In order to assess the benefits of evacuation, it is important to obtain a reliable estimate of the warning time--that is, the interval between the broadcast of a warning and the time of radionuclide release into the atmosphere. This is an important source of uncertainty. It is particularly important for early effects.

9.6.2.5 Particle-Size Distribution

This uncertainty arises because of modeling simplifications and because of a lack of knowledge of the size distribution itself. The most important impact is on deposition modeling (see Appendix D): particle size can account for a difference in the deposition velocity of up to about two orders of magnitude. Hence, any consequence that depends on the deposited level of radioactive material, such as the area of contaminated land, will be subject to uncertainty.

9.6.3 METEOROLOGICAL MODELING

A number of parameters or phenomena to which meteorological modeling is sensitive are shown in Table 9-16, which also evaluates their contributions to uncertainties in CCDFs. Those that are judged to be most important are briefly discussed below.

Table 9-16. Meteorological modeling: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs: Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage	Sensitivity studies	Relevant section of this report
Sampling of meteorological data (parameter)	Selection of weather sequences	Major (peaks only)	Low to moderate	Low	Ritchie et al. (1981b)	App. D4.1.2
Number and location of meteorological stations (parameter)	Plume trajectory	Moderate	Low	Low to moderate	Benchmark exercise Zion study[a]	9.4.3
Definition of stability categories (parameter)	Frequency of occurrence of stability categories	Low to moderate	Low	Low to moderate		9.3.1.2
Gaussian theory vs. K-theory (modeling)	Airborne concentrations	Low to moderate	Low to moderate	Low to moderate	Clarke et al. (1979) Nordlund et al. (1980)	App. D1
Choice of dispersion parameters	Airborne concentrations	Moderate to major	Moderate	Moderate	Benchmark exercise Aldrich (1979) Vogt (1981)	9.3.1.3
Changing vs. constant weather (modeling)	Weather sequences	Low[b]	Low[b]	Low[b]	Benchmark exercise McGrath et al. (1977)	9.3.2.4 App. D4
Straight line vs. trajectory vs. multipuff (modeling)	Area covered by plume and evacuation model	Major	Low	Major	Benchmark exercise	App. D4
Inversion lid (parameter)	Height of plume rise	Low	Low	Low	Sprung and Church (1977a)	9.3.1.5
Wind shear (modeling)	Lateral spread of plume	Low	Low	Low	Sprung and Church (1977b)	
Low wind speeds (modeling)	Airborne concentrations	Low	Low	Low		9.3.1.4
Surface roughness (modeling)	Dispersion parameters Deposition velocity	Low to moderate	Low	Low to moderate	Aldrich (1979)	9.3.1.3
Terrain (modeling)	Plume trajectory	Moderate	Low	Moderate	Benchmark exercise	App. D4.3

[a]Commonwealth Edison Company (1981).

[b]The contribution to uncertainties in CCDFs is assessed as low because modelers know which to choose--changing or constant weather.

9.6.3.1 Sampling of Meteorological Data

The sampling of the available meteorological data is discussed in Section D4.1.2. The uncertainties that are attributable to the sampling methods used in CRAC and CRAC2 are shown in Figure 9-17. This illustrates the considerable importance of ensuring that the data are sampled in a reliable way. (See Appendix D4 for further discussion.)

9.6.3.2 Trajectory Versus Straight Line

This uncertainty is discussed in Appendix D (Section D4) and in Section 9.6.2.3.

9.6.4 DEPOSITION

Examples of sensitivities and contributors to uncertainty in the modeling of radionuclide deposition are given in Table 9-17.

9.6.4.1 Dry-Deposition Velocity

As discussed in Appendix D3, important uncertainties arise both in the specification of a value for dry-deposition velocity v_d and in the choice of a deposition model. Kaul (1981b) gives examples of the possible ranges of airborne and deposited concentrations, given a range of values of $v_d/\bar{u}$ (see also Hosker, 1974).

It is pertinent to remark in this context that Beyea (1978a,b) automatically incorporates v_d into his models as an uncertain parameter within a range that varies with stability class as follows:

For stability classes A-D, $0.001 \leq v_d \leq 0.1$ m/sec
For stability class E, $0.001 \leq v_d \leq 0.03$ m/sec
For stability class F, $0.001 \leq v_d \leq 0.01$ m/sec

The Nuclear Power Plant Siting Study performed by Sandia National Laboratories (Strip et al., 1981) contains a sensitivity study of a variation in v_d. This study was carried out for a large, hypothetical release of radioactive material known as SST1.* The "summary evacuation" procedure was assumed,† with a 1120-MWe reactor, the Indian Point population

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*Core melt, loss of all safety systems, containment failure and radionuclide release to the atmosphere, and the following release fractions for volatiles: I, 0.45; Cs, 0.67; Te, 0.64. It is the same as the TC-γ' sequence for the rebaselined BWR (USNRC, 1981b).

†Delay times of 1, 3, and 5 hours with respective probabilities of .3, .4, and .3, and an evacuation speed of 10 mph, as described by Aldrich, Blond, and Jones (1978).

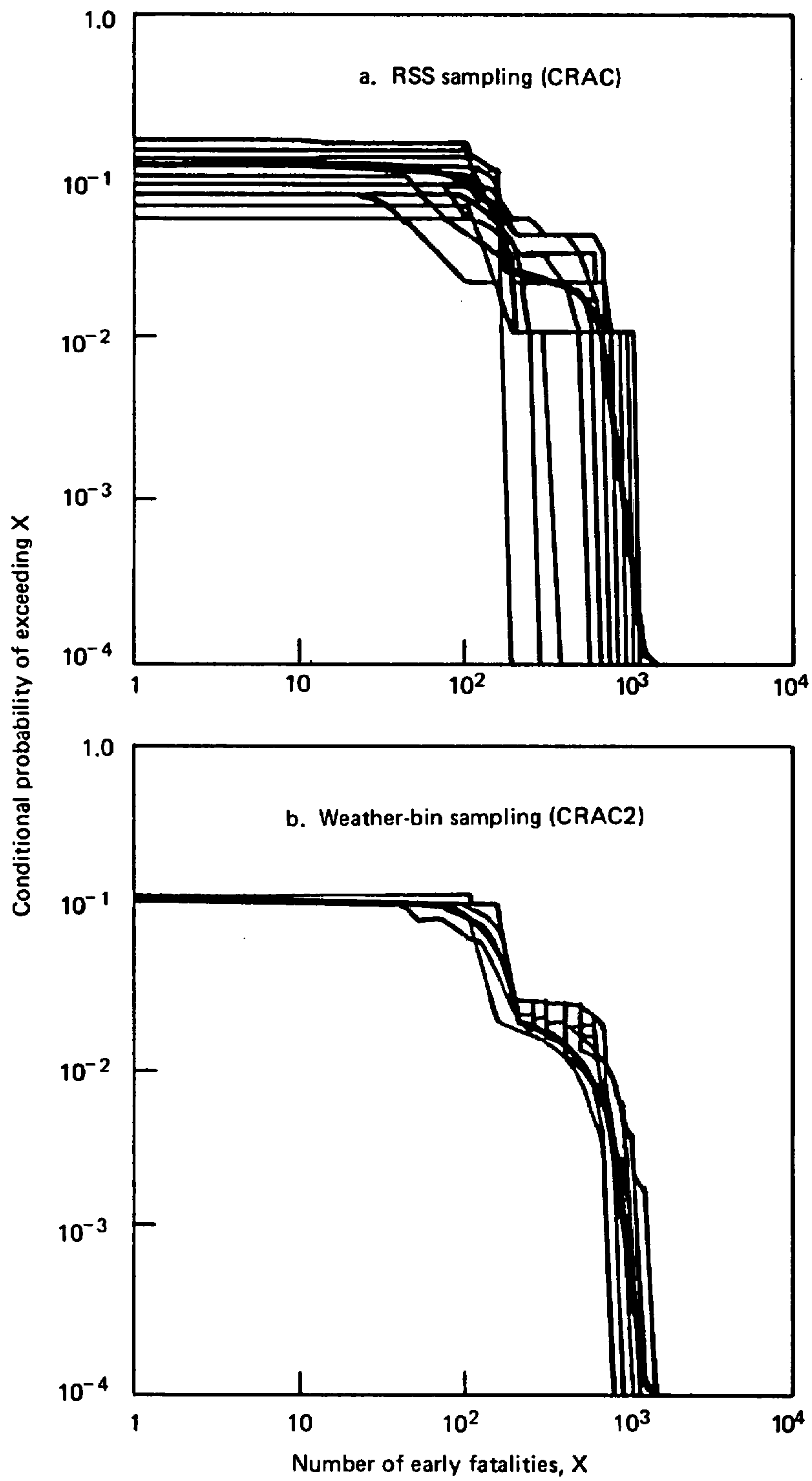

Figure 9-17. Comparison of uncertainty due to sampling by (a) the RSS technique (CRAC) and (b) the weather-bin technique (CRAC2). For each technique, 32 different sets of weather sequences are used to generate early-fatality frequency distributions for a PWR-2 release. A "best estimate," using all 8760 available sequences, is shown by the darkened line. From Ritchie et al. (1981b).

Table 9-17. Deposition modeling: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs: Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage	Sensitivity studies	Relevant section of this report
Dry-deposition velocity (modeling and parameter)	Quantity of radioactive material on ground	Moderate (major for peaks)	Moderate	Major	Benchmark exercise Beyea (1978a,b) Kaul (1981b)	App. D3
Rainfall model (modeling)	Quantity and location of deposited material	Major (peaks)	Low	Moderate	Ritchie et al. (1976, 1980)	9.3.2.3
Runoff (modeling)	Location of deposited material	Moderate	Moderate	Moderate	Ritchie et al. (1976, 1980)	9.3.2.3

distribution and wind rose, and New York City weather conditions. The deposition velocity (assumed to be the same for all radionuclides except the noble gases) was varied from 0.1 to 10 cm/sec.

Over this range, mean early fatalities varied only by a factor of 1.5. The peak early fatalities varied by a factor of 10, however. Other quantities that are significantly affected by changes in the deposition velocity are the distances within which various effects are predicted to occur: early fatalities, early injuries, and land interdiction (see Table 9-18).

9.6.4.2 Rainfall and Runoff

Section 9.3.2.3 contains a brief description of the rainfall and runoff model of Ritchie et al. (1976). When this new rainstorm and runoff model was used in CRAC, single-trial calculations (one weather sequence containing a rainstorm) yielded probabilities for early and latent fatalities that were increased or decreased by up to an order of magnitude. However, for multiple-trial calculations (91 weather sequences selected from a 1-year record by stratified sampling) mean risk estimates (approximately equal to the area under the CCDFs) were essentially unchanged when compared with the original model, principally because rain is infrequent and therefore consequences produced by weather sequences that contain rain contribute minimally to risk for the nearby public. However, the number of early fatalities predicted for the peak accident is increased because the higher rain rates of the rain cells and the small mesoscale-storm structures cause the levels of deposited radionuclides to be substantially higher over small areas at longer distances, where the chance of encountering large populations is greater.

9.6.5 ACCUMULATION OF RADIATION DOSE

Uncertainties and sensitivities are summarized in Table 9-19. The literature contains far fewer sensitivity studies in this area, but this is not to say that no uncertainties exist. For example, the treatment of weathering is largely based on a single experiment with cesium (Gale et al., 1964). The treatment of ingestion pathways is beset by uncertainties in the calculation of concentration factors, but this has not caused great concern because, for the typical mix of radionuclides that are likely to escape to the atmosphere in the event of an LWR accident, it is generally predicted that ingestion will be only a small contributor to total latent effects (see Tables 9-6 and 9-7). This conclusion would not be true if strontium-90 and cesium-137 were the main components of the release, however.

The next task of the International Benchmark Committee (see page 9-17) will be a thorough survey of chronic effects. This should provide insight into uncertainties in this area.

Table 9-18. Sensitivity of the distances to which consequences occur for various deposition velocities[a]

Dry-deposition velocity (cm/sec)	Distance (miles): Early fatalities				Early injuries				Land interdiction			
	Mean	90%	99%	Peak calculated	Mean	90%	99%	Peak calculated	Mean	90%	99%	Peak calculated
0.1	2.1	4	15	25	7.2	15	55	65	11	30	60	100
0.3	1.9	4	15	25	7.1	20	40	50	16	40	65	85
1.0	1.7	4	12	18	8.3	25	35	50	19	40	60	35
3.0	1.6	3	4	18	6.6	12	23	25	20	25	40	45
10.0	1.4	3	3	3	3.5	6	15	18	13	22	23	25

[a]From Strip et al. (1981).

Table 9-19. Accumulation of radiation dose: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs: Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage	Sensitivity studies	Relevant section of this report
Dose-conversion factors (modeling)	Radiation dose delivered by inhalation	Low[a]	Low[a]	Low[a]	Hunt et al. (1979) Kelly et al. (1979)	9.3.3.1
Methods of approximating cloudshine and groundshine (modeling)	Computing time Externally delivered radiation dose	Low to moderate	Low to moderate	Low to moderate	Van der Hoven and Gammill (1979)	9.3.3.2
Resuspension (modeling)	Long-term inhaled radiation dose	Low	Low	Low	Wall et al. (1977)	9.3.3.4
Ingestion-pathway modeling	Chronic radiation dose	Zero	Moderate	Moderate		9.3.3.3
Weathering (modeling)	Long-term externally delivered radiation dose	Low	Moderate	Major		9.3.3.2

[a]For light-water reactors.

9.6.6 MEASURES THAT CAN REDUCE PREDICTED RADIATION DOSES

Uncertainties and sensitivities in the effects of preventive countermeasures are summarized in Table 9-20. In general, the uncertainties in the evacuation model produce large uncertainties in the corresponding CCDF for early fatalities. The uncertainties in the interdiction and decontamination countermeasures lead to the greatest uncertainties in contaminated areas and property damage.

9.6.6.1 Delay Time in Evacuation Model

Delay time in evacuation is one of the most important parameters* in the consequence model, as can be seen from Figure 9-7, from Aldrich, Ritchie, and Sprung (1979). The reader is referred to the discussion in Appendix E, Sections E1 and E4.

9.6.7 HEALTH EFFECTS

Sensitivities and uncertainties in the predicted health effects are summarized in Table 9-21. This is another area in which there is a relative paucity of sensitivity studies.

9.6.7.1 Dose-Response Relationships: Thresholds

Early fatalities and early injuries are threshold effects. Clearly, assigning a threshold to a dose-risk relationship like that for early fatalities can make an important impact on the number of people who are affected at radiation-dose levels above that threshold.

9.6.7.2 Medical Treatment

Figure VI F-1 of Appendix VI of the Reactor Safety Study (USNRC, 1975) gives three dose-response curves for early mortality, assuming (1) minimal medical treatment, (2) supportive medical treatment, and (3) heroic medical treatment. The thresholds for these relationships are 150, 220, and 890 rads, respectively, and the 50-percent probability levels are at about 320, 510, and 1000 rem, respectively. Each of these dose-response curves would lead to a very wide range of the CCDFs predicted for early effects if they were used in turn in a consequence model.

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*More precisely, it is the difference between the warning time and the delay time that is important.

Table 9-20. Preventive countermeasures: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs: Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage	Sensitivity studies	Relevant section of this report
Delay time and warning time in evacuation model (modeling and parameter)	Short-term radiation dose	Major (except for peaks)	Low	Low	Aldrich, Ritchie, and Sprung (1979)	App. E1
Effective evacuation speed (parameter)	Short-term radiation dose	Moderate (low for peaks)	Low	Low	Aldrich, Ritchie, and Sprung (1979)	App. E1
Radial vs. realistic evacuation routes	Short-term radiation dose	Moderate (low for peaks)	Low	Low		App. E3
Other evacuation parameters (e.g., radius of evacuation zone)	Externally delivered radiation dose	Moderate	Low	Low	Aldrich, Ritchie, and Sprung (1979)	App. E1
Sheltering strategies and shielding factors (parameters)	Externally delivered radiation dose	Moderate (major for peaks)	Low	Low	Aldrich, Ritchie, and Sprung (1979)	App. E2
Interdiction (parameter)	Number of people relocated	Low	Moderate	Moderate to major		9.3.4.3
Decontamination (parameter)	Area of land interdicted, costs	Low	Moderate	Major		9.3.4.4
Thyroid blocking; respiratory protection; ventilation strategies (modeling)	Inhaled radiation dose	Low[a]	Low[a]	Low[a]	Aldrich and Blond (1980, 1981) Aldrich and Ericson (1977)	9.3.4.5

[a]Not usually incorporated into consequence models.

Table 9-21. Health effects: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs			Sensitivity studies	Relevant section of this report
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage		
Dose-response relationships for early effects (modeling)	Probability of occurrence of health effect	Moderate to major	Low	Low		9.3.5
Age-cohort treatment (modeling)	Inhalation dose-conversion factors and dose-response relationships	Low	Low	Low		9.3.5 9.3.3.1
Time intervals over which dose accumulates (modeling)	Dose used in dose-response relationship	Moderate	Moderate	Low		9.4.8
Dose-response relationships for cancer induction (modeling)	Probability of latent health effects	Low	Major	Low	Beyea (1978a,b) BEIR III (NAS-NRC, 1980)	9.3.5 9.4.8.4

9.6.7.3 Linear or Other Hypothesis for Cancer Induction

In the BEIR III report (NAS-NRC, 1980), estimates of the probability of cancer induction vary over an order of magnitude for low doses and dose rates. Beyea (1978a,b) accounts for this by taking the number of fatal cancers to be between 50 and 500 per 10^6 whole-body man-rem. Clearly, this could lead to an uncertainty of as much as an order of magnitude on the CCDFs for latent-cancer fatalities.

9.6.8 PROPERTY DAMAGE AND ECONOMIC COSTS

Sensitivities in the modeling of property damage and economic costs are shown in Table 9-22, which also ranks their contributions to uncertainties in CCDFs. Very few sensitivity studies, if any, have been done to estimate the width of the error bands on CCDFs for areas of interdicted land or economic costs. Possibly, further work should be carried out in this area.

9.6.9 DEMOGRAPHIC DATA

Sensitivities and uncertainties related to demographic data are displayed in Table 9-23. As discussed in Section 9.4.4, one of the most difficult problems is the treatment of diurnal variations in populations as people move between work and home. This affects not only the movement of people in relation to the plume but also evacuation and shielding strategies.

9.6.10 DISCUSSION

The uncertainties discussed above remain, for the most part, unquantified. Those studies that have attempted to quantify them have done so in a subjective manner. For example, in the Zion study (Commonwealth Edison Company, 1981), all of the uncertainties in the consequence model were simulated by a judgmental probability distribution on the calculated radiation doses. If the dose initially calculated for the best estimate was of magnitude Q, it was judged that the uncertainties could be represented by the following: (1) a probability of .1 that the dose has magnitude 2Q; (2) a probability of .35 that the dose has magnitude Q; (3) a probability of .45 that the dose has magnitude 0.5Q; and (4) a probability of .1 that the dose has magnitude 0.1Q. It is to be stressed that nobody has yet done better than this simple approach. Hence, the use of sophisticated Bayesian or classical techniques to quantify uncertainties in consequence analysis has not yet been attempted.

Future research into uncertainties in consequence modeling may well be directed into two separate channels. The first will be to reduce uncertainties in some of the important parameters. An example of this could be to take advantage of current interest in radionuclide source terms, which

Table 9-22. Property damage and economic costs: sensitivities and uncertainties

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs			Sensitivity studies	Relevant section of this report
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage		
Decontamination effectiveness (parameter)	Area of interdicted land	Low	Low to moderate	Moderate to major		9.3.4.4
Runoff and weathering (modeling)	Area of interdicted land	Low to moderate	Low to moderate	Moderate	Ritchie et al. (1976)	9.3.3.2 9.3.2.3
Cost elements in economic model (parameters)	Costs	Zero	Zero	Moderate		9.4.7

Table 9-23. Demographic data: sensitivities and uncertainties[a]

Parameter or modeling assumption	Quantity most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs: Early fatalities, injuries	Contribution to uncertainties in CCDFs: Latent-cancer fatalities	Contribution to uncertainties in CCDFs: Contaminated area, property damage	Relevant section of this report
Transient population variations	Total number of health effects	Moderate, but peaks could be major (site dependent)	Low to moderate	Low to moderate	9.4.4
Diurnal variations	Total number of health effects	Moderate to major (site dependent)	Low	Low to moderate	9.4.4
Assigning population to elements of grid	Total number of health effects	Low to moderate	Low	Low	9.4.4

[a]No sensitivity studies are known to the authors.

should lead to improved methods for predicting the magnitudes of the source terms and to an improved specification of the properties of particulate radionuclides released into the atmosphere; this ought to lead, in turn, to improved deposition modeling. The second channel will be to devise more sophisticated methods of estimating uncertainties, taking into account correlations between parameters and models.

9.7 DOCUMENTATION

This section presents a suggested outline for a final report on a consequence-modeling exercise or for a chapter on consequence modeling in the final report summarizing a complete PRA exercise. The consequence modeler is not obliged to follow this outline if he has one that he thinks is better. The main requirement is that the reader be able to follow what has been done and, in particular, be able to understand the various input parameters.

9.7.1 INTRODUCTION

The introduction should begin by stating the reason for undertaking the consequence analysis. It should then go on to say which consequence-modeling code was used and why.

9.7.2 METHODS

This section should begin by describing the consequence-modeling code that was chosen. Obviously, there is no need to duplicate the user's guide to the code or to reproduce vast portions of the Reactor Safety Study. A brief description is all that is required, with copious references if need be.

The most important part of this section should be the description of what changes, if any, have been made to the chosen code, with the reasons for making these changes. This description is necessary in order to assist in the interpretation of the final results and to answer such inevitable questions as, Why does this differ from the Reactor Safety Study?

9.7.3 INPUT DATA

The gathering and processing of input data, which were discussed at some length in Section 9.4, are the activities with which the user of consequence models can most influence the output. It is essential, therefore, that he write down clearly what he has done. This section should consist of a review of each piece or set of input data, including--

1. The source of the data, properly referenced.

2. How the data were processed.

3. If relevant, what has been done to overcome problems associated with data of poor quality (this has been mentioned particularly in the context of meteorological data).

4. Tabulation of the actual values used for input parameters, if possible.

As described in Section 9.4, the areas in which collections of input data are required include the following:

1. Source-term specification.
2. Meteorological data.
3. Population data.
4. Deposition input.
5. Economic data.
6. Health-physics data (dosimetry and health effects).
7. Evacuation data.
8. Basic radionuclide data.

Depending on the code used, other groups of data may be required. The importance of writing this section clearly and comprehensively cannot be overestimated. Without it, the reader of the final report will have grave difficulty in understanding the results.

If more than one run of the code is carried out, perhaps in order to assess sensitivities to some parameters, the range of inputs used should also be specified.

9.7.4 RESULTS AND INTERPRETATION

This section should contain the results of the study. Presumably there will be some base case that has been calculated with best estimates of all input parameters and what is thought to be the best available set of models. The output for this base case should be presented as the final results of the study; the sort of output that is possible is discussed in Section 9.5.

This set of results should be accompanied or followed by interpretation. This interpretation may include the sensitivity to various input parameters--which might be reactor-oriented (possible variations in dominant accident sequences, say) or ranges of values in a parameter like the dry-deposition velocity--or to modeling changes made by the user of the code.

The interpretation of the results may also include comparisons with the results of other studies. For example, it may be thought desirable to compare calculated CCDFs with those given in the Reactor Safety Study, in order to explain what improvements have been brought about by changes in modeling and parameters in the consequence code, or by improved event- and fault-tree analyses of the dominant accident sequences, or by changes in the design of the reactor itself. It may also be desirable to compare estimated public risks with other man-caused risks.

The section should include a discussion of the uncertainties associated with the results. As explained in Section 9.6, it will not be possible to quantify these uncertainties in any exact statistical sense. Nonetheless, they exist and cannot be ignored.

Finally, this section should contain the conclusion or conclusions of the study, clearly stated. Examples could be that the predicted risk to the public from the operation of the reactor in question has been shown to be very small or that a certain design change has been very effective in reducing public risk. The nature of the conclusion clearly depends to some extent on the purpose of the study and, as has been shown in Section 9.1.2, there are a variety of purposes for a consequence analysis.

9.7.5 MISCELLANEOUS

There will probably be a need for a final section giving information on miscellaneous items such as the assurance of technical quality, acknowledgments, and references.

9.8 ASSURANCE OF TECHNICAL QUALITY

There are two aspects to the assurance of technical quality. The first is to ensure that the user has obtained a reliable code in good working order. The second is to ensure that he has used the code in a proper manner. Nowadays, a typical consequence-modeling code is such a vast compilation of multidisciplinary models in various fields--the modeling of turbulent processes such as meteorology and buoyant plume rise, health physics, economics, social studies, biology, soil physics, etc.--that no user can hope to go through it module by module, understand every detail, and verify that it is working as claimed in the user's guide. Indeed, there is a sense in which no consequence-modeling code has been properly validated. Such codes purport to predict the consequences of large accidental releases of radioactivity into the atmosphere and to predict the number of early and latent fatalities that might occur in the surrounding population. Since there has never been an accidental release of radioactivity from a commercial reactor that has caused the death of any member of the public, or indeed any detectable injury, the predictions of such codes cannot be validated, and it is hoped that they never will be.

The code user therefore has to rely on the presumed competence of the code developer. He has to assume that the models incorporated into the various modules of the code--the meteorological model, for example--represent good practice. He should read the code manual and associated literature with a critical eye, noting the sources cited for the various models and the justifications given for their use. He should not hesitate to query the code developer if there is any reason for doubt. This is another reason why the first task of the user, described in Section 9.2.1, namely, acquiring background, is so important. Consequence models should never be used as a black box.

What the user must do is to make sure that, once he has obtained a code, it is running properly on his machine. To do this he requires samples of input and output from the code developer. These samples should cover all areas of the code that he is likely to use in his consequence analysis.

If the user alters the code he has obtained, to incorporate better modeling, he should describe the reasons for the alterations and reference them. He should also carry out in-house calculations to ascertain that the new modeling is working correctly.

If, after this, the user is still unhappy about his code, there should shortly exist a means whereby he can check the predictions of the code by carrying out standard calculations and comparing the results with those of other consequence-modeling codes. This is the Benchmark exercise, already mentioned in Section 9.2.3. The forthcoming report of this exercise will specify seven standard problems designed to exercise most parts of consequence-modeling codes. The results obtained by some 20 organizations in Europe, the United States, and Japan will also be published. The user of a consequence model should repeat these standard runs and see whether his results lie within the envelope generated by a worldwide community of consequence-modeling experts. If he does, well and good. If not, he must examine his code to determine why his results differ. If he wishes to stand by his results, he must know his code well enough to determine the reasons for the difference and to justify the modeling or parameters that cause the difference.

Once the code is put to use in a specific consequence analysis, the problem of ensuring technical quality reduces to that of justifying the input data used. This can be done by compiling the data as described in Section 9.4 and by describing it clearly as outlined in Section 9.7.3. The input data set should also be independently reviewed in order to make sure that the collected data were actually input to the code. Finally, the output of the calculation should be presented and interpreted as described in Section 9.5.

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Chapter 10

Analysis of External Events

10.1 INTRODUCTION

External events (e.g., earthquakes, floods, fires, tornadoes, aircraft impacts, and explosions) can be considered at any level of PRA, depending on the objectives and the scope of the study, as discussed in Chapter 2. This present chapter describes how external events are selected for detailed evaluation in a PRA, discusses the methods used to evaluate their hazards, and explains how the assessment of external events is integrated with the analysis of internal events in evaluating the total plant risks. An overall procedure for treating external events is discussed here; details specific to some particular external events are presented in Chapter 11.

It should be noted that the basic PRA methods and procedures presented in the preceding chapters are generally applicable to all risk contributors, including the so-called external events. However, there are valid reasons for setting aside separate chapters of this procedures guide to discuss the analysis of external events. Most important, the analysis of external events requires the use of specialized methods to address important factors not usually encountered in the analysis of the internal events. These include the assessment of frequency of occurrence versus magnitude for external events and the modeling of component and structure failure in terms of variables that describe physical interactions. Since a complete risk analysis of an external event would entail many of the PRA elements generic to any risk contributor, there is a motivation to modularize the steps in the risk-analysis procedure for external events so as to avoid overlaps. The interfaces between external event analysis and the basic event- and fault-tree logic are discussed in Section 10.3.6.

In addition to natural and man-induced external events, the scope of this chapter includes internal flooding, fire, and turbine missiles, which are not external events in the strict sense. Events like sabotage and war are not included, although it is recognized that, with appropriate refinements, the overall procedure is applicable to these events also.

In specific application, the risk-assessment methods described here have heretofore been limited to earthquakes, winds, fires, and floods. Of the methods described, some are common to all external events and others are specific to a particular event. Application to other events not yet analyzed in detail may therefore require some additional development. Furthermore, it is recognized that the degree of uncertainty in estimating risk due to accidents caused by external events tends to be greater than that associated with other accident-initiating events that have been analyzed. Greater uncertainties stem from less experience in analyzing external events, lack of data, the use of relatively new analytical techniques, and greater reliance, perhaps, on engineering judgment and expert opinion. Engineering judgment is, however, not intended to replace the concerted effort to quantify the external events. If external events are deemed to significantly contribute to the overall plant risk, work on analytical models and data collection may be encouraged in the future, and its results may eventually reduce the uncertainties that now must be assigned. In the meanwhile, a detailed peer review of the assumptions, models, and input-parameter values is necessary to achieve consistency between different PRA studies relying heavily on engineering judgment in the treatment of external events.

The greater degree of uncertainty should not be construed as a reason for excluding external events in a PRA. Indeed, assessment of the magnitude of the effects of uncertainty is a key component of the risk quantification. Consistent with the overall philosophy of PRA, the analyst has to develop a complete description of each external event phenomenon and its effect on plant risk. Such a description should include not only the best estimate of the contribution of external events to plant risk but also the uncertainty in that contribution. Hence, greater uncertainty results in wider error bands about the best estimate of plant risk.

The selection of any external event for a detailed risk analysis will depend on its frequency of occurrence, magnitude, proximity, and consequences. The results of the external event analysis will be used as input to the PRA in defining initiating events, in developing event and fault trees for accident-sequence and system analysis, and in quantifying accident sequences. The depth of analysis suggested here for external events is commensurate with the overall objectives of this document, and the procedures presented represent the current state of the art in analyzing risks from external events. Since they are still in a developmental stage, it can be expected that the methods used in the analysis of external events will undergo significant changes as the industry gains experience in the treatment of external events in PRA studies.

In considering external events, the PRA analyst faces two fundamentally different types of variability. One is fundamental to the phenomenon being represented; the other is incomplete knowledge about the representation of that fundamental variability. Throughout Chapters 10 and 11, the word "frequency" is used when the inherent randomness of variables and events is discussed, and "probability" is used to refer to the uncertainty or current level of ignorance concerning the variables and events. In order to maintain this distinction and to treat both kinds of variability consistently, the "probability-of-frequency" format proposed by Kaplan and Garrick (1981) is adopted in the discussion of external events. More details on this format can be obtained from Kaplan et al. (1981).

10.2 OVERVIEW

10.2.1 SELECTION OF EXTERNAL EVENTS

The PRA studies that have been conducted to date have treated external events to varying degrees of detail. Some studies have excluded these events altogether. Some other studies have been motivated by external events (Pacific Gas & Electric, 1977; Smith et al., 1981). Since at present the collective experience of the industry in performing PRAs for nuclear power plants is rather limited and until sufficient sensitivity studies are conducted to assess their relative contribution to plant risk, external events cannot be dismissed a priori. Hence, a formal procedure is needed to ensure that all potential external events are considered and that the significant ones are selected for detailed PRA studies.

Although a detailed risk assessment is performed only for a few selected events, it is to be understood that the final plant-risk estimate includes the contributions from all external events. The contributions from events considered insignificant may be too small to show up in the significant digits reported for the total risk estimate. For example, let us assume that the mean (or best estimate) frequency of a particular release category for earthquakes is 10^{-6} per year. If the mean frequency of the same release category for aircraft impact is calculated as 10^{-8} per year, the mean release frequency from these two events is approximately 1.01×10^{-6} per year. The analyst may choose to report this estimate as 1×10^{-6} per year, thereby masking the contribution from the aircraft impact.

The screening of external events to select the significant ones consists of several steps. First, all external events specific to the site and plant are identified (Table 10-1 on pages 10-8 and 10-9 should be reviewed to ensure that all external events are indeed considered). Screening criteria are then established. Using these criteria, each external event is reviewed to judge whether it deserves further study. The external events that are discarded as being insignificant should be documented in the PRA study report along with the reasons for not performing a detailed analysis.

10.2.2 ASSESSMENT OF RISKS FROM EXTERNAL EVENTS

As shown in Figure 10-1, the basic elements of the analysis of risk from an external event are (1) hazard analysis, (2) plant-system and structure response analysis, (3) evaluation of the fragility and vulnerability of components (structures, piping, and equipment), (4) plant-system and sequence analysis, and (5) consequence analysis. The outputs are release or damage-state frequencies and risks. The information developed in hazard analysis, response analysis, and in component-fragility evaluation is input into the overall system models described in Chapter 3 and appropriately modified for the external event under study. The accident sequences specific to this external event are then quantified, and the frequencies of different release categories are calculated. The consequence analysis discussed in Chapter 9 is performed by using a consequence-analysis model that reflects the effects of an external event on the environment (e.g., a large earthquake or a severe flood may disrupt the communications network and damage evacuation routes, so that the distribution of the population exposed to radiation may be different than that for internal events). The plant risk, expressed as a frequency of exceedence (complementary cumulative distribution function) of damage (e.g., early fatalities, latent-cancer fatalities, or property damage), is calculated using the procedures discussed in Chapter 9. At each stage of the external event analysis, the analyst(s) should quantify the uncertainty in the output (e.g., uncertainty in the frequency of exceeding different levels of hazard intensity, uncertainty in component fragilities), and these uncertainties should be appropriately propagated through the entire analysis.

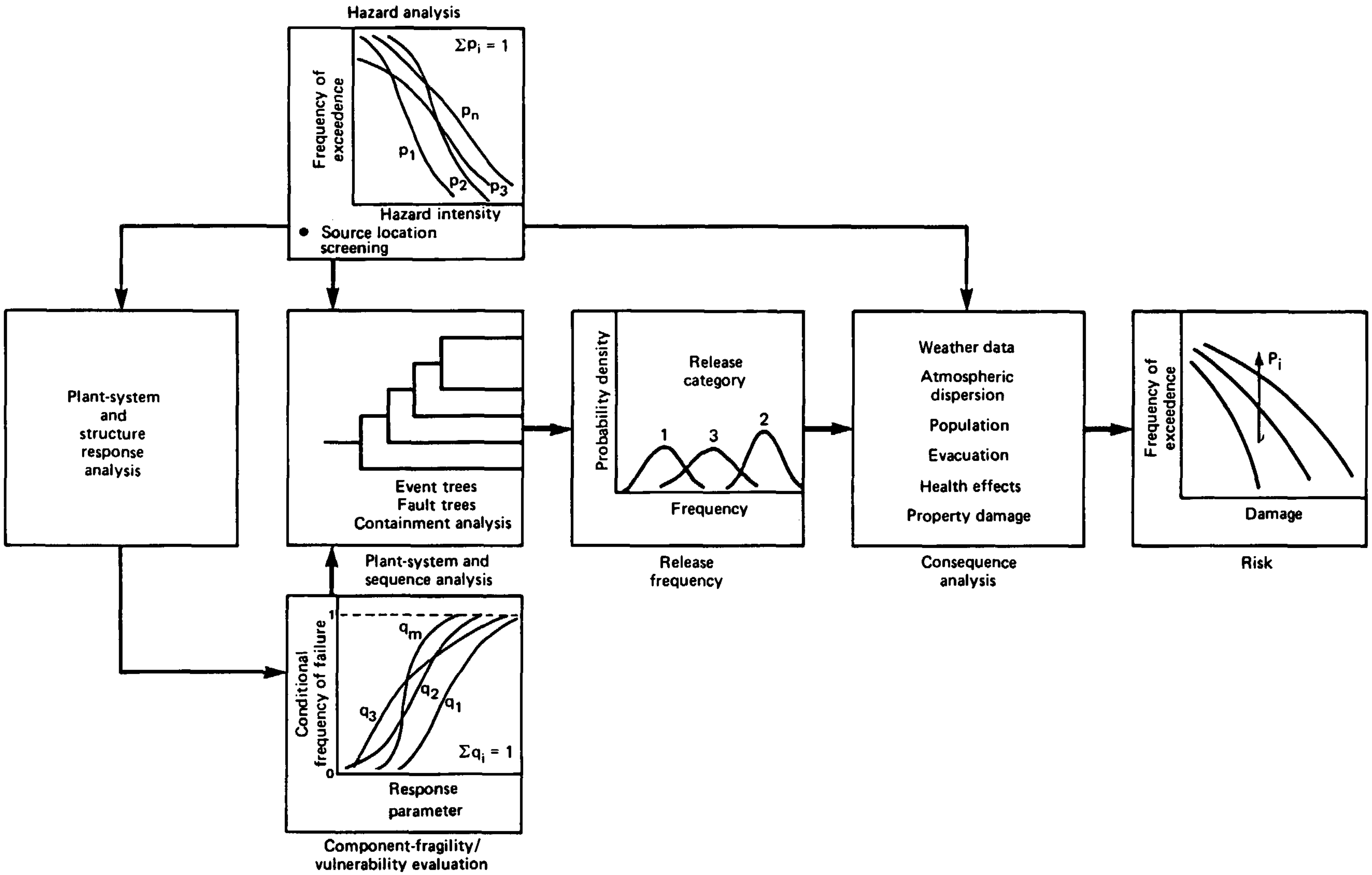


Figure 10-1. Risk-assessment procedure for external events.

A detailed risk analysis may not be warranted for some external events. The hazard-analysis results, combined with some bounding evaluation of plant damage and consequences, may indicate that the risk contribution of the external event is smaller by several orders of magnitude than those of other internal and external events. For example, the PRA study team may calculate the mean frequency of aircraft-impact damage to any one of the vulnerable structures whose failure could lead to core melt. If this frequency is much lower (e.g., 1/100) than those for other events and if the consequences of core melt from aircraft impact are comparable to other events in likelihood and magnitude, then no further detailed risk assessment for aircraft impact may be necessary.

In hazard analysis, the frequencies of occurrence of different intensities of an external event, called "hazard intensities," are calculated and presented in the form of a hazard curve. The hazard intensities could be represented by the peak ground acceleration of earthquakes, tornado intensities measured on the Fujita-Pearson intensity scale (Fujita, 1971), the sizes (weights) of aircraft, or the overspeed conditions under which turbine missiles are generated. Characterizing a complex hazard phenomenon by such a single parameter is generally inadequate. However, the other parameters that could be used to completely describe a hazard (e.g., for earthquakes, they could be duration, frequency content, etc.) are used in defining the hazard input to the response analysis. Also, the particular parameter selected to characterize an external event depends on the plant-system and sequence analysis. If the initiating events in the system event trees are related to different levels of earthquakes (e.g., 0.20g, 0.30g, and 0.40g), then the parameter of interest is the peak ground acceleration. If the initiating event is a fire in a specific area of the plant, the hazard analysis may consist of evaluating the mean rates of occurrence of fires of different sizes in various areas of the plant. The uncertainties in the hazard-parameter values and in the mathematical model of the hazard are represented by developing a family of hazard curves; a probability is assigned to each hazard curve. The summation of probabilities assigned over the family of hazard curves is unity.

In the response analysis, the response of plant systems and structures for a specified hazard input is calculated. The response of interest is generally the structural response at selected structural, piping, and equipment locations. For earthquakes, the response parameters could be spectral acceleration, moment, and deflection. For extreme winds, they could be force or moment on a structural element and deflection. For some external events (e.g., fire), no specific response analysis is performed.

In the evaluation of component fragility and vulnerability, the conditional frequencies of component failure for different values of the response parameter are calculated. Again, some differences exist between external events, depending on the plant-system and sequence analysis. For example, in a seismic risk analysis, fragilities may be expressed as functions of the local response parameter and evaluated separately for each component. In an analysis of risks from turbine missiles, the conditional frequencies of failure from turbine-missile impact are evaluated for different components in an accident sequence. These frequencies of failure depend on the location of the component with respect to the missile trajectory; the missile ricochet effects and the structural capacity of barriers are considered in calculating these frequencies. The uncertainties in the component-fragility parameters and the mathematical model are represented by developing a family of fragility curves for each component; a probability is assigned to each fragility curve. The summation of probabilities assigned over the family of fragility curves is unity. For external events that have discrete hazard-parameter values (e.g., turbine overspeeds and aircraft sizes), the component fragility is calculated at the corresponding discrete response values. The uncertainties are expressed by assigning probabilities to a vector of fragility values for a specified response value.

The plant-system and sequence analysis is performed by developing event trees and fault trees with an external event of a particular hazard intensity as the initiating event. The component fragilities are then used to compute the frequencies of failure for different safety systems. A very important consideration here is the dependences or correlations involved in the assignment of frequencies to multiple component failures. The calculated failure frequencies are conditional on the specified hazard intensity. The unconditional frequency of core melt or of radionuclide release for a given release category is obtained by integrating over the entire range of hazard intensities.

It may be logical to merge the external event analysis with the internal event analysis at the stage of plant-system and sequence analysis. In such an approach, the systems analysts should be apprised of the particular features of the external event that differ from the internal events. These features include, but are not limited to, differences in initiating events, in the event and fault trees, in the containment event trees, and in the quantification of accident sequences. The details of this interface are discussed in Section 10.3.6.

In some recent PRA studies, however, the analysts have chosen to treat the external events separately and to calculate the frequencies of release categories resulting from the external events. Several advantages are claimed for this treatment: (1) the differences of the external event analysis (e.g., initiating events, event and fault trees, containment-failure modes, and the quantification of fault trees) are made highly visible, resulting in the development of special analytical techniques; (2) the release-frequency analysis can be carried out with simplified plant-level fault trees; (3) the dependences between component failures that result from correlations between responses arising from the same loading (external event) and between component capacities arising from a common vendor and similar mounting can be handled explicitly; and (4) the contributions of different external events to core-melt frequency, release frequencies, and damage frequencies can be studied with a view to identifying the dominant events and planning optimal strategies for reducing (if needed) plant risk. The results of the external event analysis, in the form of frequencies of release categories, are then used, along with similar information from the internal event analysis, as input to the consequence analysis, if the analyst considers the differences between external and internal events in the consequence analysis to be insignificant; otherwise, the consequence analysis is carried out separately. The final product of the external event analysis is then an estimate of plant risk. The recently published Zion study (Commonwealth Edison Company, 1981) provides examples of how risks from external events are calculated.

The external event analysis should address the influence of design and construction errors and human errors due to operator action or inaction. In the PRA studies performed to date, design and construction errors have been partly accounted for by using as-built drawings and by visually inspecting existing plant conditions in a "walk-through" of the plant. Component-fragility evaluations have considered the customary tolerances in construction and manufacturing. The random equipment failures considered in these studies have included unavailability due to maintenance errors. Operator action in mitigating an accident may not be effective under extreme stress conditions (e.g., beams and walls cracking and collapsing in the control room under a large earthquake or a major fire in the control room). Operator errors of commission (e.g., turning off a wrong valve) under extreme stress were, however, not included in the past studies. The question of design and construction errors and human response deserves further study, as mentioned later in Chapter 13.

10.3 METHODS AND PROCEDURES

10.3.1 IDENTIFICATION AND SELECTION OF EXTERNAL EVENTS

An extensive review of information on the site region and plant design should be made to identify all external events to be considered. The data in the safety analysis report on the geologic, seismologic, hydrologic, and meteorological characteristics of the site region as well as present and projected industrial activities (i.e., the building of a reservoir, the construction of a road that carries hazardous materials, increases in the number of flights at an airport, etc.) in the vicinity of the plant should be reviewed for this purpose. The list of external events is to be exhaustive and is not to be constrained by any limitations on size or intensity; the screening techniques are meant to identify the significant external events to be included in the detailed risk assessment.

Table 10-1 lists the natural and man-made external events that should be considered in a PRA study. This list should be reviewed by the PRA study team to ensure that all applicable external events are included in the risk assessment. Although every attempt was made to list all possible external events, Table 10-1 should not be treated as an exhaustive set.

The external events identified as described above are screened in order to select the significant events for a detailed risk quantification. The PRA study team should formulate a set of screening criteria that should minimize the possibility of omitting significant risk contributors while reducing the amount of analysis to manageable proportions. As an example, a set of screening criteria is given below. Each of these criteria provides an acceptable basis for excluding external events from a detailed risk assessment. An external event is excluded if--

1. The event is of equal or lesser damage potential than the events for which the plant has been designed. This requires an evaluation of plant design bases in order to estimate resistance to a

Table 10-1. Natural and man-induced external events to be considered in PRA studies[a]

Event	Applicable screening criterion[b]	Remarks
Aircraft impact	--	Site specific; requires detailed study
Avalanche	3	Can be excluded for most sites in the United States
Coastal erosion	4	Included in the effects of external flooding
Drought	1	Excluded under the assumption that there are multiple sources of ultimate heat sink or that the ultimate heat sink is not affected by drought (e.g., cooling tower with adequately sized basin)
External flooding	--	Site specific; requires detailed study
Extreme winds and tornadoes	--	Site specific; requires detailed study
Fire	--	Plant specific; requires detailed study
Fog	1	Could, however, increase the frequency of man-made hazard involving surface vehicles or aircraft; accident data include the effects of fog
Forest fire	1	Fire cannot propagate to the site because the site is cleared; plant design and fire-protection provisions are adequate to mitigate the effects
Frost	1	Snow and ice govern
Hail	1	Other missiles govern
High tide, high lake level, or high river stage	4	Included under external flooding
High summer temperature	1	Ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms
Hurricane	4	Included under external flooding; wind forces are covered under extreme winds and tornadoes
Ice cover	1, 4	Ice blockage of river included in flood; loss of cooling-water flow is considered in plant design
Industrial or military facility accident	--	Site specific; requires detailed study
Internal flooding	--	Plant specific; requires detailed study
Landslide	3	Can be excluded for most sites in the United States
Lightning	1	Considered in plant design
Low lake or river water level	1	Ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms

Table 10-1. Natural and man-induced external events to be considered in PRA studies[a] (continued)

Event	Applicable screening criterion[b]	Remarks
Low winter temperature	1	Thermal stresses and embrittlement are insignificant or covered by design codes and standards for plant design; generally, there is adequate warning of icing on the ultimate heat sink so that remedial action can be taken
Meteorite	2	All sites have approximately the same frequency of occurrence
Pipeline accident (gas, etc.)	--	Site specific; requires detailed study
Intense precipitation	4	Included under external and internal flooding
Release of chemicals in onsite storage	--	Plant specific; requires detailed study
River diversion	1, 4	Considered in the evaluation of the ultimate heat sink; should diversion become a hazard, adequate storage is provided
Sandstorm	1	Included under tornadoes and winds; potential blockage of air intakes with particulate matter is generally considered in plant design
Seiche	4	Included under external flooding
Seismic activity	--	Site specific; requires detailed study
Snow	1, 4	Plant designed for higher loading; snow melt causing river flooding is included under external flooding
Soil shrink-swell consolidation	1	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard
Storm surge	4	Included under external flooding
Transportation accidents	--	Site specific; require detailed study
Tsunami	4	Included under external flooding and seismic events
Toxic gas	4	Site specific; requires detailed study
Turbine-generated missile	--	Plant specific; requires detailed study
Volcanic activity	3	Can be excluded for most sites in the United States
Waves	4	Included under external flooding

[a]Modified from ANSI/ANS-2.12-1978 (American Nuclear Society, 1978).

[b]See Section 10.3.3 for a sample set of screening criteria. The values given in this table are intended for illustration purposes only. For a specific PRA project, the analyst of external events should establish site-specific screening criteria and apply them to select the external events that may require a detailed study.

particular external event. For example, it is established that safety-related structures designed for earthquake and tornado loadings can safely withstand a 1-psi peak positive incident overpressure from explosions (USNRC, 1978). Hence, if the PRA analyst demonstrates that the overpressure resulting from explosions at a source (e.g., railroad, highway, or industrial facility) cannot exceed 1 psi, these postulated explosions need not be considered. It is assumed that the conditional frequencies of failure of structures and components for overpressures of less than 1 psi are negligible given that the safety-related structures are designed for earthquake and tornado loadings. This screening criterion is not applicable to events like earthquakes, floods, and extreme winds since their hazard intensities could conceivably exceed the plant design bases.

2. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events. For example, the PRA analyst may exclude an event whose mean frequency of occurrence is less than some small fraction of those for other events; the uncertainty in the frequency estimate for the excluded event is judged by the PRA analyst as not significantly influencing the total risk. Alternatively, the analyst may decide to compare event occurrence frequencies at some high confidence level (e.g., 95 percent). After the total plant risk is estimated, the deleted external events may have to be reviewed to ascertain that a detailed assessment would not reveal them as significant contributors to the total plant risk.

3. The event cannot occur close enough to the plant to affect it. This is also a function of the magnitude of the event. Examples of such events are landslides, volcanic eruptions, earthquake-fault ruptures (seismic motion and its effects are treated under seismic events), and explosions.

4. The event is included in the definition of another event. For example, storm surges and seiches are included in external flooding; the release of toxic gases from sources external to the plant is included in the effects of either pipeline accidents, industrial or military facility accidents, or transportation accidents.

For the sake of illustration, the above screening criteria are applied in Table 10-1 to indicate which external events may be deleted from a detailed risk assessment. It is seen that a detailed risk assessment is required for the following external events:

1. Aircraft impacts.
2. External flooding.
3. Extreme winds and tornadoes (and associated missiles).
4. Fire.
5. Accidents in nearby industrial or military facilities.
6. Internal flooding.
7. Pipeline accidents (gas, etc.).

8. Release of chemicals stored at the site.
9. Seismic events.
10. Transportation accidents.
11. Turbine-generated missiles.

The analyst is urged to use caution in applying screening criteria based solely on plant design bases. Some design-basis arguments were made to justify the exclusion of external events from early PRA studies. However, recent attempts at quantifying the risk associated with external events have led to some surprising results, as described in Chapter 11. One of the pitfalls of using criteria based on plant design bases is that emphasis is placed on comparisons of the event list with the design bases of the so-called safety-related systems and structures. However, there are important risk contributions from the so-called non-safety-related systems as well, and their capabilities and capacities are neither well defined nor documented. Moreover, the analyst should not lose sight of the possibility that the magnitude of an external event may exceed the plant design basis. It may be necessary to perform some bounding estimates of the risk contribution before a convincing case can be made for excluding any given external event from a PRA.

The screening of external events may start at the beginning of a PRA project in order to maintain the schedule. However, once an external event has been identified for a detailed PRA study, it is more efficient to perform the task after the event and fault trees for the internal events are developed so that the consequences of failures are more apparent.

The next section describes how a detailed risk assessment for these selected external events can be performed.

10.3.2 METHOD FOR ASSESSING RISKS FROM EXTERNAL EVENTS

As already mentioned, the basic elements of the analysis of risk from an external event are (1) hazard analysis, (2) plant-system and structure response analysis, (3) evaluation of component fragility and vulnerability, (4) plant-system and sequence analysis, and (5) consequence analysis. Depending on the stage at which the analyses of internal and external events are merged together, the final product of the external event analysis could be (1) results of hazard analysis, component fragilities, and modifications to system event and fault trees, and containment-failure modes; (2) probability distributions of frequencies for various release categories; or (3) probability distributions of frequencies for various damage indices (e.g., early fatalities, latent-cancer fatalities, or property damage).

The PRA of an external event can be viewed as a problem in determining $f_k(z)$, the unconditional frequency of exceeding damage level z of consequence type k, resulting from potential reactor accidents initiated by the external event. The quantity $f_k(z)$ can be expressed as

$$f_k(z) = \iint_{\mathcal{D}} \cdots \int f\left[\bigcup_{j=1}^{J} \{S_j(\underset{\sim}{y}), k(z)\}\right] h(\underset{\sim}{x})\ d\underset{\sim}{x} \qquad (10\text{-}1)$$

where

$h(\underset{\sim}{x})\, d\underset{\sim}{x}$ = frequency of occurrence of the external event with hazard intensity represented by the parameter values between $\underset{\sim}{x}$ and $\underset{\sim}{x} + d\underset{\sim}{x}$. Note that $\underset{\sim}{x}$ is a vector whose components represent different variables associated with the hazard, and the integration is carried over the entire domain of $\underset{\sim}{x}, \mathcal{D}$.

$\underset{\sim}{y}$ = vector of responses at a component location (structure, piping system, or equipment). Note that the responses are functions of hazard-intensity variables $\underset{\sim}{x}$; that is, $\underset{\sim}{y} = G(\underset{\sim}{x}, \underset{\sim}{\varepsilon})$, where $\underset{\sim}{\varepsilon}$ denotes the uncertainties in the response analysis.

$S_j(\underset{\sim}{y})$ = accident sequence j; a minimal cut set of components (1 to m_j)

$= \{1 \cap 2 \cap \cdots \cap m_j\}$.

$k(z)$ = damage level z for consequence type k.

k = consequence type k = 1,...,K (e.g., early fatalities, latent-cancer fatalities, and property damage).

$\cup$ = symbol for the union of events.

$\cap$ = symbol for the intersection of events.

In Equation 10-1, the term

$$f\left[\bigcup_{j=1}^{J} \{S_j(\underset{\sim}{y}), k(z)\}\right]$$

is the frequency of occurrence of any one of the j sequences resulting in damage level z of consequence type k. For a particular sequence j, the frequency of occurrence, $f[S_j(\underset{\sim}{y})]$, is calculated as the joint frequency of failure of components $1,2,\ldots,m_j$ in a single occurrence of the external event; it is a function of component fragility and response $\underset{\sim}{y}$. The conditional frequency of exceeding damage level z of consequence type k given the accident sequence j is $f_{k|S_j,\underset{\sim}{x}}(z)$. Note that this is a function of $\underset{\sim}{x}$, the hazard intensity. The effect of the external event on the environment is considered in this evaluation.

Since different accident sequences may involve some common components, the sequences are interdependent. The evaluation of the union of sequences in Equation 10-1 should take into account any correlations between component failures. These correlations arise from common structural models, single hazard input, and similar equipment (e.g., common vendor and identical mounting). There may also be some environmental dependence between component-failure events; for example, the collapse of a wall may damage a number of components simultaneously.

For the purposes of comparison with other external and internal events, and for merging with the risk analysis of internal events, the frequency of core melt from the external event and the frequencies of occurrence of different release categories can be calculated as shown below. The frequency of occurrence of core melt from an externally initiated accident, f_c, is expressed as

$$f_c = \iint \cdots \int f \left\{ \bigcup_{j=1}^{J_c} S_{j,c}(\underset{\sim}{y}) \right\} h(\underset{\sim}{x})\, d\underset{\sim}{x} \tag{10-2}$$

where $S_{j,c}(\underset{\sim}{y})$ is core-melt sequence j_c ($j_c = 1, \ldots, J_c$):

$$S_{j,c}(y) = \{1 \cap 2 \cap \cdots \cap m_{j,c}\}$$

The frequency of occurrence of a release category, $\gamma = 1, \ldots, \Gamma$, from an accident initiated by an external event, f_γ, is expressed as

$$f_\gamma = \iint \cdots \int f \left\{ \bigcup_{j=1}^{J_\gamma} S_j(\underset{\sim}{y}) \right\} h(\underset{\sim}{x})\, d\underset{\sim}{x} \tag{10-3}$$

where S_j is the accident sequence contributing to release category γ.

The total frequency $f_k^E(z)$ of exceeding damage level z of consequence type k resulting from all external events is approximately

$$f_k^E(z) \approx \sum f_k(z) \tag{10-4}$$

Equation 10-4 is based on the assumption that the external events are statistically independent and that the frequencies of the simultaneous occurrence of two or more external events are small. However, the PRA analyst should study the possible dependence between external events. Note that the above formulation accommodates dependence by describing the hazard intensity and response in terms of vectors. This facilitates the treatment of multiple secondary events arising from a single external event. For example, a severe storm can produce concurrent flooding, high winds and associated missiles, and dam overtopping. The effects of some dependences have been considered in past PRA studies. For example, seismically induced dam failures and pipeline failures are considered in seismic risk analysis; in the hazard modeling, certain ambient conditions (i.e., waves, snowpack, etc.) are included. Although two external events may not simultaneously exert stress on a specific nuclear plant component (structure, piping, and equipment), they may affect different components in the same accident sequence (i.e., an earthquake may fail the reactor components, whereas flooding may damage the service-water pumps in the crib house). Also, the effect of one external event may be to induce a radionuclide release as a result of a reactor accident, whereas the other external event may modify the parameters of the consequence model.

The sections that follow present methods for evaluating different elements of Equations 10-1 through 10-3.

10.3.3 HAZARD ANALYSIS

A hazard analysis estimates the frequency of occurrence for different intensities of an external event, called "hazard intensities." It may be performed by developing a phenomenological model of the event, with parameter values estimated from available data and expert opinion. Alternatively, the hazard analysis may consist of extrapolating historical data, if appropriate. It should be noted that a hazard event can be described adequately only by a multitude of variables. For example, tornado hazard is described by the rate of occurrence, tornado path width, path length, translational wind speed, tangential wind speed and vertical velocity, and the number and types of objects that are potential missiles. One or more of these variables may be probabilistically dependent on other variables. Although the hazard model may be described in terms of some of these variables, the output of the analysis is generally expressed in terms of a limited number (typically, one) of variables. The tornado hazard may, for example, be characterized by site wind speeds (i.e., frequencies of exceeding different site wind speeds). The other variables that are necessary for a "complete" description of the hazard are to be considered in the response analysis and fragility evaluation. The particular variables(s) chosen to present the results of the hazard analysis may also depend on the plant-system and sequence analysis.

Typically, the output of hazard analysis is a hazard curve of exceedence frequency versus hazard intensity. Since there may be a great deal of uncertainty in the parameter values and in the mathematical model of the hazard, it is important to represent the effects of uncertainty (see Section 10.3.4.6) through a family of hazard curves. Each curve is plotted for a postulated set of parameter values and a selected hazard model. A probability value, P_i, is assigned to each curve. An example of a family of hazard curves is shown in Figure 10-2. For a discrete event, the result of the hazard analysis would be a probability distribution of the frequency of occurrence. An example of this type of event is the turbine-generated missile. The 95-percent probability interval of the annual frequency of turbine-missile generation could be reported as 10^{-5} to 10^{-3}.

The seismic hazard analysis described in Chapter 11 is a good example of the phenomenological approach. Chapter 11 also describes the analyses for fires and floods, which have emphasized the analysis of historical data. A detailed description of an aircraft-hazard analysis is available in a report published by a committee of the American Society of Civil Engineers (1980), which also lists some significant references on the topic. Hazard analyses for extreme winds and tornadoes have been described by Abbey (1976), Fujita (1971), Wen and Chu (1973), Garson et al. (1975), Wen (1976), Twisdale et al. (1978), and Simiu et al. (1979). Hazard analyses for accidents at industrial or military facilities, pipeline accidents, and transportation accidents are described in guidelines published by the American Nuclear Society (1978), which also contain an extensive bibliography, as well as reports by Cave and Kazarians (1978) and Eichler and Napadensky (1978). Details on the analysis of turbine-missile hazards are available in reports published by the American Society of Civil Engineers (1980), Bush (1973, 1977), and the Electric Power Research Institute (1981). Hazards from the onsite storage of chemicals are evaluated on the basis of quantity, distance from the control room, and the detection capabilities of the control room.

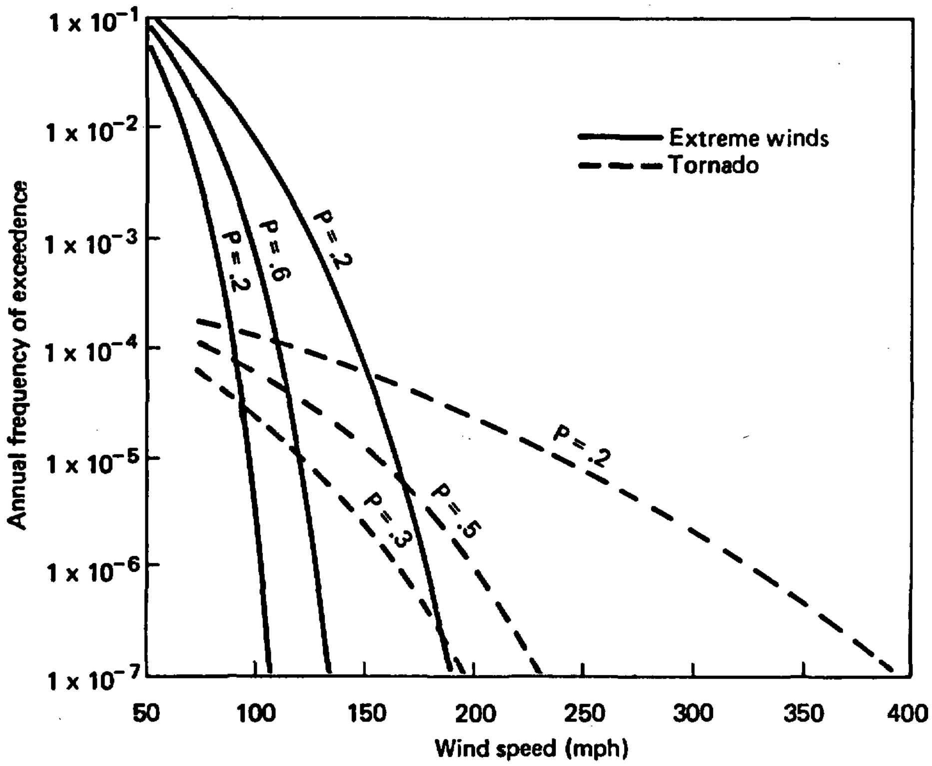


Figure 10-2. Family of hazard curves.

If the results of the hazard analysis and bounding evaluation of consequences show that the external event is not a significant contributor to risk, the PRA analyst may decide not to perform a more-detailed risk assessment. For example, if the impact of an aircraft or a turbine missile on any one of the safety-related structures has the potential for inducing a core melt but its frequency is much lower than that of the core-melt frequencies from other events, the event is ignored in further analysis. If the mean frequency for the rupture of a natural-gas pipeline near the plant is 10^{-7} per year, the effects of such a rupture (i.e., overpressure, missiles, and fire) are evaluated by using upper-bound assumptions. If the plant systems and structures are judged capable of withstanding these effects, the frequency of core melt from the pipeline rupture is assumed to be negligible.

Hazard analysis for fire and internal flooding may become intractable if all potential sources (i.e., locations) of hazard are to be considered. Procedures for screening significant source locations are discussed in Chapter 11.

10.3.4 ANALYSIS OF PLANT SYSTEM AND STRUCTURE RESPONSES

The purpose of this analysis is to translate the hazard input $\underset{\sim}{x}$ into the responses $\underset{\sim}{y}$ acting on a component. This generally involves an analysis of the structures, piping systems, and equipment. The hazard input could be a set of earthquake time histories, wind forces at selected elevations of the structures, the impact of an aircraft at a chosen location, or an incident pressure pulse due to a transportation accident. For each hazard intensity, the output of the response analysis would be a frequency distribution of the responses, such as spectral acceleration, peak moment, force, and deflection. The specific responses that are calculated depend on the failure modes of components. In the response analysis, any correlation between component responses resulting from the same hazard may be identified. When plant-design analysis information is considered appropriate, it may be used to estimate the structural responses for some external events. This circumvents the need for a detailed response analysis (Commonwealth Edison Company, 1981).

Some external events may not induce stresses in structures or components (e.g., a release of chemicals stored at the site and fire in a compartment). The response of the plant system (i.e., component and operator) needs to be considered in developing the accident sequences for such events. The propagation of fire and gases (e.g., smoke and chemicals) inside the plant determines which components and systems are affected.

10.3.5 EVALUATION OF COMPONENT FRAGILITY AND VULNERABILITY

The fragility or vulnerability of a component is defined as the conditional frequency of its failure given a value of the response parameter. For example, assume the wind hazard is characterized by the wind speed and let the wind speed be V_0. The response (e.g., force) due to this wind speed at a component location is R_0. The component's capacity to withstand the wind force is a random variable, C. The fragility of the component is calculated as

$$f = \text{frequency}\ \{C < R_0\} \tag{10-5}$$

If the component capacity is modeled as a lognormally distributed random variable with median $\check{C}$ and logarithmic standard deviation β, then f is calculated as

$$f = \Phi\left[\frac{\ln(\check{C}/R_0)}{\beta}\right] \tag{10-6}$$

where $\Phi(\cdot)$ is the standard Gaussian cumulative distribution function.

The fragility is estimated from the actual capacity of the component in any given failure mode. However, in estimating the capacity, uncertainties arise from several sources: an insufficient understanding of structural material properties and failure modes, errors in the calculated response due to approximations in modeling, and the use of generic data and engineering judgment in lieu of plant-specific data. Component fragility is therefore represented by a family of fragility curves. Each curve is developed on the basis of an assumed set of parameter values and failure modes. A probability q_i is assigned to this curve. The development of seismic fragility curves is explained in Chapter 11.

In some applications, the fragility parameter is taken as the hazard intensity; the capacity of the component, derived from design criteria and test data, is expressed in terms of the hazard intensity, using design-analysis information. The example given below shows how fragility curves are developed for a structure subject to wind loading. Let the design wind speed be V_d (for example, it could be 80 mph, corresponding to a mean recurrence interval of 100 years, at a reference height of 33 feet). The structure capacity C (mph) for wind loading can be expressed as

$$C = V_d F_W F_S \tag{10-7}$$

Here F_W is the safety factor relating the design wind pressure to the actual wind pressure on the structure; it is a function of terrain (exposure), peak pressure fluctuations, and gust response. The safety factor F_W is expressed as

$$F_W = \check{F}_W \varepsilon_{W,R} \varepsilon_{W,U} \tag{10-8}$$

where $\check{F}_W$ is the median safety factor, $\varepsilon_{W,R}$ is a random variable reflecting the inherent randomness in the wind pressure, and $\varepsilon_{W,U}$ is a random variable reflecting the uncertainty in the calculation of $\check{F}_W$. Both $\varepsilon_{W,R}$ and $\varepsilon_{W,U}$ are assumed to be lognormally distributed with logarithmic standard deviations $\beta_{W,R}$ and $\beta_{W,U}$, respectively. The values of $\beta_{W,R}$ and $\beta_{W,U}$ are taken as 0.20 and 0.30, respectively. The other quantity in Equation 10-7, F_S, is the safety factor relating the actual capacity of the structure to the calculated capacity. It is a function of the allowable stresses, the complete spectrum of load conditions for which the structure is designed, material strength variations, and approximations in structure modeling. The median value of $\check{F}_S$ is estimated as 1.5; the values of $\beta_{S,R}$ and $\beta_{S,U}$ are taken as 0.15 and 0.35, respectively.

Note that F_W and F_S are expressed as ratios of wind speeds. With these values, the median $\check{C}$ and the logarithmic standard deviations $\beta_{C,R}$ and $\beta_{C,U}$ of C are calculated as

$$\check{C} = 1.5V_d \tag{10-9}$$

$$\beta_{C,R} \approx (0.20^2 + 0.25^2)^{1/2} = 0.25 \tag{10-10}$$

$$\beta_{C,U} \approx (0.35^2 + 0.15^2)^{1/2} = 0.38 \tag{10-11}$$

Using Equations 10-9 through 10-11 and the lognormal-distribution assumption, the fragility of the structure, f', at a wind speed V, at any nonexceedence probability level Q can be derived by using the formulation given by Kennedy et al. (1980):

$$f' = \Phi\left[\frac{\ln(V/\check{C}) + \beta_{C,U}\Phi^{-1}(Q)}{\beta_{C,R}}\right] \tag{10-12}$$

where $Q = Pr[f < f'|V]$ is the probability that the true conditional failure frequency f is less than f' given a wind speed V and where $\Phi^{-1}(\bullet)$ is the inverse of the standard Gaussian cumulative distribution function. Note that Q is the sum of the probabilities assigned to all the fragilities less than f'. By this formulation, both the inherent randomness and the uncertainty are explicitly represented. Figure 10-3 shows a family of fragility curves for the structure. Such fragility curves are developed for different components whose failures are identified as either initiating an accident or contributing to any significant accident sequence that would result in a core melt or the release of radionuclides.

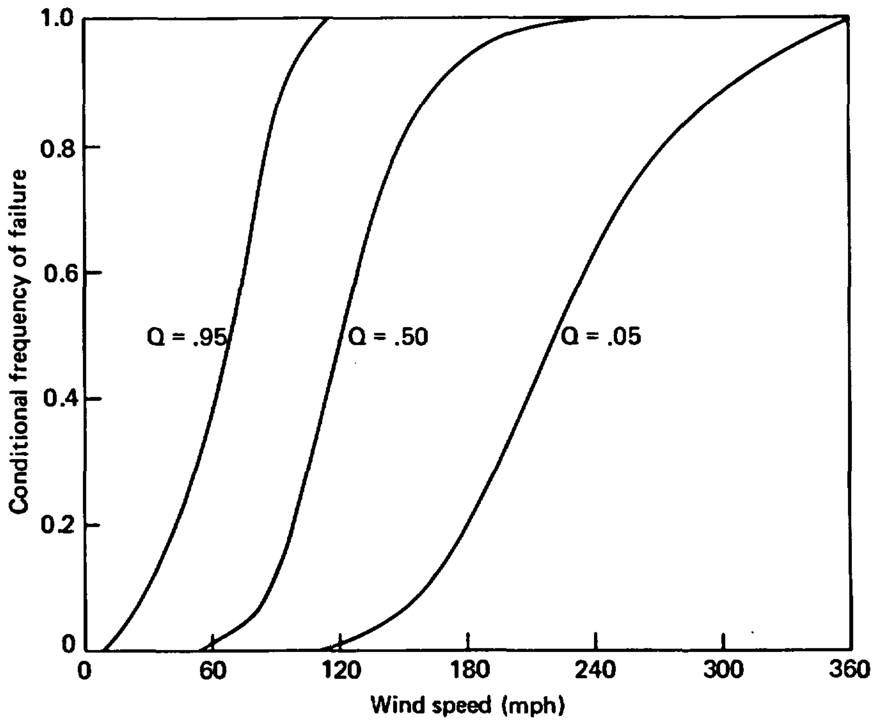


Figure 10-3. Fragility curves for wind loading.

For some external events, component fragility can be taken as 1.0 if the hazard intensity reaches a specific value (e.g., the inundation of equipment by flooding and turbine-missile impact on a vital component). The fragilities of individual components along with the information on correlation of responses and of capacities between components can be used to calculate the conditional frequencies of accident sequences consisting of a number of components. For some external events (e.g., missiles, internal flooding, and fire), the fragilities of individual components may not be meaningful; the conditional frequencies of accident sequences are directly calculated (e.g., by counting how many times a turbine-missile trajectory passed through a sequence of components).

Before the individual component fragilities can be combined in the plant-system logic, which is discussed in Section 10.3.6, it is necessary to evaluate the degree of dependence to be assumed among the failure frequencies in sets of multiple components (i.e., minimal cut sets). An incorrect assumption that the failures occur independently can lead to optimistic predictions if the multiple components appear within the same minimal cut set or to conservative predictions if they are not in the same cut set. Specific examples of the dependence or correlation of component fragilities are given in Chapter 11.

10.3.6 ANALYSIS OF PLANT SYSTEMS AND EVENT SEQUENCES

The analysis of plant systems and event sequences consists of developing event trees and fault trees in which the initiating event can be the external hazard itself or a transient or LOCA initiating event induced or caused by the external event. Various failure sequences that lead to core melt, containment failure, and a specific release category are identified. The component fragilities are then used to compute the frequencies of the various event sequences. These calculated frequencies are conditional on the specified hazard intensity. The unconditional frequency of core melt or of radionuclide release for a given release category is obtained by integrating over the entire range of hazard intensities. The consequence analysis can be carried out separately for each external event when appropriate. The output of the external event analysis would then be curves of the frequencies of damage (i.e., acute fatalities, latent-cancer fatalities, or property damage) at different nonexceedence probabilities.

If the external event analysis is merged with the internal event analysis at the stage of event-tree development, the analyst should provide the information on the initiating events for each range of hazard intensity, necessary modifications to the event trees, complete fault trees, changes to the containment event tree, and differences in the consequence-analysis results, along with the hazard curves and component-fragility curves. The component-failure dependences resulting from a common hazard intensity and similar equipment should be explicitly represented in the fault trees.

If the external event analysis is merged with the internal event analysis at the consequence-analysis stage, the analyst should provide the probability distribution of the frequency of release for each release category. This probability distribution can be calculated by event-tree and fault-tree methods. In some instances, simplified plant-level fault trees are formulated for the core melt; the type of core melt (i.e., the plant state) is decided on the functioning of fan coolers and containment sprays. The plant states are aggregated with the containment states in order to determine the release category to which they belong. A Boolean expression in terms of component failures is derived for each release category. Component fragilities are used in this expression to compute the plant-level fragility family for each release category. This family, when integrated over the hazard curves, gives the probability distribution of the frequency of release for each release category. An example of these probability distributions is shown in Figure 10-4.

If the analyst decides to keep the analysis of an external event totally separate from the analysis of internal and other external events, the probability distributions of frequency of release categories are input into the consequence-analysis model developed for the external event. The consequence-analysis modeling may depend on the external event. For example, a large earthquake or an external flood may disrupt the communications network and damage the evacuation routes (the evacuation time was increased in the Diablo Canyon Seismic Risk Study (Pacific Gas & Electric Company, 1977) to account for the effect of large seismic events on roads, bridges, structures, and communications); and extreme winds may carry radioactive materials to more distant locations. For this level of external event analysis, the output is the final risk curve plotted for a specified nonexceedence probability level. This may be compared with the risk curves from other internal or external events to judge the relative risk significance of the events under study.

The interfaces between the analysis of plant systems and event sequences and those aspects of PRA peculiar to external events--namely, the analyses of hazard and hazard response--can be seen from the following example. Suppose that a nuclear plant consists of two systems, 1 and 2, each of which protects against core damage in response to some arbitrary initiating event, denoted by X. For example, X might represent a transient event and the two systems might be the auxiliary feedwater system and the collection of components and operator actions required for "feed and bleed" cooling, respectively. Further, suppose that system 1 and system 2 each consists of two redundant subsystems, denoted by A and B for system 1 and C and D for system 2. The event tree and the list of minimal cut sets for this simple example are presented in Figure 10-5.

Accident sequences resulting from some external event can be viewed as "superimposed" on the event tree in Figure 10-5 in the following way: The external event, denoted in this example by E, can produce or contribute to an accident sequence either by causing the initiating event X to occur, or causing the failure of one or more subsystems, or a combination of these. Alternatively, accident sequence S_3, for example, can result from any combination of external and nonexternal causes that results in the failure of systems 1 and 2 and the occurrence of the initiating event.

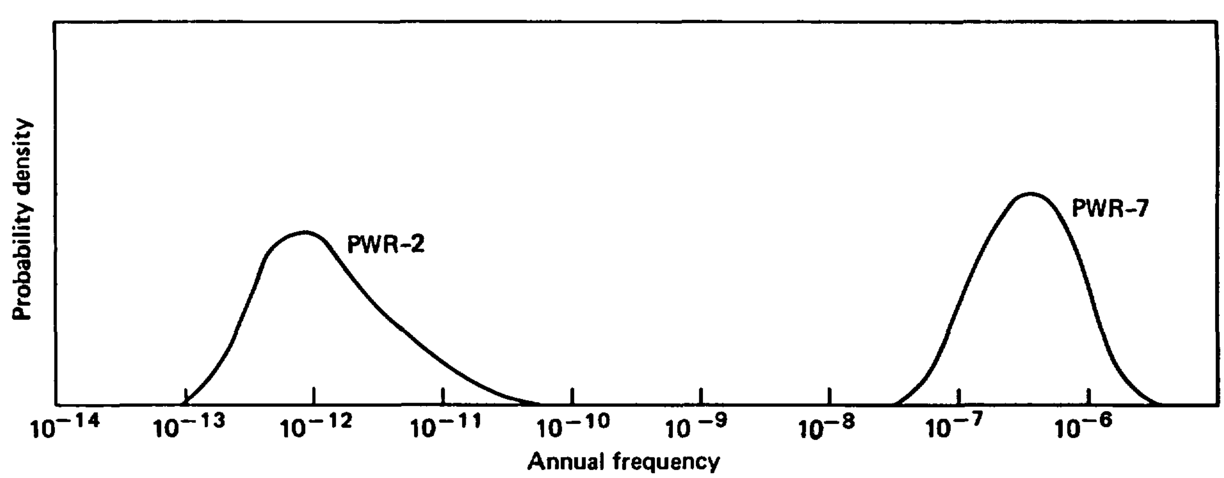
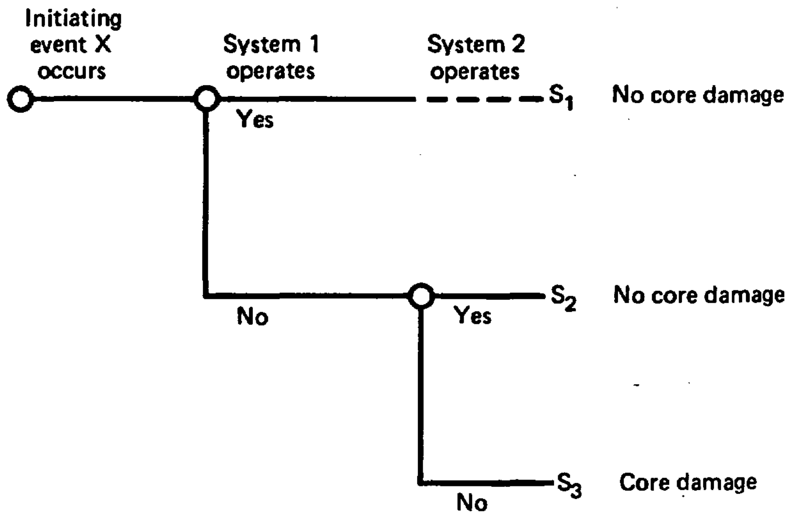


Figure 10-4. Release frequency from extreme-wind event for two release categories: PWR-2 and PWR-7.



System	Minimal cut sets
1	{A, B}
2	{C, D}

Figure 10-5. Example event tree indicating the responses of front-line systems 1 and 2 to initiating event X.

The interfaces between the event- and fault-tree methods, which are generic and applicable to all initiating events, and the analysis of hazard frequencies and responses, which are specific to external events, can be seen by structuring the accident sequences into the following sequence fragments:

1. The external event occurs.

2. Damage causally related to external event occurs.

3. The initiating event occurs.

4. Any additional failures not causally related to the external event occur.

5. Accident terminates in some damage state.

The tasks of hazard analysis and the analysis of the response of structures, systems, and components to the external events are then associated with sequence fragments 1 and 2, whereas sequence fragments 3, 4, and 5 are common to all accident sequences due to both internal and external causes.

As illustrated in Figure 10-6, there are 16 possible sequence fragments, referred to as external event damage states, that provide a complete representation of the possible effects of the external event on the four subsystems in our simple example. To the extent that the sequences in Figure 10-5 are a complete set, a complete representation of the accident sequences associated with external event E can be constructed by feeding each of the 16 sequence fragments in Figure 10-6 into the event tree in Figure 10-5, for a total of 48 sequences. (Note that, in this particular example, some of these 48 sequences would have zero frequency. For example, if external event damage state E_{16} occurs, it is impossible for sequences S_1 or S_2 to occur.)

Estimates of the frequency of core damage in the example can be made by applying Equation 10-2 in the modified form of

$$f_c = \int_{\mathscr{D}} \cdots \int f_{S,3}(\underset{\sim}{y})\ h(\underset{\sim}{x})\ d\underset{\sim}{x}$$

$$= \int_{\mathscr{D}} \cdots \int f\left[\bigcup_{\ell=1}^{16} \{E_\ell(y), S_3\}\right] h(\underset{\sim}{x})\ d\underset{\sim}{x} \qquad (10\text{-}13)$$

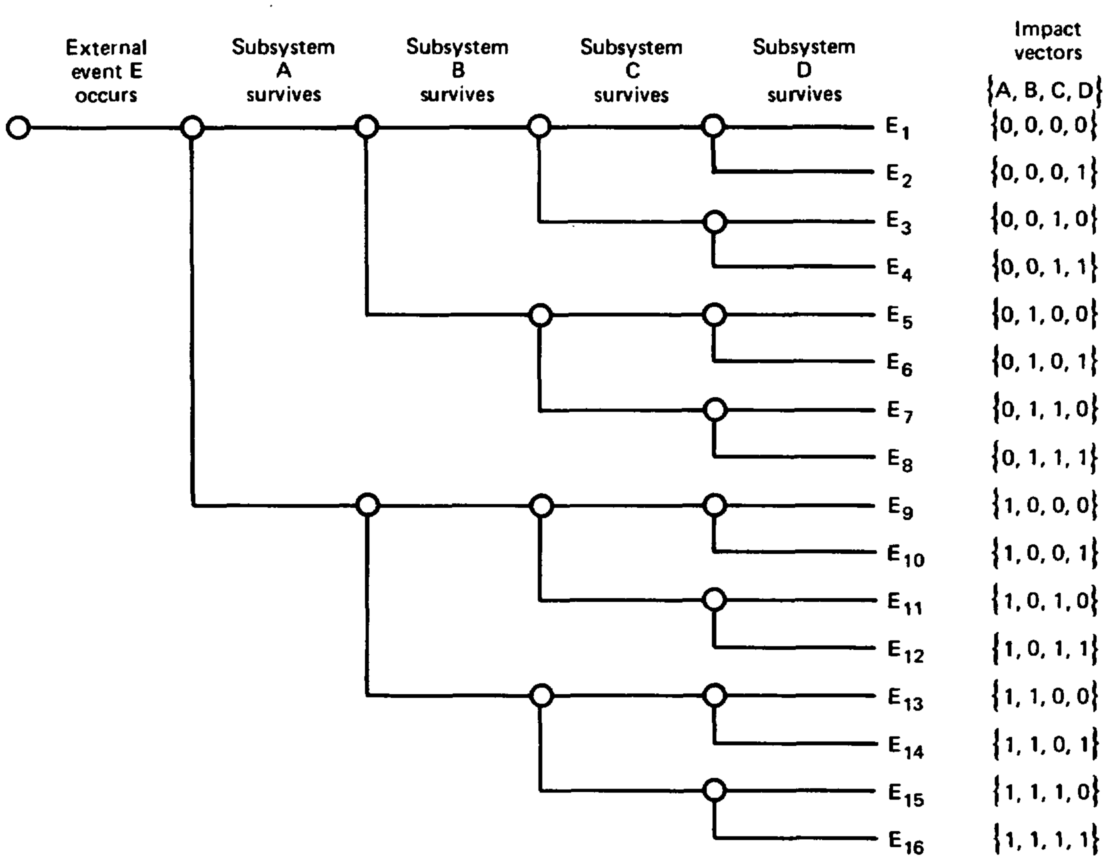


Figure 10-6. External event tree and associated impact vectors illustrating the possible damage states of four subsystems A, B, C, and D. In the heading of the tree "subsystem survives" means that the subsystem does not fail as a direct consequence of the external event. For the impact vectors, 0 indicates survival and 1 indicates a failure of the particular subsystem.

where $f[\cup_{\ell=1}^{16} \{E_\ell(\underset{\sim}{y}), S_3\}]$ is the frequency of occurrence of any one of the sequence fragments (or external event damage state) E_ℓ given response $\underset{\sim}{y}$ to the external event and resulting in the accident sequence S_3. An approximation for f_c is

$$f_c \approx \int_{\mathscr{D}} \cdots \int \sum_{\ell=1}^{16} f_{E,\ell}(\underset{\sim}{y}) \; f_{S,3|E,\ell} \; h(\underset{\sim}{x}) \; d\underset{\sim}{x} \qquad (10\text{-}14)$$

where $f_{E,\ell}(\underset{\sim}{y})$ is the frequency of sequence fragment E_ℓ given response $\underset{\sim}{y}$ to the external event and $f_{S,3|E,\ell}$ is the conditional frequency of accident sequence S_3 given external event damage state E_ℓ occurs.

Note that the assessment of the frequency of the external event damage state, $f_{E,\ell}(\underset{\sim}{y})$, involves a specific combination of failure and successes resulting from the external event. It is very important to recognize that dependences or correlations may preclude the synthesis of these frequencies as an independent combination of component fragilities.

The quantity $f_{S,3|E,\ell}$ represents the interface between the analysis of external events and the PRA analyses generic to all accident sequences. It can be estimated by using the basic event- and fault-tree methods described in Chapters 3 through 6 for all initiating events. It is extremely important, however, that the boundary conditions for quantifying the accident frequency reflect the impact of the external event. Those boundary conditions can be represented in the form of an "impact vector" (see Figure 10-6). In the quantification of the front-line event tree in Figure 10-5 for each external event damage state, E_ℓ, the impact vector denotes the subsystems that are failed as initial conditions in the quantification. For example, since external event damage state E_{16} results in the failure of minimal cut sets in both systems, it follows that $f_{S,3|E,16} = 1$.

An alternative representation that indicates the relationship between the external events and the generic aspects of PRA methods is a fault tree that is constructed for an entire accident sequence or a collection of sequences referred to as a plant-damage state or bin. Such a fault tree, constructed for the top event "core damage," is presented in Figure 10-7 for the earlier example. The fault tree indicates that each of the subsystems A, B, C, and D has both external and nonexternal failure causes and that the frequency of the initiating event depends on the external event. The quantification of this tree would produce a result equivalent to Equation 10-13. As discussed in Chapter 11, some of the seismic risk analyses have been carried out with the use of fault-tree logic to model plant-damage states similar to that in Figure 10-7.

In practical applications of external event risk analysis it is neither feasible nor desirable to provide such a complete enumeration of the possible combinations of external event damage states and nonexternal failure causes as illustrated in the above example. Indeed, it was possible to give a "complete" representation only because the example includes as few as four subsystems and only one accident sequence. To address the large number of subsystems and accident sequences postulated for a complex nuclear plant system, it is necessary to make some approximations and simplifications. One such simplification is to reduce the number of external event damage

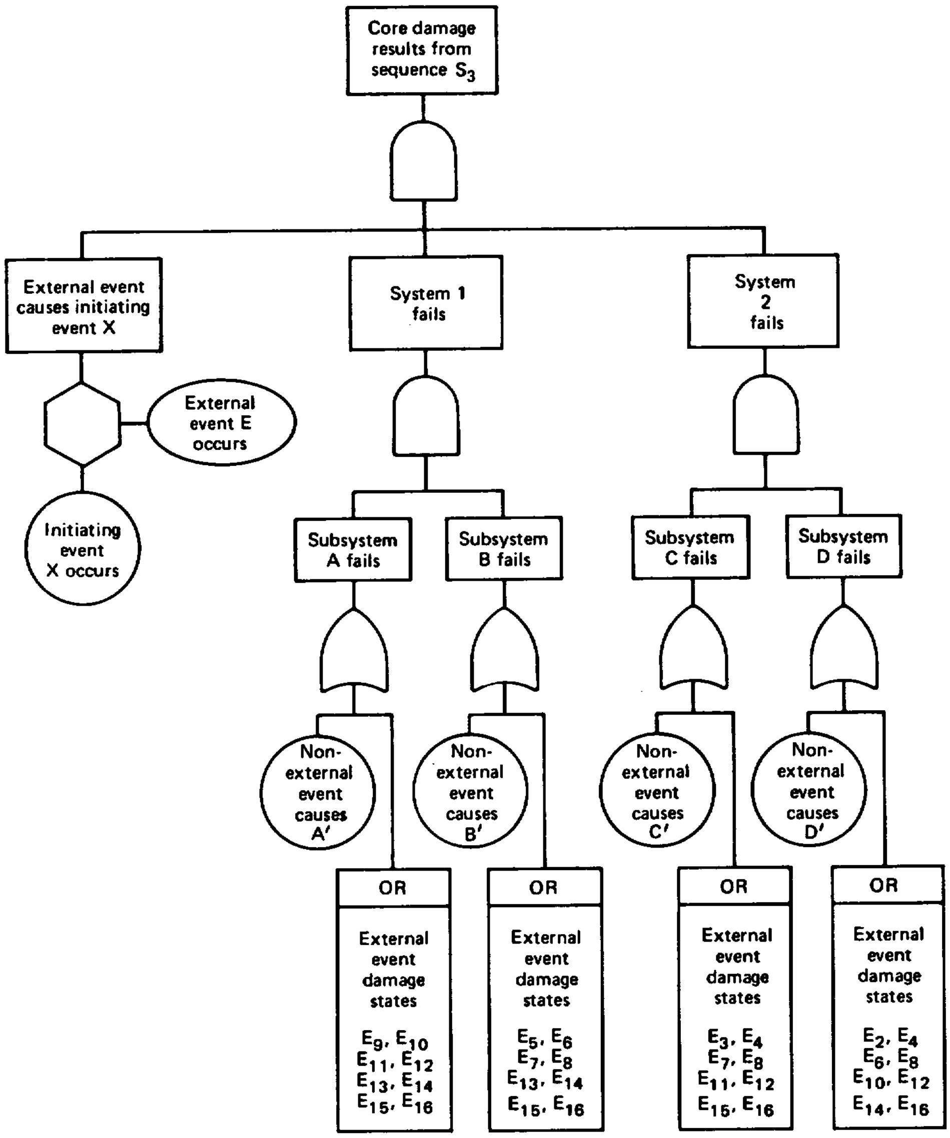


Figure 10-7. Fault tree for core damage due to external event.

states to a relatively small number in relation to the large number of cases that can be postulated. For example, suppose that in our example the external event in question is a fire at a specific location near the four subsystems, as shown in Figure 10-8. Of the 16 possible fire damage states that can be postulated, it should only be necessary to consider the following five, which are described in terms of their impact vectors:

$$E_1 = \{0,0,0,0\}; \quad E_9 = \{1,0,0,0\}; \quad E_{13} = \{1,1,0,0\}$$

$$E_{15} = \{1,1,1,0\}; \quad E_{16} = \{1,1,1,1\}$$

The remaining 11 damage states can be dismissed because the above are representative of the complete spectrum of states and would be expected to occur at a much greater frequency.

A second type of simplification is to specialize the logic of the event and fault trees to different discrete intervals of hazard or response intensity. For example, at high levels of intensity that approach or exceed the capacities of the plant components, it may not be necessary to consider nonexternal causes in the model.

A third approach to keeping the amount of data processing at a manageable level is to separately quantify the external event model, such as the event tree of Figure 10-6, and eliminate the accident-sequence fragments that make negligible risk contributions before quantifying the front-line event trees, such as that in Figure 10-5. This is made possible by the use of the impact vectors, which provide a measure of the damage of the external event in terms of subsystem-failure impact. This same impact vector is used to help model certain types of intersystem dependences, as described in Section 3.7. By comparing the frequencies and impact vectors of all the external event damage states, the total number of damage states can often be reduced by combining states with similar or symmetrical impact vectors and eliminating states that are negligible risk contributors before integrating the external event and nonexternal event portions of the event- and fault-tree logic.

10.4 TREATMENT OF UNCERTAINTY

There are many uncertainties in the analysis of external events; they arise from lack of data and analytical models. In the hazard analysis, the uncertainties to be considered are those in the frequency of occurrence of the hazard intensity, the characterization of the phenomenon (e.g., line source or point source for seismic events, path width and length models for a tornado, available sources of missiles for a tornado, and models for

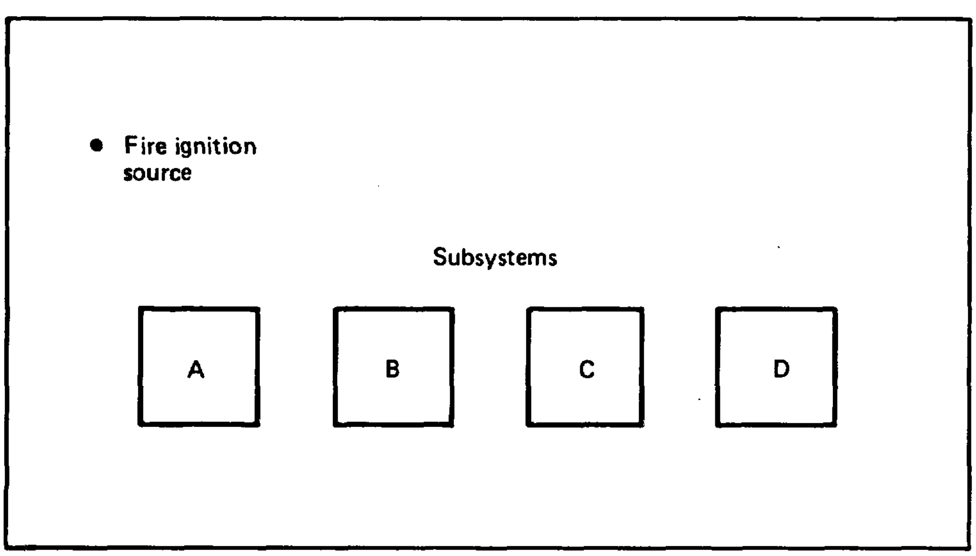


Figure 10-8. Hypothetical layout of subsystems in relation to fire.

explosive-vapor cloud transport), and the transmission of effects (e.g., overpressure, missiles, and ground acceleration) from the source to the site. In the component-fragility evaluation, uncertainties arise from an insufficient understanding of the properties and failure modes of structural materials, errors in the calculated response due to approximations in modeling, and the use of generic data and engineering judgment in the absence of plant-specific data.

At present, the quantification of uncertainty in different phases of the external event PRA is performed with a combination of limited analysis, sparse empirical data, and expert opinion. The uncertainty in hazard analysis is expressed by a family of hazard curves, each curve being drawn for a different nonexceedence-probability level. Similarly, the component fragility is expressed by a family of fragility curves, each curve being drawn for a specified nonexceedence-probability level.

One method of consistently propagating the uncertainties is to perform the risk analysis in two stages. In the first stage, the risk assessment is done by using the best-estimate hazard curve and the best-estimate fragility curve for each component. In the second stage, the risk assessment of the first stage is repeated many times, each time with a different set of hazard and component-fragility curves. These sets are sampled from the probability distributions of the hazard curves and the fragility curves reflecting their uncertainties. Since the uncertainties arise from an incomplete understanding of the phenomenon and from the use of simplified models, it is important to maintain correlations between the component fragilities in this sampling.

By performing this two-stage analysis a sufficient number of times, the probability distributions of core-melt frequency and the frequency of each release category can be obtained. A similar treatment of the uncertainties in the consequence analysis would yield the probability distribution of the exceedence of damage. Since this thorough treatment of uncertainty can become very expensive, the analyst should attempt to identify the dominant accident sequences and perform the uncertainty propagation for those sequences only.

If external events are analyzed with simplified plant-level fault trees, the uncertainties are propagated by assigning probability distributions for each component-failure frequency in the Boolean expressions. Usually, a family of curves for plant-level fragility for core melt and for each release category are obtained. Integration over the hazard-curve family then yields probability distributions for core-melt frequency and the frequency of each release category. Integration can be accomplished numerically by using discrete-probability-distribution arithmetic, the method of moments, Monte Carlo error propagation, response-surface analysis, or other statistical techniques discussed in Chapter 12.

Because of the large uncertainties present in the hazard analysis, component-fragility evaluation, plant systems, and the analyses of accident sequences, containment-failure modes, and consequences, it is important that the uncertainties be treated explicitly and consistently--and be propagated throughout the analysis in order to quantify the total uncertainty in the plant risk. Examples of available information on uncertainties are discussed in reference to seismic, fire, and flood risk analyses in Chapter 11.

10.5 INFORMATION AND PHYSICAL REQUIREMENTS

The plant design bases detailed in the safety analysis report should be reviewed for data on the site region and potentially hazardous activities in the vicinity of the plant. The analyst should ensure, however, that any conservative bias in the data is properly accounted for. This information is used with the models of external events to develop the frequencies of hazard intensities. The design criteria, applicable codes and standards, stress reports, material test data, design reports, location of plant safety systems and structures, dimensions of structural members, as well as reports on qualification and preservice tests and on periodic in-service inspection should be reviewed in order to develop the fragilities for components and systems.

10.6 DOCUMENTATION

The PRA report should contain a list of all external events that are identified as potential hazards, the screening criteria, and a table listing all excluded external events and giving the applicable screening criteria.

The report should contain a detailed description of the hazard analysis for each selected external event. The development of component fragilities, initiating events, event and fault trees, and containment event trees should be included. If the analysis is carried out independently of other external and internal events, the report should include the probability distributions of core-melt frequency, the frequencies of various release categories, and risk curves.

10.7 DISPLAY OF FINAL RESULTS

The results of the external event analysis described in this chapter will be the following:

1. The identification of external events appropriate to the site and plant.

2. The selection of the events for which a detailed risk assessment is done.

3. Hazard analysis, component fragilities, modifications to the event and fault trees and containment event trees, and modifications to the consequence model as input to the analyses described in Chapters 3 and 9.

4. Probability distributions of core-melt frequency, the frequencies of various release categories, and risk curves, if appropriate.

10.8 ASSURANCE OF TECHNICAL QUALITY

The provisions described in Chapter 2 for the assurance of technical quality are applicable to the external event analysis described in this chapter. The key elements are documentation, peer review of methods and data, and documentation of the parameters elicited from expert opinion.

NOMENCLATURE

C	capacity of a component
$\check{C}$	median capacity of a component
E_ℓ	a discrete level of damage described in terms of subsystems failed because of an external hazard (external event damage state)
F_S	safety factor relating the actual capacity of the structure to the calculated capacity
F_W	safety factor relating the design wind pressure to the actual wind pressure on the structure
$\check{F}_W$	median value of F_W
f	conditional failure frequency of a component; fragility
f'	fragility at a nonexceedence-probability level Q
f_c	frequency of core melt
f_γ	frequency of release by release category γ
$f_k(z)$	frequency of exceeding damage level z of consequence type k
$f_{k\|S,j}(z)$	conditional frequency of exceeding damage level z of consequence type k given accident sequence S_j
$f_k^E(z)$	total frequency of exceeding damage level z of consequence type k resulting from all external events
$f_{E,\ell}(\underset{\sim}{y})$	frequency of external event damage state E_ℓ
$f_{S,j}(\underset{\sim}{y})$	frequency of accident sequence S_j
$f_{S,i\|E,\ell}$	frequency of accident sequence S_i given damage state E_ℓ
$h(\underset{\sim}{x})\ d\underset{\sim}{x}$	frequency of occurrence of the external event with hazard intensity represented by parameter values between $\underset{\sim}{x}$ and $\underset{\sim}{x} + d\underset{\sim}{x}$
J_γ	number of accident sequences contributing to release category γ
k	consequence type $k = 1,\ldots,K$ (e.g., early fatalities, latent-cancer fatalities, and property damage)
m_j	total number of components in an accident sequence j
$m_{j,c}$	total number of components in a core-melt sequence

P, Q	nonexceedence-probability levels
p_i, q_i	probability assigned to hazard or fragility curve
R_0	response at a component location due to wind speed V_0
S_j	accident sequence $j = 1,\ldots,J$
$S_{j,c}$	core melt sequence $j_c = 1,\ldots,J_c$
V	wind speed
V_0	specific wind speed
V_d	design wind speed
$\underset{\sim}{x}$	a vector of hazard intensity parameters
$\underset{\sim}{y}$	a vector of responses
z	damage level of consequence type k
β	logarithmic standard deviation
$\beta_{C,R}$	logarithmic standard deviation reflecting the inherent randomness in the variable C
$\beta_{C,U}$	logarithmic standard deviation reflecting the uncertainty in C
γ	release category $\gamma = 1,\ldots,\Gamma$
$\mathcal{D}$	domain of $\underset{\sim}{x}$
$\varepsilon_{W,R}$	random variable reflecting the inherent randomness in the wind pressure
$\varepsilon_{W,U}$	random variable reflecting the uncertainty in the calculation of $\check{F}_W$
$\Phi(\cdot)$	standard Gaussian cumulative distribution function
$\Phi^{-1}(\cdot)$	inverse of the standard Gaussian cumulative distribution function

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Chapter 11

Seismic, Fire, and Flood Risk Analyses

11.1 INTRODUCTION

Chapter 10 described the overall procedure for estimating radiological risks from external events. The objective of the present chapter is to illustrate the application of this procedure to three specific external events: earthquakes, fires, and floods. As mentioned in Chapter 10, some modifications of the procedure may be necessary, depending on the external event under study.

The external event analyses discussed in this chapter illustrate different aspects of the overall procedure. The section on seismic risk analysis (Section 11.2) emphasizes the development of hazard and fragility models for predicting the occurrence frequencies of large earthquakes (i.e., earthquakes well beyond the plant design basis) and estimating the failure frequencies of components subjected to such earthquakes. The section on fire-risk analysis (Section 11.3) presents techniques used in screening for critical hazard locations and explains the details of a fire-propagation analysis. Section 11.4, which covers flood-risk analysis, highlights the uncertainties of a hazard analysis based on sparse or questionable data and describes the techniques of hazard-source screening and fragility development.

The external events discussed in this chapter as illustrations of the overall procedure have also some historical significance. For various reasons, each of them has been studied in the past, although not to the same extent. The procedures described in Chapter 10 are directly applicable to these events. For other external events, either the analytical methods have not been developed and applied or the experience in treating the events has been highly limited. The probabilistic analyses of external events discussed in plant safety analysis reports are generally restricted to one or a few stages of the overall procedure (e.g., a hazard analysis and the evaluation of structural failure frequencies) and are aimed only at calculating the frequencies of unacceptable damage as defined by NRC regulatory documents. A complete PRA has not been the objective of the studies that are reported in the safety analysis reports.

In keeping with the spirit of this guide, this chapter reflects the current state of the art in the treatment of external events in a PRA study. No new methods or improvements are suggested.

11.2 SEISMIC RISK ANALYSIS

11.2.1 INTRODUCTION

This section describes procedures for estimating radiological risks from seismic events. Its objective is to illustrate the application of the general risk-analysis procedure for external events and to highlight the similarities and differences in the analyses of seismic risk and the risk from other external events.

The analysis of seismic risk has been receiving increased attention in recent years. It is recognized that seismic excitation has the potential for simultaneously damaging several redundant components in a nuclear power plant. The basis for the conclusion in the Reactor Safety Study (USNRC, 1975) that earthquakes are not major contributors to risk has been questioned by several experts in the fields of seismology, earthquake engineering, and probabilistic risk analysis. Seismic risk studies performed since the Reactor Safety Study have indicated that the contribution of seismic risk to the overall plant risk may not be insignificant.

Following the general procedure for a probabilistic assessment of external events, the elements of a seismic risk analysis can be identified as analyses of (1) the seismic hazard at the site, (2) the responses of plant systems and structures, (3) component fragilities, (4) plant systems and accident sequences, and (5) consequences. The results of this analysis will be used as input in defining initiating events, in developing system event trees and fault trees, in quantifying the accident sequences, and in modifying the containment event trees and consequence models to reflect the unique features of seismic events. However, in the seismic risk studies done to date, the analysts have kept the seismic risk analysis separate from the analysis of internal events in the plant-system and accident-sequence analysis. The frequencies of the release categories attributable to seismic events are combined with those stemming from internal events, and a consequence analysis is performed to calculate the total plant risk.

The evaluation of seismic risk requires information on the seismologic and geologic characteristics of the region, the capacities of structures and equipment to withstand earthquakes beyond the design bases, and the interactions between the failures of various components and systems of a nuclear power plant. Empirical data available on these aspects are limited; the use of sophisticated analytical tools to calculate the real inelastic capacities of equipment and structures is expensive and has to be done on a selective basis commensurate with the uncertainties of the overall seismic risk problem. Therefore, the procedures described in this chapter call for engineering judgment based on expert opinion to supplement sparse data and limited analyses.

The output of the seismic risk analysis will depend on the stage at which the seismic event analysis is merged with the analyses of other external and internal events. If the seismic analysis is combined with the analyses of other events at the stage of accident-sequence definition and system modeling, the output will be an estimate of the seismic hazard at the site; component fragilities; initiating events; and the information needed to modify system event and fault trees, containment-failure analysis, and the frequencies of accident sequences. If the seismic analysis is combined with the analyses of other events at the consequence-analysis stage, an initial output of the seismic risk analysis could be a curve showing the probability distribution of the annual frequency of a seismically induced core melt (Figure 11-1). If this core-melt frequency is sufficiently high, further computation of the release frequencies is warranted. In that case, the final output of the seismic risk analysis is a family of probability density functions for the annual frequency of various release categories (Figure 11-2). A result of this type forms an input to the consequence analysis described in Chapter 9. Other useful outputs of the seismic risk analysis are failure frequencies for structures, systems, and equipment as well as the accident sequences that dominate the seismic risk. These permit the identification of the major contributors to core-melt and release-category frequencies.

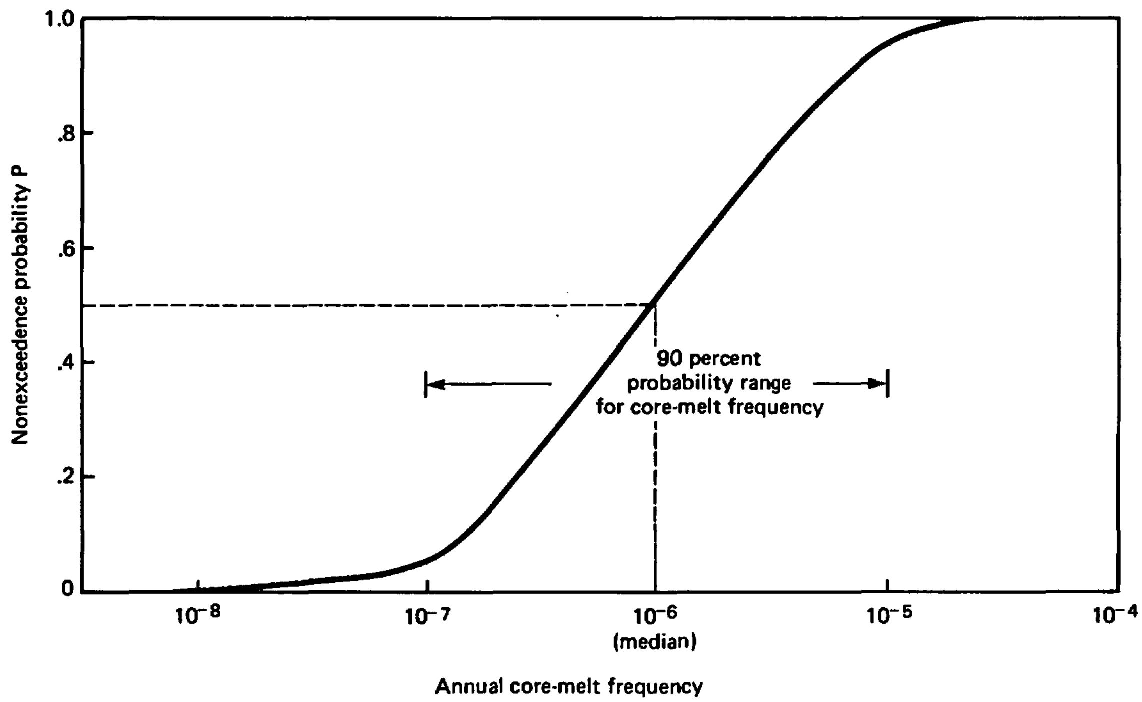

Figure 11-1. Probability distribution for the annual frequency of a seismically induced core melt in a hypothetical nuclear power plant.

11.2.2 HISTORICAL BACKGROUND

Several studies of seismic risk have been performed for nuclear power plants. The Reactor Safety Study (USNRC, 1975) examined a generic safety system consisting of two components in parallel. It used a single fragility curve based on the work of Newmark (1975) along with the seismic hazard estimates developed by Hsieh et al. (1975) for a site in the Eastern United

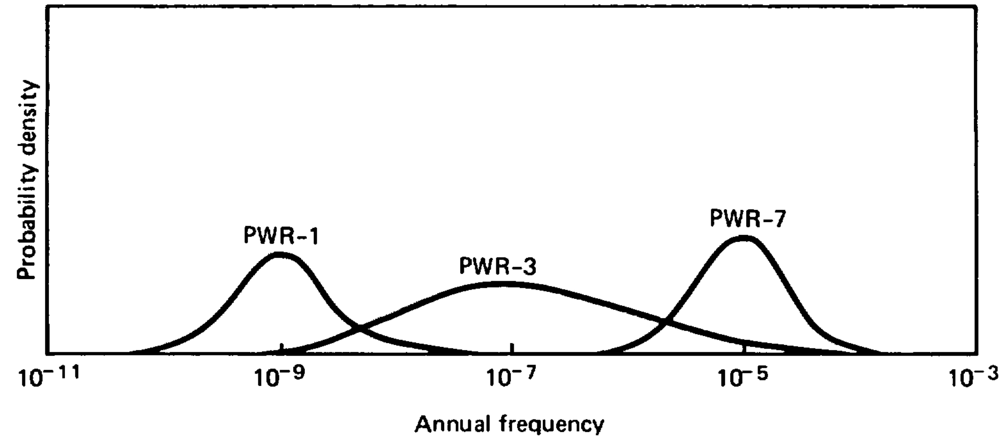

Figure 11-2. Probability density functions for release frequencies from seismic events for three release categories: PWR-1, PWR-3, and PWR-7.

States. The possibility of common-mode failures was admitted by assuming that the system-failure frequency for any given earthquake ground acceleration is not the square of the component-failure frequency (as it would be if the failures of components were independent events) but rather the component-failure frequency raised to the 1.5th power (i.e., the geometric mean of the failure frequencies of two components). The frequency of an earthquake-induced core melt in reactors designed for a safe-shutdown-earthquake (SSE) ground acceleration of 0.20g was estimated to be in the range of 1×10^{-8} to 1×10^{-6} per year. It was concluded that, in comparison with other reactor accidents, seismic events are not significant contributors to risk. This conclusion has since been seriously questioned on the grounds that in the Reactor Safety Study nuclear safety systems were modeled in a simplistic fashion, seismic safety margins were incorrectly calculated, and common-cause failures were given an approximate treatment.

A seismic risk study conducted by Anderson et al. (1975) for Canadian nuclear power plants included a generic safety system consisting of three independent components in parallel. Four fragility curves were developed to reflect differences in design practices and assumptions. The basic component-fragility curve was simply a straight line passing through the failure frequencies of 0.1 and 0.9 at response levels of 1.0 and 4.0 times the design response. Three other variations were included: the "weak" component, the "strong" component, and the "brittle" component. The frequency of system failure under earthquake conditions was calculated. The seismic contribution to risk was found to be important.

In their seismic risk study, Hsieh and Okrent (1976) considered a generic system of 10 components in series. The components were categorized into three groups of two, three, and five, on the basis of differences in the fragility attributed to degradation. Furthermore, two classes of design errors--minimal and maximal--were considered. It was shown that the potential for design error would greatly influence the frequency of system failure.

Described below are the seismic risk studies performed for the Diablo Canyon, Oyster Creek, and Big Rock Point nuclear power plants. Other generic and plant-specific seismic risk studies have been reported (Cornell and Newmark, 1978; Clinch River Breeder Reactor Plant, 1977). More-recent and ongoing studies are discussed in Section 11.2.11.

11.2.2.1 Diablo Canyon Seismic Risk Study

The first plant- and site-specific seismic risk study* was carried out in 1977 by the Pacific Gas & Electric Company for the Diablo Canyon plant (see also Ang and Newmark, 1977). The study was prompted by the realization that the Hosgri Fault could be a major seismic threat to the plant. The following five initiating events were postulated for each earthquake:

1. Transients that require a successful cooldown for the reactor-coolant system (RCS).

2. A small-small LOCA (RCS pipe break of 1/2 to 2 inches).

3. A small LOCA (RCS pipe break of 2 to 6 inches).

4. A large LOCA (RCS pipe break of more than 6 inches).

5. Reactor-vessel rupture.

The frequency of occurrence for each initiating event was determined by considering the fragilities of various components whose failures constitute the initiating event. For each initiating event an event tree was developed and fed into a containment event tree. The frequency of occurrence for each accident sequence was calculated from the failure frequencies of the safety systems. Detailed fault trees were constructed for a number of systems: containment-spray injection, emergency core-cooling injection, containment-spray recirculation and fan coolers, containment heat removal, emergency core-cooling recirculation, auxiliary feedwater, high-pressure injection, and electric power.

The plant structures and piping at Diablo Canyon were analyzed for the design earthquake, an earthquake that is double the design earthquake, and the postulated Hosgri Fault earthquake. The peak ground accelerations for these earthquakes were taken to be 0.2g, 0.4g, and 0.75g, respectively. Assuming the seismic stress to be a linear function of ground acceleration, the ground acceleration at which the piping would reach the code-allowable stress and the ultimate strength of the material were calculated, as was the ground acceleration at which structures would reach the first yield stress

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*Although this study was not part of a probabilistic risk assessment for the plant, it has the historical significance of being the first detailed seismic risk study.

and the ultimate strength. This was done for various locations in the structures and for various segments of the piping. For mechanical and electrical equipment, the seismic qualification acceleration levels were related to the ground-acceleration levels by assuming that the acceleration of the floor on which the equipment was mounted was a linear function of elevation. It may be noted that this assumption is a gross approximation for nuclear plant structures.

Component fragilities were expressed in terms of the effective peak ground acceleration. Two basic forms of fragility curves were used: a ramp-function curve and a step-function curve. For component fragilities described by a ramp function, zero frequency of failure was assumed below a specific acceleration level a, corresponding to the first yield stress in the structure, or the code-allowable stress in the piping, or to the seismic qualification level of the mechanical and electrical equipment; above a specific acceleration level b, corresponding to the ultimate strength of the structure and piping, the failure frequency was unity. For ground accelerations between a and b, the failure frequency was varied linearly between zero and unity. For component fragilities described by a step function, the frequency of failure below the acceleration value a was taken to be zero; above the value of a, it was taken to be unity. The ramp function was generally used for components, such as piping and structures, qualified by stress analysis; the step function was generally used for components, such as mechanical and electrical equipment, qualified by testing.

Variations in component quality and design and fabrication errors were accounted for by including nonzero frequencies for effective peak-ground-acceleration values smaller than a. Typical values of a and b for the containment structure were 0.9g and 1.5g, respectively; for the turbine building, 0.5g and 0.7g. For valves in the containment-spray system, the value of a corresponding to a failure frequency of unity was taken to be 4.2g; for electrical switchgear, a was 0.67g. Similarly, the values of a and b for safety-injection piping were assumed to be 2.0g and 4.3g, respectively.

Consequence calculations were performed for both seismic and internal events, using the release-category frequencies reported in the Reactor Safety Study (USNRC, 1975). In both calculations, consequence models specific to the Diablo Canyon site were used. It was concluded that the seismic contribution to the overall radiological risk from this plant is low. The turbine building, which is not classified as a Seismic Category I structure and houses emergency diesel generators, switchgear, interface heat exchangers, and the fire-protection system, was found to be the source of most of the risk due to earthquakes.

11.2.2.2 Oyster Creek Seismic Risk Analysis

A seismic risk study was conducted as part of an overall safety study for the Oyster Creek plant (Garrick and Kaplan, 1980; Kennedy et al., 1980). Much of the development work for the seismic risk analysis that is discussed in this procedures guide and has been recently applied to several PRA studies was done in the Oyster Creek seismic risk study. A distinguishing feature of this study was the development and use of uncertainty estimates for both the ground-motion occurrence frequencies and the conditional frequencies of failure for structures and components. Since the results of this study have not yet been published, the discussion has to be limited to the overall approach and is covered in Section 11.2.11.

11.2.2.3 Big Rock Point PRA Study

The Big Rock Point PRA study evaluated the contribution of seismic events to the frequency of core melt (Consumers Power Company, 1981). The study consisted of a seismic hazard analysis, a component-fragility evaluation, and an assessment of different seismically induced accident sequences as to their contribution to the core-melt frequency. The seismic hazard analysis was performed with a model of tectonic zones in the Northern and Central United States (Algermissen and Perkins, 1976) and published attenuation and intensity-acceleration relationships. Component fragilities were evaluated by assigning the components to one of three categories: (1) the acceleration levels are sufficiently low that failure is not a problem; (2) the fragility of similar components is known or can be inferred; or (3) the seismic response of the component can be estimated for any earthquake ground acceleration, and the response can be compared against the capacity of the component.

From the evaluation of a limited number of components, the study concluded that the electrical components in the power room, control room, and reactor building are most vulnerable to seismically induced failures at peak ground accelerations of less than 0.20g. At larger earthquakes (i.e., accelerations greater than 0.20g), a collapse of the emergency condenser and core-spray failures induced by circuit-breaker trips or by a collapse of the turbine building are likely. The frequency of a seismically induced core melt was estimated to be 1.2×10^{-7} per year.

The study was somewhat limited in scope. The hazard analysis did not take into account all the uncertainties (e.g., uncertainty in attenuation, source modeling, and upper-bound magnitude), and in the calculation of component fragilities a composite variability (i.e., combination of inherent randomness and uncertainty) was used.

11.2.3 SEISMIC HAZARD ANALYSIS

Seismic hazard is usually expressed by the frequency distribution of the peak value of the ground-motion parameter during a specified interval of time. The major elements of this analysis (see Figure 11-3) are as follows:

1. Identification of the sources of earthquakes, such as faults (F1, F2) and seismotectonic provinces (A1, A2, A3) (sources).

2. Evaluation of the earthquake history of the region to assess the frequencies of occurrence of earthquakes of different magnitudes or epicentral intensities (recurrence).

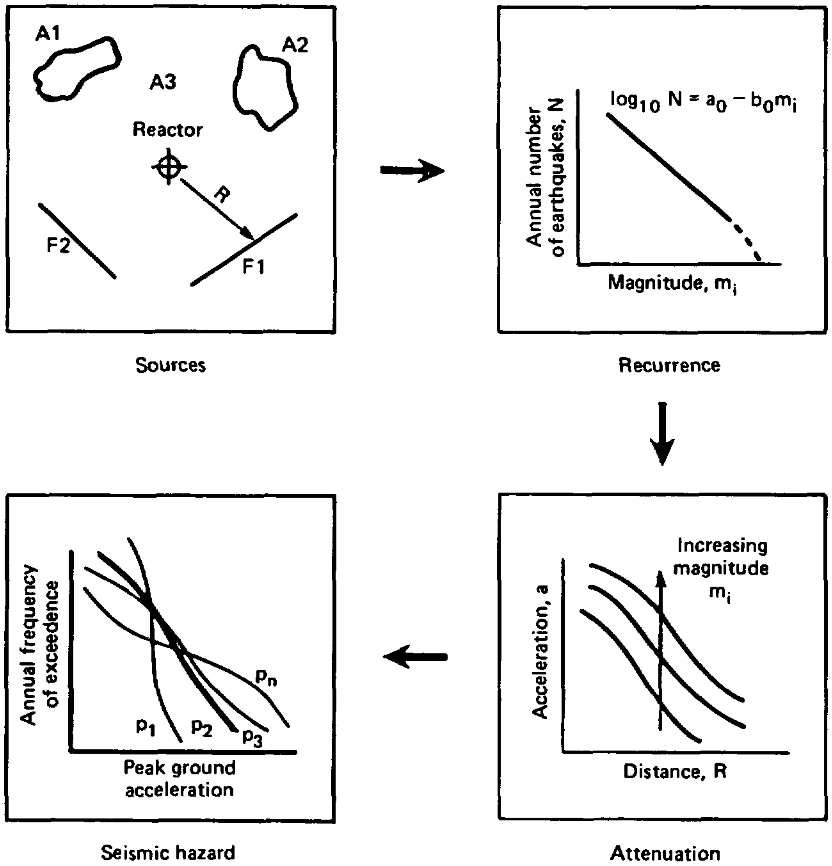


Figure 11-3. Model of seismic hazard analysis.

3. Development of attenuation relationships to estimate the intensity of earthquake-induced ground motion (e.g., peak ground acceleration) at the site (attenuation).

4. 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).

A comprehensive seismic hazard model was first proposed by Cornell (1968). Improvements to this model have been proposed by Cornell and Merz (1975), Shah et al. (1975), Algermissen and Perkins (1976), Der Kiureghian and Ang (1977), McGuire (1976), Mortgat et al. (1977), and Der Kiureghian (1981).* The next section describes the basic model with the modifications necessary for specific sites.

*Although most of these papers are called "seismic risk analysis procedures," they cover only the seismic hazard analysis as defined herein.

11.2.3.1 Seismic Hazard Model

An earthquake is a complex occurrence involving the geologic characteristics of the site, the buildup of crustal strains, slippage along fault planes, rupture surface, focus, and many other variables. The transmission of earthquake disturbance from the source to the site depends on the magnitude, distance, depth of focus, geologic characteristics of the region, type of fault movement, characteristics of earthquake waves (body and surface waves), etc. The ground motion at a site can be described by means of ground-motion parameters, such as the peak acceleration, velocity, displacement, the duration of motion, and a set of response-spectra amplitudes corresponding to the modal frequencies and dampings of the structure. Other possible descriptors are the Modified Mercalli (MM) intensity or energy-related ground-motion-intensity quantities, such as the root mean square of the ground acceleration or velocity.

Obviously, a complete description of the seismic hazard at a specific site should include all of the above variables for earthquake occurrence, wave transmission, and ground-motion input to the building. With its large number of variables, such a model would, in theory, lead to a more accurate estimate of seismic hazard, but it would be prohibitively complex. In practice, the modeling is kept tractable by retaining only a few dominant variables. The choice of a variable depends also on the type, the quality, and the amount of data. For example, the size of an earthquake is generally measured in terms of the Richter magnitude and epicentral intensity. Although other measures would perhaps be more appropriate (e.g., seismic moment and energy release), data on earthquakes have historically been gathered in terms of the Richter magnitude and the epicentral intensity on the Modified Mercalli scale.

The transmission of earthquake disturbance is generally represented in the hazard model by means of an attenuation relationship between a few significant variables (e.g., magnitude, epicentral intensity, distance, region, and soil type). The effect of other variables in the transmission process is accounted for by incorporating the observed scatter about the empirical attenuation relationship. Similarly, the earthquake ground motion may be characterized by a single parameter, such as the peak ground acceleration. The effect of other variables that are necessary for an adequate description of the ground motion may have to be included in the response analysis and fragility evaluation (e.g., using appropriate response spectra and recorded earthquake time histories).

The particular parameter or parameters that are chosen to present the results of a seismic hazard analysis depend on the plant-system and accident-sequence analysis. In the seismic risk studies done to date, the analysts have elected to characterize the seismic hazard in terms of the peak ground acceleration or some related parameter ("sustained" or "effective" peak ground acceleration).

The seismic hazard model is described in detail by Cornell (1968), Merz and Cornell (1973), and Cornell and Merz (1975). The model is used to calculate the annual mean number of events, $\nu_{\ell}(a)$, in which a ground-motion parameter A (e.g., peak ground acceleration) exceeds a value a at the site because of an earthquake on the ℓ^{th} seismic source as expressed below (Kulkarni et al., 1979):

$$\nu_\ell(a) = \sum_i \sum_j \nu_\ell f_\ell(m_i)\ f_\ell(r_j)\ f_{A|m_i,r_j}(a) \qquad (11\text{-}1)$$

where

ν_ℓ = mean annual number of earthquakes on the source; called the activity rate of the source.

$f_\ell(m_i)$ = conditional frequency of the earthquake on the source having a magnitude* equal to m_i. The product $\nu_\ell f_\ell(m_i)$ is obtained from the well-known Gutenberg-Richter (1942) recurrence relationship $\log_{10} N = a_0 - b_0 m_i$, where N is the number of earthquakes per year exceeding magnitude† m_i; a_0 and b_0 are constants that depend on the seismicity of the region.

m_i = Richter or local magnitude of earthquake i.

$f_\ell(r_j)$ = frequency with which the source-to-site distance is r_j, given an earthquake on the ℓ^{th} source.

$f_{A|m_i,r_j}(a)$ = frequency with which the ground-motion parameter A exceeds the value a given an earthquake of magnitude m_i at a distance r_j.

The term $f_\ell(r_j)$ defines the location of the site with respect to the seismic source. Actually, the seismic source is divided into a number of discrete point sources, and the distances r_j are measured from the point sources to the site. The term $f_{A|m_i,r_j}(a)$ is a function of ground-motion attenuation from the source to the site.

By summing the contributions from all seismic sources around the site, the total annual mean number of events, $\nu(a)$, in which A exceeds a at the site can be obtained:

$$\nu(a) = \sum_\ell \nu_\ell(a) \qquad (11\text{-}2)$$

The annual frequency of earthquakes in which the ground-motion parameter A is smaller than a is obtained by assuming that strong motions are Poisson events:

$$H(a) = e^{-\nu(a)} \qquad (11\text{-}3)$$

---

*The model is not limited to situations where magnitude data are available (see Section 11.2.3.2).

†In the original paper, N was the number of earthquakes with magnitude equal to m_i.

It is customary to plot the annual frequency of exceedence, $1 - H(a)$, as a function of the ground-motion parameter.

The annual frequency $h(a)$ of earthquakes in which the value of the ground-motion parameter A is between a and $(a + \Delta a)$ is given by

$$h(a) = H(a + \Delta a) - H(a) \tag{11-4}$$

This hazard estimate $h(a)$ depends on uncertain professional estimates of parameters, such as attenuation laws, upper-bound magnitudes, and the geometry of the source. Such uncertainties are included in the hazard analysis by assigning probabilities to alternative hypotheses about these uncertain parameters. A probability distribution for the frequency of occurrence is thereby developed.

The annual frequencies for exceeding specified values of the ground-motion parameter are displayed (see Figure 11-4) as a family of curves at different nonexceedence-probability levels.

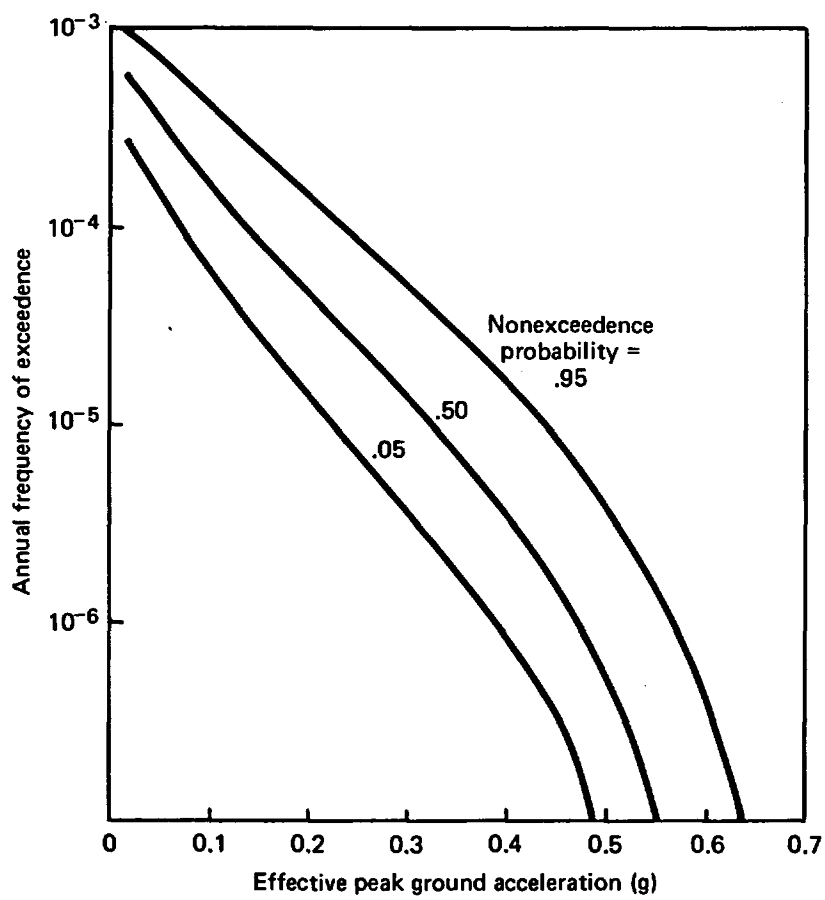


Figure 11-4. Seismic hazard curves for a hypothetical site.

11.2.3.2 Parameters of the Hazard Model

The parameters that characterize a seismic hazard model for a site are the seismic sources, the activity rate ν_ℓ, the relative frequencies of occurrence of different sizes of earthquakes on a source, the attenuation of ground motion from a source to the site, and the upper-bound magnitude or epicentral intensity of a source. In performing the seismic hazard analysis for a site, uncertainties in each of the above model parameters should be consistently treated, as explained in the sections that follow.

Seismic Sources

The seismic hazard model uses three types of seismic source: line source, area source, and point source. In fact, the numerical computation of the seismic hazard at the site is carried out by dividing the line and area sources into a number of discrete point sources. The line source is used to model faults or fault provinces. In a geographical region where recorded earthquakes cannot be related to any well-defined fault system, the concept of seismotectonic provinces is invoked, and the region is represented by a set of area sources, or seismogenic zones. A source, whether a line source or an area source, is distinguished by a uniform seismic activity; that is, the mean rate of earthquake occurrence per unit length or unit area is constant over the entire source.

The seismic sources around a site are identified by studying the epicentral locations of past earthquakes together with the geologic and geomorphologic features of the site region. The inclusion of a source in the hazard model depends on its contribution to the seismic hazard, which, in turn, depends on the activity rate, the upper-bound magnitude or epicentral Modified Mercalli (MM) intensity, and the distance to the site.

In defining the geometric configuration of a fault or a seismotectonic province, there are generally differences in interpretation among seismologists. Recent studies (TERA, 1979; McGuire, 1981) have focused on quantifying such differences in opinion through expressed degrees of belief in alternative geometric configurations for a seismic source.

Activity Rates

The mean annual rate of earthquakes over a source is known as the activity rate (ν_ℓ) of the source. This rate is estimated from the historical seismicity in that source. Historical earthquake data are generally available in magnitude or MM intensity values. It may be noted that there are many definitions of magnitude; examples are the local magnitude M (Gutenberg and Richter, 1954) and the body-wave magnitude m_b (Gutenberg, 1956). There are a number of empirical relationships between these two magnitude scales; for example, Brazee (1976) has proposed the following:

$$M = 1.34m_b - 1.71 \qquad (11\text{-}5)$$

In this chapter the term "magnitude" is used for measurement on both scales. Where a specific scale is intended, the corresponding definition and symbol are used.

Earthquakes of small magnitude (i.e., $m < 3.0$) or epicentral intensity (i.e., MM intensity < V) are not considered in estimating the activity rate because they rarely cause structural damage. For regions of high seismicity, the calculated activity rate for the fault or the seismogenic region can be considered to be stable; even for regions of low seismicity, such as the Midwestern and Eastern United States, it has been shown (McGuire, 1977a; McGuire and Barnhard, 1981) that the historical rates of seismic activity can be considered to be stable for the purposes of seismic hazard analysis. Any significant change in the seismic activity of a seismogenic region requires several centuries; therefore, the historical seismic activity is sufficiently representative for making hazard estimates for the operating life of a nuclear plant.

Relative Frequencies of Different Sizes of Earthquakes

The activity rate for a seismic source is the mean number of earthquake events regardless of their magnitudes (or intensities). The distribution of earthquakes according to their magnitudes is given by the recurrence relationship (Gutenberg and Richter, 1942)

$$\log_{10} N = a_0 - b_0 m_i \tag{11-6}$$

where N is the number of earthquakes with magnitude equal to or greater than m_i for a given source and over a given interval of time. The conditional frequency of earthquakes of magnitude m_i on the source ℓ is expressed as

$$f_\ell(m_i) = \frac{\beta_0 \exp[-\beta_0(m_i - m_0)]}{\{1 - \exp[-\beta_0(m_m - m_0)]\}} \Delta m \quad (m_0 \leq m_i \leq m_m) \tag{11-7}$$

where $\beta_0 = b_0 \ln 10$, m_0 is the magnitude below which earthquakes rarely cause structural damage (e.g., $m_0 = 3$), m_m is the upper-bound magnitude for the source, and Δm is the interval between magnitudes on the magnitude scale. The value of b_0 for a source is derived by plotting the logarithm of the number of recorded earthquakes that exceeded a particular magnitude against the magnitude and by fitting a linear relationship as in Equation 11-6. For regions of low seismicity, historical data may not be sufficient to develop a recurrence relationship like Equation 11-6 for each source zone. A single b_0 value may be appropriate for all source zones in the region. A typical average value of b_0 for the Eastern United States is 0.9 (McGuire, 1981).

Equivalent forms of Equations 11-6 and 11-7 can be developed when the earthquake data are on the MM intensity scale, which has been traditionally used to record most historical earthquakes in the Midwestern and Eastern United States. The recorded epicentral intensity I_0 is then converted to body-wave magnitude m_b. Cornell and Merz (1975) and McGuire (1977a,b) have shown how the seismic hazard analysis is performed with MM intensity data.

Upper-Bound Magnitude or Epicentral Intensity

The recurrence relationship given by Equation 11-6 predicts nonzero frequency of exceedence whatever the magnitude of the earthquake, but most seismologists believe that there is a physical limit on the size of earthquakes that can be generated by a seismic source. However, seismologists do not generally concur on a single value for this upper bound for any given source. For a well-defined fault, the upper-bound magnitude of the earthquake that the fault is capable of generating can be estimated from the rupture length (Tocher, 1958; Bonilla, 1970; Housner, 1970; Wallace, 1970; Mark, 1977).

It should be noted that the data used to derive these empirical relationships exhibit considerable scatter and that the relationship is markedly different for different regions of the world and for different types of fault movement (i.e., normal slip, reverse slip, strike slip, etc.). The hazard analyst should take into account this scatter about the mean relationships.

Wallace (1970) has developed a procedure for estimating earthquake-recurrence intervals (i.e., equivalently, frequencies of earthquakes of various magnitudes on a fault) from geologic evidence of long-term deformation rates for active faults. In a region where the recorded earthquakes cannot be correlated with any known faults, estimating the upper-bound magnitude or the epicentral intensity of a seismic source is best done by soliciting the opinion of experts (TERA, 1979). McGuire (1977a) has suggested that a probability distribution be assigned to different upper-bound magnitudes or epicentral intensities assumed for a source. Again, this probability distribution can be derived by analyzing the opinion of experts.

<u>Attenuation</u>

The decrease in the intensity of ground shaking with distance from the epicentral region is called "attenuation." Many empirical formulas have been proposed (see, for example, Donovan, 1973; Nuttli, 1973; Gupta, 1976; Murphy and O'Brien, 1977; McGuire, 1978; Campbell, 1981; Joyner and Boore, 1981). It has been observed that the attenuation of ground motion varies in different parts of the world. In the Western United States, earthquake motion attenuates more rapidly than it does in the Eastern or Midwestern United States. For the Western United States, where strong-motion instrumental data are available for a number of earthquakes, a typical attenuation formula has the form

$$a = b_1 \exp(b_2 m) \ (R + 25)^{-b_3} \qquad (11\text{-}8)$$

where a is the peak ground acceleration at the site (cm/sec^2), m is the magnitude of the earthquake (Richter or local magnitude), R is the distance to the energy center or the causative fault (km), and b_1, b_2, and b_3 are coefficients that are evaluated by using recorded strong-motion data. For example, Donovan and Borstein (1977) have reported the following values:

$$\begin{aligned} b_1 &= 2{,}154{,}000(R)^{-2.10} \\ b_2 &= 0.046 + 0.445 \log_{10} R \qquad (11\text{-}9) \\ b_3 &= 2.515 - 0.486 \log_{10} R \end{aligned}$$

An attenuation relationship like Equation 11-8 is generally a best fit to the data, which exhibit considerable scatter. This dispersion about the attenuation equation should be properly included in the hazard analysis. For example, Donovan and Borstein (1977) have reported the logarithmic standard deviation in the peak ground acceleration predicted by Equations 11-8 and 11-9 as ranging from 0.3 to 0.5.

Other empirical attenuation formulas developed for the Western United States include those by Schnabel and Seed (1972), McGuire (1974), Trifunac and Brady (1975), Blume (1977), Espinosa (1980), Campbell (1981), and Joyner and Boore (1981). The choice of any formula depends on the site geology, distance to active faults, and the availability of strong-motion data. Whichever formula is used, the analyst should take into account the dispersion in the data about the formula.

For the Eastern and Midwestern United States, where most recorded earthquake data are in MM intensity units, two approaches are available for specifying the attenuation of ground motion. In the first approach, the analyst begins by selecting an intensity-attenuation relationship appropriate for the region. An example of such an intensity attenuation is given by Gupta (1976) for the Central United States:

$$I_s = I_0 + 2.35 - 0.00316R - 1.79 \log_{10} R \quad (R \geq 20 \text{ km}) \qquad (11\text{-}10)$$

where I_s is the site intensity in MM units and I_0 is the epicentral intensity in MM units. The site intensity I_s is converted to the instrumental peak ground acceleration a_{pi} by using a relationship like that of Murphy and O'Brien (1977):

$$\log_{10} a_{pi} = 0.25 I_s + 0.25 \qquad (11\text{-}11)$$

Because of the paucity of strong-motion data for the Eastern and Midwestern United States, it may be necessary to use the intensity-acceleration relationship developed from data for the Western United States, Japan, and Europe, such as Equation 11-11.

Equations relating the site intensity to the epicentral intensity and those relating the peak instrumental ground acceleration to the site intensity are best fits to the earthquake data, which normally exhibit wide scatter. For example, the standard deviation of site intensity about the predicted value of Equation 11-10 is reported as 0.5 MM intensity unit. Murphy and O'Brien (1977) have reported that the logarithmic standard deviation of the estimate associated with Equation 11-11 is 0.36.

In the second approach, the MM epicentral intensity I_0 is converted into the body-wave magnitude m_b by using an appropriate empirical relationship (Aggarwal and Sykes, 1978; Nuttli, 1979). For example, Nuttli (1979) gives

$$m_b = 0.5 I_0 + 1.75 \qquad (11\text{-}12)$$

As before, the dispersion about this type of relationship (±0.5 magnitude unit for Equation 11-12) is to be included in the analysis.

The sustained maximum ground acceleration a_s is obtained for an earthquake of body-wave magnitude m_b at a distance R from the site by using a suitable attenuation relationship. For example, from Nuttli's theory (1979) McGuire (1981) has derived the following attenuation equation for the Central United States:

$$a_s = 0.584 \exp\left[-0.427 \exp(-0.444m_b) + 1.098m_b\right] \quad (R < 10 \text{ km})$$

$$a_s = 3.98R^{-5/6} \exp\left[-0.0427R \exp(-0.444m_b) + 1.098m_b\right] \quad (R \geq 10 \text{ km}) \tag{11-13}$$

McGuire (1981) also suggests a value of 0.6 as the logarithmic standard deviation about the mean value of a_s obtained with Equation 11-13.

The specific attenuation approach and the formulas used in the hazard analysis depend on the site region and the availability of intensity and strong-motion data. Cornell et al. (1979) have discussed the variabilities introduced in the predicted response by different attenuation approaches. Their nominal results for the standard deviation include, however, both natural dispersion and the systematic bias introduced by substitutions like Equation 11-10 into Equation 11-11.

In some seismic risk studies, such as the Zion PRA (Commonwealth Edison Company, 1981), the analysts have required that the seismic hazard be expressed in terms of the effective peak ground acceleration for compatibility with the component fragilities, which are derived in terms of the effective peak ground acceleration. Described below are two candidate procedures for expressing the seismic hazard in terms of the effective peak ground acceleration.

In the first procedure, the instrumental peak ground acceleration is reduced by an appropriate factor to obtain the sustained maximum ground acceleration a_s. The quantity a_s is defined as the level of acceleration corresponding to the third highest peak in the acceleration time history (Nuttli, 1979). Kennedy (1981) has suggested that the effective peak ground acceleration a_D can be taken as 1.25 times the sustained maximum ground acceleration.

The second procedure--proposed by McCann and Shah (1979), Mortgat (1979), and Vanmarcke and Lai (1980)--uses the root-mean-square acceleration A_{rms} as the ground-motion parameter of interest. The effective peak ground acceleration is related to the rms acceleration by

$$a_D = K_p A_{rms} \tag{11-14}$$

where K_p is a function of the acceptable exceedence frequency p for each individual peak of the time history:

$$K_p = \frac{\ell n\ (1/p)}{\sqrt{2}} \tag{11-15}$$

Mortgat (1979) has shown how A_{rms} is used as the ground-motion parameter in a seismic hazard analysis. Whether this procedure can be used depends on the availability of strong-motion data, which are needed to develop attenuation relationships like Equation 11-8 in terms of the rms acceleration.

11.2.3.3 Other Models for Hazard Analysis

The basic hazard model described in Section 11.2.3.1 has been developed over the last 15 years and has had the benefit of improvement through specific applications (Donovan, 1973; Shah et al., 1975). At present, the hazard models actually used in specific site applications differ only in the parameter values. Several investigators have shown how the seismic hazard parameters can be evaluated by using Bayesian techniques to augment sparse data (Benjamin, 1968; Cornell and Vanmarcke, 1969; Esteva, 1969; Mortgat, 1976; Campbell, 1977; Eguchi and Hasselman, 1979; TERA, 1980) and using expert opinion (TERA, 1979).

The basic model assumes that earthquake events follow the Poisson model. The assumption that earthquakes are independent in time, as implied in this model, has been questioned by some seismologists. A Markovian assumption of one-step memory in time may be more valid, but the Poisson assumption for large events does not introduce major errors (Gardner and Knopoff, 1974). Der Kiureghian and Ang (1977) have proposed a line-source model that considers earthquakes originating as slips along geologic faults and assumes that the shortest distance to the slipped area is the important parameter in estimating the ground-motion intensity at the site. The use of this model for sites in regions with seismic activity concentrated on geologic faults may lead to better estimates of the seismic hazard.

11.2.3.4 Sensitivity Studies

Cornell and Vanmarcke (1969), Cornell and Merz (1975), Donovan and Borstein (1977), and McGuire (1977a, 1981) have studied the sensitivity of seismic hazard estimates to variations in the model-parameter values. The results of seismic hazard analysis are found to be especially sensitive to the mean attenuation function and to the dispersion about this function. Hence, the analyst must ascertain that the attenuation relationship is appropriate to the site region and that the scatter in the data is consistently accounted for.

An accepted procedure for including the uncertainties of the parameters in the hazard analysis is to postulate a set of hypotheses. Each hypothesis will consist of, for example, a specified configuration of the seismic sources, a value of the Gutenberg-Richter slope parameter b_0, a value of the upper-bound magnitude or epicentral intensity for each seismic source, and a cutoff value for the effective peak ground acceleration (Kennedy, 1981). A probability value is assigned to each of these hypotheses, based on the analyst's degree of belief and expert opinion. A seismic hazard curve representing the annual frequency of exceeding a specified effective peak ground acceleration is generated for each hypothesis. Some studies (see example, Cornell and Merz, 1975; TERA, 1980) have used such approaches in calculating the mean frequencies of exceeding different acceleration levels. For PRA applications, it is more appropriate to present the seismic hazard at the site as a family of hazard curves with different nonexceedence-probability levels (Figure 11-4).

11.2.3.5 Computer Codes

The following computer codes can be used in analyzing the seismic hazard at the site of a nuclear power plant:

1. SRA (Seismic Risk Analysis), developed by C. A. Cornell at the Massachusetts Institute of Technology, 1975.

2. EQRISK, a Fortran code developed by McGuire (1976). It is available from the National Information Service for Earthquake Engineering, University of California, Berkeley.

3. FRISK, a code for seismic risk analysis using faults as earthquake sources, developed by McGuire (1978).

4. Seismic Risk Analysis Program by C. P. Mortgat, the John A. Blume Earthquake Engineering Center, Stanford University, Stanford, California, 1978.

5. HAZARD, developed at the Lawrence Livermore National Laboratory, Livermore, California, 1980.

11.2.3.6 Case Studies

Cornell and Merz (1975) have described the analysis of the seismic hazard at a site in the Eastern United States. They discuss the process of selecting the parameter values and the sensitivity of hazard estimates to variations in these values. Applications to sites in the Western United States that are exposed to line sources have been described by Shah et al. (1975) and Donovan and Borstein (1977). For recent applications in seismic risk studies for nuclear power plants, the reader is referred to the Zion PRA (Commonwealth Edison Company, 1981) and to reports by TERA (1980) and Chung and Bernreuter (1981).

11.2.4 ANALYSIS OF PLANT-SYSTEM AND STRUCTURE RESPONSES

In order to calculate failure frequencies for structures, equipment, and piping, it is necessary to obtain the seismic responses of these components to various levels of the ground-motion parameter (e.g., peak ground acceleration). The breadth and depth of the response analysis depend on the information existing on analyses performed during the design stage and on the method used to develop component fragilities. For older nuclear plants (those built in the 1960s), seismic design procedures and criteria would have been much different from the current ones, and not all of the seismic design information (e.g., structural and piping analysis models, stress reports, and equipment-qualification reports) may be available. For such plants, it may be necessary to develop structural and piping analysis models and to calculate the responses for critical components. Some amount of iteration and interaction between the structures analyst and the systems analyst would reduce the amount of response analysis by concentrating on the critical structures and components. For a "newer" plant, the analyst can rely on the design-analysis information.

If a detailed response analysis is needed, the following procedure is used. Design drawings and as-built conditions are reviewed to develop structural analysis models for the critical structures. If the analyst thinks that the effects of soil-structure interactions are important, such effects can be incorporated by using a direct method that models the soil and the structure together or by using a substructure approach (Johnson, 1980). Since correlations between the components are needed to estimate the joint failure frequencies for a set of components in an accident sequence, structure and piping-system analyses are performed by means of time-history methods. The variability in the input ground motion is incorporated by simulating a set of time histories consistent with the hazard curve. For example, in the Seismic Safety Margins Research Program (SSMRP--Smith et al., 1981), a hazard model was developed in order to select a set of time histories each of which simulated a particular peak spectral acceleration and spectral shape. Subsystem responses (i.e., piping responses) are determined by using a multisupport time-history analysis. The subsystems may consist of valves, nozzles, and pumps as well as piping nodes.

Although some component failures may involve inelastic responses, most current analyses are limited to linear dynamic analyses of structures and subsystems. Nonlinear response effects are accounted for by estimating the inelastic-energy-absorption capacity for the component under study; the ductility-factor approach of Newmark (1977) is used for this estimate.

The output of the response analysis is the frequency density function of the peak response (e.g., moment, stress, and deformation) of each critical component and the covariances between component responses. Variabilities in the input parameters (e.g., soil shear modulus and damping, and structure and subsystem response frequencies and damping) are incorporated by using an appropriate sampling technique, such as the Latin hypercube (Iman et al., 1980). By separating the variability of each parameter into randomness and uncertainty, and by assigning probabilities to express uncertainties, a probability distribution on the cumulative distribution of component response is derived.

Alternatively, the analyst may decide to estimate the actual component response for a given level of seismic input from the available design-analysis information. A response factor of safety is derived from a linear dynamic analysis of the structure or equipment. In most cases, the response factor of safety can be estimated from the results of response analyses performed for the design-earthquake levels (e.g., operating-basis and safe-shutdown earthquakes) and ground-response spectra. This factor of safety depends on the safety factors involved in the selection of ground-response spectra, the procedure used to include the effects of soil-structure interactions, the selected damping levels, the modeling of structures and piping, and the method of analysis. The safety factors are treated as random variables, and their statistical parameters, such as the median and the logarithmic standard deviation, are estimated by using available data and engineering judgment.

While this approach circumvents the need for a detailed response analysis, it does consider important variables that might affect the responses of structures and equipment. It is expected that the overall variability in response predicted by this approach will be higher than that obtained by a detailed modeling and analysis of structures. Furthermore, any correlation between component responses can be treated only approximately because in the safety-factor evaluation approach equipment, structural elements, piping, cable trays, etc., are examined separately and not as an assemblage.

The responses calculated as described above are related to the responses of structural elements, piping, and any on-line equipment (e.g., valves, nozzles, and pumps). Some equipment that is mounted on the floor or attached to walls may not be included in the dynamic analysis models of main structures and subsystems. The structural analysis will yield the floor spectra (more specifically, a frequency distribution for the floor spectra) for the particular equipment. The actual response of the equipment depends on its dynamic characteristics and how it is qualified.

The frequency density function of the equipment response is derived by modifying the floor-response frequency density function by a multiplicative factor called the equipment-response factor F_{RE}. This factor is a random variable that accounts for variabilities due to (1) the equipment-qualification method, (2) modeling error (i.e., frequency and mode shape), (3) damping, (4) modal response combination, and (5) earthquake-component combination. As before, the equipment-response factor is described by a set of frequency density functions, each with an assigned probability value.

11.2.4.1 Computer Code

The Seismic Safety Margins Research Program developed a computer code called SMACS (Seismic Methodology Analysis Chain with Statistics) for calculating the seismic responses of structures, systems, and components. This code links the seismic input in the form of ensembles of acceleration time histories with the calculations of soil-structure interactions, the responses of major structures, and the responses of subsystems. Since SMACS uses a multisupport approach to perform the time-history response calculations for piping subsystems, the correlations between component responses can be handled explicitly.

11.2.5 FRAGILITY EVALUATION

The fragility of a component is defined as the conditional frequency of its failure given a value of the response parameter, such as stress, moment, and spectral acceleration.

11.2.5.1 Failure Modes

The first step in generating fragility curves like those in Figure 11-5 is to develop a clear definition of what constitutes failure for each component. This definition of failure must be acceptable to both the structural analyst, who generates the fragility curves, and the systems analyst, who must judge the consequences of a component's failure in estimating plant risk. It may be necessary to consider several modes of failure (each with a different consequence), and fragility curves are required for each mode.

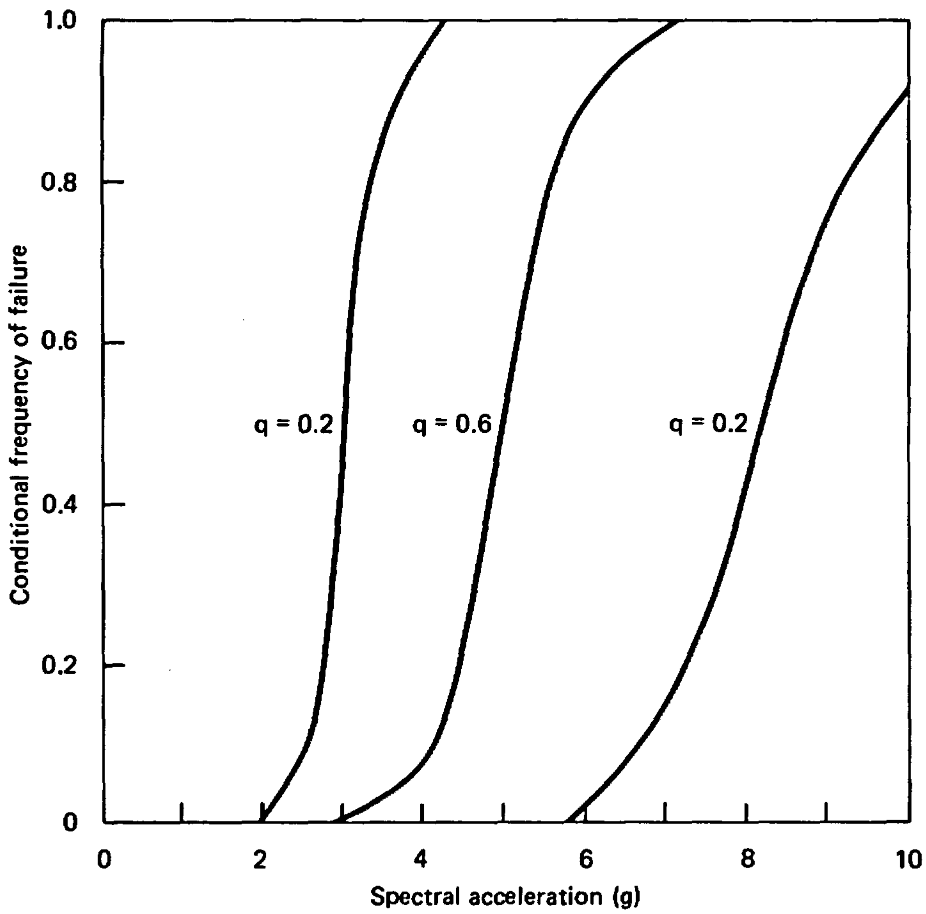


Figure 11-5. Fragility curves for a component.

For example, a motor-operated valve can fail in any of the following ways (Kennedy et al., 1980):

1. Failure of power or controls to the valve (generally related to the seismic capacity of the cable trays, control room, and emergency power). These failure modes are most easily handled as failures of separate systems linked in series to the equipment since they are not related to the specific piece of equipment (i.e., a motor-operated valve) and are common to all active equipment.

2. Failure of the motor.

3. Valve binding due to distortion.

4. Rupture of the pressure boundary.

By reviewing the equipment design, it may be possible to identify the failure mode that is most likely to be caused by the seismic event and to consider only that mode. Otherwise, in developing fragility curves it is necessary to use the premise that the component can fail in any one of all potential failure modes.

The identification of credible modes of failure is based largely on the analyst's experience and judgment. A review of plant design criteria, calculated stress levels in relation to the allowable limits, the results of qualification tests, and failures reported in licensee event reports, including fragility tests, is useful in this task. Piping, electrical, mechanical, or electromechanical equipment vital to the safety of a plant is considered to fail when it cannot perform its designated function. In some PRA studies, relay chatter and trip were considered to temporarily interrupt the component function or considered to be corrected manually. It was therefore assumed that the electrical components would not fail in this mode. However, in judging relay failures as recoverable, the analyst should consider the possibility of spurious alarms and incorrect actions by the operator. For piping, a failure of the support system or a plastic collapse of the pressure boundary is considered to be the dominant failure mode.

Structures can be considered to fail functionally when the inelastic deformations under seismic loads are estimated to be sufficient to potentially interfere with the operability of safety-related equipment attached to the structure or fractured sufficiently for equipment attachments to fail. These failure modes represent a conservative lower bound on the seismic capacity because a nuclear plant structure has a considerably greater margin of safety against collapse. However, a structural collapse should generally be assumed to result in the failure of all safety-related equipment or systems housed inside the portion of the structure that is judged to have failed; that is, the structural failure results in a common-cause failure of multiple safety systems if they are housed in the same structure. The event and fault trees should appropriately reflect this condition.

Consideration should also be given to the potential for soil failure in various modes: liquefaction, toe-bearing pressure failure, slope failures, and base-slab uplift. For buried structures (i.e., piping and tanks), failure due to lateral soil pressures may be important. Both structures and equipment may be damaged through an earthquake-induced impact by another structure or equipment (e.g., a crane). A seismically induced dam failure, if any dams are present nearby, should also be investigated, as it may result in flooding or a loss of cooling source.

11.2.5.2 Calculation of Component Fragilities

Component fragility, which is defined as the conditional frequency of failure for a given value of the response parameter, is calculated by developing the frequency distribution of the seismic capacity of a component and finding the frequency for this capacity being less than the response-parameter value.

Seismic Capacity

The seismic capacity of a structure, piping system, or a piece of equipment is calculated by considering both the strength (i.e., ultimate strength or strength at loss of function) and the capacity for inelastic energy absorption; the latter term refers to the fact that an earthquake is a limited energy source, and many structures and equipment are capable of absorbing, without loss of function, substantial amounts of energy beyond yield.

In estimating the seismic capacity of a component, it is convenient to work in terms of an intermediate random variable called the capacity factor of safety, F_C. This factor is defined as the ratio of the capacity of the component to the magnitude of the fragility (local response) parameter specified for the reference earthquake (e.g., safe-shutdown earthquake), A_{SSE}. The quantity F_C is expressed as

$$F_C = F_S F_\mu \tag{11-16}$$

where F_S is the strength factor--that is, the ratio of ultimate strength (or strength at loss of function) to the stress calculated for A_{SSE}--and F_μ is the inelastic-energy-absorption factor, whose evaluation is discussed later in this section. For active components, the operability limits are likely to govern, and hence the median value of F_μ may be smaller than it is for structures. In calculating the strength factor F_S, the nonseismic portion of the total load (stress) or response acting on the component is subtracted from the strength, as shown below.

$$F_S = \frac{S - P_N}{P_T - P_N} \tag{11-17}$$

where S is the strength of the component, P_N is the normal operating load (stress), and P_T is the total load on the component--that is, the sum of the seismic load (SSE) and the normal operating load. For higher levels of earthquake, other transients (e.g., the discharge of safety relief valves and turbine trip) may have a high likelihood of occurring simultaneously with the earthquake and then the definition of P_N will be extended to include the loads from these transients. In combining the dynamic responses from earthquakes and transients to calculate P_N and P_T, a realistic procedure like the method of the square root of the sum of squares (SRSS) is used.

Sometimes, the strength S of a component is expressed as a function of a number of variables. For example, the shear strength of a concrete shear wall is a function of the compressive strength of the concrete, the yield strength of steel, the steel reinforcement ratio, and the like. The mean and standard deviation of the strength can be calculated by using first-order approximations (Benjamin and Cornell, 1970).

A complete description of the seismic capacity should include the variabilities due to both inherent randomness and uncertainty in the parameters of the model for capacity. Therefore, the appropriate seismic capacity for a specific failure mode is described by a set of frequency density functions (representing inherent randomness), each with an assigned probability q_i (representing the uncertainty in the parameter values). A model that considers both types of variability is explained below (Kennedy et al., 1980). The seismic capacity C is expressed as

$$C = \check{C}\,\varepsilon_{C,R}\varepsilon_{C,U} \tag{11-18}$$

where C is the median capacity, $\varepsilon_{C,R}$ is a random variable reflecting the inherent randomness in the capacity, and $\varepsilon_{C,U}$ is a random variable reflecting the uncertainty in the calculation of C. Both $\varepsilon_{C,R}$ and $\varepsilon_{C,U}$ are assumed to be lognormally distributed with unit median and logarithmic standard deviations $\beta_{C,R}$ and $\beta_{C,U}$, respectively. Recalling that the seismic capacity is expressed as the capacity factor of safety F_C times the reference value of the fragility parameter A_{SSE}, we use Equation 11-16 and the properties of the lognormal distribution to obtain

$$\check{F}_C = \check{F}_S\check{F}_\mu \tag{11-19}$$

$$\beta_{C,R} = \left(\beta_{S,R}^2 + \beta_{\mu,R}^2\right)^{1/2} \tag{11-20}$$

$$\beta_{C,U} = \left(\beta_{S,U}^2 + \beta_{\mu,U}^2\right)^{1/2} \tag{11-21}$$

where $\check{F}_C$, $\check{F}_S$, and $\check{F}_\mu$ are the median values of F_C, F_S and F_μ, respectively, $\beta_{S,R}$ and $\beta_{\mu,R}$ are the logarithmic standard deviations reflecting the inherent randomness in F_S and F_μ, and $\beta_{S,U}$ and $\beta_{\mu,U}$ are the logarithmic standard deviations reflecting the uncertainties in the median value of F_S and F_μ.

The inelastic-energy-absorption factor F_μ is a function of the allowable ductility ratio μ. Newmark and Hall (1978) have suggested that, for frequencies within the amplified acceleration range of the ground-response spectrum, the factor F_μ on capacity can be estimated by

$$F_\mu \approx (2\mu - 1)^{1/2}\varepsilon \tag{11-22}$$

where ε is a random variable to account for the uncertainty associated with the use of Equation 11-22 to define F_μ. The quantity ε is assumed to be lognormally distributed with unit median and logarithmic standard deviation β_ε, which is estimated to be 0.20 (Kennedy et al., 1980). The median ductility ratio $\check{\mu}$ and the logarithmic standard deviation β_d for the component are estimated from a review of the relevant literature and experimental data. Equation 11-22 and the values of $\check{\mu}$, β_d, and β_ε are then used to calculate the median and logarithmic standard deviations of F_μ.

The frequency distribution for seismic capacity is now developed from the values of $\check{C}$, $\beta_{C,R}$, and $\beta_{C,U}$. Figure 11-5 shows an example of fragility curves for a component. Plotted are the conditional frequencies of failure versus the spectral acceleration. The conditional frequency of failure at any given spectral acceleration is the frequency that the seismic capacity is less than or equal to the spectral acceleration. This is calculated from the parameters $\check{C}$, $\beta_{C,R}$, and $\beta_{C,U}$ along with the lognormal-distribution assumption. A set of fragility curves is developed. To each curve, a probability value q_i is assigned to reflect the uncertainty in the seismic capacity parameters.

The seismic capacities of some components may be correlated because the components are supplied by the same manufacturer or mounted in the same way. In such situations, a correlation-coefficient matrix of seismic capacities may be developed in order to calculate the joint failure frequencies of components in an accident sequence (Collins and Hudson, 1981). However, this is not done in routine PRA studies for lack of data on such correlations.

11.2.5.3 An Alternative Formulation of Component Fragility

In this formulation, the fragility of a component is expressed as the conditional frequency of failure for a given peak ground acceleration. Data on seismically induced fragilities are generally not available for equipment and structures. Fragility curves must therefore be developed primarily from analysis supplemented with engineering judgment and limited test data. In view of this, maximum use is made of the response-analysis results obtained at the plant design stage.

The component fragility for a particular failure mode is expressed in terms of the ground-acceleration capacity A. The fragility is therefore the frequency at which the random variable A is less than or equal to a specified value, a. The ground-acceleration capacity is, in turn, modeled as

$$A = \check{A}\, \varepsilon_{A,R}\varepsilon_{A,U} \tag{11-23}$$

where $\check{A}$ is the median ground-acceleration capacity, $\varepsilon_{A,R}$ is a random variable (with unit median) representing the inherent randomness about A, and $\varepsilon_{A,U}$ is a random variable (with unit median) representing the uncertainty in the median value.

It is assumed that both $\varepsilon_{A,R}$ and $\varepsilon_{A,U}$ are lognormally distributed with logarithmic standard deviations of $\beta_{A,R}$ and $\beta_{A,U}$, respectively. The advantages of this formulation are as follows:

1. The entire fragility curve and its uncertainty can be expressed by only three parameters: $\check{A}$, $\beta_{A,R}$, and $\beta_{A,U}$. With the limited data

available on component fragility, it is necessary to estimate only three parameters rather than the entire shape of the fragility curve and its uncertainty.

2. The product form in Equation 11-23 and the lognormal-distribution assumption make the fragility computations mathematically tractable.

The lognormal distribution can be justified as reasonable (Kennedy et al., 1980) because it can adequately represent the statistical variation of many material properties and seismic response variables, provided one is not primarily concerned with the extreme tails of the distribution. In addition, the central limit theorem states that a distribution of a random variable consisting of products and quotients of several variables tends to be lognormal even if the distributions of the individual variables are not lognormal. For estimating failure frequencies on the order of 1 percent or higher, this distribution is considered to be reasonably accurate. However, if the lognormal distribution is used for estimating the very low failure frequencies associated with the tails of the distribution, the approach is considered to be conservative: the low-frequency (probability) tails of the lognormal distribution generally extend farther from the median than the actual structural resistance or response data might extend since the data on material strength or response show cutoff limits beyond which there is essentially zero frequency of occurrence.

Using Equation 11-23 and the lognormal-distribution assumption, the fragility (i.e., the frequency of failure, f') at any nonexceedence probability level Q can be derived as

$$f' = \Phi\left[\frac{\ell n(a/\check{A}) + \beta_{A,U}\,\Phi^{-1}(Q)}{\beta_{A,R}}\right] \tag{11-24}$$

where $Q = P(f < f'|a)$ is the probability that the conditional frequency f is less than f' for a peak ground acceleration a. The quantity $\Phi(\cdot)$ is the standard Gaussian cumulative distribution function, and $\Phi^{-1}(\cdot)$ is its inverse. For displaying the fragility curves, the nonexceedence-probability level Q is used. Subsequent computations are made easier by discretizing the probability distribution of frequency, Q, into values q_i associated with different values of the failure frequency f. A family of fragility curves, each with an associated probability q_i, is developed.

For example, let the fragility parameters of a component be $\check{A} = 0.73g$, $\beta_{A,R} = 0.30$, and $\beta_{A,U} = 0.28$; then, from Equation 11-24, the conditional failure frequency that is not exceeded with a 95-percent probability for a ground acceleration of 0.5g is found to be 0.60. At a 90-percent nonexceedence probability, the conditional failure frequency for a ground acceleration of 0.5g is approximately 0.52.

In estimating the fragility parameters, it is convenient, as before, to work in terms of an intermediate random variable known as the factor of safety F. This is defined as the ratio of the ground-acceleration capacity A to the safe-shutdown-earthquake (SSE) acceleration used in plant design.

The median factor of safety $\check{F}$ can be directly related to the median ground-acceleration capacity $\check{A}$ as

$$\check{F} = \frac{\check{A}}{SSE} \tag{11-25}$$

The logarithmic standard deviations $\beta_{F,R}$ and $\beta_{F,U}$ for F are identical with those for the ground-acceleration capacity.

For structures, the factor of safety is modeled as the product of three random variables:

$$F = F_S F_\mu F_{RS} \tag{11-26}$$

where F_S is the strength factor, F_μ is the inelastic-energy-absorption factor, and F_{RS} is the structure-response conservatism factor. The strength factor represents the ratio of the ultimate strength (or strength at loss of function) to the computed response level. The structure-response factor recognizes the variability in (1) ground motion and the associated ground-response spectra for a given peak acceleration, (2) soil-structure interactions, (3) energy dissipation (damping), (4) structural modeling, (5) the method of analysis, (6) the combination of modes or time-history analysis results, and (7) the combination of earthquake components.

For equipment and other components, the factor of safety is modeled as

$$F = F_S F_\mu F_{RE} F_{RS} \tag{11-27}$$

The factors F_S and F_μ together represent the capacity factor of safety for the equipment relative to the floor acceleration used for the equipment design. The factor F_{RE} represents the safety inherent in the computation of equipment response, and F_{RS} is the factor of safety in the structure-response analysis that resulted in the floor spectra for equipment design.

Median, $\check{F}_{(.)}$, and variability, $\beta_{(.)R}$ and $\beta_{(.)U}$, estimates are made for each of the parameters affecting the capacity and response factors of safety. Using the properties of the lognormal distribution, these median and variability estimates are then combined to obtain the overall median factor of safety $\check{F}$ and the variability, $\beta_{F,R}$ and $\beta_{F,U}$, estimates required to define the fragility curve for a structure or a component. It should be noted that $\beta_{(.)R}$ represents the sources of dispersion in the factor of safety that cannot be reduced by a more detailed evaluation or by gathering more data. These sources include but are not limited to (1) the variability in an earthquake time history and thus in structure response when the earthquake is defined only in terms of the peak ground acceleration and (2) the variability in material properties (structure, soil, and equipment), such as strength, inelastic energy absorption, and damping.

The dispersion represented by $\beta_{(.)U}$ arises from (1) the variability due to an insufficient understanding of structural material properties, (2) errors in the calculated response that result from using approximate modeling for the structure and inaccuracies in mass and stiffness representations, and (3) the use of engineering judgment in lieu of complete plant-specific data on the fragility levels of equipment and on responses.

Examples showing how fragility curves are developed for structures and equipment can be found in a paper by Kennedy et al. (1980) and in the Zion PRA study (Commonwealth Edison Company, 1981).

Although fragility curves are developed independently for different components, some dependence is likely to exist between the earthquake-induced failures of components, particularly for structural elements and equipment located inside the structures. This is so even though the failure events are conditional on the peak ground acceleration. If the components are on the same elevation of the structure, are made by the same manufacturer, and are oriented in the same direction, then perfect dependence between them may be assumed; if none of these conditions are met, then perfect independence may be assumed. However, if one or two of these three conditions exist, the analyst, lacking other means to establish the extent of dependence, may assume independence or dependence, whichever gives conservative results. In some instances, these assumptions may result in large dispersions in the plant-risk estimates, which would require further in-depth studies and modeling (Smith et al., 1981).

The information required for estimating component fragilities includes as-built layouts and dimensions of members; material-strength test data for concrete, reinforcing steel, and structural steel; plant design bases; design calculations; stress reports; and qualification procedures and test reports for equipment. A more detailed list of the information needed for developing component fragilities is given in Section 11.2.12. For mechanical and electrical equipment, fragility curves are based on design-analysis data, shock-test results (i.e., by the U.S. Army Corps of Engineers), and expert opinion (Vagliente, 1981).

11.2.5.4 Selection of Components for Response and Fragility Evaluation

The selection of components or systems for fragility evaluation is an iterative process and calls for a close interaction between the systems analyst and the structural analyst. The systems analyst provides a list of structures, systems, and components whose failure may lead to radiological consequences. He may be guided in this selection by the accident sequences identified for the internal events and by other published seismic risk studies (e.g., Diablo Canyon, Zion, SSMRP, and Big Rock Point). For a typical nuclear plant, this list may include from about 100 up to 300 components, depending on the detail employed in the plant-system and sequence analysis. In some studies (Smith et al., 1981; Commonwealth Edison Company, 1981) it has been necessary to group the equipment into generic categories. The structural analyst develops the response frequency distributions and fragility curves for significant failure modes for each of these structures, systems, and equipment. After reviewing plant design criteria, stress reports, and equipment-qualification reports and performing a walk-through inspection of the plant, he may add to, or delete from, the list of components.

In the process of developing fragility curves, the structural analyst may identify components that have low fragilities even at extremely high ground accelerations (e.g., six to eight times the SSE acceleration); for these components, further refinements in the form of detailed response analyses and data collection may not significantly influence the calculations of seismic risk. However, such refinements may be necessary for those components that are calculated to have significant frequencies of failure at ground accelerations of 1.5 to 3 times the SSE acceleration. The need for a detailed fragility evaluation finally rests on the significance of the components in an accident sequence and the contribution of that sequence to the plant seismic risk. By the procedure described here, the analysts in some PRA studies have reduced the number of major components in the plant logic for seismic events to as few as 10.

11.2.6 PLANT-SYSTEM AND ACCIDENT-SEQUENCE ANALYSIS

The frequencies of core melt and radionuclide releases to the environment are calculated by using the plant logic combined with component fragilities and seismic hazard estimates. Event and fault trees are constructed to identify the accident sequences that may lead to core melt and a release of radionuclides.

In the performance of plant-system and accident-sequence analysis, the major differences between seismic and internal events are in--

1. The identification of initiating events.

2. The increased likelihood of multiple failures of safety systems requiring a more detailed event-tree development.

3. More pronounced dependences between component failures as a result of correlation between component responses and between capacities.

11.2.6.1 <u>Initiating Events</u>

The first step in the plant-system and accident-sequence analysis is the identification of earthquake-induced initiating events. To this end, the initiating events postulated for the internal events (see Chapter 3) are reviewed to identify those that are relevant to the seismic risk study. For example, the following initiating events are used in the Seismic Safety Margins Research Program for a PWR plant (Smith et al., 1981):

1. Reactor-vessel rupture.

2. Large LOCA (rupture of a pipe larger than 6 inches in diameter or the equivalent).

3. Medium LOCA (rupture of a pipe 3 to 6 inches in diameter or the equivalent).

4. Small LOCA (rupture of a pipe 1.5 to 3.0 inches in diameter or the equivalent).

5. "Small-small" LOCA (rupture of a pipe 0.5 to 1.5 inches in diameter or the equivalent).

6. Transient with the power-conversion system (PCS) operable.

7. Transient with the PCS inoperable.

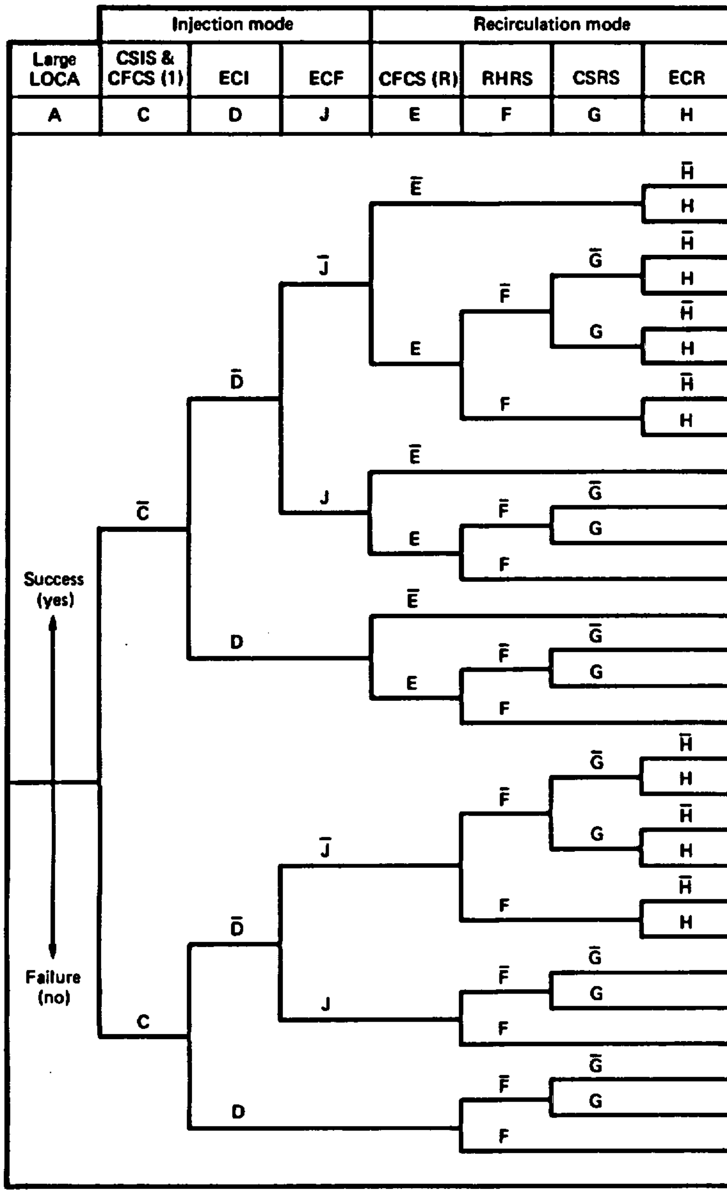
The conditional frequency of each initiating event is calculated for different levels of earthquakes. These levels are established on the basis of ranges in the peak ground acceleration (e.g., 0 to 0.15g, 0.15g to 0.30g, 0.30g to 0.45g). Given an earthquake in the acceleration range 0.15g to 0.30g, the analyst calculates a joint frequency distribution of the responses at different critical locations in the piping whose failure would lead to an initiating event (e.g., small LOCA). The convolution of the frequency distribution with the fragilities yields the conditional frequency of the initiating event.

In identifying the initiating events caused by earthquakes, the analyst may have to look beyond the single initiating events studied for internal events. For large earthquakes, multiple initiating events may occur at the same time, with markedly different effects on the engineered safety features (ESFs). For example, when a small LOCA that occurs without a complete blowdown is coupled with a loss of main feedwater, the effects on the capability of certain ESFs may be different from those of a loss of main feedwater or a small LOCA occurring as different events separated in time (Collins and Hudson, 1979). Another example would be the simultaneous break of a main-steam line and a LOCA. The inclusion of these and other initiating events in event trees depends on their conditional frequencies. Once the dominant initiating events have been identified, they can be arranged into a hierarchical order and grouped as described in Chapter 3.

11.2.6.2 Event Trees

The development of event trees for earthquake-induced initiating events follows essentially the methods described in Chapter 3. From these event trees, core-melt accident sequences are identified. Each of these core-melt sequences is followed by a containment sequence that establishes the release sequence. Figure 11-6 shows an event tree for a large LOCA in a PWR (Smith et al., 1981); it contains 23 core-melt sequences, and each sequence can lead to a release through the potential containment-failure modes designated α, β, γ, δ, and ε.

In developing the event trees, the analyst should be aware of the increased likelihood of multiple failures of safety systems under earthquake conditions. The systems that are essentially guaranteed to be available for mitigating accidents initiated by internal events may fail under earthquake conditions. For example, in a risk study for a PWR, the analyst may judge that, since three of five containment fan coolers will provide sufficient cooling for the containment in the event of core melt, the fan-cooler system is always available for mitigation. But a large earthquake may damage all five fan coolers, and this possibility has to be reflected in the seismic event trees. Also, if the failure of an ESF can lead to an initiating event, that ESF cannot appear on the corresponding event tree as available



Sequence	Seq.	Containment failure mode	Reactor Safety Study equivalent
A $\bar{C}$ $\bar{D}$ $\bar{J}$ $\bar{E}$ $\bar{H}$*	1		
A $\bar{C}$ $\bar{D}$ $\bar{J}$ $\bar{E}$ H	2	$\alpha\beta\epsilon$	AH
A $\bar{C}$ $\bar{D}$ $\bar{J}$ E $\bar{F}$ $\bar{G}$ $\bar{H}$*	3		
A $\bar{C}$ $\bar{D}$ $\bar{J}$ E $\bar{F}$ $\bar{G}$ H	4	$\alpha\beta\epsilon$	AH
A $\bar{C}$ $\bar{D}$ $\bar{J}$ E $\bar{F}$ G $\bar{H}$*	5		
A $\bar{C}$ $\bar{D}$ $\bar{J}$ E $\bar{F}$ G H	6	$\alpha\beta\gamma\delta\epsilon$	AHF
A $\bar{C}$ $\bar{D}$ $\bar{J}$ E F $\bar{H}$	7	$\alpha\delta$	AG
A $\bar{C}$ $\bar{D}$ $\bar{J}$ E F H	8	$\alpha\beta\delta\epsilon$	AHG
A $\bar{C}$ $\bar{D}$ J $\bar{E}$	9	$\alpha\beta\epsilon$	AE
A $\bar{C}$ $\bar{D}$ J E $\bar{F}$ $\bar{G}$	10	$\alpha\beta\epsilon$	AE
A $\bar{C}$ $\bar{D}$ J E $\bar{F}$ G	11	$\alpha\beta\delta\epsilon$	AEF
A $\bar{C}$ $\bar{D}$ J E F	12	$\alpha\beta\delta\epsilon$	AEG
A $\bar{C}$ D $\bar{E}$	13	$\alpha\beta\epsilon$	AD
A $\bar{C}$ D E $\bar{F}$ $\bar{G}$	14	$\alpha\beta\epsilon$	AD
A $\bar{C}$ D E $\bar{F}$ G	15	$\alpha\beta\epsilon$	AF
A $\bar{C}$ D E F	16	$\alpha\beta\delta\epsilon$	ADG
A C $\bar{D}$ $\bar{J}$ $\bar{F}$ $\bar{G}$ $\bar{H}$*	17		
A C $\bar{D}$ $\bar{J}$ $\bar{F}$ $\bar{G}$ H	18	$\alpha\beta\epsilon$	ACH
A C $\bar{D}$ $\bar{J}$ $\bar{F}$ G $\bar{H}$*	19		
A C $\bar{D}$ $\bar{J}$ $\bar{F}$ G H	20	$\alpha\beta\epsilon$	ACH
A C $\bar{D}$ $\bar{J}$ F $\bar{H}$	21	$\alpha\delta$	ACG
A C $\bar{D}$ $\bar{J}$ F H	22	$\alpha\beta\delta\epsilon$	ACHG
A C $\bar{D}$ J $\bar{F}$ $\bar{G}$	23	$\alpha\beta\epsilon$	ACE
A C $\bar{D}$ J $\bar{F}$ G	24	$\alpha\beta\epsilon$	ACE
A C $\bar{D}$ J F	25	$\alpha\beta\delta\epsilon$	ACEG
A C D $\bar{F}$ $\bar{G}$	26	$\alpha\beta\epsilon$	ACD
A C D $\bar{F}$ G	27	$\alpha\beta\epsilon$	ACD
A C D F	28	$\alpha\beta\epsilon\delta$	ACDG

Acronyms: CSIS, containment spray injection system; CFCS, containment fan cooler system (I = injection; R = recirculation); ECI, emergency coolant injection; ECF, emergency core-cooling function; RHRS, residual heat removal system; CSRS, containment spray recirculation system; ECR, emergency core-cooling recirculation.

Key to containment-failure modes:

α = CRSVE = containment rupture due to reactor vessel steam explosion

β = CL = containment leakage

γ = CR-B = containment rupture due to hydrogen burning

δ = CR-OP = containment rupture due to overpressurization

ϵ = CR-MT = containment rupture due to meltthrough

Figure 11-6. Event tree for a large LOCA in a PWR plant. An asterisk indicates no core melt.

to mitigate the consequences of the initiating event. An example would be the failure of the component-cooling-water system, which supplies cooling water to the seals of the reactor-coolant pumps. The failure of these seals may lead to a small LOCA. However, a failure of the component-cooling-water system causes a failure of the emergency core-cooling system (ECCS) because of the functional dependences between these two systems. The result is a LOCA with no ECCS available to mitigate the initiating event. Thus, the ESF cannot appear on the event tree as being available (Collins and Hudson, 1979).

In some risk studies, the analysts have preferred to develop a plant-level seismic fault tree to identify different core-melt and release sequences (Zion Probabilistic Safety Study--see Section 11.2.11.1).

11.2.6.3 Fault Trees

The major difference between earthquakes and internal events lies in the quantification of the fault trees. The frequencies of failure estimated for each component in a seismic fault tree are comprised of both the seismic fragility-related failure frequency and the random unavailability of the component. Each fault tree is expressed as a union of minimal cut sets. Calculation of the failure frequency for the system should consider the joint frequency distribution of the seismic responses and capacities of all components in the minimal cut set.

11.2.7 CONSEQUENCE ANALYSIS

A consequence analysis for seismic events differs from that for internal events in that some parameters of the consequence-analysis model may be influenced by the earthquake. For example, a large earthquake may disrupt the communications network and damage the roads that would be used for evacuation. It may also invalidate the consequence-modeling assumption that people will seek shelter in nearby buildings from external irradiation by gamma rays. In the presence of multiple hazards (i.e., earthquake and reactor accident), people may react differently than they would if faced with a reactor accident alone. For such reasons, the spatial distribution of population exposed to radiation effects in a seismically induced reactor accident is expected to be different from that for internal events. Similarly, there are some differences in the expected property damage for the two events. The consequence analyst should recognize these differences in building the consequence-analysis model for seismic events. The final output of the consequence analysis is a family of risk curves (Figure 11-7).

In recent seismic risk studies that included a consequence analysis, the consequence modeling has not been consistently different for seismic and internal events. This modeling approach was justified on the grounds that the large uncertainties assigned to the parameters of the consequence model are assumed to cover the differences between internal and seismic events.

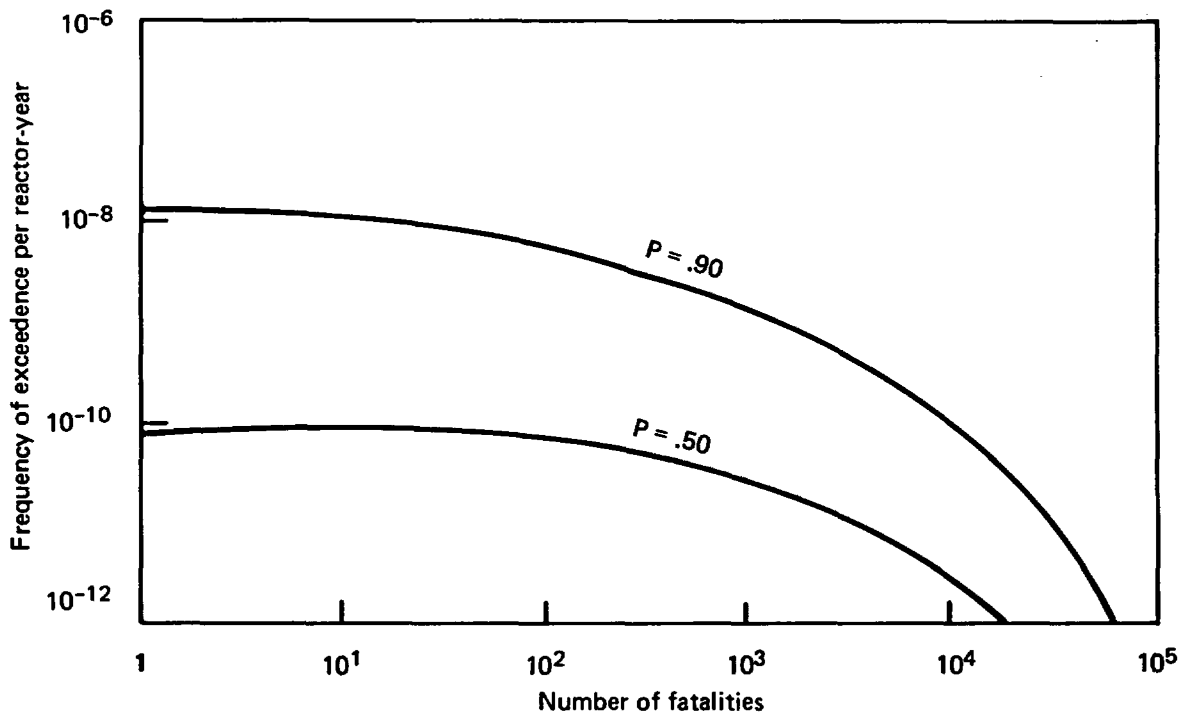


Figure 11-7. Seismic risk curves.

11.2.8 TREATMENT OF UNCERTAINTY

11.2.8.1 Sources of Uncertainty

Uncertainties in the analysis of seismic risk arise mainly from lack of data and analytical models. The sources of uncertainties are further grouped according to the stage of the analysis.

Seismic Hazard Analysis

The parameters of a model for seismic hazard analysis are separated into those that represent the inherent randomness of the seismic hazard and those whose values are uncertain. This distinction is also made for the sake of convenience in presenting the results of the hazard analysis and is based on the analyst's judgment. The first category contains the parameters whose values are estimated from empirical data. Examples are the activity rates of seismic sources, the mean attenuation relationship along with the dispersion, and relationships between intensity and acceleration, magnitude and rupture length, as well as intensity and magnitude with their respective dispersions in the data. The parameters whose values are uncertain include the geometric configuration of a seismic source, the value of the Gutenberg-Richter slope parameter b_0 for the source, the upper-bound magnitude or epicentral intensity for the source, and a cutoff value for the effective peak ground acceleration. The uncertainty in these parameters depends heavily on the specific site region. As such, no ordering of these parameters according to their contributions to the total uncertainty can be made.

Analysis of System and Structure Responses

In this portion of the seismic risk analysis, there are uncertainties in the seismic input and in the description of the dynamic behavior of the soil, structures, and subsystems (Johnson et al., 1981). In the definition of seismic input, uncertainties arise from using a limited number of parameters (e.g., peak ground acceleration) to describe the complex process of earthquake ground motion. Uncertainties are present in the definition of the ground-response spectra for a given peak free-field ground acceleration. Uncertainties in soil-structure interactions come from the idealization of the soil-structure system, the estimates of in-situ soil properties, and the details of the solution process. Uncertainties in structure responses result from variations in material properties (which affect structural frequencies, damping, etc.), variations in the details of construction, and the assumptions made in idealizing the structure. Uncertainties in piping-system responses arise from variations in material and geometric properties as well as modeling assumptions (e.g., linearity, gapless rigid supports, and the assumption that piping analysis can be decoupled from the structural analysis).

Component-Fragility Evaluation

The uncertainties in the component-fragility evaluation arise from an insufficient knowledge of material properties (e.g., strength and capacity for inelastic energy absorption), the definition of failure modes, the use of engineering judgment and generic data in lieu of complete plant-specific data, the lack of fragility test data for equipment, and the lack of data on the correlation between component capacities.

Plant-System and Accident-Sequence Analysis

The sources of uncertainty in the plant-system and accident-sequence analysis are the incomplete identification of all potential accident sequences, the lack of data on the physical interactions between components, and the modeling of dependences between component failures.

Consequence Analysis

In consequence analysis, the source of uncertainty specific to the seismic risk analysis is the lack of models for predicting the effects of large earthquakes on the parameters of consequence-analysis models (e.g., evacuation time, population distribution, and public response). Even if such models are available in other fields (e.g., lifeline earthquake engineering), the nuclear plant PRAs that have been conducted to date did not make explicit use of them.

11.2.8.2 Procedures for Uncertainty Analysis

At present, the quantification of uncertainties arising from different sources has to be done by a combination of limited analysis, sparse empirical data, and engineering judgment based on expert opinion. References that demonstrate how this quantification is accomplished include the Zion

Probabilistic Safety Study (Commonwealth Edison Company, 1981) and the reports of the Seismic Safety Margins Research Program (Smith et al., 1981; Johnson et al., 1981).

In the seismic risk studies performed so far, two essentially similar approaches to the propagation of uncertainties have been followed. One approach is to perform the risk analysis in two stages. In the first stage, the risk assessment is done with the best estimates of the parameters (about which there is uncertainty) of the seismic hazard analysis, response analysis, fragility evaluation, plant-system and accident-sequence analysis, and consequence analysis. This best-estimate analysis also assists in identifying the dominant accident sequences. Sensitivity studies with different parameter values are used to identify the significant parameters. In the second stage, a risk assessment of dominant sequences is repeated many times, each time with a different set of values for the significant parameters. These sets are sampled from the probability distributions of the parameters. By performing this two-stage analysis a sufficient number of times, one obtains the probability distributions for core-melt frequency, the frequency of each release category, and the frequency of exceeding various damage indices.

The other approach to the propagation of uncertainties is to assign discrete probability distributions (DPDs) to the parameters and then to use DPD arithmetic along with the Boolean expressions for the dominant accident sequences derived from fault trees to obtain a family of plant-level fragility curves for core melt and for each release category. Integration over the hazard-curve family yields probability distributions for core-melt frequency and the frequency of each release category. The same approach is extended into the consequence analysis to obtain a family of risk curves.

In performing the uncertainty propagation, it is important to maintain the "correlation" along the acceleration axis. A consistent way of doing this is as follows: The seismic hazard is represented by a set of hazard curves; associated with each curve is a probability (or "weight") p_i that reflects the analyst's degree of belief in the particular hypothesis. The entire seismic risk analysis is to be made conditional on one seismic hazard curve and is repeated for other curves. Similarly, component fragility is expressed by a set of fragility curves, each with an associated probability q_i. The integration over the entire range of acceleration values is performed for one fragility curve at a time. This ensures that the conditional frequency of failure does not decrease when the acceleration increases.

11.2.8.3 Available Information on Uncertainty Evaluation

In the Zion study (Commonwealth Edison Company, 1981), uncertainties in the parameters of the risk model were propagated throughout to obtain the probability distributions for core-melt frequency, the frequencies of various release categories, and the frequencies of exceeding various damage indices (early fatalities, thyroid cancers, etc.). The 10- to 90-percent probability range for the annual core-melt frequency is approximately 1×10^{-7} to 1×10^{-5}; for the annual occurrence frequency of release category 2R (i.e., delayed overpressure failure of containment without sprays), it is 2×10^{-7} to 2×10^{-5}.

It is observed that the uncertainty in the seismic hazard drives the total uncertainty in the seismic risk. The uncertainty in the seismic hazard stems mainly from differences in opinion between seismologists as to the upper-bound earthquake magnitude, the seismogenic source boundary, and the value of the Richter-Gutenberg slope (b). The many alternative hypotheses considered and the subjective probabilities assigned to them increase the uncertainty in the seismic hazard and thereby the uncertainty in the seismic risk. In contrast to the other portions of the seismic risk analysis (e.g., seismic fragility evaluation), the variables that introduce uncertainty in seismic hazard estimates are known, but the values of these variables are uncertain.

Some preliminary studies on uncertainties were done in phase I of the Seismic Safety Margins Research Program. A detailed investigation of the uncertainties in seismic risk analysis coupled with sensitivity studies is being undertaken in phase II.

11.2.9 FINAL RESULTS OF A SEISMIC RISK ANALYSIS

The final results of a seismic risk analysis may take several forms, depending on the site and the specific objectives of the PRA. If the site is in a low-seismicity region, it may be sufficient to calculate the frequency of an earthquake-induced core melt for comparing with the frequency of core melt from internal events and other external events. The final results of a seismic risk analysis can then be presented as shown in Figure 11-1 for a hypothetical plant. In this figure, the median annual frequency of an earthquake-induced core melt is 1×10^{-6}. The 5- to 95-percent probability ("confidence") interval for the annual core-melt frequency is 1×10^{-7} to 1×10^{-5}.

If the annual frequency of an earthquake-induced core melt is significant in comparison with other internal and external events, the objective of the PRA study may be extended to estimate the radiological risk from seismic events. An intermediate result of the seismic risk analysis can then be presented as a probability density function for the frequency of each release category (Figure 11-2). Again, these frequencies can be compared with the release frequencies from other internal and external events to judge the seismic event contribution to plant risk. Finally, the frequencies of exceeding different damage levels (e.g., number of early fatalities) can be presented at different probability levels (Figure 11-7).

If the contribution from the seismic risk dominates the total plant risk and the latter is considered to be high, it is necessary to review the components and systems in the release sequences that had high frequencies of occurrence. It may be necessary to reevaluate the seismic hazard curve, component responses, and component fragilities by using more-detailed models and by gathering additional data. Such a review may also uncover weak links in the methods of the seismic design (Smith et al., 1980).

In recent years, many PRA studies have been initiated for operating nuclear power plants as well as those under construction. Since these studies are still in their preliminary stages and not all results have been published, the relative contribution of seismic events to plant risk and the relative significance of each component and system in the seismic safety of the plant are not yet established. Specific guidelines for the appropriate level of effort in the seismic risk analysis cannot therefore be given at this time.

11.2.10 REQUIREMENTS FOR SEISMIC RISK ANALYSES

A review of the seismic risk studies performed so far indicates that they vary in completeness and sophistication, mostly because of differences in their objectives and scopes. As already mentioned, the analysis of seismic risk has not reached a stage where definitive guidelines as to the level of detail can be provided. However, it may be advisable to outline a few general requirements in order to promote uniformity and consistency in PRA studies. For a seismic risk study to be acceptable, it is not sufficient to just meet these requirements: the study must be conducted by a qualified team and be thoroughly reviewed by peers.

The requirements are as follows:

1. The analysis should consider all the plant systems and components whose failures might contribute importantly to the frequency of release.

2. The analysis should include all significant variables contributing to the seismic hazard, to the responses of structures and equipment, to the fragilities of components and systems, to the release frequencies, and to the plant risk.

3. At each stage of the analysis, the analyst should not only make a best estimate of each variable but also record the uncertainty in the estimate. The seismic hazard analysis should reflect variations in professional opinions regarding the values of different variables (e.g., upper-bound earthquake magnitude, seismic source boundaries, and the value of b_0 in Equation 11-6). The uncertainties in different variables should be consistently propagated so that the confidence in the output (e.g., core-melt frequency and risk estimates) can be quantified.

4. The risk-analysis method should not be constrained by the plant licensing criteria. For example, the inelastic capacities of structures and equipment should be estimated, although the plant design criteria may require that the structures and equipment be within yield levels under the safe-shutdown earthquake. In the seismic hazard analysis, the entire range of earthquake levels should be studied, even though the plant may have been designed for a conservatively selected safe-shutdown earthquake.

5. Since seismic events have the potential to affect a number of components, it is essential that any correlation between the failures of different components be properly handled. Correlation arises both in the responses of different components to a single earthquake and a common structural model, and in component capacities (because of a common manufacturer or similar mounting). The correlation due to a common earthquake level is automatically handled by integration in Equation 10-1, where the product of the frequency of occurrence of any earthquake level and the conditional frequency of failure for a sequence of components given the hazard is integrated over the entire range of hazard intensity.

6. The possibility of equipment failure from nonseismic causes (e.g., random failures) during a seismic event should be recognized. For example, the ceramic insulators on offsite-power transformers have very high frequencies of failure at moderate ground accelerations (0.20g to 0.30g). Given such a failure and the resulting loss of offsite power, the unavailability of emergency diesel generators in the time required to repair or replace the insulators should be considered. It may be greater than the unavailability attributable to the seismic fragility of diesel generators at low to moderate earthquake levels (0.20g to 0.40g).

11.2.11 CURRENT METHODS

Two methods are currently available for estimating seismic risks. They generally fulfill all of the requirements outlined in Section 11.2.10. The major difference between them is the level of detail in the seismic response analysis and the plant-system and accident-sequence analysis. The first method was developed and applied in the Oyster Creek PRA study (Garrick and Kaplan, 1980; Kennedy et al., 1980). It has since been improved and applied to estimate seismic risks for the Zion plant (Commonwealth Edison Company, 1981) and the Indian Point plant (PASNY, 1982). Called here "the Zion method" for short, it is now being used in estimating seismic risks for the La Salle, Oconee, Browns Ferry, Midland, and Pilgrim plants. The second method was developed in an NRC-funded research program at the Lawrence Livermore National Laboratory--the Seismic Safety Margins Research Program (Smith et al., 1981); it is called "the SSMRP method" in the discussion that follows. By coincidence, the Zion Nuclear Generating Station was selected as a reference plant for the SSMRP study.

The Zion method relies heavily on the use of engineering judgment to supplement sparse data and limited analysis, whereas the SSMRP method emphasizes extensive component and system modeling as well as a detailed seismic response analysis. Engineering judgment is, of course, used in the SSMRP method in estimating the seismic hazard, deriving component fragilities, and performing the plant-system analysis. For a routine PRA study of a nuclear power plant, the Zion method offers a procedure that takes into account all the important features of the seismic risk and involves relatively less effort. However, the risk estimates derived by this procedure may have larger variabilities associated with them. If the Zion method shows that the seismic risk contribution dominates the total plant risk or that a particular plant safety system is a dominant risk contributor, a more refined estimate of the seismic risk can be obtained by following the SSMRP method.

11.2.11.1 The Zion Method

Seismic Hazard Analysis

The procedure for the seismic hazard analysis is essentially as outlined in Section 11.2.3. In the Zion study, the seismic hazard model was based on the tectonic provinces described in the work of TERA (1979), in which a number of nationally recognized seismicity experts made judgments of the seismicity in various regions of the United States. The earthquake data available in Modified Mercalli intensity units were converted to the body-wave magnitude m_b by means of Equation 11-12. The attenuation relationship between m_b and the sustained maximum ground acceleration a_s as given in Equation 11-13 was used. The uncertainty in the maximum m_b that the seismic sources are capable of producing was accounted for by assigning probabilities of .28, .44, and .28 to the m_b values of 5.6, 5.8, and 6.0, respectively. Similarly, three different tectonic province assumptions were made; probabilities of .5, .3, and .2 were assigned to these hypotheses (McGuire, 1981).

In accordance with the work of Kennedy (1981), the effective peak ground acceleration was expressed in terms of the sustained maximum ground acceleration, and the maximum cutoff values established for this parameter were 0.44g to 0.65g. Some sensitivity analyses have revealed, however, that the risk estimates are not too sensitive to the maximum cutoff values.

Analysis of Plant-System and Structure Responses

In the Zion method, structural and equipment fragilities are expressed in terms of a ground-motion parameter (e.g., effective peak ground acceleration). A response factor of safety is derived from a linear dynamic analysis of the structure or equipment. In most cases, the results of response analyses performed for the design-earthquake levels (e.g., operating-basis and safe-shutdown earthquakes) and ground-response spectra can be used to estimate the response factor of safety. As already mentioned, this factor of safety depends on the safety factors involved in the selection of ground-response spectra, the procedure used to include the effects of soil-structure interactions, the selection of damping levels, the modeling of structures and piping, and the method of analysis. The safety factors are treated as random variables, and their statistical parameters, such as the median and the logarithmic standard deviation, are estimated by using available data and engineering judgment.

Fragility Evaluation

Section 11.2.5.3 describes the development of component fragilities by the Zion method. Examples of the use of this method for deriving component fragilities can be seen in the Zion study (Commonwealth Edison Company, 1981) and in a recent report by Ravindra (1982).

Plant-System and Accident-Sequence Analysis

The plant-system and accident-sequence analysis used in the Zion method can be summarized as follows:

1. For each initiating event, the analyst constructs fault trees reflecting (a) failures that could initiate an accident sequence and (b) failures of key system components or structures that could mitigate the core-melt sequence.

2. The fragility of each such component (initiators and mitigators) is estimated.

3. Fault trees are used to develop Boolean expressions for core-melt sequences that lead to each of the various plant-state frequencies.

4. Considering possible core-melt sequences and containment mitigating systems (e.g., fan coolers, containment sprays, and containment), Boolean expressions are developed for each release category.

The plant model is described through a seismic fault tree like the one shown in Figure 11-8 for a PWR plant. According to this fault tree, an earthquake-induced core melt occurs if any of the initiating events occurs together with a loss of safety injection or loss of cooling; that is,

$$M_S = M_1 \vee M_2 \vee M_3 \qquad (11\text{-}28)$$

where the Boolean-algebra symbol $\vee$ means "or"; in the sequel, the symbol $\wedge$ is used to signify "and." For each of these events, M_1, M_2, and M_3, fault trees are constructed in terms of the primary component failures. Figure 11-9 shows a typical fault tree for a small LOCA with a loss of safety injection or cooling.

The termination of a fault tree at any basic component failure level depends on the fragility of the component and on the maximum ground acceleration possible at the site. All components that have a slight chance of failure (e.g., 5 percent) at the upper-bound effective peak ground acceleration are included in the fault trees. These components are identified by reviewing the fragility descriptions developed earlier.

The frequency of core melt is calculated by combining the plant-level fragility with the seismic hazard curves. This is done by first translating the seismic fault trees into a Boolean expression. For example, the Boolean expression for an earthquake-induced core melt is

$$M_S = ① \vee ② \vee ⑤ \vee ⑥ \vee ⑦ \vee ④ \vee [(⑨ \vee ⑩ \vee ⑧) \wedge ③] \qquad (11\text{-}29)$$

The components denoted by numbers in circles are listed in Table 11-1 generally in order of increasing capacity. Plant-level fragility curves are obtained by aggregating the fragilities of individual components according to Equation 11-29 and using DPD arithmetic (Kaplan, 1981). An example of the

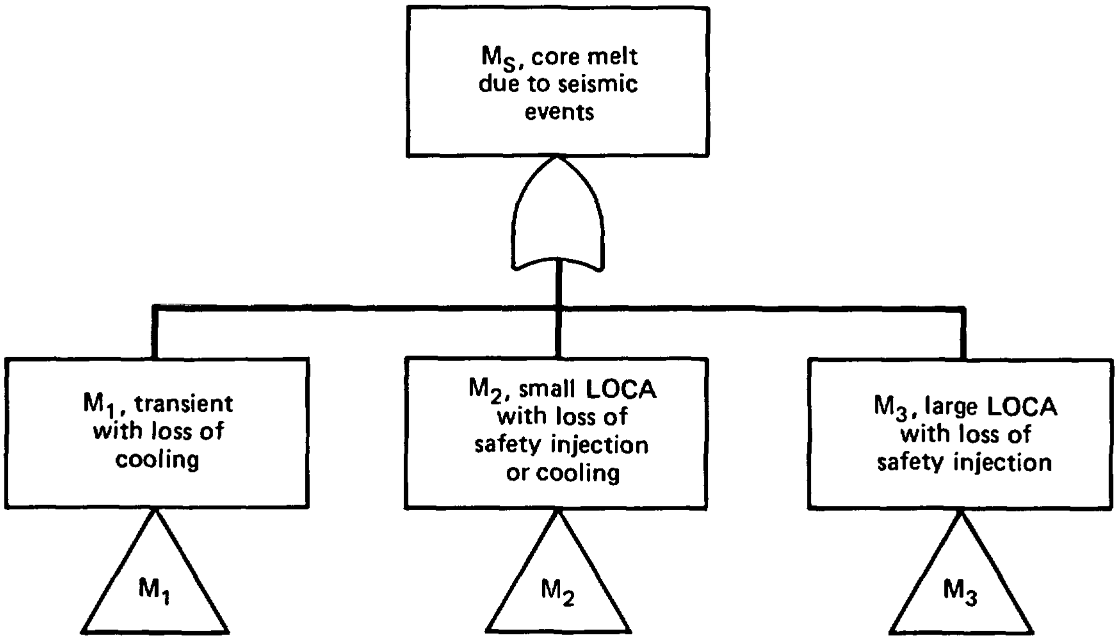


Figure 11-8. Seismic fault tree for a PWR plant.

plant-level fragilities is shown in Figure 11-10. For purposes of comparison, a family of fragility curves for a particular component is also plotted in Figure 11-10. We can observe a shift in the fragility curves from the component level to the plant level.

This shift to the left in the plant-level fragility illustrates an important feature of the seismic risk problem. Since an earthquake can simultaneously affect a number of redundant components, the plant-level fragility (i.e., the conditional frequency of failure given an acceleration value) is higher than the fragility of any component.

The core-melt frequency f_c is calculated as follows:

$$f_c = \sum_i h(a_i)\ f_S(a_i) \qquad (11\text{-}30)$$

where $f_S(a_i)$ is the occurrence frequency of a system failure that leads to a core melt for effective peak ground accelerations of less than or equal to a_i and $h(a_i)$ is the annual frequency of occurrence of earthquakes with an effective peak ground acceleration between a_i and $(a_i + \Delta a)$. The summation is carried over the entire range of accelerations.

The seismic hazard and the plant-level fragility are each represented by a family of curves plotted for different nonexceedence probabilities. Equation 11-29 and DPD arithmetic are then used for a probabilistic multiplication of these curves to obtain the probability distribution for the core-melt frequency (Figure 11-1).

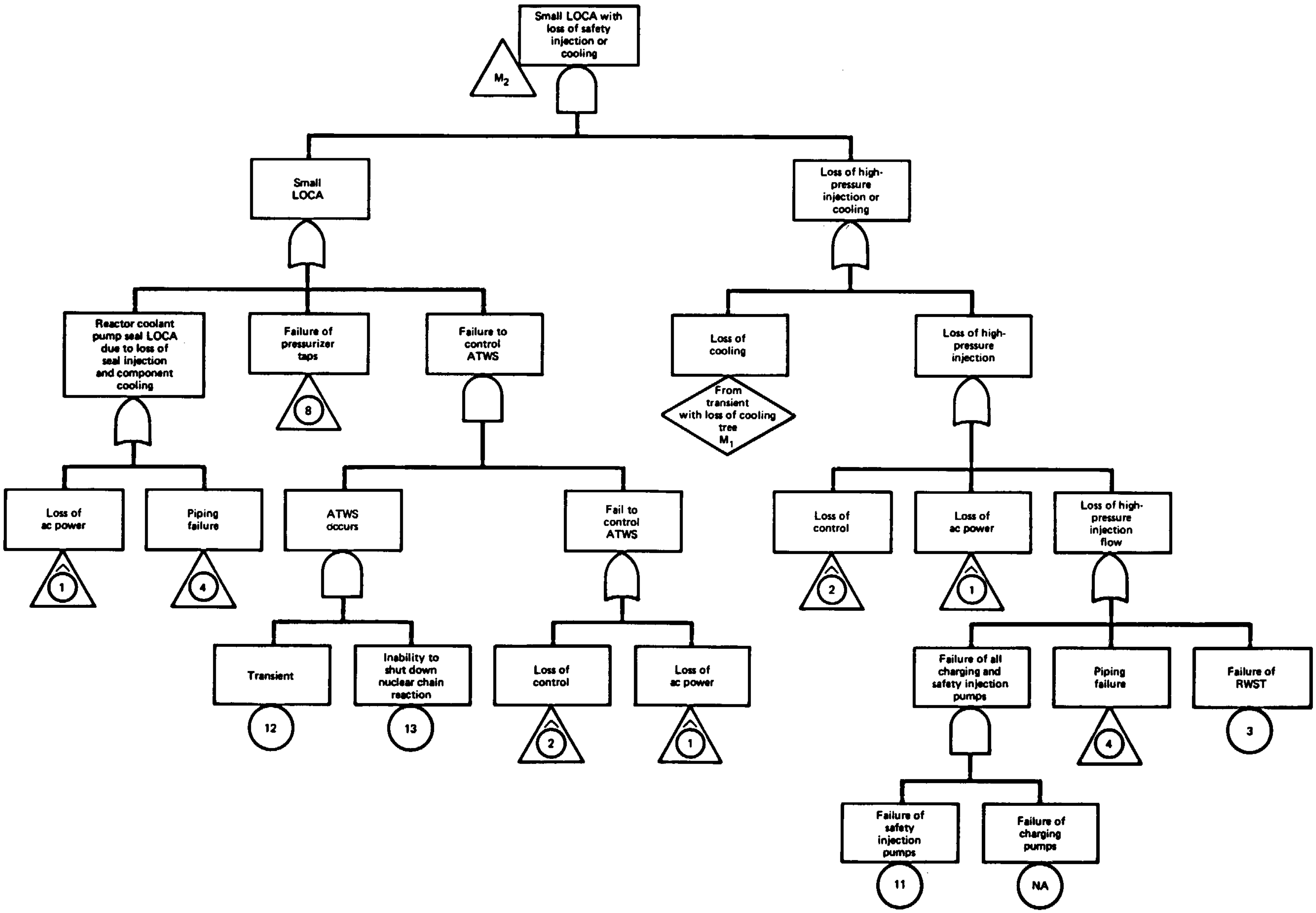


Figure 11-9. Fault tree for a small LOCA with loss of safety injection or cooling in a PWR plant. The numbers in circles correspond to the primary events of Table 11-1. The label "NA" (not applicable) means that this failure is not induced by the range of possible seismic events.

The frequencies of various release categories are based on the type of core melt (i.e., plant state) and the containment state (failure or no failure). The plant state would be dependent on the top event of the fault tree (i.e., M_1, M_2, or M_3) and on the functioning of the containment fan coolers and containment sprays. Boolean equations are developed for different plant states. Each plant state combined with the containment state is assigned to a particular release category. Therefore, a Boolean equation for each release category is derived to express the logical relationships between component failures. Using the Boolean equation along with component-fragility families, a fragility family for each release category is derived. By integrating this family of fragility curves over the family of seismic hazard curves, a probability distribution is obtained for the frequency of the release category (see Figure 11-2).

Table 11-1. List of critical structures and equipment in the seismic fault tree for a typical PWR[a]

Number[b]	Structure or equipment
1	Service-water pumps
2	Auxiliary building--failure of concrete shear wall
3	Refueling water storage tank
4	Interconnecting piping/soil failure beneath reactor building
5	Collapse of pump enclosure roof in cribhouse
6	125-volt dc batteries and racks
7	Service-water system, 48-inch buried pipe
8	Collapse of pressurizer-enclosure roof
9	Condensate storage rack
10	20-inch piping, condensate storage tank
11	Safety injection pumps
12	Offsite-power ceramic insulators
13	Core geometry

[a]See Figure 11-9 and Equation 11-29.
[b]Shown inside circles in Equation 11-29 and Figure 11-9.

A computer code called SEIS has been developed (Kaplan, 1981) to perform the probabilistic calculation of core-melt and release frequencies. The seismic hazard curves and the component-fragility families are the inputs to this code.

Consequence Analysis

In the Zion probabilistic study (Commonwealth Edison Company, 1981), the consequence model developed for internal events was employed for analyzing the

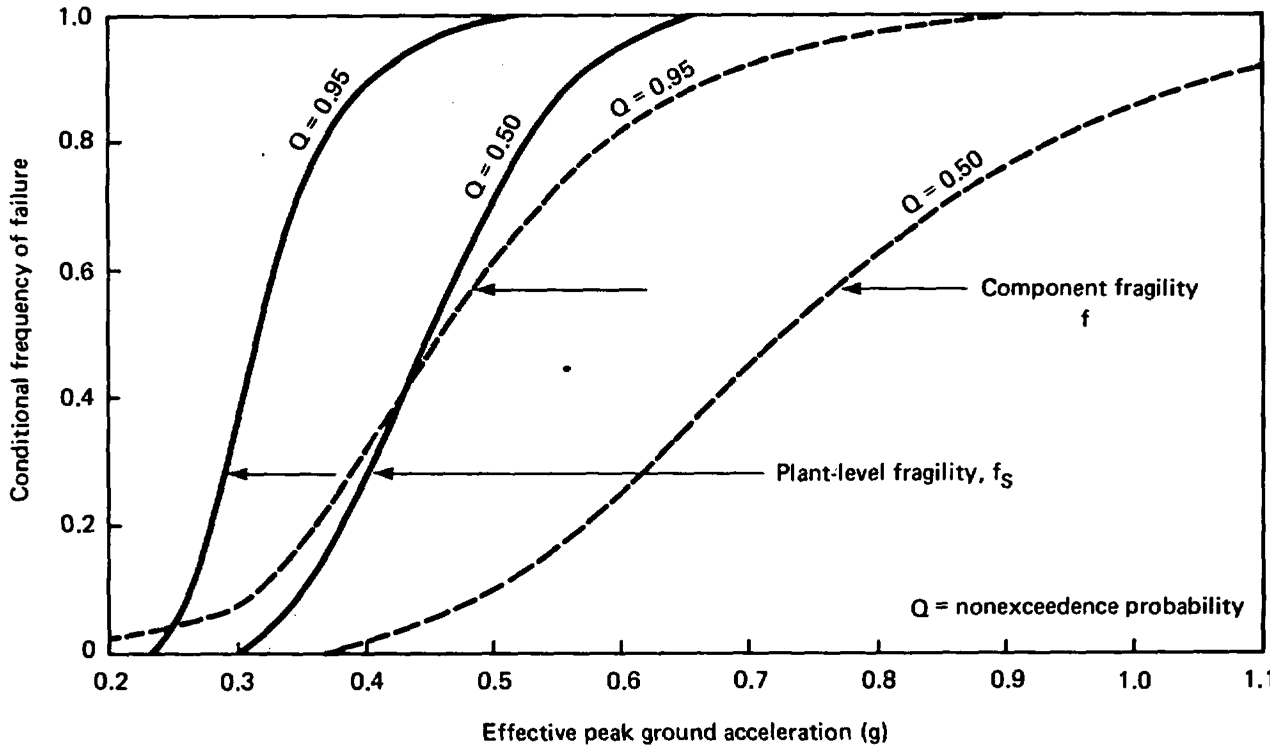


Figure 11-10. Component and plant-level fragility curves.

consequences of seismic events. The reasoning was that the large uncertainties assigned to the parameters of the consequence model would cover the variations due to seismic events.

11.2.11.2 The SSMRP Method

Seismic Hazard Analysis

The procedure for the seismic hazard analysis is essentially as outlined in Section 11.2.3. In the first phase of SSMRP, the peak ground acceleration was used as the hazard parameter. Historical data and expert opinion were used in calculating the annual frequencies of exceeding different values of the peak ground acceleration.

Analysis of Plant-System and Structure Responses

In the SSMRP method, structural and equipment fragilities are expressed in terms of local response parameters, such as stress, moment, and spectral acceleration. Therefore, given an earthquake, the conditional failure frequency of a structure or equipment is obtained by a convolution of the frequency distribution of the response for that ground acceleration and the frequency distribution of the resistance of the structure or equipment.

A major emphasis of the SSMRP method lies in the computation of structural and equipment responses. Some of the distinguishing features of the SSMRP response-analysis method as developed and applied in phase I of the program are summarized below.

As mentioned above, the seismic hazard parameter is the peak ground acceleration. Variability in the ground motion is incorporated into the analysis by simulating a set of time histories consistent with the hazard curve; each time history is developed to simulate a particular spectral shape.

A detailed analysis of soil-structure interactions is made, using the substructure approach. For structures, a detailed finite-element analysis is performed. Subsystem responses are determined by using a multisupport time-history analysis. The subsystems modeled in phase I of the program consisted of a number of valves, nozzles, and pumps as well as critical piping nodes.

Uncertainties in the input parameters (e.g., soil shear modulus and damping, and structures and subsystem frequencies and damping) are included by using a Latin-hypercube technique to sample different parameter values for each earthquake time-history simulation (Iman et al., 1980).

By such detailed modeling and analysis, phase I of the SSMRP study was able to derive the peak responses of structural elements, equipment, piping nodes, etc., and the correlation between them. This extensive response analysis was needed because the phase I study was a first attempt at quantifying the probabilistic response with all the input variables included and because the reference plant was not designed to present-day criteria and standards. Furthermore, the objective of the program was to establish the relative importance of various methods used in current seismic design practice.

Phase II of the SSMRP will develop methods for estimating the ratio of the actual response to the design response for structures and equipment in nuclear power plants designed according to the procedures of the NRC Standard Review Plan (Bumpus et al., 1980). This should circumvent the need for a detailed response analysis and should facilitate an optimal use of plant-design information. Phase II will also concentrate on estimating the sensitivity of response to different input parameters in order to improve the efficiency of response computations and will investigate the significance of assuming perfect dependence or perfect independence between component responses in lieu of a detailed correlation analysis.

Fragility Evaluation

As discussed before, in the SSMRP method the fragility of a component is anchored to local response parameters (e.g., moment, stress, and spectral acceleration). Therefore, the fragility description for a component includes only the variability of its seismic capacity. In this the SSMRP method differs from the Zion method, which expresses the fragility of a component as a function of both seismic response and capacity. Apart from this, the procedure for developing component-fragility curves is the same as that described in Section 11.2.5.2. The fragilities of structures are based on the peak ground acceleration at which inelastic structural deformations would interfere with the operation of safety-related equipment housed in the structure. This failure acceleration is computed as the SSE acceleration times a factor of safety that accounts for original conservative estimates of material strength and conservative design analysis. The factor of safety also includes a ductility factor that allows nonlinear failure criteria to be related to the linear responses calculated as described in Section 11.2.4.

Details of how the fragility curves for structural elements were established in phase I of the SSMRP study are given by Wesley and Hashimoto (1980). Equipment and piping fragility was described by a random variable C that represents the seismic capacity expressed in terms of force, moment, or spectral acceleration. As before, the random variable C was modeled as

$$C = F_C A_{SSE} \tag{11-31}$$

where A_{SSE} is the magnitude of the fragility (local response) parameter specified for the safe-shutdown earthquake and F_C is an equipment-capacity factor that accounts for both strength and ductility. The development of fragilities for equipment and piping has been described by Campbell et al. (1981).

In phase I, randomness and uncertainty were not treated separately; instead a composite fragility curve was used for each component. The total variability was approximated as

$$\beta_C = \left(\beta^2_{C,R} + \beta^2_{C,U}\right)^{1/2} \tag{11-32}$$

Correlations among component fragilities (owing to the same manufacturer or identical mounting), if specified by the analyst, can be consistently handled by the SSMRP computation scheme.

<u>Plant-System and Accident-Sequence Analysis</u>

The SSMRP method consists of identifying initiating events, developing event and fault trees, and finding the dominant accident sequences for various release categories. In phase I of this program, the occurrence frequencies of initiating events were calculated for various peak ground accelerations. An event tree was constructed for each initiating event, and from these event trees 148 core-melt sequences were identified. Each core-melt sequence was followed by a containment event tree that established the release sequence. Figure 11-6 shows an event tree for a large LOCA in a PWR (Smith et al., 1981); it contains 23 core-melt sequences, and each sequence can lead to a release through the potential containment-failure modes designated α, β, γ, δ, and ε.

A fault tree was used to evaluate the occurrence frequency of system failure at each branch of an event tree. Fault trees were developed for the auxiliary feedwater system, the service-water system, the emergency core-cooling system (comprising the safety injection system), the residual-heat-removal system, the charging system and the accumulators, and the electric power system. Each fault tree had several hundreds of components and was represented by a union of cut sets. The frequencies of primary events in a fault tree, and hence the frequency of system failure, depend on component-failure frequencies. In calculating the component-failure frequencies, both random failures and earthquake-induced failures were taken into account. The frequencies of earthquake-induced failures were calculated by using a multivariate peak-response distribution developed by the SEISIM code from the output of the SMACS code and the component fragility developed as described in Section 11.2.5.

The computer code SEISIM (Seismic Evaluation of Important Safety Improvement Measures) was developed to compute the occurrence frequencies of structural failures, component failures, system failures, and radionuclide releases (Hudson and Collins, 1979). It uses as input the seismic hazard curves, the component-fragility curves, and the structure and equipment responses calculated by the SMACS code. Event and fault trees are represented by Boolean expressions. A unique feature of SEISIM is the consistent treatment of the correlation between component responses to a given earthquake and between component capacities.

The SEISIM code computes the conditional occurrence frequency of a single radionuclide-release sequence (for a given earthquake-acceleration range) as the product of four occurrence frequencies: (1) the frequency of an earthquake producing the given input ground-motion level, taken from the seismic hazard curve; (2) the frequency of the necessary initiating event, given the input motion level; (3) the frequency of the core-melt accident sequence, given the input motion level and the initiating event; and (4) the frequency of the containment-failure mode producing the specific radionuclide release, given the input motion level, the initiating event, and the accident sequence. The SEISIM code accumulates these conditional frequencies of release sequences into each of the seven radionuclide-release categories identified in the Reactor Safety Study (USNRC, 1975). Finally, the unconditional frequency of release for a given release category is obtained by integrating over all acceleration ranges. SEISIM is also structured to perform sensitivity analyses to determine which components or systems dominate in the failure- and release-frequency computations.

In the first phase of the SSMRP, the uncertainties identified in the seismic hazard analysis and in the development of component fragilities were not completely propagated. Extensive sensitivity studies and uncertainty propagation are planned for the second phase. The two-stage analysis described in Section 11.2.8 and illustrated by Collins and Hudson (1981) may be used for this purpose.

Consequence Analysis

In the first phase of the Seismic Safety Margins Research Program, the frequencies of radionuclide releases were the end products. An analysis of consequences was not performed. If the analyst elects to use the SSMRP method, he may follow the consequence-analysis method described in Section 11.2.7.

11.2.12 INFORMATION AND PHYSICAL REQUIREMENTS

11.2.12.1 Information Requirements

The information needed to perform a seismic risk analysis for a nuclear power plant consists of plant-design details as well as generic information.

Plant-Design Details

The needed information includes the following:

1. Description of plant systems, including the location of structures and components, and sizes of structural members; a set of general arrangement, structural, piping, electrical, and equipment drawings.

2. Design criteria, applicable codes, and applicable standards.

3. Safety analysis report, especially the chapter on the geologic and seismic characteristics of the region in which the site of the plant is located.

4. Material-strength test reports (i.e., concrete-cylinder test data and steel-mill certificates).

5. Design reports for plant-specific equipment, the nuclear steam supply system, and engineered safety features.

6. Specifications for the seismic design of equipment.

7. Reports on qualification and preservice tests as well as periodic inservice inspections.

8. Stress reports, including structural and subsystem models for the seismic response analysis.

9. Seismic Qualification Review Team (SQRT) reports for equipment, if available.

Generic Information

The list of needed generic information includes the following data and reports:

1. High seismic zone qualification reports for identical and similar equipment.

2. Seismic capacities of similar valves. (In the probabilistic risk assessments performed to date, plant-specific information on valves

was not available, and therefore it was necessary to use generic information.)

3. Shock-test reports from the U.S. Army Corps of Engineers or other sources.

4. Topical reports.

5. Reports of published PRA studies.

11.2.12.2 Personnel and Schedule

In addition to system analysts and reliability engineers, the PRA study team should include a qualified seismologist and engineers experienced in seismic hazard analysis, seismic structural and subsystem analyses, structural and mechanical design, and seismic qualification testing. Since the hazard analysis and seismic fragility evaluation call for engineering judgment to supplement the results of simplified analyses and sparse test data, the PRA study team may benefit from seeking advice from outside experts in these fields.

Seismic hazard analysis and fragility evaluation may start at the beginning of a PRA project; however, plant-system and accident-sequence analyses and the seismic risk assessment can be efficiently done after the event and fault trees for internal events are developed.

Computer requirements would not be different from those for the internal events. If a more detailed response analysis and release-frequency analysis are attempted, nonproprietary computer codes like SMACS and SEISIM can be used.

11.2.13 PROCEDURES

The recommended task-by-task procedure for performing the seismic risk analysis of a nuclear power plant is given below.

Task 1: Collection of Information

The collection of information on plant design, regional seismology, test reports, etc., as described in Section 11.2.12.1, forms the starting point for the seismic risk analysis.

Task 2: Establishment of Objectives and Scope

The objectives of the seismic risk analysis need to be established; it could be a part of the routine PRA done for the plant, or the seismic risk analysis may have been necessitated by a situation not envisioned during plant design (e.g., the discovery of a potential fault close by). The scope of the analysis and the refinement of the analytical model will depend on the objectives of the seismic risk analysis. A more refined procedure may be used if the seismic safety of the plant is to be proved in light of new information (e.g., a potential fault).

The project team should agree on a description of the ground-motion parameter (e.g., effective peak ground acceleration and instrumental peak ground acceleration). This will establish proper communication between the seismologist, the structural engineer, and the systems analyst. Similarly, different failure modes for structures and equipment should be defined and quantified.

A format for reporting the results of intermediate tasks is to be established. A possible format is the probability of frequency. The seismic hazard, the component fragility, and the release frequency can all be presented in this format to ensure consistency between different tasks.

The method chosen for the seismic risk analysis should meet the requirements outlined in Section 11.2.10.

Task 3: Plant Familiarization

Before a detailed analysis can begin, it is necessary for the PRA team to become familiar with the design, operation, and maintenance of the plant. Plant-design criteria, stress reports, design and as-built drawings, qualification procedures for equipment, the functions of various plant systems, and consequences of failures should be reviewed to aid in identifying initiating events and in constructing the models for plant-system and accident-sequence analyses and for the consequence analysis. A walk-through inspection of the plant is essential to identify the status of component supports (equipment and piping) and any visible deviations from the as-built drawings.

Task 4: Seismic Hazard Analysis

The seismology and past earthquake history of the site region should be reviewed. The information documented in the safety analysis report is a good starting point. A seismic hazard model identifying all seismic sources in the site region should be developed. The parameters of this model, such as source boundaries, activity rates, recurrence relationships, the upper-bound magnitude or the epicentral intensity of each source, attenuation relationships, and the correlation between intensity and ground acceleration, should be established on the basis of site-specific data, applicable regional data, and the expert opinion expressed in professional papers and reports.

Alternative hypotheses reflecting professional uncertainty on the values of such significant variables as source boundaries, the upper-bound earthquake magnitude, and the Richter-Gutenberg slope (b) should be postulated. A probability (or weight) should be assigned to each hypothesis.

Task 5: Analysis of Plant-System and Structure Responses

If a method like the Zion method is chosen, no new response analyses for structures and equipment need be performed; the responses for earthquakes different from the safe-shutdown earthquake can be obtained by extrapolation from the design responses. When a method like the SSMRP method is used, it may be necessary to develop detailed analytical models of structural systems and piping subsystems if no such models exist or the design analysis models are not adequate; the response analysis is performed with a computer code like SMACS.

Task 6: Fragility Evaluation

Depending on the risk-analysis method that is chosen, component fragilities will be developed as a function of a global ground-motion parameter or a local response parameter. A list of safety-related structures, piping, and mechanical and electrical equipment should be provided by the systems analyst to the structural engineer assigned to this task. By reviewing the plant design bases, the structural engineer will estimate the median inelastic safety factor implied by the component design over the SSE acceleration (response) for the particular mode of failure. He would also express the variability in the safety factor by the values of β_R and β_U, inherent randomness and uncertainty. The fragility curve is therefore represented by the median ground acceleration (or local response) capacity and the values of β_R and β_U.

Task 7: Plant-System and Accident-Sequence Analysis

This analysis begins with the identification of the earthquake-related initiating events, such as a large loss-of-coolant accident, a small loss-of-coolant accident, and transients. Event trees and/or fault trees showing core-melt and radionuclide-release sequences are developed for each initiating event. If event trees are used, the failure at each branch of the event tree is represented by generating a fault tree. The core-melt frequency and the frequencies of release categories are calculated from the seismic hazard estimates, the component fragilities, and the event and fault trees. Dependences between component failures should be properly accounted for in the analysis.

Task 8: Consequence Analysis

The consequence analysis specific to the seismic event is performed with a consequence model that may be a modification of the model used to analyze the consequences of internal events to reflect the effects of earthquakes on the evacuation of people, public response, etc.

Task 9: Development and Display of Results

The results of a seismic risk analysis consist of seismic hazard curves, component fragilities, probability distributions for the occurrence frequency of earthquake-induced core-melt accidents and for the occurrence frequencies of various radionuclide-release categories, and risk curves. Other useful results include failure frequencies for structures, systems, and equipment and the accident sequences that dominate the seismic risk.

11.2.14 METHODS OF DOCUMENTATION

The chapter of the PRA report that discusses the seismic risk analysis should include the sections described below.

Description of the Site Region

The geographic location of the plant as well as the seismic and geologic characteristics of the site and the region surrounding the site should be described.

Analytical Method

The reasons for choosing a particular risk-analysis method should be discussed, demonstrating that the chosen method meets the requirements of the seismic risk analysis (Section 11.2.10).

Seismic Hazard Analysis

The report should describe the hazard model; the seismic sources, their activity rates, and upper-bound magnitude or epicentral intensity; recurrence relationships and available intensity or magnitude data; the attenuation relationship selected for the analysis; and the correlation between intensity and magnitude and/or acceleration if used. Uncertainties in these parameters of the hazard model should be discussed in detail, along with the methods used to quantify them (see Section 11.2.8 for a discussion of the treatment of uncertainties). Sources of data, professional papers, and opinion surveys should be included.

The final result of the hazard analysis should be presented as a family of annual exceedence-frequency curves plotted against the values of a ground-motion parameter. The choice of the ground-motion parameter should be substantiated.

Analysis of Plant-System and Structure Responses

If the structural and subsystem responses for earthquake-acceleration levels higher than those of the safe-shutdown earthquake were obtained by a linear extrapolation, a brief description of the design analysis should be given. If a more detailed structural and subsystem analysis was performed, a description of the analytical models used for the seismic input and for soil, structures, and subsystems should be given. The input parameter values selected in this analysis (e.g., soil properties, structural damping, and ductility) should be reported. Variability in the response as a result of uncertainties in the input parameters and in the analytical models should be quantified.

Fragility Evaluation

The report should describe the failure modes of the structures and the equipment for which the fragility curves were developed. The discussion should include both the sources of data and the methods used in developing fragilities for structures and equipment. A tabulation of safety-related structures and equipment as well as their fragility parameters should be provided.

Plant-System and Accident-Sequence Analysis

Starting with a discussion of the initiating events selected for analysis, this section should describe how their frequencies of occurrence were estimated, describe the event and fault trees for each initiating event, and list all identified accident sequences. It should also describe how the core-melt frequency and the frequencies of release categories were calculated and document how dependences between component failures were accounted for in the analysis.

Consequence Analysis

The parameters of the consequence-analysis model that are different from those traditionally used for internal events (see Chapter 9) should be described, substantiating the values chosen.

Final Results

The results of the seismic risk analysis should include the seismic hazard curves, families of component fragilities, and modifications to the event and fault trees of internal events. If the seismic risk analysis is separately carried further, the probability distribution for the occurrence frequency of an earthquake-induced core-melt accident and probability density functions for the occurrence frequencies of various radionuclide-release categories should be presented. Finally, the seismic risk curves should be presented for selected damage indices, examples being early fatalities, latent-cancer fatalities, and property damage. The accident (release) sequences and the systems and/or components that are the dominant contributors to public risk should be identified.

11.2.15 DISPLAY OF FINAL RESULTS

The results of a seismic risk analysis consist of the following:

1. Seismic hazard curves (Figure 11-4).

2. Component fragilities (Figure 11-5).

3. Probability distribution for the occurrence frequency of earthquake-induced core-melt accidents (Figure 11-1).

4. Probability density functions for the occurrence frequency of radionuclide-release categories attributed to earthquakes (Figure 11-2).

5. Seismic risk curves (Figure 11-7).

11.3 RISK ANALYSIS OF FIRES

11.3.1 INTRODUCTION

The early applications of risk analysis to nuclear power plants, including that presented in the draft report of the Reactor Safety Study (RSS), did not include a quantitative assessment of accidents initiated by major fires. The reason for this omission was twofold: (1) it was judged that fires were not likely to be dominant contributors to risk (RSS final report--USNRC, 1975) and (2) the state of the art in risk analysis had not yet developed an approach to covering fires. The importance of fire as a potential initiator of multiple-system failures took on a new perspective after the cable-tray fire at Browns Ferry in 1975. Although various experts have disagreed as to how close that fire came to an accident resulting in core damage and a major release of radioactive material, it is clear that its impact was extensive when measured in terms of the failure of redundant and diverse safety-related systems. It is not surprising, therefore, that risk analyses performed after the Browns Ferry fire have tended to include fires in the quantification of risk.

One of the first attempts at numerically estimating the risks due to fires appeared in the final report of the Reactor Safety Study, published later in the same year (1975) as the Browns Ferry fire. An estimate was made of the conditional probability of core melt given the specific damage state induced in the Browns Ferry systems by the fire (USNRC, 1975). The unconditional frequency of fire-induced core melt, calculated by averaging out the observed frequency of the Browns Ferry type of fire over the experience of U.S. commercial nuclear power plants, was found to be 1×10^{-5} per reactor-year, which is about 20 percent of the total core-melt probability estimated in the Reactor Safety Study. Kazarians and Apostolakis (1978) performed the same type of calculations under different assumptions and concluded that the frequency of core melt could be higher by a factor of 10. Both of these analyses appropriately point out that the results apply only to the specific circumstances of one particular fire and should not be construed as an estimate of the total contribution of fires to risk.

A more detailed risk analysis of fires was included in the Clinch River Breeder Reactor (CRBR) Risk Assessment Study (1977). A failure modes and effects analysis was used to identify important fire locations for a wide variety of combustibles, including cables, oil, and sodium. Its estimate of the frequency of fire-induced core melt, 5×10^{-7} per reactor-year, is substantially below the estimates discussed above. At least part of the difference in the estimates can be attributed to the vastly greater physical separation of cables and equipment in the CRBR design.

Further contributions to the risk analysis of fires were made in a risk-assessment study for the high-temperature gas-cooled reactor (HTGR) (Fleming et al., 1979). In addition to a qualitative screening procedure similar to that employed in the CRBR study, the HTGR study made use of a quantitative bounding method to screen for important fire locations.

Nuclear plant experience data were analyzed in detail to obtain estimates of fire-occurrence frequency and probability distributions for fire severity as measured by the duration of burn and the size of the fire-damage area. These data were used in a simple fire-propagation model to estimate the probabilities of location-dependent common-cause failures. A major finding of this study, which is independent of the unique characteristics of HTGRs, is that the contribution of fires to risk cannot be expressed simply in terms of core-melt frequency, as in earlier studies, because the conditional probability of containment failure given a core melt may be greater for fires than for other initiators. Fire-induced core-heatup accident sequences were found to dominate the HTGR risk-assessment curve at accident frequencies below 1×10^{-7} per reactor-year.

The Rensselaer Polytechnic Institute examined, for the U.S. Nuclear Regulatory Commission, various aspects of fire risk for light-water reactors. In Gallucci's work (1980) nuclear plant data are analyzed and categorized in the HTGR study. The source of data extended beyond licensee event reports to include insurance company records, and therefore the sample size was somewhat larger. The result was a more complete data base, particularly with regard to fires during construction.

In his doctoral dissertation at Rensselaer, Gallucci (1980) developed a risk-analysis method and applied it to a representative design for a large BWR. The probabilistic aspects of fire propagation were modeled in terms of an event tree that explicitly models various stages of ignition, detection, suppression, and propagation. The application of this technique is described in Section 11.3.3. The frequency of core damage due to fires was estimated to be about 2×10^{-4} per reactor-year, with an upper bound of about 1×10^{-3} per reactor-year. In this study, three types of combustibles at each of 11 plant locations were analyzed in the quantification of risk.

Recent advancements in the risk analysis of major fires have been made by Apostolakis, Kazarians, and Siu in projects carried out at the University of California at Los Angeles and as part of the Zion (Commonwealth Edison Company, 1981) and Indian Point (PASNY, 1982) risk studies.* Specific advancements in this work include the development of a physical model for fire propagation and suppression, a method for propagating uncertainties through this model, and the use of Bayes' theorem in estimating plant-specific and location-specific fire-occurrence frequencies. The Zion, Indian Point, and Big Rock Point studies have included detailed analyses of fire-induced accident sequences.

The trend in the risk analysis of fires is clear. There is a growing body of evidence to suggest that fires cannot and should not be dismissed as important risk contributors on a generic basis. In certain applications of risk analysis, such as those performed during the conceptual or detailed design stage, it may not be practical to attempt a fire-risk analysis

*See Apostolakis and Kazarians (1980), Apostolakis et al. (1982), Kazarians and Apostolakis (1978, 1981), Siu (1980), and Siu and Apostolakis (1981).

because of the need for factoring in details of the physical layout and construction. However, whenever the results of a risk study are to be interpreted on an absolute scale, the omission of fires appears to create a high risk of overlooking potentially dominant accident sequences. Fortunately, the methods discussed in this section include those that allow most fires to be screened out without the need for detailed investigation.

11.3.2 OVERVIEW

The purpose of the analytical method developed in the next section is to identify a list of the dominant accident sequences that are initiated by fire and then to assess the frequency of occurrence for each. This process requires information about several important aspects of a fire (e.g., ignition, progression, detection and suppression, characteristics of materials under fire conditions) as well as the plant safety functions and their behavior under accident conditions. Considerable uncertainties exist in the analysis because of gaps in the required knowledge. Nevertheless, the current analytical technique shows significant improvements over that available only a few years ago.

Fires are generally treated as external events, although they are generated by plant equipment and personnel. It is assumed that fires are initiated at certain frequencies within various plant compartments; the analyst is to determine what sequences follow and with what frequency.

Following the standard format for the analysis of external events, the fire analysis is divided into four parts: a hazard analysis; a fire-propagation analysis, which is somewhat analogous to a component-fragility analysis; a plant and system analysis; and a release-frequency analysis. The hazard analysis develops the frequency and magnitude of the "externally imposed stress," where "stress" is in terms of potential fire-induced accident sequences. The propagation analysis investigates the resistance of the plant to fire damage by studying the propagation of the fire and the effectiveness and timing of suppression. The last two analyses evaluate the response of plant systems to the accident sequence triggered by the fire; the first considers core damage, while the second is concerned with the release of radioactive material from the containment. This division is by no means unique; it simply provides a familiar structure that can be used to examine the methods of fire-risk analysis, which are described below.

11.3.3 METHODS

The object of the fire-risk analysis is to estimate the frequency of fire-induced radionuclide releases of varying magnitudes. Because of the inherent variability of fire phenomena and the relatively primitive understanding of these phenomena, the large uncertainties in the models leading to these release-category frequency estimates should be treated explicitly throughout the analysis.

The method can be divided into four somewhat independent steps: the fire-hazard analysis, which identifies critical plant areas and estimates the frequency of fires; the fire-propagation analysis, which models the behavior of fires in the critical areas; the plant-system analysis, which estimates the likelihood of the fires leading to plant-damage states; and the release-frequency analysis, which uses the results of the preceding analyses to derive the frequencies of accident sequences leading to radionuclide releases.

The sections that follow discuss the merits of various models available for each analysis.

11.3.3.1 Fire-Hazard Analysis

11.3.3.1.1 Location Screening

Theoretically, the fire-risk analyst should study the potential contributions to risk of fires anywhere in the nuclear power plant. By screening out unimportant locations, however, he can greatly reduce the amount of work required without sacrificing significant confidence in his results. The purpose of the fire-hazard analysis is to identify the locations that are important to the fire-risk analysis.

For the purposes of initial analysis, fire locations are usually considered to be coincident with the fire zones defined by the utility in its fire-protection review, issued in response to the NRC's Technical Position 9.5-1. The fire zones consist of one or more compartments and are separated from other zones by rated fire barriers. The spread of fire between zones is generally unlikely, although flames did spread through an improperly sealed cable penetration in the well-known Browns Ferry fire. A more detailed analysis may show that only limited areas within the fire zones contain critical equipment (Commonwealth Edison Company, 1981), in which case a number of fire locations may be defined within these zones.

The "importance" of a fire location is measured by its contribution to the frequency and the nature of a release of radioactive material. Since this cannot be determined until at least the first iteration of the fire-risk analysis has been completed, more approximate measures are employed. The primary measures are the type and the quantity of fire-vulnerable safety equipment at the location of interest. This information can be obtained directly from the fire-protection reviews. Other factors that may be used in the screening process are the frequencies of fires, the types and the amounts of combustible materials, and the available fire-suppression systems. Information on the last three factors can also be obtained from the fire-protection reviews.

Three basic methods for determining the importance of plant locations with respect to fire risk are described below. The first considers the presence of fire-vulnerable safety equipment, the second employs a failure modes and effects analysis (FMEA), and the third uses an FMEA coupled with additional factors to account for the likelihood of severe fires.

Location Analysis: Method 1

The location of interest is considered important if it contains enough safety equipment so that a severe fire could fail one or more safety systems (e.g., shutdown heat removal), which may or may not be in the same division. The loss of only one division of safety equipment means a loss of redundancy and does not necessarily lead to core damage and a release of radionuclides; nevertheless, the analyst may decide that this is an event that should be quantified.

The fire-protection reviews and a recent study at the Rensselaer Polytechnic Institute (Gallucci, 1980) employ such a screening approach. Because there are many rooms that contain some safety equipment, these studies consider fire occurrences in many locations. However, only a small number of these locations contribute significantly to the risk in most power plants: the rooms that contain many divisions of safety equipment. The less critical locations are screened out by performing an FMEA.

Location Analysis: Method 2

As in method 1, the locations containing fire-vulnerable safety equipment are identified. The loss of all equipment at that location is then postulated. If it is found that an initiating event (LOCA or transient) will not occur, the location is eliminated from consideration. (Note that a reactor trip, which is a transient by definition, will almost certainly be induced by a fire severe enough to disable many items of safety equipment.) Given a LOCA or a transient, a number of safety functions are required for safe shutdown. If the loss of all equipment in the location of interest prohibits the performance of any or all required functions, the location is tabbed for further analysis.

This screening method, described by Kazarians and Apostolakis (1981), was employed in the fire-risk portions of the Zion (Commonwealth Edison Company, 1981) and Indian Point studies (PASNY, 1982). In these analyses, the fire-induced loss of control of safety systems is judged to dominate fire-induced hardware losses, and therefore the critical locations that were investigated contain electrical cables and/or switchgear for many safety (and nonsafety) systems.

Given this assumption, and a fire that has caused an initiating event, the analysts then determine whether the same fire can induce failures that will prevent--

1. Reaching and maintaining a condition of negative reactivity.

2. Removing decay heat.

3. Monitoring and controlling the inventory and pressure of the reactor-coolant system (RCS).

Because of the fail-safe logic of the reactor-scram circuitry, the analysts assume that the reactor trip is successful. Therefore, critical fires will affect the implementation of actions 2 and 3. If no one fire can do this, the fires that can prevent either action 2 or action 3 are considered. However, these fires are less likely to be significant contributors to risk, because at least one independent failure in an unaffected safety system is required for core damage and radionuclide release to occur.

One feature of method 2, as executed by Kazarians and Apostolakis (1981), is that sets of fire locations are not considered for evaluation. For instance, if two adjacent rooms each contains one train of safety equipment, fires that spread from one room to the other and disable both trains are not studied. It is felt that a very large and long-burning fire is needed to penetrate most power-plant compartment walls, whether or not they are fire-rated walls. These fires are low-frequency events, and their contributions to risk are likely to be dominated by the contributions from fires burning in rooms that contain both trains.

If interzone fire propagation is considered to be important, a more complicated screening procedure that looks at groups of adjacent locations can be employed. One such approach uses a component-level fault tree in which the components are assigned location identifiers. "Core melt" is typically the top event, although it need not be. Minimal cut sets for the fault tree are derived in terms of the location identifiers. Several computer codes are available for this purpose. (For a discussion of such qualitative search procedures, see Section 3.7.) The WAMCOM code (Putney, 1981) is especially useful because it can identify cut sets with up to two locations.

One disadvantage of this method is the potential for omitting fire sequences leading partway to core meltdown but requiring additional component failures to result in the top event. For example, a sequence initiated by a fire in one location plus a dependent failure of components in each of two other locations would be screened out by this method, although it is questionable that the probability of such a sequence is really very low.

The screening approach of method 2 is illustrated here by example. The location of interest is the outer cable-spreading room of an imaginary power plant. The analyst must determine whether a fire in this location can cause not only an initiating event but also prevent the removal of decay heat, prevent the monitoring and control of RCS coolant inventory and pressure, or do both.

The outer cable-spreading room contains control and power cables for the motor-driven pumps of the auxiliary feedwater system, for the power-operated relief valves, and for the safety-injection pumps; control cables for the containment sprays and fan coolers; and control and power cables for many other systems.

A fire in this room has the potential to cause a LOCA by spuriously activating an isolation valve. If this does not happen, the presence of a large number of control and instrument cables in the room virtually ensures that a transient will occur. Thus, the first screening criterion of method 2 is met. Since the loss of the equipment mentioned above will clearly affect the removal of decay heat and the control of the RCS coolant inventory and pressure, criteria 2 and 3 are satisfied at least partially, and so this room is a candidate for an in-depth analysis. The presence of control cables for the containment sprays and fan coolers further emphasizes the room's importance: a fire that leads to core damage can also inhibit the performance of these containment-protection systems.

Of course, the cable-spreading room is an intuitive choice to begin with, not to mention the fact that the Browns Ferry fire involved this room. Most of the locations selected by this screening method are indeed "obvious" danger spots. However, the method provides a systematic means for their selection, sometimes identifies rooms that are not quite so obvious, and even rejects rooms that may seem to be obvious choices but actually do not contain enough critical equipment.

A more serious criticism is that this method does not account for the possibility that the fire-caused simultaneous failures of many instruments and/or nonsafety systems may initiate accident sequences. One might question the importance of this weakness when a power plant has single rooms that contain redundant safety trains and is vulnerable in the manner considered above. Furthermore, this weakness is common to all screening procedures described here and to the overall methods that are currently used for fire-risk analyses.

Location Analysis: Method 3

This method employs additional measures of importance to supplement either method 1 or method 2. In one model of this type, each room's fuel loading, fuel type, and fire-suppression effectiveness are combined with the safety-equipment inventory by a judgmental ranking procedure (Hockenbury and Yater, 1980).

A more elaborate fire location and progression analysis (FLPA) is described by Fleming et al. (1979). In addition to the effects of a fire on the components inside a room and the subsequent plant response, this method takes into account the inventories of combustible materials, the characteristics of adjacent locations, fire-brigade access, and ventilation systems; it also uses qualitative assessments of the likelihood of fire initiation and progression. Since the characteristics of adjacent compartments are explicitly considered, the possibility of fire spread from rooms containing large inventories of combustibles to compartments containing safety equipment is not overlooked.

Fleming et al. (1979) also discuss a method where the frequency of a particular radionuclide-release category due to all initiating events except fire, divided by the conditional frequency of that release category, given the loss of all components in the zone of interest, is compared with a rough estimate of the frequency of fires for that zone. If the release-category frequency ratio is greater than the fire frequency, the location is judged to be an insignificant contributor. This method requires a prior or concurrent assessment of other initiating events.

Clearly, the screening procedures of method 3 require more information and analysis than those of method 2. It is worthwhile to consider the merits of the additional complexity. The selection of a method and its implementation should be based on the objective of minimizing the chances that important fire-source locations will be overlooked in balance with the objective of minimizing the expenditure of effort on unimportant locations.

11.3.3.1.2 Fire-Occurrence Frequency

Once the analyst has identified the locations where a fire has the potential to initiate an accident sequence leading to a release of radionuclides, it is natural to ask how often these fires occur. Although an internal event, the fire is viewed as an external stress imposed on the plant at random times. The rate of occurrence can be established from the historical record.

Data from more than 900 reactor-years of U.S. experience are available for evaluation when construction and preoperational testing periods are included with plant operating periods. Considerable effort has been spent in evaluating the fire events cataloged in licensee event reports and data from the American Nuclear Insurers (see, for example, Hockenbury and Yater, 1980; Fleming et al., 1979). Data extracted from both sources are shown in Tables 11-2 and 11-3. The nature and the frequency of fires at nuclear power plants change dramatically, however, between construction, preoperational testing, and plant operation. Consequently, only the plant operating histories are suitable for assessing the risk from plants at power.

A number of issues arise in using the available data in estimating the rates of location-dependent fire occurrence. These include the possible reduction in the frequency of fires that results from an increased awareness

Table 11-2. Frequency of fires by reactor type

Reactor type	Status or mode of operation	Number of events	Percentage of total[a]
Boiling water	Construction	37	15.7
	Preoperational testing	6	2.5
	Operation	25	10.6
	Hot shutdown	0	0.0
	Cold shutdown	1	0.4
	Refueling/extended outage	3	1.2
Pressurized water	Construction	61	25.9
	Preoperational testing	15	6.4
	Operation	25	10.6
	Hot shutdown	4	1.7
	Cold shutdown	4	1.7
	Refueling/extended outage	1	0.4

[a]Total includes 22 events in fuel-fabrication facilities, 1 event in a reprocessing plant, 27 events in research and educational reactors, 2 events in a high-temperature gas-cooled reactor, and 3 events in a fast breeder reactor.

Table 11-3. Summary of fire-experience data

Number of reactor units	65
Operating experience (reactor-years)[a]	372
Number of fires (in operation)	49
Mean rate of occurrence per reactor-year	0.13
Diameter of fire damage (ft)	
Mean	8.6
Maximum	67
Time to put out fire (hr)	
Mean	1
Maximum	24

[a]From first electricity generation through April 1978.

of the danger of fire (Apostolakis and Kazarians, 1980; Gallucci, 1980), the discrepancy between the actual number of fire occurrences and the number of reported fires (Hockenbury and Yater, 1980; Hockenbury et al., 1981), and the question of applying industry-wide fire data to a particular power plant (Commonwealth Edison Company, 1981). These problems give rise to large uncertainties in the interpretation of the historical data.

Apostolakis and Kazarians (1980) model the frequency of fires for various compartments, using a probability-of-frequency framework to consistently treat the uncertainties. Starting with broad prior distributions to model their weak state of knowledge, they employ the statistical evidence given in Table 11-4 and use Bayes' theorem to derive the fire frequencies shown in Table 11-5. Their procedure is roughly outlined below.

The gamma probability distribution, with the parameters α and β given in Table 11-5, is chosen to represent the prior distributions for the various compartments. The gamma distribution is

$$\Pi(\lambda) = \frac{\beta\alpha}{\Gamma(\alpha)} \lambda^{\alpha-1} \exp(-\beta\lambda)$$

where $\Pi(\lambda)$ is the probability density function of fire frequency λ. The likelihood of the data, the probability of γ fires in T reactor-years (see Table 11-4), given fire frequency λ, is modeled as

$$L\left(\frac{E}{\lambda}\right) = \exp(-\lambda T) \frac{(\lambda T)r}{r!}$$

Bayes' theorem then states that the updated probability distribution for fire frequency, given evidence E, is

$$\Pi'\left(\frac{\lambda}{E}\right) = \frac{\Pi(\lambda)\ L(E/\lambda)}{\int_0^\infty d\lambda\ \Pi(\alpha)\ L(E/\lambda)}$$

Table 11-4. Statistical evidence of fires in light-water reactors[a,b]

Area	Number of fires, r	Number of relevant years, T
Control room	1	288.5
Cable-spreading room	2	301.3
Diesel-generator room	10	543.0
Containment	5	337
Turbine building	9	295.3
Auxiliary building	10	303.3

[a]From Apostolakis (1980).
[b]As of May 1, 1978.

An analytical evaluation of this expression shows that $\Pi'(\lambda/E)$ is also a gamma distribution, but with characteristic parameters $\alpha' = \alpha + r$ and $\beta' = \beta + T$. These updated values are also given in Table 11-5. For example, in the cable-spreading room from Table 11-5, the values of α and β (0.182 and 0.96) yield a mean frequency of .21, while the posterior distribution α' and β' (2.182) and 302.26) yields a mean frequency of .0072.

This same procedure can be used to update the given distributions for fire frequencies, when further reactor experience is accumulated.

Table 11-5. Distribution of the frequency of fires[a]

Area	Parameters of gamma distribution		Frequency of fires per room-year			
	α	β	λ_{05}	λ_{50}	λ_{95}	$\langle \lambda \rangle$
Control room						
Prior	0.182	0.96	5.0×10^{-8}	0.015	1.0	0.21
Posterior	1.182	289.46	3.1×10^{-4}	0.003	0.012	0.0041
Cable-spreading room						
Prior	0.182	0.96	5.0×10^{-8}	0.015	1.0	0.21
Posterior	2.182	302.26	1.4×10^{-3}	0.0062	0.017	0.0072
Diesel-generator room						
Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
Posterior	10.32	543.29	1.1×10^{-2}	0.018	0.03	0.019
Containment						
Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
Posterior	5.32	337.29	6.2×10^{-3}	0.014	0.028	0.016
Turbine building						
Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
Posterior	9.32	295.59	1.7×10^{-2}	0.03	0.05	0.032
Auxiliary building						
Prior	0.32	0.29	2.1×10^{-4}	0.30	5.0	1.11
Posterior	10.32	303.59	1.9×10^{-2}	0.033	0.053	0.034

[a]From Apostolakis (1980).

It should be noted that the fire-frequency distributions derived are generic. To account for plant-to-plant variability, similar distributions based on each plant's experience can be constructed, either by using the above procedure or by updating the generic distributions in Table 11-5.

11.3.3.1.3 Fire-Propagation Analysis

The purpose of a fire-propagation analysis is to determine the likelihood and extent of various levels of damage in a compartment, given that a fire has occurred. Three different approaches have been used to date. The first employs a statistical model based on past experience (Fleming et al., 1979), the second uses a multistage event-tree model (Gallucci, 1980), and the third requires the construction of physical models (Siu, 1980; Siu and Apostolakis, 1981).

The analyst should be aware that the existing fire-growth and fire-suppression models do not span the set of all possible scenarios and that even the existing models exhibit large uncertainties. Every attempt should be made to quantify the effects of these uncertainties.

Fire-Propagation Analysis: Method 1

This method is based on deriving equations for (1) the distance of fire spread or volume affected versus the time to fire control and (2) the probability of control versus the time to fire control (Fleming et al., 1979). From the two equations a curve for conditional probability versus fire size can be obtained. Here "fire size" is defined as that size within which components are failed. The equations are derived from linear regression analyses of fire data from nuclear power plants. Different correlations are developed for different combustibles (e.g., electrical fires versus lubricating-oil fires).

Method 1 is relatively easy to implement and with some conservative assumptions can be very effective at screening out unimportant locations. It has not yet been applied to fires that penetrate fire barriers. This approach glosses over the specifics of plant design: it assumes that average fire-occurrence frequencies derived from the operating histories of many plants apply to the plant under study. In its application so far, method 1 has assumed that fire has an equal probability of starting anywhere in the location studied--as would happen if transient combustibles were the dominant sources of most fires or if the permanent combustibles were uniformly distributed throughout the location.

Fire-Propagation Analysis: Method 2

Method 2 uses event trees (Gallucci and Hockenbury, 1981) to divide the fire model into four elements: (1) ignition, (2) detection, (3) suppression, and (4) propagation. Each element heads a column of the event tree. The fire is assumed to start in one component and potentially propagate to the next. The use of these four elements is illustrated in Figure 11-11 (Gallucci, 1980) by a two-stage event tree for two redundant components in the location (more stages may be required for more components). Submodels,

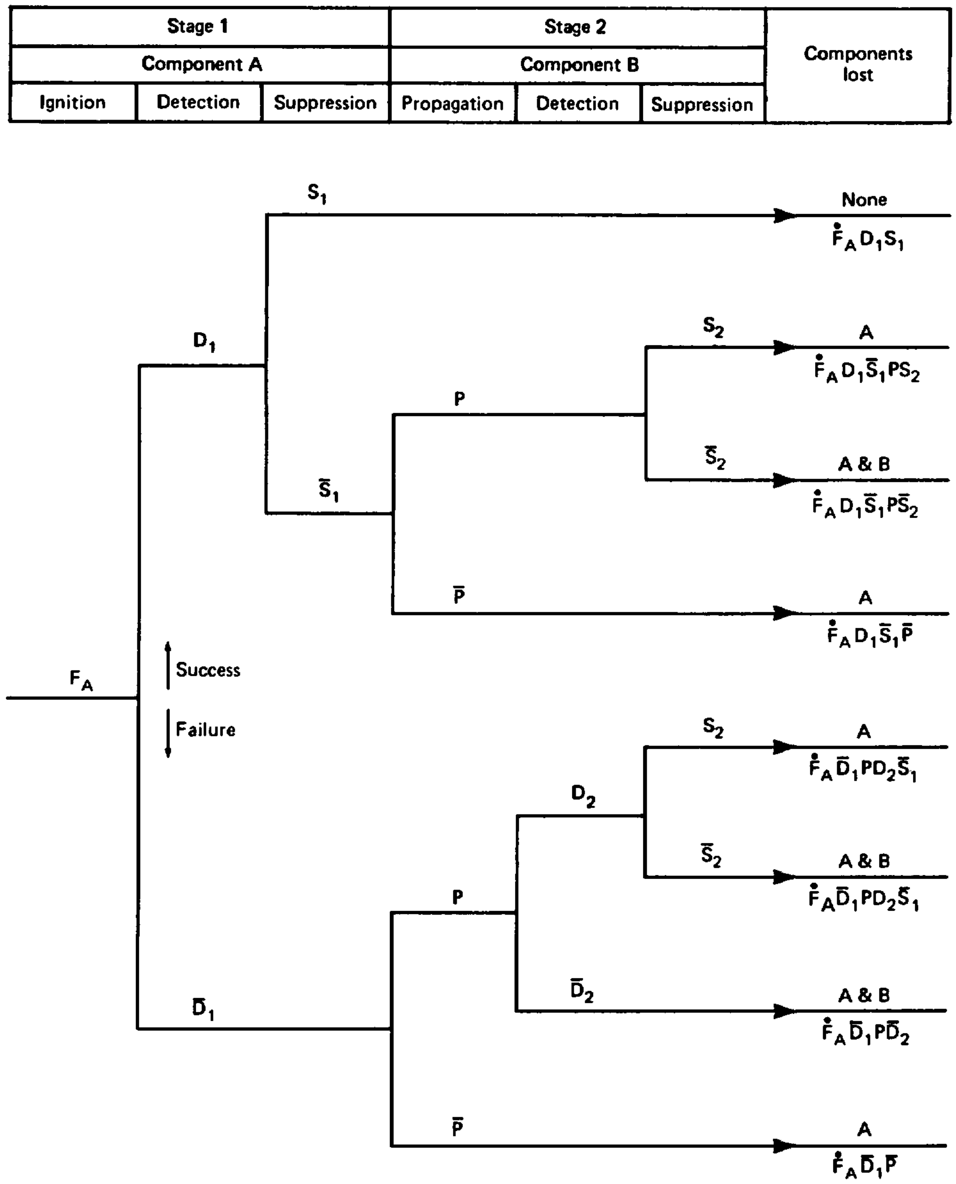


Figure 11-11. Illustrative two-stage event tree for two redundant components.

sometimes employing fault trees, are used to quantify the conditional branching probabilities of the event tree. Plant-design specifics, particularly for the detection and suppression elements, are accounted for in the submodels, as warranted. Much effort is placed on establishing the reliability of the fire-protection system (Moelling, 1979). Automatic and manual means of detection are included in the detection submodel.

Fire-Propagation Analysis: Method 3

In this approach, fire growth and suppression are viewed as competing time-dependent processes (Siu, 1980; Siu and Apostolakis, 1981). One or more representative fire-growth scenarios are developed for each location, depending on the physical configuration of the area. The distribution for the analyst-defined characteristic spread time is then compared against the distribution for the suppression time, to obtain the conditional frequency of fire growth, given the fire scenario.

For example, assume that two horizontal cable trays, one stacked over the other, contain critical power and control cables. In the representative fire scenario, a fire initiated on the lower tray spreads to the upper one in t_g minutes. The mean fire-suppression time is t_s minutes. Note that t_s includes the time to detect the fire, which often requires human response. Therefore, the distribution of the fire-spread frequency is the distribution of the frequency with which t_s exceeds t_g. The fire-spread time is computed by using physical models, while t_s is estimated from statistical data; their distributions describe the state of knowledge concerning fire processes.

The keys to this approach are the explicit use of simple physical models for fire (Siu, 1980), which enables the analyst to properly account for the extremely strong dependence of fire behavior on the physical configuration of the fuel bed and its surroundings, and the consistent treatment of the large uncertainties in the model outputs.

The simple physical model (Siu, 1980) is used to calculate the heat transferred from a fire to its surroundings, the time to ignition or damage for affected materials, and the subsequent rate of fire growth. Its predictions are subject to uncertainty, of course, because of statistical uncertainties in the behavior of fires, uncertainties caused by basic modeling assumptions, and uncertainties in the numerical values of the input parameters. The last-named source of uncertainty is propagated through the model by response-surface techniques, and the statistical uncertainties are often left unquantified, since they are generally dominated by the state-of-knowledge uncertainties. To treat the basic modeling uncertainty, the output of the model is treated as an expert's opinion, and a probability distribution for the accuracy of the model is constructed, based on available data and the judgment of the analyst.

The physical model of Siu (1980), called the "deterministic reference model," or DRM, was used in the Zion (Commonwealth Edison Company, 1981) and Indian Point (PASNY, 1982) studies. It focuses on predicting radiative and convective heat transfer from a fire to an object. This object may be another portion of the fuel bed, a noncombustible component, or even a fire barrier. In the last case, the heat flux leaving the barrier is also computed, to determine the effect on objects that are not directly exposed to the flames. Using a simple ignition- or damage-threshold temperature criterion, the impact of the fire on its surroundings is then computed as a function of time.

For many configurations, the DRM consists of only a few equations, and the needed calculations can be done by hand. For more complex configurations, where the interaction of several burning fuel elements is important, the computer code COMPBRN (Siu, 1980) is useful.

COMPBRN was developed to analyze fairly general fire-growth scenarios in a compartment. Widely varying configurations, fuel types, initiating-fire characteristics, and room properties can be studied. Essentially, the primary limitation on the complexity of the simulation is economic, since the storage requirements of COMPBRN increase greatly with the number of fuel elements modeled. COMPBRN has been used to model a number of small experimental fires, with generally good results (Siu, 1980). Its predictions for very large fires approaching flashover may be subject to greater uncertainties.

The distribution for the fire-suppression time is estimated in the Zion and Indian Point studies from information presented by Fleming et al. (1979). This distribution is somewhat dependent on the size of the fire, the degree of dependence being assessed judgmentally.

Because the behavior and the effects of fire do depend strongly on the layout of the location of interest, the physical modeling of method 3 appears to be the favored approach for modeling fire propagation. Uncertainties in the estimates obtained with models like COMPBRN will decrease when the models are upgraded to reflect experimental results and advances in fire research field.

11.3.3.2 Plant-System Analysis

Once the frequencies of fire-induced component losses are assessed, it is possible to estimate the frequency of fire-initiated accident sequences leading to core damage.

As with other initiating events, separate event trees may be constructed for fires because the operator, rather than automatic actions, may be responsible for shutting down the plant in response to a fire. Often the analyst simply modifies the front end of existing event trees for other initiating events to specialize them for fires. The postulated fire can be assumed to coincide with another initiating event, in which case the original event-tree structure would be retained. The conditional branching probabilities would be altered to reflect the dependence on the fire. However, if fires are to be treated as a separate event, care should be taken that data from which basic component-failure rates are determined do not double-count these failures from fires.

As noted above, human intervention plays an important role in the accident. Not only can the operators extinguish the fire and operate equipment manually, they may make repairs and jury-rig replacement equipment as well. Then again, they may be misled by fire-caused faulty information and may actually exacerbate the situation. A variety of operator actions are modeled in the Zion and Indian Point fire-risk analyses, depending on the scenario considered, but the modeling is extremely crude at this stage. Some other issues that have to be addressed in the analysis of fire-induced accident sequences are smoke propagation, effects of fire-suppression activities, fires outside the plant, and the failure of fire barriers.

The quantification of accident sequences involving fires follows the lines described in the more general discussion of accident-sequence quantification; the interfaces are described in Section 10.3.6. Particular attention must be paid to the intersystem dependences introduced by fires. Fires as causes of component failures may be included as "house" events directly in the system fault trees as a function of location and size. System reliabilities can then be evaluated conditional on the occurrence of the postulated fire.

Another point to be emphasized is that the uncertainties in the analysis of fire frequency and propagation must be combined with the event-tree uncertainties. A rational comparison of various sources of risk (e.g., fires, floods, hardware, etc.) requires the recognition and consistent treatment of uncertainties.

Once the sequences involving fires are delineated and frequency distributions quantified, the assessment of plant-system response proceeds as with other initiating events. Besides direct impacts on system components, fires have other deleterious impacts: the flooding that results from attempts to extinguish the fire; smoke, which may hinder personnel access; the generation of ignition sources for other flammable products; the possible boiling of water inside pipes passing through the fire; and the like. These impacts have not been addressed in detail by any of the fire studies to date. The dependences of fire as a secondary event to some other external event (e.g., fire initiated by an earthquake) have also not been evaluated, although the methods described here and in Chapter 10 are fully applicable to these dependences.

11.3.3.3 Release-Frequency Analysis

The purpose of this analysis is to derive the distributions for the various categories of radionuclide releases from the containment. The comments of the preceding section apply here as well, although the end result of the accident sequences is release rather than core damage. The release-category analyses should take into account that the same fire that damaged the reactor core may well have damaged containment mitigating functions also. A careful investigation of the entire accident sequence, and not just the portion following core damage, is required.

11.3.4 INFORMATION REQUIREMENTS

The information required to perform a risk analysis of fires can be summarized as follows:

1. Description of plant systems, including the location of components and systems within structures. Especially important are routings for safety-related power and control cables.

2. Fire-protection report, which contains information on transient and permanent fuel loadings, suppression systems, ventilation systems, and safety-equipment inventories for each fire zone as well as a simplified FMEA for some zones.

3. Reports on the fire qualification of components. Physical data for electrical cables and trays are very useful.

4. Results of the plant-system analysis for internal initiating events, especially accident-sequence descriptions. Accident-sequence frequencies are also useful for screening purposes.

5. A compilation of licensee event reports involving fires at nuclear power plants.

11.3.5 PROCEDURE

The methods described in Section 11.3.3 are summarized below in the form of a task-by-task procedure for performing a risk analysis of fires.

<u>Task 1: Fire-Hazard Analysis</u>

1. Construct simple systems model of plant.

2. Identify locations of safety equipment.

3. Identify critical fire-impact locations, using a simple FMEA.

4. Identify locations adjacent to critical locations containing large quantities of combustibles.

5. Evaluate the distributions for fire frequency for each location.

<u>Task 2: Fire-Propagation Analysis</u>

1. Define representative fire-growth scenarios for each location.
2. Determine distribution for fire-growth time for each scenario.
3. Determine distribution for fire-suppression time for each scenario.
4. Compute distribution of frequency of growth.

<u>Task 3: Plant and Systems Analysis</u>

1. Develop event- or fault-tree logic that links component damage to one or more core-damage states.

2. Apply component and system failure boundary conditions to the event- or fault-tree logic.

3. Develop the distributions for the frequency of fires resulting in each core-damage state.

<u>Task 4: Release-Frequency Analysis.</u> Proceed as in Task 3, but carry out to release categories.

<u>Task 5: Iterate.</u>

11.4 RISK ANALYSIS OF FLOODS

11.4.1 INTRODUCTION

This section describes methods and procedures for assessing the consequences of reactor accidents involving external or internal floods. The methods and procedures described here should be used in conjunction with the external hazards screening criteria described in Chapter 10 and are conceptually similar to those presented in Sections 11.2 and 11.3 for the analysis of seismic and fire risk, respectively.

In comparison with some other external hazards, particularly fires and earthquakes, floods have received less attention as a potential cause of reactor accidents in the PRA studies carried out so far. As a consequence, there are no well-established methods for the analysis of either external or internal floods. The implied perception is that floods are less likely than fires and earthquakes to induce accidents that might contribute significantly to the overall risk of a nuclear plant. This perception is supported by the view that the flood-protection measures required for licensing have resulted in extremely low frequencies of floods that produce significant damage. In addition--and this is especially true of most external floods--even when floods of extreme severity are postulated to occur, there should often be ample warning time to safely shut down the reactor before significant damage in important systems and structures can occur.

However, there are several reasons for not excluding floods as potentially important risk contributors in PRA studies. First, there are large uncertainties in the estimated frequencies of external floods of extreme severity and in the associated fragilities of plant structures and components. Second, some causes of flooding, such as the failure of an upstream dam or a large rupture, inside the turbine building, in the circulating-water system may not provide significant warning time. Third, many of the design and operational features required to protect against external floods may not provide the same degree of protection against internally initiated floods. In fact, the experience with flooding at nuclear power plants indicates that internal floods may have a relatively greater potential to cause a reactor accident with nonnegligible risk. For example, Table 11-6, which is taken from a report by Verna (1982), summarizes the U.S. nuclear experience with turbine-building floods from internal sources. One of the more serious of these internal flood events is described in the paragraph that follows.

In June 1972, at Quad Cities Unit 1, a rupture in the circulating-water system caused the rapid flooding of a room containing a number of pumps in different systems. The equipment damaged by the flood included four service-water pumps for residual heat removal, two diesel-generator cooling-water pumps, four condensate-booster pumps, and three condensate-transfer pumps. In addition, the floor-drain sump pumps, the hypochlorite system analyzer, and condensate-pressure gauges were damaged. Although the reactor was not damaged, the impact of this flood in terms of the failure of multiple components and systems was extensive. Modifications were made at Quad Cities to enhance the physical separation of the safety-related pumps and thus protect against the recurrence of a flood in the same room.

Table 11-6. Turbine-building flooding in U.S. nuclear power plants[a,b]

Date of occurrence	Plant	Affected safety component	Spill rate	Remarks
		SOURCE: SERVICE WATER		
June 1975	Surry 2	Service-water valve		Pump developed seal leak
October 1977	Surry 2	Service-water valves of all redundant trains		Personnel forgot to close valves that were opened for maintenance
October 1978	E. Hatch 1	Service-water valve		Valve body blew out during repair
October 1979	Dresden 2	Diesel-generator control cabinet		Fire-water leak
		SOURCE: CONDENSER CIRCULATING WATER		
January 1979	Crystal River 3		Large	Solenoid valve failed open and led to flooding
April 1977	Three Mile Island		Large	Circulating-water pump casing split 360°
October 1976	Oconee	Emergency feed-water pumps	Large	Pneumatic isolation valve opened when condenser manhole was open and spilled lake water into turbine building
October 1978	Surry 2	Service-water valves	Small	Intentionally flooded during maintenance
June 1972	Quad Cities 1	Many redundant and diverse safety-related components	Very large	Valve closed inadvertently, and water hammer ruptured expansion joint

[a]Data on incidents from the start of commercial power operation up to July 1981.
[b]From Verna (1982).

As a result of NRC followup, various modifications were also made at 10 other plants to enhance protection against the flood-induced loss of safety functions (Verna, 1981).

A similar flood occurred at Three Mile Island Unit 1 in April 1977; it was caused by a leak in the circulating-water system at the casing of one of the circulating-water pumps. However, because of the plant's layout, damage was confined essentially to the six circulating-water pumps and did not affect any other systems (Verna, 1981).

These events and other incidents involving flooding indicate that, at least for certain nuclear power plants, internal floods may be an important cause of multiple, dependent failures. It is also apparent that differences in design features, such as provisions for physical separation, and plant layout can give rise to significant differences in the plant's response to the same flooding condition.

In summary, there are a number of reasons why flooding from external and internal causes should be considered for analysis in PRA studies. The operating experience of reactors includes floods that have resulted in the coincident loss of multiple components and even multiple systems. Attempts to estimate the frequency of severe external floods have resulted in the identification of large uncertainties. Finally, flooding is one of the failure mechanisms associated with common-cause failures--that is, multiple concurrent failures due to the same cause--which have long been recognized as an important factor in risk assessment (see Section 3.7).

Because of the relatively low emphasis given to external events in general and to floods in particular, the guidance given here on assessing the risk associated with flooding does not benefit from the same degree of experience in the development and application of PRA methods as that set forth for transients and events that initiate LOCAs. Among the external events, floods rank behind earthquakes and fires in terms of the PRA-relevant work that has been carried out.

This is not to say, however, that the current state of the art is insufficient to quantify the risk associated with flooding. As will be shown in this section, the analysis of such risk can be structured around the same basic hazard-fragility-systems approach that has been successfully applied to earthquakes, missiles, and other external events. In addition, some of the methods used in location-dependent common-cause analyses of fires are applicable to floods as well. The existing methods for calculating the frequencies of flood-induced accident sequences (especially for external floods) have resulted in wide uncertainties. It is recommended that, regardless of the magnitude of the uncertainty, the PRA should include a flooding analysis and attempts should be made to quantify the effects of uncertainties to the extent that this can be done. For these reasons, this section of the procedures guide was set aside for the risk analysis of flooding.

11.4.2 OVERVIEW

The probabilistic analysis of reactor accidents involving flooding can be viewed as a problem in determining $f_k(\underset{\sim}{z})$, the unconditional frequency of exceeding damage level $\underset{\sim}{z}$ of consequence type k, resulting from potential reactor accidents initiated by floods.

It is convenient to expand the external event risk equation (Equation 10-1) to the following form for the risk analysis of floods:

$$f_k(\underset{\sim}{z}) = \int_{\mathcal{D}} \cdots \int \sum_{j=1}^{J} \sum_{\ell=1}^{L} f_{E,\ell}(\underset{\sim}{y}) f_{S,j|E,\ell} f_{k|S,j}(\underset{\sim}{z}) h(\underset{\sim}{x}) d\underset{\sim}{x} \qquad (11\text{-}34)$$

where $f_{E,\ell}(\underset{\sim}{y})$ is the frequency of flood-damage state E_ℓ given response $\underset{\sim}{y}$ to flood level $\underset{\sim}{x}$, $f_{S,j|E,\ell}$ is the frequency of accident sequence S_j given flood-damage state E_ℓ, and the quantities $f_k(\underset{\sim}{z})$, $F_{k|S,j}(\underset{\sim}{z})$, and $h(\underset{\sim}{x})\, d\underset{\sim}{x}$ are defined as in Equation 10-1.

Note that the magnitude ($\underset{\sim}{x}$) and response ($\underset{\sim}{y}$) are vectors to accommodate the multivariate aspect of floods. For example, a single flood event, such as a hurricane, can produce multiple effects, such as winds, wind waves, and high water levels.

Certainly, all the above defined parameters should be expressed by a probability distribution, depicting the uncertainties in the estimation processes and underlying data. These distributions can be derived by the methods given in Chapter 12 on the treatment of uncertainties.

Depending on the range of possible damage resulting from the causes of flooding under consideration, the term $\underset{\sim}{z}$ in Equation 11-34 could be either a discrete level of plant damage (e.g., core melt), a measure of the magnitude of a release of radionuclides, or an estimate of the health effects expected in the population at risk.

Risk analysis for flooding is performed along lines similar to those followed for other external events like earthquakes and fires. The steps include a flooding-hazard analysis, a fragility and vulnerability evaluation, a plant and system analysis, and a release-frequency analysis. A flooding-hazard analysis consists of identifying the site-dependent causes (e.g., dam failures) and associated failure mechanisms (e.g., submersion) and estimating the flood-hazard intensities $h(\underset{\sim}{x})$ for each flooding variable. Uncertainties in estimating the hazard intensities can be expressed by providing a family of curves, each with a state-of-knowledge probability assigned in the same manner as is done in seismic analysis.

The task of estimating the frequency of various flood-damage states for each failure mechanism, $f_{E,\ell}(\underset{\sim}{y})$, is referred to as the "analysis of flooding fragility and vulnerability." It entails the definition of a suitable set of flood-damage states that may include, for some external floods, damage to plant structures. In this case, it is necessary to estimate the frequencies of structural failures and associated state-of-knowledge probabilities that describe the level of uncertainty. The method of arriving at these estimates is much the same as that used in the seismic risk analyses described in Section 11.2, except that different failure mechanisms should be addressed, as will be explained below. If components are submerged, the fragility might reduce to a simple step function with a transition in the frequency of failure from zero to unity as the flood-height parameter reaches or exceeds the elevation of the component.

The tasks of plant and system analysis and release-frequency analysis are performed to complete the risk assessment of flooding and to quantify the terms $F_{S,j|E,\ell}$ and $f_{k|S,j}(\underset{\sim}{z})$. This is where the connection is made between the elements of the risk modeling that are unique to flooding and those generic to all the other initiating events analyzed in a PRA study, such as transients and LOCAs. Stated another way, the above terms include the event- and fault-tree logic that relates the various states of flood damage to the public risk from accidental releases of radioactivity. The details of this interface are discussed in Section 10.3.6.

If the particular flood in question has only plant-hardware implications and does not influence the calculated offsite consequences, as would happen in the case of an internal flood confined to a room, it may be desirable to carry out the flood risk analysis only to some intermediate "pinch point." One convenient pinch point is the frequency of occurrence of core damage or melt. However, if the flood is seen to influence the transport of radioactive material or to hinder the evacuation of people, an evaluation of flood-specific environmental consequences may be more appropriate.

It should be emphasized that a comprehensive evaluation of the risk from flooding--one that includes a complete quantification of each term in Equation 11-34--has not yet been carried out in a PRA for a nuclear plant. However, a substantial amount of technical work has been done in quantifying various elements of Equation 11-34 from which to synthesize an overall method for quantification; this work is summarized in Section 11.4.3. It is clear that more developmental research, as well as attempts at application, is necessary to bring the level of flood-risk analysis to the current state of the art for the analysis of seismic and fire risks.

Nonetheless, it would be a mistake to forego the inclusion of floods in a plant-specific risk analysis just because of the formative state of the methods, including those used for the probabilistic quantification of uncertainties. A relatively undeveloped method for quantification simply gives rise to greater uncertainties. From the perspective of enhancing design and licensing decisionmaking through risk analysis, the decision to include or exclude a candidate risk contributor should be based solely on its perceived contribution to risk. As already argued in Section 11.4.1, there is insufficient evidence to dismiss flooding as a potential risk contributor on a generic basis.

11.4.3 METHODS

A flood-risk analysis follows the general procedure described in Chapter 10. It consists of a flooding-hazard analysis, a component-fragility evaluation, a plant and system analysis, and a release-frequency analysis. Certain details, particularly in the areas of hazard and fragility analysis, are different, depending on whether the flood results from external or internal causes. Before describing these differences, it is instructive to briefly review the relevant literature and to set out the criteria for an acceptable probabilistic analysis of flood-induced accidents.

11.4.3.1 Relevant Literature

The first comprehensive assessment of accident risks in U.S. commercial light-water reactors, the Reactor Safety Study (RSS), considered certain external events. Unfortunately, the assessment of floods, as can be inferred from the presentation of the results (USNRC, 1975), did not include a quantification of this risk contributor, nor did it include a quantification of any of the terms in Equation 11-34. The qualitative assessment did, however, provide some insights of interest here. It showed, for example, that the basic PRA methods employed in the Reactor Safety Study are applicable to floods as well as to other external events. More specifically, the topology of accident sequences displayed in the RSS event trees was found to be applicable to floods. It also indicated that the frequency (referred to in the Study as "probability") of a core melt induced by floods at a river site should be expected to be extremely low--because structures enclosing essential safety-related equipment are specifically designed to survive a hypothetical flood called the probable maximum flood (PMF). Because of the conservatism in the way in which the PMF is calculated, it is argued that the frequency of the PMF is very low, as is the conditional frequency of failure for the associated structures. The qualitative assessment obtained from these calculations indicates that floods represent negligible risks. A similar conclusion and rationale are presented for coastal sites subjected to impulsive generated waves (tsunamis) and wind waves or the high water levels, waves, winds, and erosion due to tropical storms (hurricanes) and extra-tropical storms.

Wall (1974) has reviewed methods for estimating the frequency (return period) of floods at a river site. The approach emphasized in his paper is the statistical analysis of river-discharge data by means of various curve-fitting techniques. Different types of distributions are fitted to the same 44 years of river-discharge data, including log-Pearson type III, lognormal, and one of the extreme-value distributions, which is fitted by using maximum-likelihood estimators. Wall noted that excessive extrapolations of these curves beyond the range of the data would be required for an estimate of the frequency of floods approaching the magnitude of the PMF established for the site in question. An evaluation was made of the consequences of the PMF in terms of the warning time, damage to offsite-power supplies, and the role of watertight barriers. Wall concluded that the risk of a serious reactor accident due to rising water levels is negligible and proposed that the design-basis flood be redefined as that having a frequency of exceedence of 5×10^{-4} over the next 50 years, or 1×10^{-6} per reactor-year.

In addition to the need for excessive extrapolation, a major problem with estimating river-flood frequencies from historical data is the possibility that the historical data may have been rendered inapplicable by natural or man-made alterations to the hydrologic characteristics of the drainage basin. This observation has recently led to the suggestion of an alternative approach of calculating flood levels as a function of sequences of natural events and other factors relevant to flood levels (D. W. Newton, unpublished work). The statistical analyses are then performed on the subordinate events--intensity and duration of precipitation, relative sequence of successive storms, snowpack, temperature variations, and the like. Examples of an application to a sequence of rainstorms have been given by Alexander (1963). A model for predicting the frequency of hurricanes was developed by Mogolesko (1978). One of the limitations of Newton's approach is that statistical independence among the flood variables, often assumed in these applications, cannot be ensured without an extensive statistical analysis. For example, if successive rainstorms tend to cluster together in time, this assumption will be violated. The problem of changes in site hydrology affecting the applicability of historical flood data was also addressed in the development of the NRC's FLOE code, in which the curve fits to the data are subjectively modified by expert opinion applied with a Bayesian updating procedure.

One cause of river flooding that needs to be treated separately is the failure of an upstream dam. The frequencies and risks of earthquake-induced dam failures in California have been estimated in a study performed at the University of California at Los Angeles (Okrent et al., 1974).

Wagner et al. (1980) developed a method for estimating the effects of a river flood on the availability of systems. A computer code called NOAH was developed for estimating the fragility of a system--that is, the conditional frequency of failure--given the submergence of equipment at various flooding heights. The method was applied to the auxiliary feedwater system of Surry Unit 1, one of the plants analyzed in the Reactor Safety Study. The frequency of the initiating flood was not assessed in this study, but rather treated parametrically. It was found that, if the frequency of the postulated flood is assumed to exceed 10^{-4} per year, significant increases result in the estimated frequency of some of the dominant accident sequences identified in the Reactor Safety Study. Although several aspects of the flood equation (Equation 11-34) were not included--such as flood-hazard analysis, flood failure mechanisms other than submergence, and the quantification of uncertainties--this study is the best risk analysis of external floods performed so far, particularly with regard to how the terms $f_{E,\ell}(y)$, $F_{S,j|E,\ell}$ and $f_{k|S,j}(z)$ should be estimated.

In summary, the literature on flood-risk analysis includes statistical analyses of phenomena that contribute to floods, some qualitative assessments that indicate a low degree of risk from floods, and for external floods a computer-aided method for analyzing the location-dependent aspects of system fragilities. Although the literature does not yet include a comprehensive risk analysis of floods, several such studies are under way, including those for Midland and Oconee. There has been sufficient progress in specific elements of the methods to set forth criteria for an acceptable assessment of flood-induced accidents. These are presented below.

11.4.3.2 Acceptable Methods

In view of the current state of the art of PRA in general and flood-risk analysis in particular, it is recommended that the methods for analyzing the risk of flood-induced accidents meet the following criteria:

1. The methods should provide reasonable assurance that all sources of flooding, both external and internal, that are applicable to the site have been considered. Internal causes include leaks and breaks in major water systems, the overfilling of tanks, sump-pump malfunctions, and the backing up of drains. External causes include river flooding, dam failure, excessive precipitation, hurricanes, tsunamis, seiches, wind waves, and surges. Special attention should be paid to flood-protection provisions, their failure probabilities, and possible methods for flood termination (important to internal floods).

2. The methods should ensure completeness in the coverage of all relevant mechanisms of failure for structures and components that could affect risk. The following mechanisms should be considered at a minimum: loss of structural integrity through collapse, sliding, overturning, ponding, excessive impact and hydrostatic loads; flooding and wetting of equipment from seepage through walls and

roof; flow through openings; sprays, thermal shocks, missile impacts; and the blockage of cooling-water intakes by trash.

3. The analysis of flood frequency and flood-induced damage to plant structures, components, and systems must be properly integrated with the definition of accident sequences in the event trees, taking due account of the dependences associated with the flood. These dependences include the beneficial effects of warning time and the detrimental effects of increases in the frequencies of multiple concurrent failures.

4. In estimating the fragilities of structures and components, the failure criteria should be based on realistic assumptions and should not be considered synonymous with design limits.

5. In estimating the offsite consequences of flood-induced accidents, the impact of flood conditions on radionuclide transport and on evacuation should be considered. Regional emergency influences on the plant should also be taken into account, such as the loss of offsite power, the loss of communications, the loss of access, and various human factors. This is especially important for major floods from external causes.

6. The methods should ensure that all sources of uncertainty in the risk estimates are identified and their effects quantified if possible, including the uncertainties associated with sparse or inadequate data, uncertainties in the models used to calculate flood variables, uncertainties and variabilities in the failure limits of components and structures, uncertain increases in component-failure rates in abnormal flooding environments, and other uncertainties associated with risk estimation (see Chapter 12).

7. There are unique aspects of human interactions that must be taken into account in flood-risk analysis. They include the effects of warning time, if any, to shut the plant down, the conflicts between flood mitigation and plant operations, and the effects of stress.

11.4.3.3 Flooding-Hazard Analysis

The objective of a flooding-hazard analysis is to establish the relationship between the frequency and the magnitude of each flood variable to be analyzed. In the context of the flood-risk equation (11-34), this entails the development of the flood-hazard intensities $h(\underset{\sim}{x})$. Since the approaches to quantification are somewhat specific to the various causes, the methods used for external and internal floods are discussed separately.

External Floods

The first step in analyzing the hazards of external floods is the selection of the causes of flooding and the appropriate flood variables for the site. An exhaustive list of external flood causes is normally found in Chapter 2 of the safety analysis report in the section on hydrology. Depending on the site, these causes may include the following:

1. River flooding.
2. Upstream dam failure.
3. Failure of dikes and levees.
4. Tsunamis.
5. Surges.
6. Seiches.
7. Wind waves.
8. Precipitation.
9. Snow melt.

Item 2, upstream dam failures includes all secondary causes (e.g., earthquakes, overtopping, antecedent dam failures), and item 8, precipitation, includes hurricanes and sequences of storms.

Of the variables associated with a flood that can be related to an assessment of the damage to plant structures, flood height is the most important since little flood-induced damage can be postulated unless the flood height exceeds some minimum level. One possible exception is the blockage of cooling-water intakes with trash, which could result from floods of moderate heights. This minimum level might be the grade elevation of the plant structures, the minimum level at which offsite-power supplies might be damaged, or the minimum level at which a flood-protection system would fail. Certainly, for the latter case the failure of the flood-protection systems should be expressed probabilistically. Hence, a risk quantification of external floods would usually be expected to include an assessment of the hazard curve for flood height.

At least part of the flood-height distribution can be estimated by a statistical analysis of data. In the case of river sites, data are usually analyzed in terms of flow rate instead of height since rarely are data available at the precise location of the reactor site along the river. An example of data on maximum annual flows at a river site is given in Table 11-7 (Wall, 1974). In Figures 11-12 and 11-13, distributions are fit to the data, Figure 11-12 showing the lognormal and log-Pearson type III distributions, and Figure 11-13 showing one of the extreme-value distributions. The differences among the various fits within the range of the data (i.e., for river flows of less than about 50,000 cfs) are negligible in comparison with the magnitude of uncertainties normally encountered in risk assessment. However, when each of these curves is extrapolated for the river flows predicted to exceed the design-basis flood at a frequency of 10^{-4} per year, the following results are obtained (Wall, 1974):

Type of fit	Flow (cfs)	Water elevation (ft)
Lognormal (fitted by eye)	100,000	922
Log-Pearson type III	71,000	918
Extreme value (fitted by maximum-likelihood method)	77,000	919

Table 11-7. Maximum daily discharge at St. Cloud, Minnesota, for water-years 1927 to 1970[a]

Water-year	Flow (cfs)	Water-year	Flow (cfs)
1927	16,885	1949	8,857
1928	13,055	1950	31,920
1929	15,910	1951	19,717
1930	17,076	1952	37,900
1931	10,371	1953	23,110
1932	8,334	1954	20,500
1933	13,409	1955	13,370
1934	5,707	1956	18,460
1935	6,531	1957	19,620
1936	12,139	1958	6,609
1937	13,239	1959	16,375
1938	24,840	1960	14,290
1939	15,598	1961	9,860
1940	14,199	1962	25,500
1941	20,755	1963	12,345
1942	20,835	1964	15,570
1943	27,374	1965	46,780
1944	25,400	1966	26,350
1945	24,216	1967	23,253
1946	26,275	1968	17,746
1947	18,574	1969	39,366
1948	19,286	1970	19,780

[a]From Wall (1980).

Hence, the extrapolation of the fitted curves to a 10^{-4} exceedence frequency results in a difference of as much as 30 percent in the river flow rate, but an increase in water elevation of only 3 feet. Although a difference of 3 feet may be small, such a difference could be important in view of the threshold effect of rising water levels.

The grade elevation for the plant at the reference site is 935 feet, and the water level corresponding to the probable maximum flood was set at 939.2 feet (365,000 cfs). Clearly, the extrapolation of the fitted curves in Figures 11-12 and 11-13 to these extreme flood levels would be excessive. Hence, the curve-fitting techniques alone are in most, if not all, cases insufficient for estimating the hazard frequencies associated with floods exceeding the design-basis floods. The tails of the distribution must be estimated by using some sources of information other than the flood data.

One approach to estimating the tails of the hazard curve is to estimate the total frequency of the sequence of events that is used to define

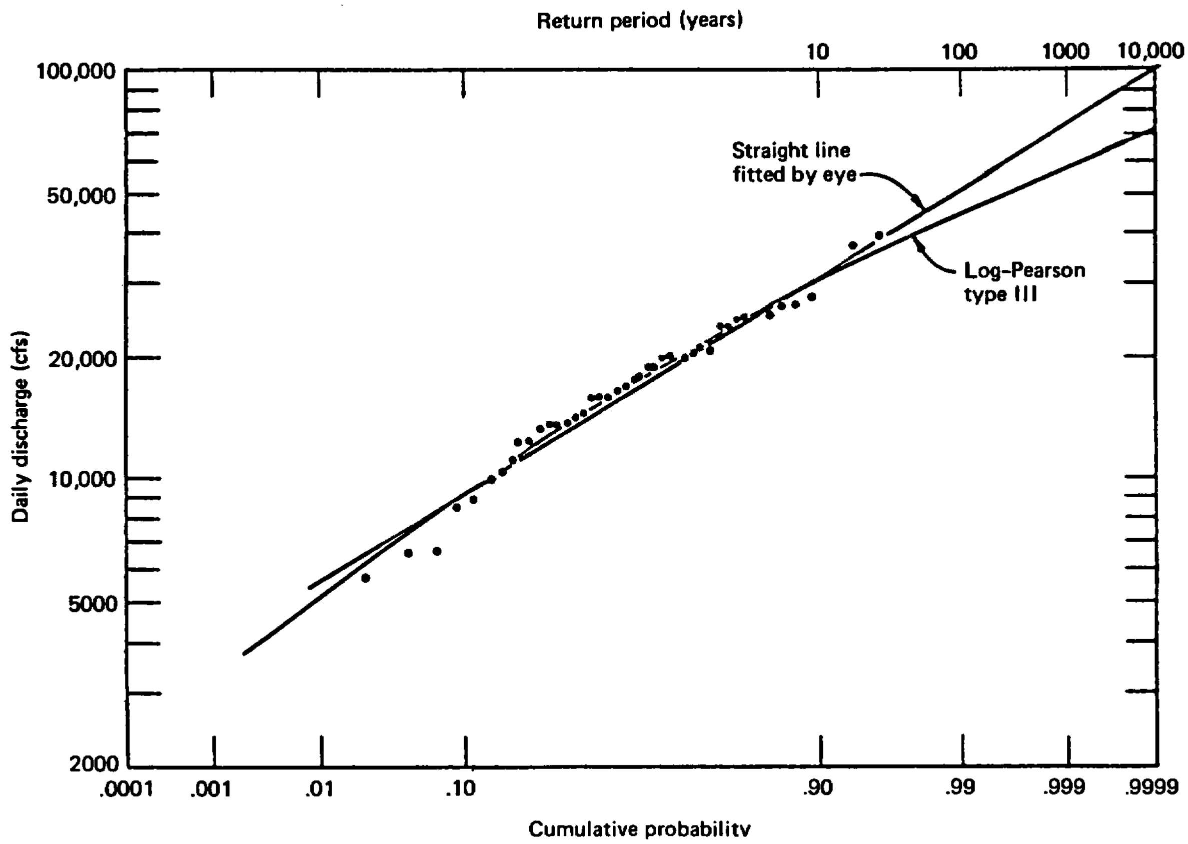


Figure 11-12. Flood data of Table 11-7 plotted on lognormal probability paper. From Wall (1974).

the probable maximum flood (PMF). For the reference site this estimate was made under the assumption of a 100-year snow cover, followed by the maximum historical temperature sequence and the occurrence of the probable maximum precipitation (PMP). The frequency of this sequence could be estimated by

$$\phi(\ell_i^*) = \phi(S)\ \Pr\{T|S\}\ \Pr\{R|S,T\} \tag{11-35}$$

where

$\phi(\ell_i^*)$ = frequency of exceeding the PMF.

$\phi(S)$ = frequency of exceeding the snow cover assumed in the PMF.

$\Pr\{T|S\}$ = conditional probability of a maximum temperature sequence or worse, given the snow cover S.*

$\Pr\{R|S,T\}$ = conditional probability of a PMP, given the sequence S,T.*

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*The temperature and precipitation must occur within a specified period of time to produce the PMF.

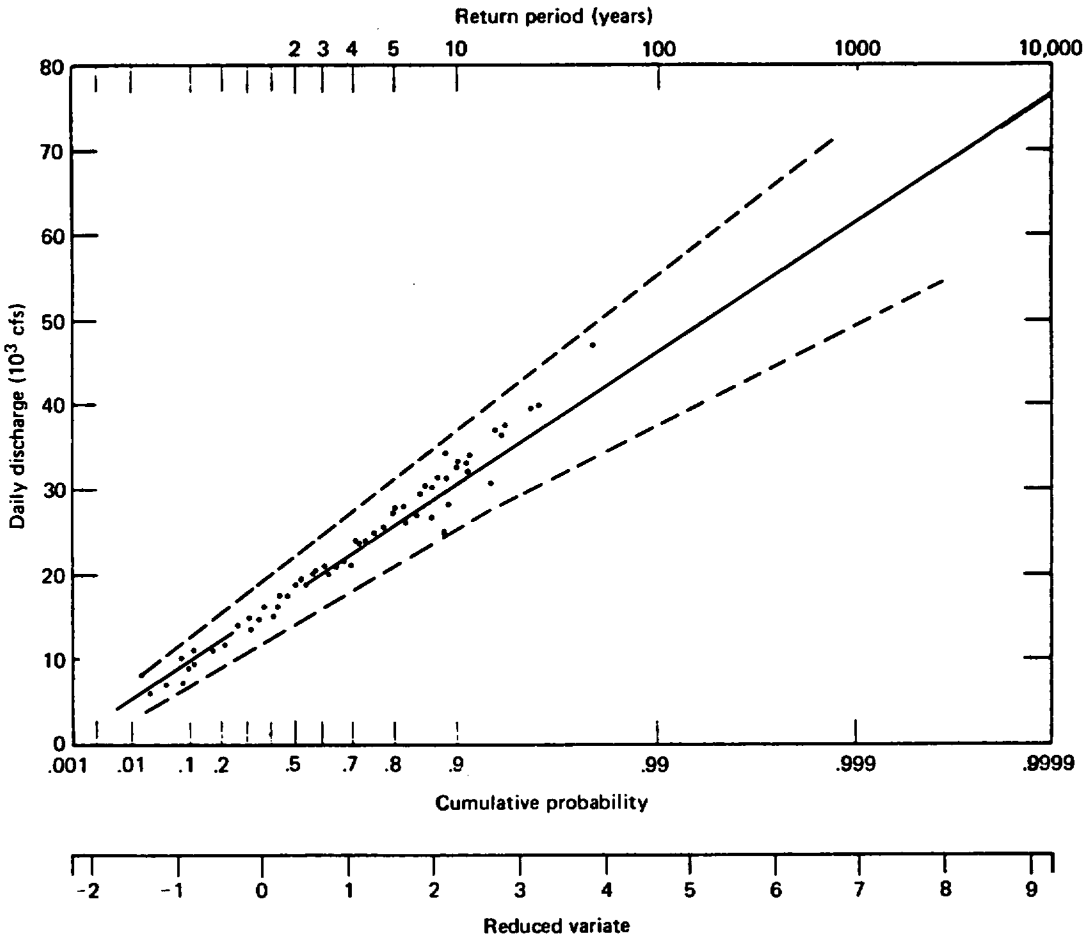

Figure 11-13. Extreme-value graph of flood data of Table 11-7 showing control curves. From Wall (1974).

Data on S, T, and R can be obtained, and the same type of statistical analysis can be performed as described above for river-flow data if it can be assumed that the snow levels, temperature sequence, and precipitation are statistically independent events (see footnote on page 11-80). Statistical independence of events in the sequence causing the flood may not be a good assumption, however, if the sequence includes multiple rainstorms occurring in succession.

Figure 11-14 shows the synthesis from statistical data of a flood-hazard distribution on the flood variable and an estimate of an extreme value on the tail, obtained by using the approach described above. An implied assumption is that there are no sequences of events other than the one used to define the PMF that would produce the flood magnitude ℓ_i^* at frequencies comparable to $\phi(\ell_i^*)$. Since the PMF is defined as the maximum flood resulting from a large number of combinations of candidate event sequences (USNRC, 1977), this is probably a good assumption. The method of synthesizing the two sources of information in Figure 11-14 is simply to draw a smooth curve connecting the fitted curve to the extreme point estimated from Equation 11-35. The indicated probability intervals represent uncertainties in developing the curve.

Sources of uncertainty include those associated with the curve fitting of the data; these can be calculated in terms of confidence limits on the parameters of the fitted line (see Figure 11-13). Similarly, confidence limits can be estimated for each term in Equation 11-35 and appropriately combined by moment-propagation, Monte Carlo, or discrete-probability-arithmetic methods (see Chapter 12) to obtain probability intervals for $\phi(\ell_i^*)$. Other sources of uncertainty that should be incorporated into the above intervals must be estimated subjectively. They include all sources of uncertainty not represented in the data base, such as the possible inapplicability of data because of changes in site hydrology and undefined sequences of natural events that might change the tails of the curve.

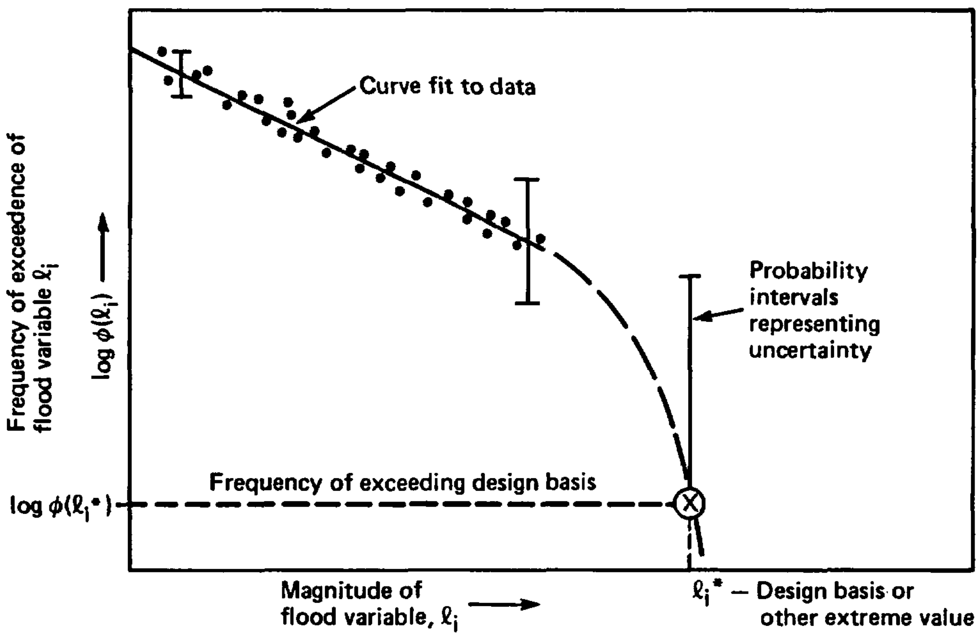

Figure 11-14. Results of a hypothetical hazard analysis of flood variable ℓ_i.

<u>Internal Floods</u>

The analysis of internal flood hazards is conceptually similar to that described in Section 11.3 for the analysis of fire hazards. The hazard analysis in both cases consists of a qualitative phase, in which specific cases are selected for quantification, and a quantitative phase, which provides an estimate of the frequency of exceeding various levels of magnitude. One of the most significant differences between floods and fires from the perspective of how the analysis is carried out is that the specific sources of flooding can be more easily and completely enumerated and floods are very likely to propagate to adjacent compartments, whereas fires are generally confined to rather small areas. These aspects are taken into account in the flood-location screening methods described below.

The identification of important locations must be made from two perspectives. It is necessary to identify both the <u>source locations</u> (i.e., the locations where floods are most likely to start) and the critical <u>impact</u>

locations (i.e., the locations where the existence of a flooding condition would have the greatest impact on the availability of key safety-related systems). Both types of location must be considered because of the possibility that the flooding will propagate from one location to another.

One method for ranking locations in terms of the impact of a flooding condition on the risk of reactor accidents is to perform, for the major plant systems, a special type of qualitative fault-tree analysis that takes into account the location of system components. This procedure, illustrated in Figure 11-15, starts by constructing a fault tree for the top event "core melt due to internal flood." The fault tree is developed under the assumption that a postulated flood causes a transient or an initiating event for a

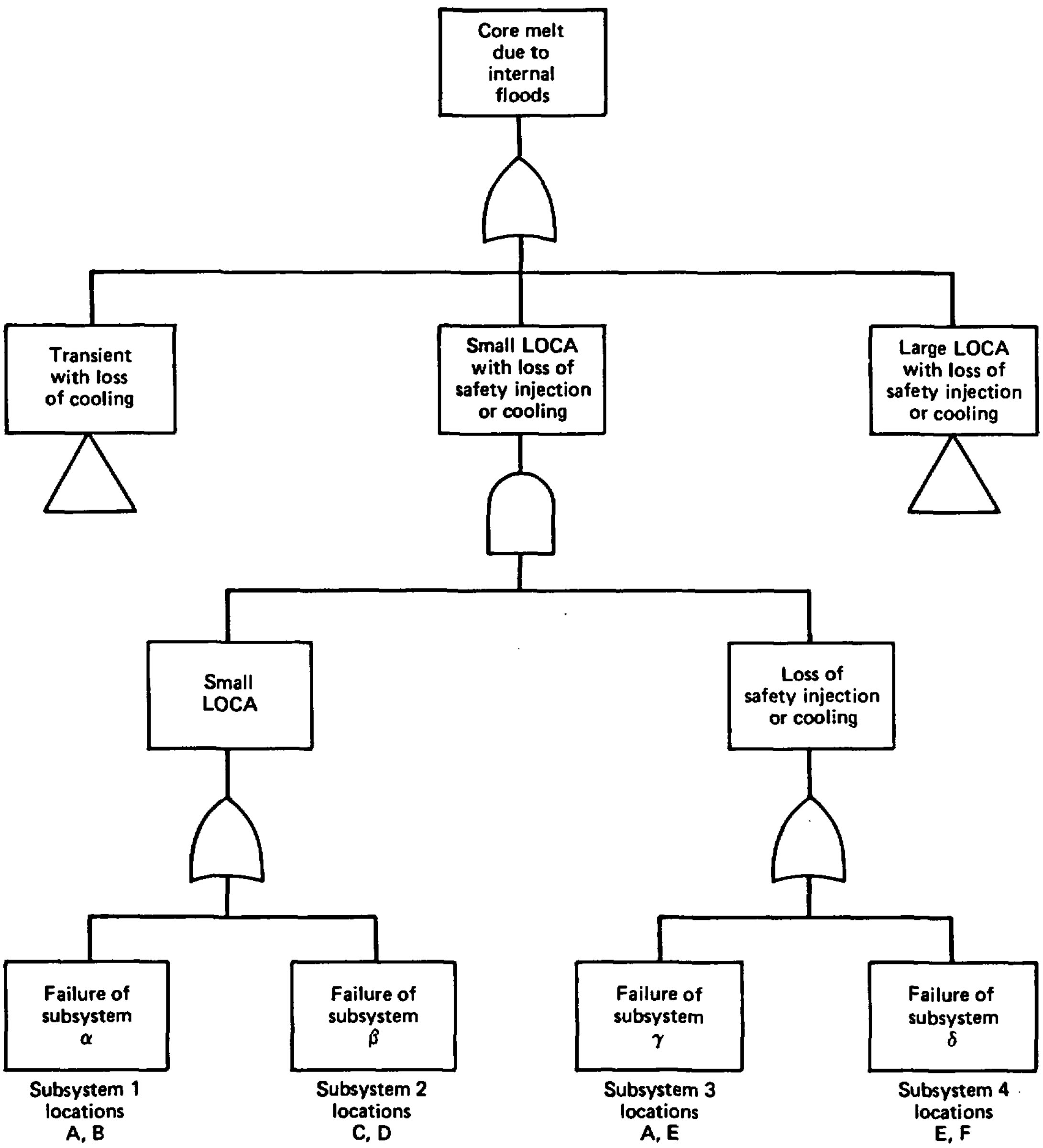

Figure 11-15. Fault tree for identifying important flood-impact locations. The triangles represent events not developed in this example.

small or a large LOCA, or that it causes the failure of a system or subsystem required to mitigate the initiating event, or a combination of these. The tree need be developed only to the level of major subsystems (e.g., high-pressure-injection train A) so that the singular effect of the loss of groups of components in specific locations can be resolved. It is necessary, however, to include support systems whose failure has a significant impact on the availability of the main safety-related systems.

The example of Figure 11-15 is a simplified illustration: the subtree for a small LOCA is developed to identify four subsystems denoted α, β, γ, and δ. In a practical application of this method, the tree would be much larger, and the subsystems would typically consist of redundant trains of components; hence, there would be more AND gates near the bottom of the tree. After the tree has been developed to the subsystem level, the locations of all components in each subsystem are itemized. In this simple example, the four subsystems are seen to have components in a total of six locations labeled A through F.

The next step in this approach is to determine the minimal cut sets of the fault tree, which in this case are

$$(\alpha,\gamma);\ (\alpha,\delta);\ (\beta,\gamma);\ (\beta,\delta)$$

The relative importance of each location is determined by postulating that a flooding condition exists in each location, one at a time, and assuming that all the subsystems in that location are failed with a probability of unity. Conditional minimal cut sets for each flood location are then determined in two steps: (1) by modifying each of the original cut sets to remove the subsystems associated with each location and (2) by reducing the remaining cut sets to minimal cut sets by eliminating the supersets. The set (β,γ) is a superset of (γ), for example. The cut-set analysis for this example is presented in Table 11-8.

Table 11-8. Fault-tree quantification to determine the impact importance of flood locations[a]

Subsystem-level cut sets	Conditional cut sets given flood-induced failure at location j					
	A	B	C	D	E	F
(α,γ)	(1)	(δ)	(α,δ)	(α,δ)	(α)	(α,δ)
(α,δ)	(δ)	(δ)	(α,δ)	(α,δ)	(α)	(α)
(β,γ)	(β)	(β,δ)	(δ)	(δ)	(β)	(β,δ)
(β,δ)	(β,δ)	(β,δ)	(δ)	(δ)	(β)	(β)
Conditional minimal cut sets	(1)	(γ), (δ)	(γ), (δ)	(γ), (δ)	(α), (β)	(α), (β)
Conditional top-event frequency	1	10^{-2}	10^{-2}	10^{-2}	10^{-1}	10^{-1}

[a]See Figure 11-15.

The simplest way to rank locations is by the size of the minimal cut sets remaining after postulating the flood. In this case location A would simply rank above all the rest since the location failure itself produces the top event. The remaining locations cannot be distinguished since each results in two single-event minimal cut sets. Note that the correlation between cut-set size and the frequency of failure is only approximate.

If system unavailabilities from causes independent of the flood are known, a more effective ranking can be made by estimating the conditional frequency of core melt given each failed location. In the example, the following subsystem unavailabilities are assumed: 10^{-1} for α, 10^{-3} for β, 10^{-2} for γ, and 10^{-4} for δ. The quantification of the fault tree gives greater resolution in ranking the impact locations, as would be expected. The quantitative approach results in the following importance ranking from most to least important:

1. A
2. E, F
3. B, C, D

Note that, if a bounding estimate can be obtained for the frequency of a flooding condition in each of the locations, a bound on the risk due to flooding can be obtained at this point. Such an estimate would conservatively neglect fragility--that is, the conditional frequency of failure given a flooding condition in each location. Several computer codes are available to aid in the qualitative analysis of locations in fault trees. Discussed in Sections 3.7 and 6.6, these codes include COMCAN, BACFIRE, and WAMCOM.

After the important impact locations have been determined, it is necessary to evaluate the source locations where floods can start. The analyst starts by listing the major sources of water at the plant, including the major tanks and systems that supply, circulate, and process water. Such systems would include, for example, the circulating-water, condensate, feedwater, service-water, component-cooling water, makeup and purification, spent-fuel pool, reactor-coolant, safety injection, and decay-heat-removal systems. A qualitative evaluation should be performed on each system and flooding source to identify and select those for quantification.

One useful technique to aid in the selection of source locations is a failure modes and effects analysis (FMEA) structured especially for this application. A similar approach has been successfully applied to the evaluation of important fire locations. An FMEA format specialized for floods is shown in Figure 11-16. The source locations found to have the relatively greatest potential for propagating to one or more important impact locations are selected for quantitative analysis.

The hazard analysis for internal flooding is completed by estimating the frequency of flood initiation, at each source location selected for analysis by the FMEA procedure, as a function of flood severity. A particular initiating flood might produce various degrees of flooding, depending on, for example, the timing and the success or failure of various mitigating actions, such as the shutting down of pumps or the closing of isolation

System or flood source	Subsystem or component	Location	Flood rate range (gpm)	Inventory of maximum estimated flood quantity (gal)	Colocated components, systems	Flood barriers and mitigating actions	Flood pathways to important impact locations
Circulating-water system	Expansion joint	Near condenser at el. 105 ft in turbine building	10,000	300,000 if terminated in 30 min	Condensate pumps, condensate booster pumps		

Figure 11-16. Example of FMEA format for evaluating flood-source locations.

valves. A distribution of flood magnitudes at a given location can be developed either directly from the data base or with the aid of a specialized event tree that is conceptually illustrated in Figure 11-17.

There is a substantial body of statistical data that is applicable to the hazard analysis of internal floods. Nuclear Power Experience (Verna, 1981) lists about 60 incidents at U.S. nuclear power plants that involved

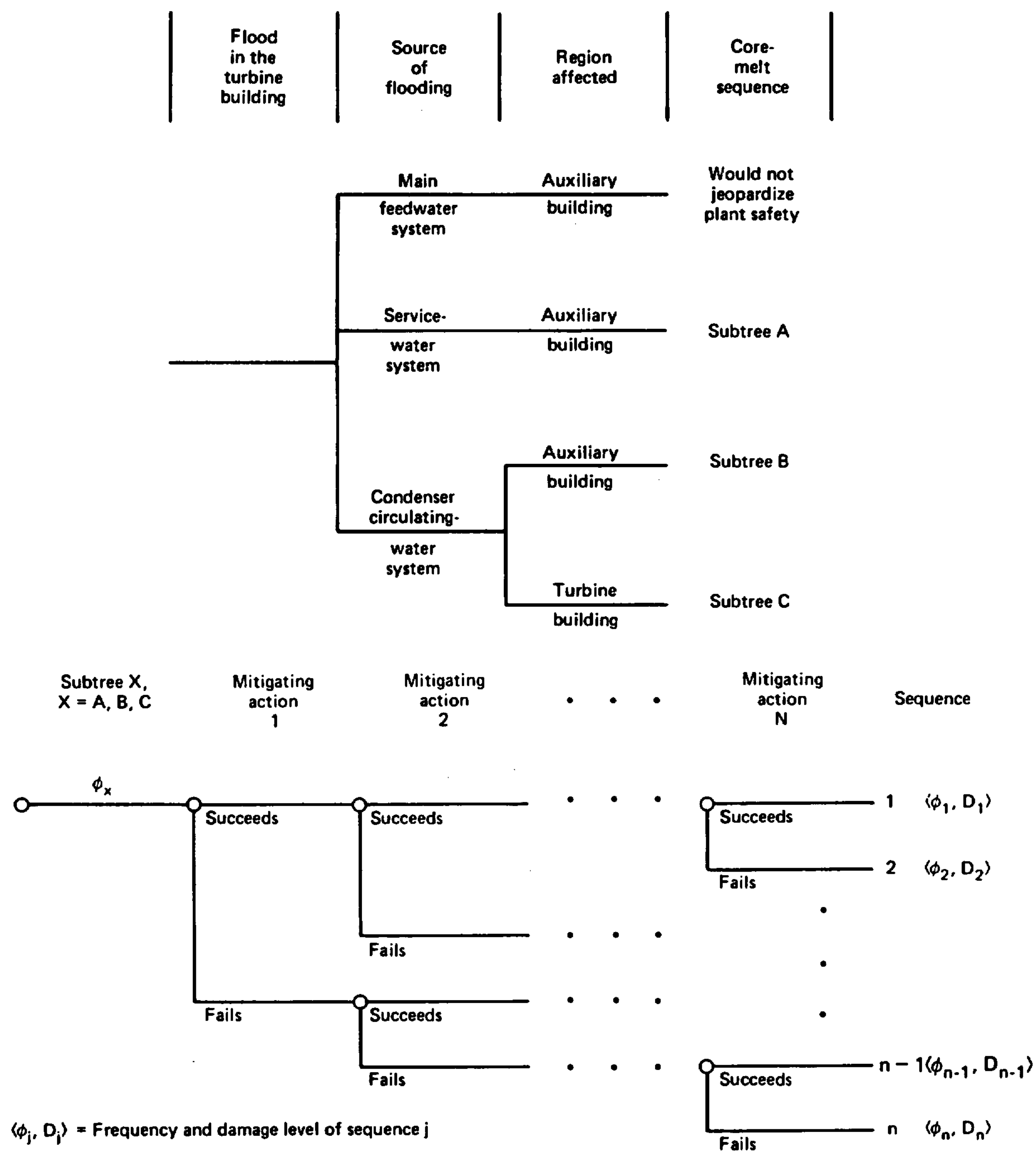

Figure 11-17. Event tree for developing frequency-magnitude estimates for the hazards of internal floods.

Table 11-9. Flooding frequencies for turbine and auxiliary buildings

Location	Severity level	Flooding frequency (per reactor-year) 5th percentile	Median	95th percentile	Mean
Auxiliary building	Small	2.0×10^{-6}	3.4×10^{-4}	1.0×10^{-2}	3.1×10^{-3}
	Moderate	1.6×10^{-4}	7.4×10^{-3}	3.1×10^{-2}	1.5×10^{-2}
	Large	1.0×10^{-6}	4.5×10^{-4}	2.0×10^{-2}	6.3×10^{-3}
	Moderate and large	2.5×10^{-5}	3.3×10^{-3}	1.6×10^{-2}	1.6×10^{-2}
Turbine building					
Service-water source	Moderate to large	2.9×10^{-7}	1.4×10^{-3}	2.5×10^{-2}	4.9×10^{-3}
Circulating-water source	Moderate to large	2.2×10^{-3}	1.2×10^{-2}	1.3×10^{-1}	2.8×10^{-2}

flooding of some sort. Other data of interest are the component-failure and pipe-failure data that have been analyzed and used in the assessment of loss-of-coolant accidents (see Chapter 5). These data can be analyzed to provide an estimate of the frequency of floods as a function of magnitude at specific locations. Shown in Table 11-9 are the results for floods in the auxiliary building and the turbine building. The following definitions of flood severity were used to categorize the data:

1. Small. Flooding on the order of hundreds of gallons (e.g., valve pit flooding, flooding of an instrument, or flooding within a component).

2. Moderate. Flooding on the order of several thousands of gallons (e.g., a few feet of water on the floor of a typical pump room).

3. Large. Flooding on the order of tens of thousands of gallons (e.g., a few feet of water in large rooms, very deep water [more than 10 feet] in a typical pump room).

4. Very large. Flooding on the order of hundreds of thousands of gallons (e.g., floods involving circulating-water or service-water piping).

A Bayesian procedure, identical with that described in Chapter 5 for the analysis of component and initiating-event data, can be used as indicated in Table 11-9 to express the level of uncertainty in the frequency estimates. As indicated in Section 11.4.3.3 for external floods, the quantification of hazard curves for internal floods must include an assessment of uncertainty by providing a family of curves, each of which is assigned a probability describing the level of uncertainty or state of knowledge.

Special attention should be given to component failures and whether the flooding entails sprays of water or just a rising pool. For example, spurious actuations are possible if an electrical cabinet is sprayed with water. Otherwise, the electrical circuits within the cabinets will be deenergized. Furthermore, special attention should be given to flood termination possibilities (in the case of most internal floods) and recovery of failed systems via local manual actuations. The latter entails a human-error analysis, which is discussed in Chapter 4.

11.4.3.4 Fragility Evaluation

The objective of fragility evaluation is to estimate the frequency of producing each of a number of flood-damage states E_ℓ as a function of the flood intensity $\underset{\sim}{x}$ and response $\underset{\sim}{y}$, denoted by $f_{e,\ell}(\underset{\sim}{y})$. For convenience and simplicity, it is assumed that the continuum of flood-damage states can be adequately approximated by a finite number of discrete states. This does not differ from the practice of using a finite number of release categories to approximate the continuum of possible release magnitudes that could result from a reactor accident.

The first step in fragility evaluation is to define the flood-damage states for analysis. These could be expressed as specific combinations of structural failures that might result from external floods or the occurrence of flooding at various combinations of important impact locations determined from the hazard analysis of internal floods. In the latter case the natural combinations of impact locations would be adjacent locations or those that would most likely be linked by propagation pathways emanating from the flood source. The flood-damage states used by Wagner et al. (1980) consisted of submergence to different flood heights. As noted earlier, however, special attention should be given to sprays of water. This is because under a spray condition the failure mode of some components, such as electrical cabinets, could be quite different from their failure mode under a rising-pool condition.

Although there have been no known attempts to do so, the method described in Section 11.2 for estimating the fragility of structures in terms of safety factors incorporated into the design should be applicable to floods as well. Confidence that this is indeed true is supported by the observation that a similar method has been successfully used in the Indian Point PRA (PASNY, 1982) to estimate the fragility of structures subjected to extreme winds and wind-generated missiles. The major difference in applying this technique to floods is that the calculation of structural loads and integrity must account for failure mechanisms unique to floods, which include wave runup and impact forces, erosion, missile strikes, liquefaction, ponding, overturning, sliding, hydrostatic loading, and leakage. Another failure mechanism is the blockage of cooling-water intakes by trash, which can occur in floods less severe than the probable maximum flood.

In addition to the fragility of structures, it is necessary to consider, for those event sequences in which the pertinent structures do not fail, the fragility of components inside the critical impact locations identified in the hazard analysis. A conservative approach is to assume that the components inside a room are failed if a flooding condition propagates to that room or location. The term "flooding condition," as used here, includes full or partial submergence, spraying, seepage into, or the wetting by any other mode of equipment anywhere inside the room. If this conservative approach is taken for internal floods, the entire fragility evaluation can be incorporated into the event tree of Figure 11-17. A less conservative approach was that taken by Wagner et al. (1980), who assumed that components fail only when submerged.

11.4.3.5 Plant and System Analysis

The objective of the plant and system analysis is to estimate the frequency of core-damage or core-melt sequences initiated at each of the flood-damage states defined in the flood-hazard analysis. This phase of the flooding-risk analysis uses the basic event- and fault-tree method described in Chapter 3. It is important, however, to ensure that this basic method of analysis accounts for the boundary conditions associated with each flood-damage state.

There are several different approaches to the plant and system analysis, each of which uses event trees, fault trees, or both. Variations on these approaches are described in Section 10.3.6.

11.4.3.6 Release-Frequency Analysis

The objective of release-frequency analysis is to estimate the conditional frequency of exceeding levels of accident consequences, given the occurrence of each flood-damage state. In the notation of Equation 11-34, this quantity is $f_{k|s,j}(z)$. The methods described in Chapters 7 and 9 for analyzing the containment event tree and the consequences of core-melt accidents should be fully applicable to flood-induced accidents, with the exception that dependences between the cause of the flood and certain factors that might affect offsite consequences must be taken into account. These dependences include weather conditions, the effects of the flood on emergency plans and evacuation, and liquid pathways for radionuclides. In the case of internal floods, there is no need to carry out a special analysis of release frequencies because these dependences would not apply.

11.4.4 INFORMATION REQUIREMENTS

The information required to perform a risk analysis of flooding consists of the following:

1. Description of plant systems, including the location of components and systems within structures; a set of general arrangement, structural, piping, electrical, and equipment drawings.

2. Safety analysis report, especially the chapters on the hydrologic characteristics of the site, the meteorological and topographic

features of the region, design criteria, applicable codes and standards.

3. Reports on qualification and preservice tests as well as inservice inspections.

4. Results of the plant and systems analysis for internal accident initiators, including data on the component-failure rates.

5. Statistical data from the National Weather Service, U.S. Army Corps of Engineers, and other sources on precipitation conditions contributing to floods. A summary of these data requirements for various types of sites is presented in Table 11-10.

6. A compilation of licensee event reports involving flooding at nuclear power plants like that provided by Verna (1982).

Table 11-10. Statistical data requirements for the analysis of external floods at various types of site

	Applicability of data type to site			
Data-source description	River site	Atlantic Coast or Gulf of Mexico site	Great Lakes site	West Coast site
Local historical point-precipitation information	X	X	X	X
Tropical storm history	X	X	X	X
Storm-surge history and potential		X	X	X
Seiche history and potential			X	
Characteristics of historical river floods	X			
History of astronomical tides (including initial rise effects)		X		X
Snow pack and melt characteristics	X		X	
Wind-wave and wave-setup potential	X	X	X	X
Basic hydrosphere characteristics	X	X	X	X
Historical geoseismic activity and potential				X
Locations and characteristics of dams, levees, etc.	X	X	X	X

11.4.5 PROCEDURE

The methods described in Section 11.4.3 are summarized below in the form of a step-by-step procedure for analyzing the risk from flooding.

Task 1: Flood-Hazard Analysis

1. Identify external and internal causes of flooding.

2. Identify important flood variables and failure mechanisms.

3. Determine the critical flood-impact locations.

4. Estimate the frequency-of-exceedence curve for each flood variable.

5. To the extent feasible, quantify probabilistically the uncertainties in frequency estimates, stating clearly which sources of uncertainty have been addressed.

Task 2: Fragility and Vulnerability Evaluation

1. Define flood-damage states.

2. Determine the susceptibility of components and structures to each flood-failure mechanism.

3. Identify actions to mitigate flood damage.

4. Estimate the frequency of each flood-damage state as a function of flood magnitude.

5. To the extent feasible, quantify probabilistically the uncertainties in frequency estimates, stating clearly which sources of uncertainty have been addressed.

Task 3: Plant and System Analysis

1. Develop the event- or fault-tree logic that defines the flood-damage states and relates them to core-damage states.

2. Apply component- and system-failure boundary conditions to the event- and fault-tree logic.

3. Estimate the frequency of core-damage accidents initiated by each flood-damage state.

4. To the extent feasible, quantify probabilistically the uncertainties in frequency estimates, stating clearly which sources of uncertainty have been addressed.

Task 4: Release-Frequency Analysis

1. Identify dependences between floods, meteorological conditions, and evacuation procedures.

2. Develop data for liquid-pathway modeling.

3. Estimate the frequency-of-exceedence curve for the consequences resulting from each flood-damage state.

4. To the extent feasible, quantify probabilistically the uncertainties in frequency estimates, stating clearly which sources of uncertainty have been addressed.

11.5 ASSURANCE OF TECHNICAL QUALITY

The provisions described in Chapter 2 for the assurance of technical quality are applicable to a seismic risk analysis as well. The sources of data, design reports, and material and qualification test results should be documented. Since the risk analysis is based heavily on assumptions and engineering judgment, it is essential to have all aspects of the analysis thoroughly reviewed by peers.

NOMENCLATURE

A	seismic capacity of a component expressed in terms of ground acceleration; a random variable
$\check{A}$	median ground-acceleration capacity
A_{rms}	root-mean-square acceleration
A_{SSE}	local response parameter specified for the reference earthquake (e.g., safe-shutdown earthquake)
a	specific value of the random variable A
a_0	a parameter in the recurrence relationship by Gutenberg and Richter (1942)
a_D	effective peak ground acceleration
a_{pi}	instrumental peak ground acceleration
a_s	sustained maximum ground acceleration
b_0	Gutenberg-Richter slope parameter
b_1, b_2, b_3	constants in the attenuation equation
C	seismic capacity of the component
F	factor of safety
$\check{F}$	median factor of safety
F_C	capacity factor of safety
F_{RE}	factor of safety in equipment-response computations
F_{RS}	factor of safety in the structural response analysis
F_S	strength factor
F	inelastic-energy-absorption factor
f'	frequency of component failure at nonexceedence-probability level of Q
$F_{A\|m_i,r_j}(a)$	frequency with which the ground-motion parameter A exceeds a value a given an earthquake of magnitude m_i at a distance r_j
f_c	core-melt frequency
$f_\ell(m_i)$	conditional frequency with which an earthquake at the source has a magnitude equal to m_i

$f_{\ell}(r_j)$	frequency with which the source-to-site distance is r_j given an earthquake on the ℓ^{th} source
$f_s(a)$	conditional frequency of plant-system failure leading to core melt for an effective peak ground acceleration equal to a
H(a)	cumulative annual frequency of occurrence of earthquakes that cause ground-motion parameter values less than or equal to a
h(a)	annual frequency of earthquakes with ground-motion parameter values between a and a + Δa
I_0	epicentral intensity (MM) of the earthquake site intensity
I_s	site intensity
K_p	A function of acceptable frequency p, relating the value of a_D to A_{rms}
ℓ	a seismic source
M_s, M_1, M_2, M_3	system-failure events
m	Richter (local) magnitude; m_i a specific value
m_0	earthquake magnitude below which damage rarely occurs
m_b	bodywave magnitude, can be related to m
m_m	upper-bound magnitude for the source
N	number of earthquakes per year exceeding magnitude m_i
P, Q	nonexceedence probability
P_N	normal operating load or stress
P_T	total load (stress) on the components (i.e., the sum of the seismic load and the normal operating load)
R	distance of the earthquake energy center (epicenter) from the site, a random variable
r_j	a specific value of R
β_0	b_0 ℓn 10, where b_0 is the Gutenberg-Richter slope parameter in the recurrence equation
$\beta_{(.),R}$	logarithmic standard deviation of the variable reflecting inherent randomness
$\beta_{(.),U}$	logarithmic standard deviation of the variable reflecting uncertainty

$\beta(.)$	logarithmic standard deviation representing total variability
$\varepsilon(.),R$	random variable (with unit median) representing the inherent randomness in the variable designated in the parentheses
$\varepsilon(.),U$	random variable (with unit median) representing the uncertainty in the median value of the variable (.)
μ	allowable ductility ratio
ν_ℓ	mean number of earthquakes per year on the ℓ^{th} source; activity rate of the source
$\nu(a)$	total mean number of earthquakes per year in which A exceeds a at the site
$\nu_\ell(a)$	mean number of earthquake events per year in which A exceeds a at the site because of an earthquake on the ℓ^{th} seismic source
$\Phi(.)$	standard Gaussian cumulative distribution function
$\vee$	OR symbol
$\wedge$	AND symbol

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Chapter 12

Uncertainty and Sensitivity Analysis

12.1 INTRODUCTION

The preceding chapters of this guide have described, for particular parts of a PRA, the sources of uncertainties, suggested procedures for uncertainty analysis, and available information on calculated uncertainties. This chapter provides details of the uncertainty-analysis methods referenced by some of the other chapters. It also explains how the uncertainties evaluated for each part of a PRA can be combined to display the overall uncertainty in the estimates of risk. The techniques described here have generic applicability in a PRA; the specialized approaches to uncertainty analysis that have been used for certain parts of a PRA are discussed in the other chapters.

Uncertainty analysis is performed either to estimate the uncertainty in the final results--that is, the risk to public health and safety--or to estimate the uncertainty in some intermediate quantity, such as the frequency of core melt or radionuclide releases to the environment. The identification, evaluation, and comparison of uncertainties are important; they provide a deeper insight into the risk analyses, add to the credibility of the results, and aid in the process of decisionmaking.

Uncertainty analysis can be performed qualitatively or quantitatively. Both approaches are described. Sensitivity analysis, often a useful adjunct to uncertainty analysis, is also discussed.

The field of uncertainty analysis for PRA has not been fully developed. In particular, there is no generally accepted rigorous mathematical basis for uncertainty analysis. The theory of statistics, with which uncertainty analysis is often identified, can provide valuable tools and guidelines for dealing with uncertainty, but it is generally too restrictive to satisfy the needs of the uncertainty analyst.

Risk analysts are only at the threshold of performing comprehensive uncertainty analyses. A variety of techniques that have been used or proposed are described in this chapter. However, many of these techniques are still being developed, and the methods have not been applied in all their combinations to all parts of a PRA. Consequently, in performing such analyses, the analyst may be breaking new ground. Considerable work remains to be done to establish a rigorous basis for the techniques. Therefore, the reader is cautioned that, although this chapter describes the present state of the art in uncertainty analysis, there is no generally accepted approach, and, indeed, any effort to perform a comprehensive uncertainty analysis as part of a PRA would likely involve work on methods development.

The section that follows, 12.2, provides an overview of uncertainty analysis and introduces some useful concepts. Qualitative approaches to uncertainty analysis are described in Section 12.3. Possible frameworks for a quantitative analysis are presented in Section 12.4, which discusses commonly used measures of uncertainty, sensitivity analysis, the determination of input uncertainties, and the propagation of these uncertainties through the risk analyses; it also explains how the uncertainties in intermediate outputs can be combined into overall uncertainties in the estimates of risk. The ways in which these uncertainties can be displayed are covered in Section 12.5. A summary of available sources of information on uncertainties in risk estimates is provided in Section 12.6. A set of procedures for performing an uncertainty analysis for a PRA is given in Section 12.7, and measures for the assurance of technical quality are discussed in Section 12.8.

12.2 OVERVIEW

The evaluation of uncertainties in a PRA involves four elements:

1. Evaluation of uncertainties in the input to each of the tasks of a PRA.

2. Propagation of input uncertainties through each task.

3. Combination of the uncertainties in the output from the various tasks.

4. Display and interpretation of the uncertainties in the PRA results.

A comprehensive evaluation of uncertainties requires the consideration of uncertainties in all parts of a PRA.

This section explains how the concept of uncertainty is defined in this chapter and describes the types of uncertainty that arise in the performance of a risk assessment. Also discussed is the basic philosophy of the approaches that can be taken to treat these uncertainties either qualitatively or quantitatively. Finally, levels of uncertainty analysis are described with respect to the degree of quantification performed in assessing uncertainty.

12.2.1 DEFINITION OF UNCERTAINTY

Historically, in the context of PRAs, the term "uncertainty" has been used to describe two different concepts:

1. Random variability in some parameter or measurable quantity.

2. An imprecision in the analyst's knowledge about models, their parameters, or their predictions.

The difference between these two concepts can be explained by considering the example of predicting the failure rate of valves. Assume that there is a valve-failure data base that contains data from several plants and that a model for the failure rate has been developed from these data. The failure rate predicted by this model for a particular valve may be uncertain for two reasons:

1. The model is intended to describe a randomness that is due to plant-to-plant variations (concept 1).

2. There are inadequacies in the model and its parameters have been estimated from a limited data base (concept 2).

The essential difference between these two concepts is that an enlargement of the data base may improve precision in concept 2 but cannot affect the fundamental random variability (concept 1), although a numerical assessment of that variability can be made more precise (see the discussion of tolerance and confidence intervals in Section 12.4.1.3). For clarity the term "uncertainty" will be used in this chapter to mean concept 2.

The distinction between these two concepts is important for decisionmaking because it indicates where, on the one hand, an increased effort in data gathering can improve the quality of decisionmaking by reducing uncertainty and, on the other hand, where it would be ineffective. Furthermore, as pointed out by Apostolakis and Kaplan (1981), whether one is concerned with random variability or uncertainty affects the way in which the propagation of the relevant measures is performed.

However, it is not always easy to separate the two concepts. The complexity of the calculations sometimes leads analysts to combine both variabilities into one measure. This was done in the Reactor Safety Study (USNRC, 1975) with the reliability data and in the Zion PRA (Commonwealth Edison Company, 1981) with variability in the magnitude of the source term, for example. Indeed, in the absence of data for a particular plant, the analyst may use a measure of the random variability in some characteristic of a population of power plants as the measure of uncertainty for this characteristic if the plant in question is believed to belong to the general population. In a Bayesian analysis, random distributions that originate in plant-to-plant variability have been used to define the prior distributions. These particular aspects are discussed in more detail in Section 12.4.1.

The distinction between uncertainty and random variability in parameter values is an area of uncertainty analysis where there is substantial room for improvement over current practice. This improvement can be achieved if the problems being solved and the probabilistic models used to solve them are defined more clearly, so that one knows at the outset which variabilities are being addressed and how. Current practice generally does not distinguish between uncertainty and random variability in the uncertainty analysis, which makes it impossible to separate the contributions from random variability and uncertainty in the final uncertainty bounds. Given the complexity of PRA procedures, this is not surprising. However, it is a goal toward which the PRA community should strive. At present, uncertainty analysis must be understood to mean the analysis of how both random variability in parameter values and uncertainties, as defined above, propagate through the PRA to give a single uncertainty/variability measure for the results of the PRA.

12.2.2 TYPES OF UNCERTAINTY

The uncertainties that arise in risk assessments can be of three types: uncertainties in parameter values, uncertainties in modeling, and uncertainties in the degree of completeness (see examples in Table 12-1). Parameter uncertainties arise from the need to estimate parameter values from data. Such uncertainties are inherent because the available data are usually incomplete, and the analyst must make inferences from a state of incomplete knowledge. Modeling uncertainties stem from inadequacies in the various models used to evaluate accident probabilities and consequences, and from the deficiencies of the models in representing reality. Completeness uncertainties are related to the inability of the analyst to evaluate exhaustively all contributions to risk. They refer to the problem of assessing what has been omitted and might be regarded as a type of modeling uncertainty, although a very special one.

Depending on the specific part of the risk assessment being performed, the type of uncertainty that dominates at each stage of the analysis can be different. Parameter, modeling, and completeness uncertainties contribute to the uncertainty in the final plant risk at each stage in a risk assessment (i.e., system analysis, containment analysis, and consequence analysis). To date, PRAs have given more attention to parameter uncertainties than to modeling and completeness uncertainties because parameter uncertainties can be treated more straightforwardly.

12.2.3 SOURCES OF UNCERTAINTY

Uncertainties of the various types described in the preceding section can arise in all parts of a PRA. Each of the chapters in this guide contains a section that describes these sources of uncertainty for the PRA tasks covered by that chapter.

12.2.4 MEASURES OF UNCERTAINTY AND RANDOM VARIABILITY

If the above-mentioned uncertainties and variabilities in inputs are to be propagated through the analyses of a PRA, it is essential to have some quantitative measures of uncertainty and random variability that can be manipulated in a consistent way. Measures of random variability and uncertainty are suggested by the theory of distributions and the theory of statistics. However, not all the uncertainties encountered in a PRA lend themselves to a statistical treatment, because of a lack of relevant data. Thus, while the theories of statistics can give guidance in constructing a formalism for measures of uncertainty (see Section 12.4.1), the results of an uncertainty analysis may not in general have a statistical interpretation.

There are two main approaches to statistics: (1) the frequentist, or classical, approach and (2) the Bayesian, or subjectivist, approach. A Bayesian approach was adopted in the Reactor Safety Study (USNRC, 1975), and such approaches have been advocated for the treatment of uncertainty in PRAs by Apostolakis (1978), Parry and Winter (1981), and Kaplan and Garrick (1981). However, there is no general agreement that this approach should be adopted (see, for example, Easterling, 1981; Abramson, 1981), and consequently both approaches are discussed in this guide. It is important to stress that the choice of one approach over the other affects all aspects of statistical inference, so that both point estimates and statistical measures of uncertainty can be significantly different.

Table 12-1. Types of uncertainties

Category	Examples
Parameter	Data may be incomplete or biased. In licensee event reports, for example, are we sure that all relevant failures are listed, and do we know the number of trials? Do the available data apply to the particular case? This raises the question of generic vs. site-specific data. Is the method of data analysis valid? Do the data really apply to the situation being studied? For example, are all pumps in all plants in the data base expected to have the same failure rate, or should they be regarded as variable?
Modeling	Is the model adequate? For example, do the binary event-tree and fault-tree models represent the continuous process adequately? Is uncertainty introduced by the mathematical or numerical approximations that are made for convenience? If the model is valid over a certain range, is it being used outside that range?
Completeness	Have the analyses been taken to sufficient depth? Have all human errors and all common-cause failures been considered? Have all important physical processes been treated? Have all important accident sequences been considered?

The most commonly used measures of uncertainty are probabilistic statements about the values of parameters, but the concept of probability is interpreted differently by classical and Bayesian analysts. Since this difference in interpretation is important when comparing classical and Bayesian measures of uncertainty, the two interpretations are briefly described in the next section.

12.2.5 THE INTERPRETATION OF PROBABILITY AND ITS CONSEQUENCES FOR THE QUANTIFICATION OF UNCERTAINTY

12.2.5.1 The Interpretation of Probability

Formally the classical theory of probability is the theory of additive and nonnegative set functions, and the mathematical theory was developed in terms of measure theory by Kolmogorov (1950). In his generally accepted treatment, probability is introduced as a primitive notion--a quantity P, associated with an event E, that is a possible outcome of an experiment. Although there is no formal connection between the classical theory of probability and the real world, the empirical observation that the outcomes of real-world experiments can be described by the results of the classical theory of probability validates the applicability of the theory to the real world.

The key problem in the theory of statistics is to estimate the probability P of an event E. It is a theorem in the theory of probability (the law of large numbers) that the observed relative frequency of the event E in a large number of repetitions of the experiment tends to approximate P. This result is the theoretical basis for the use of the observed relative frequency of E as a point estimate of P.

There are several different subjectivist interpretations, but the essential hallmark is that they all view as meaningful the use of the probability $P(H|E)$ of a hypothesis H given evidence E. Thus, a subjectivist interprets probability as a degree of belief in some hypothesis. Given this interpretation, it has been possible to construct a theory of subjective probability that includes an operational method for determining numerical values for probability $P(H|E)$.

As discussed in Section 12.4 and in Chapter 5, the choice of a particular interpretation of probability and the associated theory of statistics affects the choice of analytical tools that will be used by the analyst.

12.2.5.2 The Quantification of Uncertainty

The quantification of uncertainty is made at many different levels in a PRA. At the first level, the estimation of fundamental parameters, quantification may be achievable through an application of statistics, given the existence of relevant data from which inferences can be made. The frequentist and the subjectivist differ here in that they would use different theories of statistics. However, given sufficient data and the same modeling assumptions, they would usually get numerically similar results for best estimates and uncertainty bounds. (The interpretation of these uncertainty bounds would nonetheless differ, as discussed in Section 12.4.1.4.)

In many cases, however, it is necessary to make estimates, either of basic parameters or of the outcomes of hypothetical events, that cannot be based on data but must be based on experience in related areas, engineering analyses, and/or engineering judgment. Both the frequentist and the subjectivist may characterize their uncertainties in qualitative terms. The subjectivist, with his interpretation of probability as a degree of belief, will in general find it easier to express the uncertainties quantitatively but since his assignment of probabilities is subjective, he may have difficulty in convincing others to accept his assignment.

12.2.6 LEVELS OF UNCERTAINTY ANALYSIS

Each type of uncertainty (i.e., parameter, modeling, and completeness) can be characterized either qualitatively or quantitatively. Various levels of uncertainty analysis may be performed, depending on the extent to which each type of uncertainty is quantified. For example, an uncertainty analysis may consist almost entirely of a qualitative treatment of uncertainty; an example is the analysis performed in the Limerick PRA (Philadelphia Electric Company, 1981). The next level might be a quantitative treatment of data uncertainty with a qualitative treatment of modeling and completeness uncertainties; an example is provided by the German Risk Study (Gesellschaft fuer Reaktorsicherheit, 1980; EPRI, 1981). A more complete analysis would be given by a quantitative treatment of both data and modeling uncertainties with a qualitative treatment of completeness uncertainties. Finally, an analysis of all three uncertainty types, including a quantitative estimate of completeness uncertainties, may be attempted; a PRA that claims to have done this is the Zion study (Commonwealth Edison Company, 1981). Various levels of uncertainty analysis can thus be characterized by the degree to which each type of uncertainty is quantitatively analyzed.

12.3 QUALITATIVE UNCERTAINTY ANALYSIS

The simplest level of uncertainty analysis entails only the qualitative evaluation of the impact of input uncertainties on the intermediate and final output of a PRA. Such an analysis may be performed as a prelude to a quantitative analysis of uncertainty. One possible approach to qualitative uncertainty analysis is proposed here.

In general, the principal contributions to overall uncertainties come from areas where experimental or operating data are lacking and in areas where modeling uncertainties prevail. Some of these areas include the following:

1. Data base (especially in the case of human factors).

2. Treatment of common-mode and common-cause failures (system interactions).

3. Treatment of external events.

4. Modeling of core-melt phenomena.

5. Modeling of steam and hydrogen explosions.

6. Calculations of radionuclide release fractions and dispersion.

7. Health-effects models.

8. Completeness of the analyses.

The quantification of uncertainties in many of these areas will be dominated by subjectivity owing to the lack of data and the limited knowledge of the processes.

The objective of this section is to outline a procedure for the qualitative assessment of uncertainties--a procedure that is simple and systematic. Figure 12-1 presents a concise tabular format for a qualitative uncertainty analysis. The first three columns address the details of the PRA areas analyzed. Typical of these areas are system analysis, containment analysis, and the analysis of environmental transport and consequences. Each area is divided into a number of segments, which are then divided into a number of tasks that can be further broken down into subtasks. The next column, "major assumptions," states the assumptions and limitations associated with models, data input, and results. The fifth column indicates whether the subtask affects primarily the estimates of probabilities and frequency (P) or consequences (C). The "rank" column labels the uncertainties associated with each subtask as having a major impact (M), an intermediate impact (I), or a minor impact (m) on the total uncertainties for the task under consideration. The three categories are based on a subjective evaluation of the uncertainty contribution of each major assumption or subtask feature to the overall uncertainty of the subtask and task. A limited sensitivity analysis on the subtask and task level may be required to justify the proper ranking.

Sensitivity analysis should address the assumptions suspected of having a potentially significant impact on the task results. These assumptions are generally in areas where knowledge is lacking or where data are sparse, requiring heavy reliance on the judgment of the analyst. Sensitivity analysis can then be accomplished by formulating alternative assumptions and evaluating their individual impacts on the results; examples include the substitution of realistic success criteria for conservative ones, taking credit for recovery actions or not, and extremes in the operating environment under accident conditions. In the case of sparse data, plausible upper and lower bounds should be identified by the analyst to define the range of variation for sensitivity analysis.

Finally, the "remarks" column provides supplementary information the analyst may like to include. Such information can be related to sources of conservatism in the assumptions and the use of sensitivity analysis for ranking, as well as other issues related to modeling and model input adequacy, limitations, and completeness of the analyses within the subtask.

The analyst can define tasks and subtasks in each segment to fit his needs. He can also modify the format shown in Figure 12-1 to suit his objectives. To optimize the effort required for the preparation of such a table, the analyst will have to exercise his judgment. Areas, tasks, and subtasks are to be well defined, but only assumptions relevant to the

	Area					
Segment	Task	Subtask	Major assumptions	Impact[a]	Rank[b]	Remarks
. . .	. . .	. . .	. . .	. . .	. . .	. . .
Event-tree development	Identification of system-failure criteria	Success criteria	(Brief statement of the major assumptions involved and method of derivation of the criteria)	P	X	(Statement of 1. Conservatism 2. Effect of plant age and system models 3. Basis for selecting X)
	Delineation of accident sequences	Completeness of accident sequences	(Brief statement about basis for selecting or deleting sequences in the quantification)	P	X	
		. . .	. . .	. . .	. . .	
	Selection of initiating events	Initiator frequencies	(Brief statement about method and major assumptions used in frequency assignment)	P	X	(Same as above plus 4. Role of previous experience 5. List of initiators with high degree of subjectivity in frequency assignment owing to lack of data (e.g., ATWS)

[a]Key: P = probability or frequency; C = consequence.
[b]X can be major (M), intermediate (I), or minor (m).

Figure 12-1. Example of format for summarizing areas of uncertainties with potential effects on the partial results.

objectives of this section should be included. The detail provided in the table should be sufficient for a peer review.

A table in the format of Figure 12-1 can serve a dual purpose--as an uncertainty-assessment tool and to some degree as a vehicle for the assurance of technical quality. It will provide a fair amount of detail about the relative importance of uncertainties. This information is derived and based on individual task or subtask levels. However, the tasks and subtasks are not strictly independent of each other, and uncertainty ranks may change when viewed from the perspective of the overall result. The higher level of assessment that is required is accommodated by a table in the format of Figure 12-2, which is used to present the information of Figure 12-1 in a form modified to reflect the influence of task uncertainties on the overall results. Only M and I ranks are addressed in this table, and rank identification should be based preferably on a global (overall) level of sensitivity analyses.

In summary, two hierarchical levels of qualitative analysis are required: a detailed (task/subtask) level supported by a local limited sensitivity analysis to rank the uncertainties (Figure 12-1) and a higher (segment/area) level supported by a global (overall) limited sensitivity analysis to assess the impact of the uncertainties on the final PRA results (Figure 12-2).

The final step in a qualitative uncertainty assessment is to supplement the information in Figures 12-1 and 12-2 by discussing the tabulated findings and identifying dominant accident sequences and any features that appeared to be important or unique to the uncertainty assessment.

12.4 QUANTITATIVE UNCERTAINTY ANALYSIS

A quantitative treatment of uncertainty may involve, in varying degree, some or all of the steps defined in Section 12.2:

1. Evaluation of uncertainties in the input to each of the tasks of a PRA.

2. Propagation of input uncertainties through each task.

3. Combination of the uncertainties in the output from the various tasks.

4. Display and interpretation of the uncertainties in the PRA results.

The extent to which steps 1 through 3 are involved depends on many factors, including the level of analysis (Section 12.2.6) and the procedures that are adopted. This will become clearer in later sections. However, common to all tasks is the necessity for some quantitative measure of uncertainty and random variability. Some commonly used measures of uncertainty and random variability are discussed in Section 12.4.1. The evaluation of input

Area				
Segment	Task	Major assumptions	Impact	Remarks

Figure 12-2. Example of format for summarizing areas of uncertainties with major effects on the overall results.

uncertainties, methods for propagation, and methods for combination are discussed in Sections 12.4.2 through 12.4.4. The display of uncertainties is discussed in Section 12.5.

12.4.1 MEASURES OF RANDOM VARIABILITY AND UNCERTAINTY

This section describes measures derived from the theory of distributions or the theory of statistics to describe random variability or uncertainty in the values of the parameters of models or the values of some measurable quantity. The purpose of describing these measures, which have a limited applicability when compared to the scope of a full uncertainty analysis in a PRA, is to establish a terminology and the basis of a mathematical structure on which an uncertainty analysis can be founded.

12.4.1.1 A Simple Interval Measure

The simplest quantitative measure of variability in a parameter or a measurable quantity is given by an assessed range of the values the parameter or quantity can take. This measure may be adequate for certain purposes (e.g., as input to a sensitivity analysis), but in general it is not a complete representation of the analyst's knowledge or state of confidence and generally will lead to an unrealistic range of results if such measures are propagated through an analysis. The mathematics of random variables and the theory of statistics do provide measures of variability that can be more complete in displaying the degree of knowledge. These are discussed in the sections that follow.

12.4.1.2 Measures of Random Variability

The most complete characterization of a random variable x is given by the distribution function F(x). The value of the distribution function for a particular value of the variate x_1 is the probability that a randomly chosen x will have a value less than x_1. If the distribution function is known completely, it can be described by a particular functional form with specified parameters.

It is sometimes more convenient to characterize the variability by a single number rather than the population function itself. A commonly used measure is the variance, which is defined in terms of the density function f(x) associated with the distribution function as

$$\mathrm{var}(x) = \int_R f(x)(x - \mu)^2 \, dx \qquad (12\text{-}1)$$

where the mean μ of the distribution is given by

$$\mu = \int_R x \, f(x) \, dx \qquad (12\text{-}2)$$

and $f(x)\,dx$ is the probability that the random variable has a value in the range $(x, x + dx)$. Here $f(x)$ is regarded as continuous and normalized, and R is the range of the variable x.

The standard deviation s is given by

$$s = \left[\mathrm{var}(x)\right]^{1/2} \tag{12-3}$$

Both the distribution function and the variance are measures of variability that can be propagated by some of the methods explained in Section 12.4.3. In the case of the method for propagating variance, when input variables are not independent, it is necessary also to define a covariance, as discussed in Section 12.4.3.

In general, neither the form nor the parameters of the distribution function of a random variable will be known completely, and it will be necessary to make estimates based on data. The two most commonly estimated distribution parameters are the mean and the variance. Given a set of observed values of a random variable x $(x_1, \ldots, x_n)$, the mean and the variance of the population can be estimated by

$$\bar{x} = \sum_{i=1}^{n} \frac{x_i}{n} \tag{12-4}$$

and

$$v^2 = \frac{1}{n - 1} \sum_{i=1}^{n} (x - \bar{x})^2 \tag{12-5}$$

However, these estimates are subject to some uncertainty. This uncertainty can be described by confidence intervals on the parameters. An alternative is to construct tolerance intervals on the population. These concepts are described in the next section.

12.4.1.3 Tolerance and Confidence Intervals

There are two commonly used statistical interval estimates: tolerance intervals and confidence intervals. The differences between them are shown by the following example. Suppose there exists a density $f(x;\theta)$ on a random variable x with a parameter vector θ (f might be the Gaussian distribution and $\theta = (\mu, \sigma)$ in standard notation). Suppose, further, that a random sample of n values of the variate $(x_1, \ldots, x_n)$ is obtained. Statistically, it is possible to use this information in two distinct ways:

1. To obtain point estimates of the values of the parameters θ. The uncertainty in these estimates can be expressed by an associated confidence interval. Roughly speaking, the confidence level associated with this interval is an estimate of the probability that the value of the parameter lies within the interval. The exact

statement depends on whether the analyst is using classical or Bayesian statistics.

2. To estimate the proportion of the population of possible x values contained in an interval (x_L, x_U). This interval is a tolerance interval if the limits x_L and x_U are constructed as functions of the random observations to guarantee that the interval (x_L, x_U) will cover at least a prescribed proportion of the population with a probability equal to or larger than a prescribed confidence level. Tolerance intervals are discussed in great detail for both classical and Bayesian statistical frameworks by Guttman (1970).

Confidence intervals are an expression of the uncertainty in the estimate of a parameter. Tolerance intervals are confidence statements on a proportion of the population. Which of the two concepts is applicable depends on both the data and the objectives of the analysis. For example, suppose that the plant for which the PRA is being done is new. There are therefore no data for estimating plant-specific component-reliability parameters, and it is necessary to use generic data sources. If it is believed that there are significant plant-to-plant variations (or even component-to-component variations) but the plant in question is expected to be a member of the population of plants for which data are available, then the measure of uncertainty that is adopted should reflect this. The correct measure for this sort of variability is a tolerance interval. On the other hand, if it is believed that a simple model, like the exponential model of the time to failure (i.e., a fixed failure rate), is valid for the whole population of components and that the variability of the time to failure is adequately expressed by the randomness inherent in the exponential model, then it is only necessary to use a confidence interval to express uncertainty on the estimate made by pooling all the available data.

As relevant data are gathered, confidence intervals, for a given level of confidence, become narrower. However, since tolerance intervals are basically measures of random variability, they will not decrease indefinitely. They may, however, become known with a greater degree of confidence.

There has been very little explicit use of tolerance intervals in PRAs, largely because the literature on tolerance limits is still very theoretical and widely dispersed through the statistical literature, and the concept is much less appreciated than that of confidence. More work has been done on tolerance limits for the normal distribution than for other distributions. Moreover, there are some theoretical and practical problems with propagating tolerance intervals (Parry et al., 1977, 1981) in a meaningful way. The detailed derivation of tolerance intervals is therefore not discussed here. However, they are potentially useful when the random variability of input parameters is expected to be an important factor, as in the example above.

Historically, the idea of a confidence interval has been much more widely used. Its construction and interpretation are different in the classical and Bayesian statistical frameworks, and this is the subject of the next section.

12.4.1.4 Classical and Bayesian Confidence Intervals

A classical statistical confidence interval (L,U) on a parameter λ has the property that the probability (interpreted in the classical sense) that the true value of λ lies within the interval is α, the confidence level:

$$\text{Prob}(L < \lambda < U) = \alpha$$

The confidence interval is a random interval that is a function of the data. If the experiment is envisioned as being repeated many times, then the confidence level approximates the fraction of the times the confidence interval will include the parameter λ.

The determination of classical confidence bounds is discussed in many books on statistics. Since there are many different probabilistic models for which parameter estimates may be required, it is impractical to try to list the relevant results here. Green and Bourne (1972) and Mann et al. (1974) give the derivation of confidence bounds on the parameters of many of the commonly used distributions. It should be noted that the confidence limits depend not only on the form of the distribution whose parameters are being estimated but also, as shown in Chapter 5 of the book by Mann et al., on the way in which data are collected (i.e., the censoring scheme or stopping rule). The confidence limits are not generally in closed form; they are obtained from tables of distributions.

In the Bayesian, or subjectivist, framework, a Bayesian probability interval (L,U) is constructed from a probability distribution on the parameter value. This distribution represents the analyst's degree of belief about the possible values of the parameter and reflects the state of his knowledge about that parameter. The probability or confidence associated with the interval is the fraction of the distribution that lies between the two limits. So if the analyst's state of knowledge about the values of a parameter λ is characterized by a density $f(\lambda)$, the interval (L,U) is an α probability interval, or a Bayesian confidence interval, if

$$\int_L^U f(\lambda)\, d\lambda = \alpha$$

Chapter 5 of this guide describes how both the classical and Bayesian methods are applied to failure rates and failure probabilities. In the Bayesian approach, the analysis of uncertainty is an integral part of the estimation process in that the distributions from which the intervals are determined are used throughout. It is interesting that, in some applications, the degree of belief as expressed by the prior is dominated by an observed or assessed random variability. Examples are the prior distributions discussed by Apostolakis et al. (1980), which are representations of, among other things, plant-to-plant variability. These priors, however, are then specialized, using plant-specific data in Bayes' theorem to produce plant-specific distributions on parameters. (This is also the approach of the Zion study (Commonwealth Edison Company, 1981).) It is assumed that, for that plant, the failures of all like components are governed by the same failure rate or failure probability. The posterior distribution in this case becomes a Bayesian confidence statement--that is, it reflects the analyst's state of knowledge about the value of the parameter for that particular plant.

Of course, in the Bayesian, or subjectivist, framework it is sufficient to give a probability distribution on the parameter value in order to express confidence. This has been a popular way of representing uncertainty since the Reactor Safety Study (USNRC, 1975). The Zion study used this approach throughout. The use of distributions on input and output parameters is appealing because of the ease with which they can be manipulated. However, the provision of distributions for all parameters is not an easy task.

12.4.2 INPUT UNCERTAINTIES

12.4.2.1 Quantifiability

Each separate part of the PRA has its own particular input uncertainties. These may be as follows:

1. Uncertainties in the values of input parameters unique to that part of the PRA. An example is the deposition velocity in the consequence model.

2. Uncertainties in inputs that are outputs from another part of the PRA. An example is the frequency of an accident sequence and its input to the consequence model for risk evaluation.

3. Uncertainties due to modeling or completeness issues.

Whether these uncertainties are regarded as quantifiable depends on the level of analysis, and the degree of quantifiability that is necessary depends on the method of propagation. For example, if the overall uncertainties were to be treated only as bounds with no probability assignment, then it might be felt that it would be adequate to provide bounds on the input variables, and thus all parameter uncertainties should be quantifiable in the sense of constructing bounds. Some effort involving sensitivity analyses would be needed in defining bounds for models, and completeness uncertainties would need special treatment. Such an approach to uncertainty analysis would not, however, provide sufficient detail for some applications of PRA.

Uncertainties on input parameters, whether they are parameters of the model or inputs from another part of the PRA, may be quantifiable by one of the measures described in the preceding section. However, uncertainty in a parameter that is the output of another analysis is only as quantifiable as the methods of that analysis allow.

The treatment of modeling uncertainties is really part of the propagation task. The only input would be to decide on some weighting to be applied to different models that might be used. Since a model is an expression of the analyst's understanding of the phenomena being modeled, it is possible to interpret the weighting in a Bayesian sense as a degree of belief in the particular model.

When uncertainties are evaluated on a purely subjective basis, there tends to be an underestimate of the uncertainty, as discussed in Chapter 5. To aid the peer review, such subjective assessments and the reasoning behind them should be well documented.

The quantification of uncertainty on the completeness of a PRA is a difficult and paradoxical problem. The problem is difficult because it requires the quantification of all possibilities for incomplete descriptions and models and their probabilities within an already complex PRA calculation; it is paradoxical because the logical assessment of what one knows and what one does not know is not formally well structured (DeFinetti, 1970). For example, if an expert must examine how much is not known, then some information must be "known" about "not knowing" in order to form a judgment. This issue of completeness uncertainty is addressed in the literature under such titles as "incompleteness" (Kaplan and Garrick, 1981; Suppe, 1977), "assessment difficulties" (Fischhoff et al., 1981), "accuracy" (Holloway, 1979), "credibility" (Watanabe, 1969), and "robustness" (Lindley, 1972).

No one has formally solved this problem scientifically because it examines the limits of knowledge from only the one side where something is known.

The quantification of completeness is not feasible for PRA calculations. Individual judgments of completeness probably cannot be supported by evidence or wide consensus of other experts. Only through a thorough analysis and peer review can possible uncertainties be minimized.

12.4.2.2 Quantification

One of the first tasks in a quantitative uncertainty analysis is to decide which of the many sources of uncertainty are to be addressed. This can be decided by performing a sensitivity analysis or by using the results of such analyses performed by others and reported in the literature.

12.4.2.2.1 Sensitivity Analysis

Sensitivity analysis entails the determination of how rapidly the output of an analysis changes with respect to variations in the input. Sensitivity studies do not usually incorporate the error range or uncertainty of the input. This distinguishes sensitivity analysis from uncertainty analysis since the latter incorporates the input uncertainties with their sensitivities into output uncertainties. Sensitivity studies can be particularly useful for assessing the impacts of different models, system-success criteria, and the like. Sensitivity studies can be accomplished by the straightforward application of statistical designs (Mazumdar et al., 1975, 1976). A more sophisticated adjoint sensitivity approach has been proposed recently by Oblow (1978).

12.4.2.2.2 Parameter Uncertainties

Measures for quantifying uncertainties on parameter values on the basis of data are discussed in Section 12.4.1 and in Chapter 5. In Chapter 5 the discussions are specific to the reliability parameters of certain models, but the methods are generally applicable. When plant-specific data are available, it is recommended that one of these methods be used. The particular method chosen depends on the overall approach to uncertainty (i.e., classical or Bayesian).

When generic data are to be used, it may not be necessary to perform an analysis, but information may be taken from the literature directly. However, if uncertainties are quoted, care should be taken to understand what they mean. Suppose, for example, that the LER summary reports (Sullivan and Poloski, 1980a,b; Hubble and Miller, 1980) were to be used to provide generic estimates of component-failure rates. The uncertainties quoted on failure rates and failure probabilities are based on a common failure rate for all components. They would be an incorrect measure of generic plant-to-plant variability.

12.4.2.2.3 Modeling Uncertainties

The quantitative treatment of modeling uncertainties in PRAs is still very much in its infancy. It is an area that has not received much attention. Nevertheless, some recent attempts have been reported.

Baybutt et al. (1981a,b) in one example associated a discrete variable with alternative models. They assign a variance to the discrete variable. This variance does not have the same physical interpretation as the variance of a continuous variable, but is an intuitive estimate of its uncertainty compared with the uncertainty of other variables. This variance was then propagated. In another approach used by the same authors, a subjective probability distribution is given to the discrete variable associated with the spectrum of models. The discrete variable is used to keep track of the different models and their results.

An alternative approach is to short-cut the propagation of modeling and input-data uncertainties, assessing a distribution directly on the output. To be meaningful, such an assessment must be based on sensitivity studies or an intermediate uncertainty analysis. The rationale for the assessment must be recorded, and, as discussed in Chapter 5, the analyst should be aware of, and try to avoid, the tendency to underestimate uncertainty in a subjective assessment. This approach to modeling uncertainty was adopted for the uncertainties on source terms in the Zion PRA, but the rationale behind the choice of probability distributions was not given.

12.4.3 PROPAGATION METHODS

A number of methods have been developed to treat and propagate the uncertainty measures discussed in Section 12.4.1. They include integration methods and various techniques based on moments. The former methods include analytical integration, numerical integration, and Monte Carlo simulation, while the latter include the method of moments, Taylor expansion approximation, and response-surface approximation. What follows is a brief description of each of these techniques as well as a concise critique stressing their underlying assumptions. Sections 12.4.3.1 and 12.4.3.2 describe the methods that can be used when uncertainties are regarded as being characterized by distributions and as such are more applicable in the Bayesian framework. Techniques for the propagation of classical confidence intervals are discussed in Section 12.4.3.3.

12.4.3.1 Integration Methods

The integration methods described in this section include analytical as well as numerical approaches. Also discussed is Monte Carlo simulation.

12.4.3.1.1 Analytical Integration

In this method the joint probability density function of the input variables (x_i) is assumed to be known and represented by

$$f_x = f(x_1, x_2, \ldots, x_i, \ldots, x_n)$$

The integration of this function leads to an analytical expression for the output-variable probability density function.

In general, analytical integration is applicable in cases involving a limited number of independent variables. Moreover, when the input-variable joint probability density function is not known, the analyst is left with the choice of assuming independence or introducing dependence for the variables that are known to interact statistically. In this sense, a degree of judgment is exercised and introduced into the analysis. Other limitations of this method include complexity and difficulty in finding a closed-form solution for the integrals, which defeats the most attractive features of this technique and forces the analyst to use approximations or simulation methods.

12.4.3.1.2 Discrete Probability Distribution Method

In this numerical integration method, the input uncertainties are characterized by a discrete probability distribution (DPD) on parameter values. Suppose the output variable z is a function ϕ of the input variables $x_1, \ldots, x_n$:

$$z = \phi(x_1, x_2, \ldots, x_i, \ldots, x_n)$$

Let $x_{i1}, x_{i2}, \ldots, x_{ij}, \ldots, x_{im}$ denote a set of discrete values of x_i and let $P_{i1}, P_{i2}, \ldots, P_{ij}, \ldots, P_{im}$ be the probabilities associated with these values such that

$$\sum_{j=1}^{n} P_{ij} = 1$$

The DPD is then defined as the set of doublets that approximate the x_i continuous probability density function as

$$\langle P_{i1}, x_{i1} \rangle \ \langle P_{i2}, x_{i2} \rangle \ldots \langle P_{ij}, x_{ij} \rangle \ldots \langle P_{im}, x_{im} \rangle \qquad (i = 1, \ldots, n; \ j = 1, \ldots, m)$$

The corresponding DPD for the output variable z is given by

$$\langle P_{\alpha, \beta, \ldots, \theta}, z_{\alpha, \beta, \ldots, \theta} \rangle$$

where, for independent variables,

$$P_{\alpha, \beta, \ldots, \theta} = P_{1\alpha} P_{2\beta} \cdots P_{n\theta}$$

and

$$z_{\alpha, \beta, \ldots, \theta} = \phi(x_{1\alpha}, x_{2\beta}, \ldots, x_{n\theta})$$

As an example, consider the simple case of $z = x_1 + x_2$, where x_1 and x_2 are assumed to be independent. The DPDs are given by

$$x_1 = \{\langle .4, -2 \rangle, \langle .4, 1 \rangle, \langle .2, 2 \rangle\}$$

$$x_2 = \{\langle .3, 4 \rangle, \langle .7, 6 \rangle\}$$

$$z = \{\langle .12, 2 \rangle, \langle .12, 5 \rangle, \langle .06, 6 \rangle, \langle .28, 4 \rangle, \langle .28, 7 \rangle, \langle .14, 8 \rangle\}$$

The method is conceptually simple and can be applied to continuous distributions after discretization. Unlike direct integration or the method of moments, it does not involve analytical computations. The DPD procedure is specially straightforward when the x_i variables are statistically independent. However, it may require an excessive number of manipulations when the number n of the x_i variables is large. To avoid serious problems with computer storage and running time, an aggregation operation may be required. The discretization of continuous distributions may lead to optimistic results because of tail truncation. Some pitfalls of this technique and methods of dealing with them, especially in the case of dependences among the x_i variables, are outlined in the Zion PRA study (Commonwealth Edison, 1981, section on methods).

12.4.3.1.3 Monte Carlo Simulation

The Monte Carlo method presents the most direct approach to the problem of uncertainty propagation when input uncertainties are represented as distributions on parameters. It involves an evaluation of the output of a computer code or other analytical model for many sets of combinations of the input parameters. These combinations of input values are obtained by a random sampling from the distributions assigned to the input variables. Monte Carlo simulation thus constructs an approximation to the output-variable probability distribution.

Many codes have been written to perform Monte Carlo computations, including SAMPLE (USNRC, 1975), STADIC (Cairns and Fleming, 1977), and SPASM (Leverenz, 1981). The particular features of these codes are summarized in Table 6-5. Examples of the application of Monte Carlo methods are given by Wakefield and Barsell (1980) and Wakefield and Ligon (1981).

One of the potential limitations of the Monte Carlo technique is its cost: some cases require a large number of computer runs to generate an accurate representation of the output-variable probability distribution, even when suitable variance-reducing techniques are used. However, this will not be a problem if only a range for the output variable(s) is required. Another limitation of the currently used codes is that they do not provide any indication as to which subsets of the input variables are the major contributors to the uncertainty in the output variable. A problem common to other techniques as well is the process of assigning probability distributions to the input variables, which introduces an additional element of uncertainty. Moreover, in currently used codes the sensitivity of the output distribution to variations in the input-variable distributions can be assessed only by further independent Monte Carlo simulations at a greater cost. Finally, there is a limitation that results from dependences among the input-variable distributions (Apostolakis and Kaplan, 1981). The existence of these dependences adds complexity to the process of sampling the input distributions, but is not a serious problem.

12.4.3.2 Moments Methods

Let us assume that the input-output relationship for a certain model can be represented by the functional relationship

$$z = \phi(x_1, x_2, \ldots, x_i, \ldots, x_n) \tag{12-6}$$

where x_i ($i = 1, \ldots, n$) are the input variables and z is an output variable.

The moments methods are applicable when sufficient information is available to generate estimates of the first few moments of the x_i variables. This information is used to generate the corresponding moments for the output variable z.

If we further assume that the joint probability density function of the input variables is known and given by

$$f(x_1, \ldots, x_n) \tag{12-7}$$

then the mean and the variance of the output variable z are defined by

$$E[z] = \mu_z = \int x_1 \int x_2 \ldots \int x_n \, \phi(x_1, x_2, \ldots, x_n)$$

$$f(x_1, x_2, \ldots x_n) \, dx_1 \, dx_2 \ldots dx_n \tag{12-8}$$

$$V(z) = \sigma_z^2 = E[z^2] - \mu_z^2 \tag{12-9}$$

Unfortunately, sufficient information is usually not available to define the function 12-7, and therefore several analytical complexities arise in the process of evaluating Equations 12-8 and 12-9. However, a number of special cases of interest exist and are discussed in the sections that follow.

12.4.3.2.1 Method of Moments

The method of moments (Murchland and Weber, 1972; Apostolakis and Lee, 1977) treats problems of combining the input-variable moments to generate the corresponding moments for the output variables in fault-tree applications. Cases 1 and 2 summarize the results for simple OR and AND gates.

Case 1: OR Gate

In the special case

$$z = \sum_{i=1}^{N} x_i$$

the mean and the variance of z are given by

$$\mu_z = \sum_i \mu_i$$

and

$$V(z) = \sigma_z^2 = \sum_i \sigma_i^2 + 2 \sum_{i=1}^{N-1} \sum_{j=i+1}^{N} \mathrm{cov}(x_i, x_j)$$

In the special case where the x_i variables are assumed to be independent, the covariance term becomes zero.

Case 2: AND Gate

The case of the AND gate is not as simple as that of the OR gate unless independence is assumed among the x_i variables. Here

$$z = \prod_{i=1}^{N} x_i$$

and μ_z and σ_z^2 can be expressed as

$$\mu_z = \prod_{i=1}^{N} \mu_i$$

$$V(z) = \sigma_z^2 = \prod_{i=1}^{N} (\sigma_i^2 + \mu_i^2) - \prod_{i=1}^{N} \mu_i^2$$

In the special case where N = 2 and x_1 and x_2 are assumed to be dependent,

$$\mu_z = \mu_1\mu_2 + \text{cov}(x_1,x_2)$$

and

$$\sigma_z^2 = \sigma_1^2\sigma_2^2 + \mu_1^2\sigma_2^2 + 2\mu_1\mu_2\,\text{cov}(x_1,x_2) - \text{cov}^2(x_1,x_2) + \text{cov}(x_1^2,x_2^2)$$

Deleting the terms including covariances from the last two equations is equivalent to the assumption that x_1 and x_2 are independent.

A step-by-step approach can be used to propagate the means and the variances, starting with basic failure events at the bottom of the fault tree, until the corresponding moments are determined for the top event. It is important to note that μ_z cannot be calculated by simply substituting the corresponding μ_i variables in Equation 12-6 unless the x_i variables are found to be uncorrelated. Extreme care should be exercised in identifying those correlated input variables because the covariance terms can affect the moments calculated for the top event.

12.4.3.2.2 Taylor Expansion Method

The method of moments described in the last section can be used for a few simple cases. However, for more complex functional dependence of the output z on the variables x_i (see Equation 12-6), the derivation will be extremely complex. For this reason, a procedure that provides a good approximation for the mean and variance is required. Such a procedure is made possible by the use of the Taylor expansion method (Shooman, 1968).

In this method, the function ϕ of Equation 12-6 is expanded about a nominal point given by

$$x_1 = \bar{x}_1, x_2 = \bar{x}_2, \ldots, x_i = \bar{x}_i, \ldots, x_n = \bar{x}_n$$

This point can be selected for convenience or to represent the mean value of the x_i variables ($\bar{x}_i = \mu_i$). The first few terms of the Taylor expansion can be expressed as

$$z \cong \bar{z} + \sum_{i=1}^{n} \left.\frac{\partial \phi}{\partial x_i}\right|_{\bar{x}_i} \delta x_i + \frac{1}{2} \sum_{i=1}^{n} \sum_{j=1}^{n} \left.\frac{\partial^2 \phi}{\partial x_i \, \partial x_j}\right|_{\bar{x}_i} \delta x_i \, \delta x_j \qquad (12\text{-}10)$$

or equivalently

$$z \cong \bar{z} + \sum_{i=1}^{n} b_i \, \delta x_i + \frac{1}{2} \sum_{i=1}^{n} \sum_{j=1}^{n} c_{ij} \, \delta x_i \, \delta x_j \qquad (12\text{-}11)$$

where

$$\delta x_i = x_i - \bar{x}_i$$

which is taken as

$$\delta x_i = x_i - \mu_i$$

$$b_i = \left.\frac{\partial \phi}{\partial x_i}\right|_{\bar{x}}$$

and

$$c_{ij} = \left.\frac{\partial^2 \phi}{\partial x_i \, \partial x_j}\right|_{\bar{x}}$$

Using the above equation, it is straightforward to show that the expected value and variance relative to $\bar{z}$ are given by

$$\mu_z \cong \bar{z} + \frac{1}{2} \sum_{i=1}^{n} c_{ii} \sigma_i^2 + \sum_{i=1}^{n-1} \sum_{j=i+1}^{n} c_{ij} \operatorname{cov}(x_i, x_j)$$

and

$$V(z) \cong \sum_i \sum_j b_i b_j \operatorname{cov}(x_i, x_j)$$

noting that terms of the third and higher orders were dropped for consistency. The last two equations will provide a reasonable approximation for the first and second moments of the variable z whenever the function ϕ (Equation 12-6) is well behaved (exhibits weak nonlinearity). However, in cases where this function is highly nonlinear, higher-order expansions will be needed in Equation 12-10, and the derivation of μ_z and V(z) will generally be very complex.

12.4.3.2.3 Response-Surface Technique

In the preceding section the Taylor expansion method was used to approximate the function ϕ (see Equation 12-6) when available in a complex analytical form. The response-surface method is a similar technique that can be used in case ϕ represents a long-running computer code (Steck et al., 1980; Metcalf and Pegram, 1981; Ronen et al., 1980; Baybutt et al., 1981a,b). The output variable z is approximated by

$$z \cong \bar{z} + \sum_{i=1}^{n} b_i \, \delta x_i + \frac{1}{2} \sum_{i=1}^{n} \sum_{j=1}^{n} c_{ij} \, \delta x_i \, \delta x_j \qquad (12\text{-}12)$$

where $\delta x_i = x_i - \bar{x}_i$, $\bar{x}_i$ is the nominal value for x_i, and the b_i, c_{ij} terms are sensitivity coefficients.

The sensitivity coefficients in Equation 12-12 represent first-order and second-order derivatives of the function ϕ. The coefficients b_i and c_{ij} can be determined by using a statistical design and applying a least-squares fitting procedure to the code outputs resulting from a set of code runs. Different sampling techniques can be employed to improve the coverage of the multivariable sampling space associated with the x_i variables. Among the various sampling procedures available are the Monte Carlo technique, factorial sampling, and Latin-hypercube sampling. One of the major factors that has to be considered in selecting a sampling technique is the cost of the required computer runs. Steck et al. (1980) and Metcalf and Pegram (1981) have presented discussions and comparisons between the alternative methods.

Equation 12-11 can be used directly to approximate the mean and variance of the output variable z. In cases where z behaves linearly or exhibits a weak nonlinearity as a function of the x_i variables, Equation 12-11 will yield a good approximation. However, the existence of strong nonlinearities can cause these equations to be inadequate for a valid uncertainty analysis unless they are expanded to include higher-order terms. If variance analysis is based on approximate expressions like those of Equation 12-11, there may be a tendency to underestimate the overall output variance when higher-order terms are neglected. It should be noted that the ability of a response surface to represent well a complex computer code could possibly be improved by using a representation other than polynomial functions. For example, the use of trigonometric functions for at least some independent variables is a possibility.

Another critique of the response-surface technique is that Equation 12-11 may represent a proper approximation of the output variable z over a limited range of x-variables. Extrapolations beyond this range will, in themselves, invalidate the uncertainty analysis.

12.4.3.3 Methods for Propagating Uncertainties in the Classical Framework

Sections 12.4.3.1 and 12.4.3.2 discussed methods for evaluating function uncertainty when the argument uncertainties are expressed as probability distributions. This section considers the situation in which classical statistical (data-based) estimates of the function arguments are available. As in Chapter 5, the classical statistical methods used to assess the resulting uncertainty of the function output are those aimed at obtaining standard errors and confidence intervals. Mechanically, some of these methods are the same as those given in Sections 12.4.3.1 and 12.4.3.2 for probabilistic uncertainty analyses, and therefore the same computer codes can be used.

Mathematically, the problem can be expressed as follows. Let

$$Q = h(\theta_1, \theta_2, \ldots, \theta_k)$$

denote the function of interest and let θ_i^* denote (as in Chapter 5) an estimate of θ_i. Then Q is estimated by

$$Q^* = h(\theta_1^*, \theta_2^*, \ldots, \theta_k^*)$$

$$= h(\theta^*)$$

If the sampling distributions of the estimators θ_i^* were known, the resulting sampling distribution of Q^* could be derived or approximated by the methods of Sections 12.4.3.1 and 12.4.3.2. These distributions will not be known, because they involve the unknown parameters θ, but if they can be estimated, the sampling distribution of Q^* can then be estimated and used to obtain a standard error of Q^* and approximate confidence intervals on Q.

12.4.3.3.1 Bootstrap Method

Efron (1979) coined the term "bootstrap" for an analysis in which the sampling distributions of the estimators θ_i^* are estimated and then propagated by Monte Carlo methods (see Section 12.4.3.1) to obtain an estimated sampling distribution of Q^*. In this analysis the input sampling distributions are not specified subjectively by the analyst--they are specific functions of the data. For the common models discussed in Chapter 5, the bootstrap distributions are specified as follows:

1. Binomial. Bootstrap values of p^* are given by x/n, where x is obtained by sampling from a binomial distribution with parameters n and p equal to the observed $p^* = f/n$.

2. Poisson. Bootstrap values of λ^* are given by x/T, where x is obtained by sampling from a Poisson distribution with the parameter λT equal to f, the observed number of failures in T time units.

3. <u>Lognormal</u>. Bootstrap values of μ^* are obtained by sampling from a normal distribution with a mean of $\bar{t}$, the observed mean, and a variance of s_t^2/n, the observed variance divided by the sample size. Bootstrap values of σ^2 are given by $s_t^2 v/(n - 1)$, where v is obtained by sampling from a chi-squared distribution with n - 1 degrees of freedom.

Repeatedly sampling from the bootstrap distributions of the estimates θ_i^* and calculating the resulting Q^* provides an estimate of the sampling distribution of Q^*. The square root of the variance of the bootstrap-sample functions Q^* yields a standard error associated with Q^* and percentiles from the Monte Carlo distribution of the Q^* functions provide approximate confidence limits on Q. How well these approximate statistical confidence limits perform depends on the nature of the h-function and the available data. It may be advisable to carry out an auxiliary investigation of this question.

12.4.3.3.2 Taylor's Series

For functions of interest that can be differentiated, a Taylor's series expansion can be used to obtain a standard error for Q^*. Let h_i^* denote the derivative of $h(\theta^*)$ with respect to θ_i^*, evaluated at θ_i. Then the first-order Taylor's series expansion of Q^* is

$$Q^* = h(\theta) + \sum h_i^*(\theta_i^* - \theta_i)$$

and the variance of Q^* is

$$\mathrm{var}(Q^*) = \sum h_i^2 \ \mathrm{var}(\theta_i^*) + \sum_i \sum_j h_i h_j \ \mathrm{cov}(\theta_i^*, \theta_j^*)$$

where $\mathrm{cov}(\theta_i^*, \theta_j^*)$ denotes the covariance of θ_i^* and θ_j^*. Thus, different parameters can be estimated from the same or related data and thus not be independent.

Note: In the model for Q, there may be distinct, independent events, say two different valve failures, whose probabilities are estimated by the same data. These estimates should appear in $h(\theta^*)$ as a single θ_i^*, not as distinct θ_i^* based on (apparently) distinct data.

For simplicity, consider the case of statistical independence for which

$$\mathrm{var}(Q^*) = \sum_{i=1}^{k} (h_i^*)^2 \ \mathrm{var}(\theta_i^*)$$

By estimating the right-hand terms in this expression, one obtains an estimate of the left-hand side. Chapter 5 provides standard errors of θ_i^* for the cases in which θ_i^* is an estimated failure rate, failure probability, or expected repair time. The squares of these standard errors are estimates of $\mathrm{var}(\theta_i^*)$. Depending on the complexity of $h(\theta^*)$, the derivatives would be obtained analytically or numerically and then estimated by replacing the parameters θ_i by their estimates θ_i^*. Thus, the Taylor's series standard error (s.e.) of Q^* is given by

$$\text{s.e.}(Q^*) = \left\{\sum_{i=1}^{k} (h_i^*)^2 [\text{s.e.}(\theta_i^*)]^2\right\}^{1/2}$$

Note that the k terms in the sum identify the portion of the overall imprecision of Q^* attributable to each θ_i^*. Note also that response-surface methods, such as those discussed in Section 12.4.3.2.3, might be used to obtain derivatives.

Both the Taylor's series and the bootstrap analyses can yield unduly optimistic results when the data for estimating failure rates or probabilities consist of zero failures in T time units or demands. The standard error of θ_i^* in these cases becomes zero, and the bootstrap-sampling distributions are degenerate (they yield λ^* or $p^* = 0$ with probability 1). Whether these zeros have other than a negligible effect on the standard error of Q^* depends on where the respective θ_i terms appear in the model and the nature of the data pertaining to the other θ_i terms. Sometimes an inspection of the model, $h(\theta)$, and the data will indicate possible appreciable nonconservatism in using zero-failure data directly. One approach to checking for such possibilities is to add additional fractional failures, say 1/4 or 1/2, in these cases and see how the standard error is affected. Standard errors obtained this way are conservative.

The absence of any data pertaining to some θ_i terms means that a classical statistical standard error for Q^* cannot be obtained (nor can a classical statistical estimate of Q be obtained in the first place). There are several options an analyst might follow:

1. The θ_i terms for which there are no data could be held fixed at selected values and then the statistical analysis of Q^*, considering the data available pertaining to the other θ_i terms, would be conditioned on these fixed values. The analysis might be repeated at various settings--say optimistic, nominal, and pessimistic--of those parameters for which there are no data in order to convey the uncertainty associated with these parameters. If there are many such parameters, this analysis becomes unwieldy, but techniques from the statistical design of experiments, such as fractional factorials, might be used to simplify the analysis.

2. Subjective standard errors, representing the analyst's "uncertainty" about the θ_i terms for which there are no data, could be inserted into the Taylor's series analysis.

3. Alternatively, the analyst might represent his uncertainty about θ_i as pseudo-data and use either the Taylor's series or the bootstrap analyses; that is, he might interpret his knowledge of θ_i as being analogous to the information yielded, say, by f_i failures in T_i hours. The results in Chapter 5 pertaining to standard errors and confidence intervals could be used to guide the choice of the pseudo-data.

Having to use pseudo-standard errors or pseudo-data (or prior distributions--a natural conjugate prior distribution can be interpreted as pseudo-data) is not particularly satisfying. It softens the analysis. Nevertheless, in order to convey the overall uncertainty in a point estimate such devices may be necessary. The analyst should convey to his audience the extent to which his results depend on subjective assessments.

12.4.3.3.3 System Reduction

Consider the specific problem of obtaining confidence limits on a system-failure probability based on binomial component data. Algorithms by which the component data are reduced to effective binomial system data, say f^* failures in n^* demands, are available in a recently published handbook (Maximus, Inc., 1980). Then the binomial distribution methods of Chapter 5 can be used to obtain a standard error of $Q^* = f^*/n^*$ and to obtain confidence limits on Q, the system-failure probability. The reduction rules in the handbook treat series-parallel arrangements of components and also the use of the same data to estimate failure probabilities for different components. The case of zero failures poses no special problem, and the confidence limits obtained are conservative; that is, 95-percent confidence limits, for example, have at least a 95-percent chance of including the system-failure probability.

Though the Maximus handbook pertains only to binomial data and the estimation of system-failure probability, the methods can be readily extended to Poisson data and failure-rate estimation when the "$\lambda\tau$ approximation" is justified. For example, consider a diesel generator that is to start and run for 8 hours. Suppose the available data pertaining to failure to start are 30 failures in 1700 attempts and the fail-to-run data are 6 failures in 1600 hours. Since the 1600 hours amounts to 200 eight-hour "demands," we can treat this diesel-generator "system" as a series system of two "components" (fail to start and fail to run for 8 hours) with respective data of 30/1700 and 6/200. These data yield

$$Q^* = \frac{30}{1700} + \frac{6}{200}$$

$$= .048$$

The Maximus method for a series system is to take the effective system demands, n^*, to be the minimum of the component demands, 200 in this case, and the effective number of failures to be $f^* = Q^*n^*$, 9.5 in this case. Statistical confidence limits on Q would then be based on data of 9.5 failures in 200 demands; for example, the resulting upper 95-percent confidence limit on Q is .082.

Another approach to system reduction can be used in conjunction with a Taylor's series or a bootstrap analysis. Both of these analyses yield an estimated variance associated with the estimate Q^*. Suppose that Q is a failure probability. If Q could be estimated directly from binomial data, say f failures in n demands, then, using the methods described in Chapter 5, the following estimates would be obtained:

$$Q^* = f/n$$

$$\text{var}^*(Q^*) = \frac{f(n - f)}{n^3}$$

If the left-hand estimates, obtained from the data pertaining to the parameters that go into Q and either a Taylor's series or a bootstrap analysis, are equated to the right-hand functions of f and n, then these equations can be solved for f and n. The solutions are as follows:

$$n^* = \frac{Q^*(1 - Q^*)}{\text{var}^*(Q^*)}$$

$$f^* = n^*Q^*$$

Thus, the data used to obtain Q^* and $\text{var}^*(Q^*)$ are equivalent to binomial data, f^* occurrences in n^* trials, in the sense that the same point estimate and standard error would be obtained. Approximate statistical confidence limits on Q can then be obtained by using the Chapter 5 methods for binomial confidence limits.

For the above diesel-generator example, either a Taylor's series or a bootstrap analysis yields

$$\text{var}^*(Q^*) = 1.6 \times 10^{-4}$$

The resulting effective system data are 13.7 failures in 285 demands, from which confidence limits on Q could be obtained. These would be somewhat tighter than those yielded by the Maximus analysis, but of course the latter's conservatism is not guaranteed.

Alternatively, a PRA may be aimed at estimating the rate λ_E at which an accident occurs, rather than its probability in some specific time period. Often, an accident sequence is modeled as the occurrence of an initiating event, at a constant rate λ, followed by some sequence of component and system failures that occurs with conditional probability p. Thus, the accident would occur at a rate $\lambda_E = \lambda p$. The analyses described above lead to a point estimate λ_E^* and an estimated variance $\text{var}^*(\lambda_E^*)$. From the results given in Chapter 5, equivalent Poisson data pertaining directly to λ_E would be obtained by solving

$$\lambda_E^* = \frac{f}{T}$$

$$\text{var}^*(\lambda_E^*) = \frac{f}{T^2}$$

for f and T. These solutions are

$$T^* = \frac{\lambda_E}{var^*(\lambda_E^*)}$$

$$f^* = T^* \lambda_E^*$$

Using these equivalent data in the Chapter 5 expressions for Poisson confidence limits yields approximate statistical confidence limits on λ_E. For the small rates and probabilities generally encountered in nuclear plant PRAs, this analysis differs negligibly from the binomial analysis described in the preceding paragraphs.

12.4.3.3.4 Other Statistical Methods

Other methods for obtaining statistical confidence limits on a system-failure probability are discussed by Mann et al. (1974). A method called the "jackknife" is discussed by R. G. Easterling in a paper to be published in the Proceedings of the 1981 DOE Statistical Symposium. (This paper also discusses and illustrates the bootstrap and Taylor's series analyses for problems other than estimating a system-failure probability.) The jackknife method does not appear to be well suited for estimating failure probabilities or rates, but it might be used effectively in evaluating the uncertainty of consequence-model estimates, given data pertaining to the parameters in the consequence model.

12.4.4 METHODS FOR COMBINING UNCERTAINTIES

The uncertainties from each of the different parts of the PRA must be combined to provide an overall quantitative measure of the uncertainty on risk. As illustrated by Figure 12-3, the information flow between the different parts of the PRA is complex, and, given the complexity of the individual parts themselves, a rigorous uncertainty evaluation is out of the question. However, by making judgments based on the results of sensitivity analyses or detailed uncertainty analyses on the individual parts of the PRA, it is possible to construct bounds on the results of the PRA. Whether these bounds may be interpreted in some sense as confidence intervals depends on the philosophy adopted by the analyst.

In order to have a mathematically well-defined overall measure of uncertainty, it is important to make sure that the measures of uncertainty employed throughout can indeed be combined in a meaningful way. This does not mean that all the measures have to be expressed in the same way. For example, having a probability distribution on the frequency of an accident sequence is compatible with merely giving a range for the associated source term. However, if it has been decided to attempt to quantify the effects of both modeling and statistical uncertainties in one measure, then the measures of modeling uncertainties and statistical uncertainties must be compatible. The problem

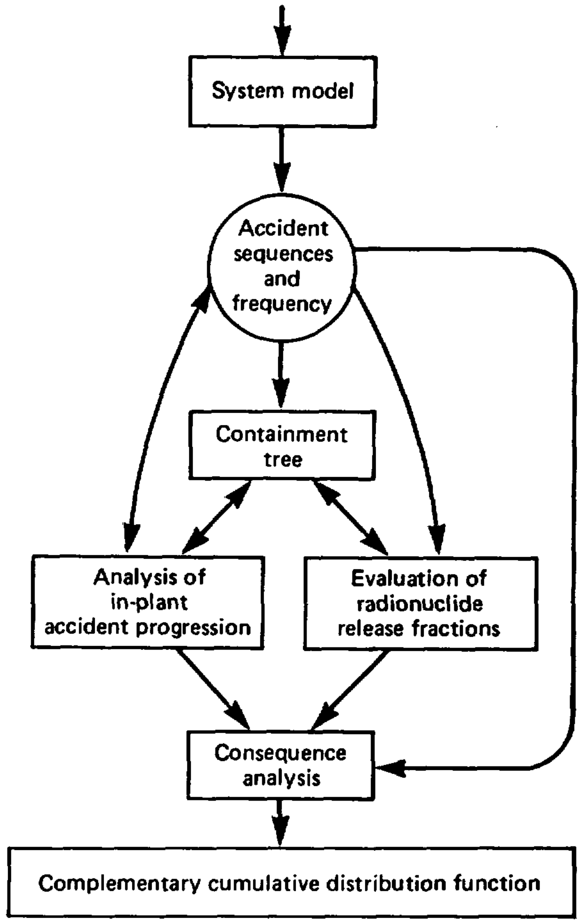


Figure 12-3. PRA information flow.

of compatibility is discussed by Baybutt et al. (1981a,b), who give some suggestions as to how it can be handled. The methods for combining uncertainties should be the same as those for propagation, discussed in the preceding sections. However, as mentioned above, the complexity of the PRA procedures means that at present a mathematically rigorous, all-embracing uncertainty analysis along these lines is impractical.

Since the Reactor Safety Study (USNRC, 1975), two different approaches have been used to tackle the problem of compatibility in measures of uncertainty. Both are highly subjective. One is a global modification of an existing PRA; the other is an integral part of the PRA.

The first approach is that of Erdmann et al. (1981), who examined an existing PRA--the Reactor Safety Study--to identify the key factors that were believed to have the potential of significantly affecting either the magnitude of the risk estimate or its uncertainty. The magnitude of these effects was estimated subjectively, and these subjective estimates were combined as if each effect were characterized by a lognormal distribution. Thus, given that there are n factors to be considered, each characterized by a median m_i and an error factor E_i ($i = 1,\ldots,n$), the overall uncertainty factor UF and the median M are given by

$$UF = \exp\left\{\left[\sum_{i=1}^{n} (\ell n\ E_i)^2\right]^{1/2}\right\}$$

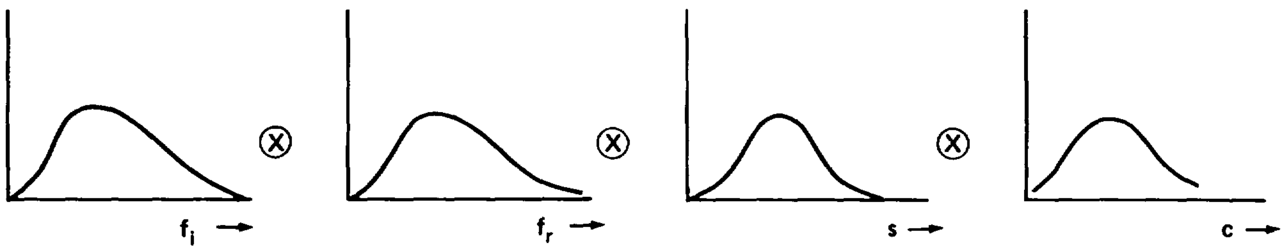
$$M = \prod_{i=1}^{n} m_i$$

The significance of the numerical results thus obtained is difficult to assess. This approach has the advantage that it is possible to handle the uncertainties from many different sources. However, since many effects are nonlinear and correlated with others, it is difficult to see how one can have confidence that all these different effects have been accounted for correctly. A very detailed understanding of all the processes is therefore necessary, and a careful documentation of all reasoning must be supplied if this approach is to have much credibility. In the Limerick study (Philadelphia Electric Company, 1981), the step of giving subjective assessments of the significance of the various key factors was not taken, and the discussion was qualitative. However, a total uncertainty band was constructed subjectively. The basis for that uncertainty band is not well explained.

The second approach is that of the Zion study (Commonwealth Edison Company, 1981). Here the calculational procedure, which is adopted in most PRAs, associates accident sequences and containment-failure modes with source terms. This procedure allows the analyst to partition the problem so that at each stage a single parameter can be used to characterize the output, which is then the input to the next stage. This is represented in the Zion PRA as a matrix multiplication to cover all sequences. The uncertainty analysis is represented schematically in Figure 12-4. The distributions represent the uncertainty of the relevant parameters. The total uncertainty in the PRA results is evaluated simply by combining the uncertainties from each part of the analysis.

The Zion approach appears to be formally neater than the previous method and gives the possibility of quantifying the probabilities associated with the bounds. Again, a great deal of analysis must go into understanding the phenomena so that the parameters that best characterize the input and output of the various stages can be chosen and probabilities can be attached to the different values. Another problem is that of partitioning uncertainty. In the Zion application, all the uncertainty on the source term is expressed as a probability (or state of knowledge) distribution. It is probable that at least some of the variability should be associated with the frequency since the phenomena would produce a different source term in different accidents. Whether this is possible, or whether it has much effect, cannot be answered at present.

In conclusion, the quantitative combination of uncertainties throughout the PRA is a relatively new art and at present is possible only through exercising a great deal of informed judgment. The credibility of such an exercise depends on the expertise of those making the judgments. In the



f_i frequency of core melt
f_r conditional frequency of a particular release category
s magnitude of associated source term
c magnitude of consequence

Figure 12-4. Schematic representation of the uncertainty analysis used in the Zion PRA (Commonwealth Edison Company, 1981).

interests of clarity and traceability, analysts should avoid the unsubstantiated use of subjective opinion on uncertainties.

12.5 DISPLAY OF UNCERTAINTIES IN THE RISK RESULTS

A concise method of displaying the uncertainties in the overall results of a PRA is to present a series of complementary cumulative distribution functions (CCDFs).* These different CCDFs could represent, for instance, the best estimate and an upper and lower bound. If a full uncertainty analysis were done, it would be possible to produce a series of curves at different probability levels. Then the best representation would be as in Figure 12-5, with the probabilistic assignment being the cumulative probability.

Since it is concise, this representation of uncertainty does not allow a ready appreciation of the principal sources of uncertainty. To provide greater insight into the sources of uncertainty, it is suggested that this be supplemented by a table that identifies the important sources and gives at least a qualitative assessment of their effects, and, if possible, a quantitative estimate. Depending on the level of PRA uncertainty analysis, this table might be the only form possible for the display of results. An example of such a table is provided in the Limerick PRA (Philadelphia Electric Company, 1981).

The most complete treatment of uncertainties so far has been provided by the Zion Probabilistic Safety Study (Commonwealth Edison Company, 1981).

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*It should be noted that there are many other important results from a PRA for which uncertainties can be calculated (e.g., core-melt frequency). Some of these are discussed in Chapter 13.

A subjectivist framework is used, and the uncertainties are displayed as in Figure 12-5.

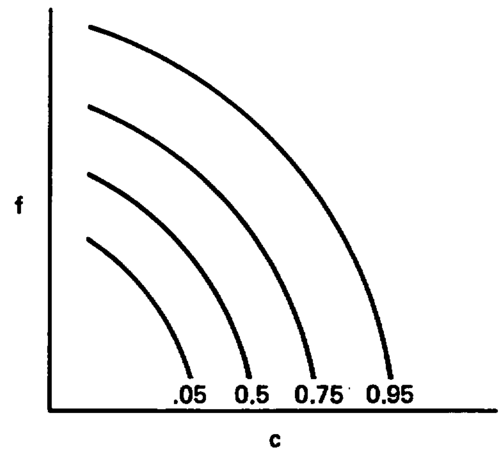

Figure 12-5. Display of uncertainties in a complementary cumulative distribution function (an f/c curve, where f is frequency and c is consequence).

An additional piece of information forthcoming from this method for uncertainty treatment is that of the "cut curve." A vertical line is drawn at some consequence level x_1. The intersection of this line with the family leads to a cumulative probability density function, as shown in Figure 12-6. This curve can be differentiated to yield a curve that expresses the state of knowledge or belief about the frequency with which events of level x_1 or greater can occur. Such cut curves can also be drawn to show the contributions to the risk family from various sources of risk.

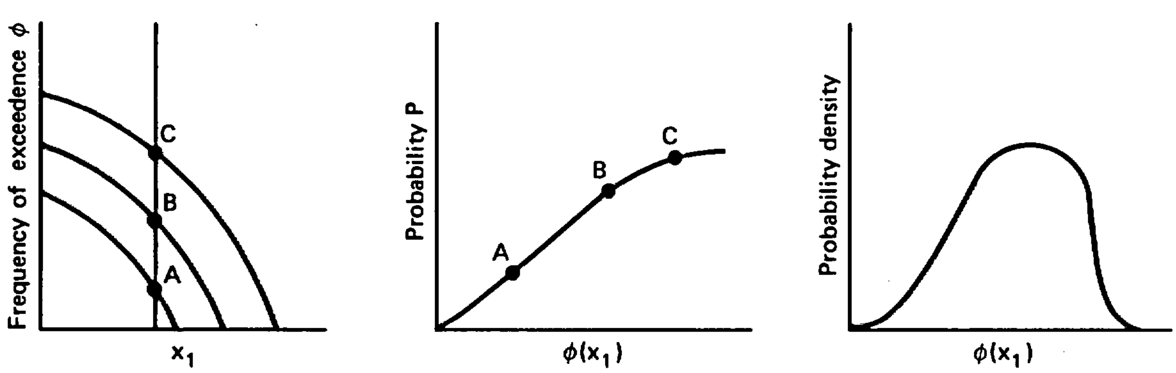

Figure 12-6. Development of cut curves from a family of risk curves.

12.6 AVAILABLE SOURCES OF INFORMATION ON UNCERTAINTIES IN RISK ESTIMATES

To date, PRAs have quantified uncertainties to varying degrees. Information on uncertainty treatments can be found in the Reactor Safety Study (USNRC, 1975), the German Risk Study (Gesellschaft fuer Reaktorsicherheit, 1980; EPRI, 1981), the Limerick PRA (Philadelphia Electric Company, 1981), the Zion PRA (Commonwealth Edison Company, 1981), and the Indian Point PRA (PASNY, 1982).

Some of the data used in these analyses, or indeed some of the results, may be of value in performing uncertainty analyses for new PRAs. For example, if the similarities are sufficient, it may be possible to use uncertainty estimates from an existing PRA in a new PRA.

12.7 SUGGESTED PROCEDURES

Tasks for evaluating uncertainties in a PRA are listed below. It should be noted that an uncertainty analysis cannot be performed simply by following the tasks listed below step by step. Some iteration among steps is likely to be needed, and in some cases it may not be possible to perform each step completely.

1. Determine level of analysis to be performed. Uncertainty analyses can be performed either qualitatively or quantitatively. It is usually preferable to quantify uncertainties, but the selection of the analysis level depends on the objectives of the PRA, what is feasible for a particular risk assessment, and the preference of the analyst.

2. Select treatment and depth of analysis for the uncertainties to be included. A choice should be made for data, model, and completeness uncertainties.

3. Select parts of PRA to be included in the analysis. A comprehensive analysis may not be needed. For example, it may suffice to evaluate uncertainties in either accident probabilities or consequences.

4. Identify sources of uncertainty. For those parts of the PRA to be included in the analysis, all sources of uncertainty of the types selected in step 2 should be identified by reviewing the calculational procedures. Sources of uncertainty have been discussed in Chapters 3 through 11 of this guide.

5. Decide on statistical framework. Decide where to use classical and/or Bayesian methods.

6. (Optionally) perform sensitivity analysis. Before performing an uncertainty analysis, the analyst may wish to evaluate sensitivities to obtain some insight into what is important in controlling the output of the risk analyses. This process can help in deciding what should be included in an uncertainty analysis. Methods are described in Section 12.4.

7. Estimate input uncertainties. Techniques are described in Sections 12.4.1 and 12.4.2.

8. Propagate input uncertainties through risk analyses. Choose from the methods described in Section 12.4.3.

9. Combine intermediate uncertainties. Methods are described in Section 12.4.4.

10. Display uncertainties in risk results. Approaches are described in Section 12.5.

12.8 ASSURANCE OF TECHNICAL QUALITY

Little formal work has been done to develop methods of ensuring that PRAs or uncertainty analyses, in particular, are performed correctly. The reader is referred to Chapter 2 for some general guidelines on the assurance of technical quality.

One special issue in regard to the assurance of technical quality arises in a Bayesian uncertainty analysis, where the choice and the form of the prior are important, particularly when the prior is based largely on expert opinion and when it is expected that it will not be modified much by data. Some thoughts on this very complex subject are presented in Chapter 5. The important point from the standpoint of ensuring technical quality is that, as stated in "Recording Expert Opinion," the important procedural and substantive factors in that evaluation should be recorded. Although this should be done for any prior, its importance is greater in the case of expert judgment.

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Chapter 13

Development and Interpretation of Results

The preceding chapters have described in some detail the steps to be performed in accomplishing a probabilistic risk assessment and various methods for performing these steps. This chapter discusses the development of quantitative results for PRAs of various levels of scope, the analysis of uncertainties, and the interpretation of these results through the development of engineering insight. The concluding remarks provide perspective on the objectives of this procedures guide as well as the status, role, and utility of probabilistic risk assessments.

13.1 DEVELOPMENT OF QUANTITATIVE RESULTS

As discussed in Chapter 2, the results of a PRA depend on the scope of the analysis. This section describes how results are developed for PRAs of various levels of scope.

13.1.1 LEVEL 1 PRA

A level 1 PRA consists of an analysis of plant design and operation, focused on the accident sequences that could lead to a core melt, their basic causes, and their frequencies. The analysis may or may not include external events. The quantitative results of this analysis consist of the frequencies of each core-melt accident. They can be used to derive the core-melt frequency by simply summing the frequencies of the individual sequences.

The quantitative results of a level 1 PRA flow directly from the analysis described in Chapter 6, "Accident-Sequence Quantification." The products of that analysis consist of the frequencies of the various accident sequences. If event trees consisting of sequences leading to core melt have been analyzed, the results of the analysis correspond directly to the level 1 quantitative results.

These results can be displayed in a table giving each accident sequence and its frequency. Alternatively, the frequencies for each sequence could be shown on the event tree. Often only the most probable sequences are displayed as the final quantitative results. A hypothetical example of such a display is presented in Table 13-1.

In addition to the accident-sequence frequencies, information pertaining to the plant-damage state associated with each sequence may be developed as part of the quantification process. Such information can be displayed in the form of a matrix containing the frequencies of each plant-damage state given each initiating event; an example is shown in Table 13-2. The particular sequences contributing to each plant-damage state for a given initiating event are delineated in the material pertaining to the particular event tree.

13.1.2 LEVEL 2 PRA

A level 2 PRA consists of an analysis of the physical processes of the accident and the response of the containment in addition to the analysis performed in a level 1 PRA. Besides estimating the frequencies of core-melt sequences, it predicts the time and mode of containment failure as well as the inventories of radionuclides released to the environment. As a result, core-melt accidents can be categorized by the severity of the release. External events may or may not be included in the analysis. The quantitative results of a level 2 PRA represent an integration of the results obtained in system analysis and in containment analysis. Event trees reflecting consequence distinctions are constructed and quantified in this analysis. As in a level 1 PRA, the product of the sequence-quantification task is a frequency for each event-tree sequence. In addition, the frequency of each plant-damage state may be estimated.

Table 13-1. Hypothetical sequence-frequency table

Core-melt sequence	Frequency (per reactor-year)
V	3.3×10^{-6}
S_1D	6.5×10^{-5}
S_2D	1.0×10^{-4}
TML	4.0×10^{-5}
TKQ	1.1×10^{-6}
Total	2.1×10^{-4}

In turn, the analysis of physical processes constructs and quantifies containment event trees for each accident sequence or for each plant-damage state. The associated radionuclide release is assessed, and release categories are generally defined. The results of these analyses may be simply a list of containment-failure modes, release categories, and their probabilities for each sequence, as illustrated in Table 13-3. Alternatively, the results may be in the form of a containment matrix presenting the frequency of a given release category for each plant-damage state, as illustrated in Table 13-4.

Table 13-2. The plant matrix for internal initiating events (mean value)[a,b]

		Plant event sequence category[c,d]																				
	Initiating event	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	VE
1	Large LOCA	0	0	0	0	0	0	0	0	0	0	0	0	1.40-3	1.10-7	2.24-6	3.90-9	5.20-3	3.87-7	3.77-7	2.60-10	0
2	Medium LOCA	0	0	0	0	0	0	0	0	0	0	0	0	4.64-4	8.89-8	6.51-6	7.31-9	5.20-3	3.86-7	4.66-8	8.15-12	0
3	Small LOCA	8.52-7	2.20-8	3.93-7	3.65-9	4.58-4	3.08-8	2.39-6	2.75-8	0	0	0	0	0	0	0	0	0	0	0	0	0
4	Steam-generator tube rupture	4.25-13	3.05-17	8.66-17	2.59-18	1.93-14	1.51-18	1.04-16	8.98-18	6.68-6	4.51-9	2.30-6	2.05-7	0	0	0	0	0	0	0	0	0
5	Steam break inside containment	7.45-7	5.33-11	1.52-10	4.55-12	2.62-6	3.94-9	2.21-6	1.95-8	6.47-7	2.39-10	1.14-7	1.02-8	0	0	0	0	0	0	0	0	0
6	Steam break outside containment	7.45-7	5.23-11	3.12-10	2.87-11	2.62-6	3.11-9	1.19-6	1.08-8	6.45-7	1.67-10	5.79-8	6.57-8	0	0	0	0	0	0	0	0	0
7	Loss of main feedwater	7.53-7	4.90-11	3.11-10	2.84-11	3.44-8	6.12-12	2.64-10	2.01-10	4.52-8	1.04-10	5.96-8	1.26-9	0	0	0	0	0	0	0	0	0
8	Trip of one MSIV	0	0	0	0	1.75-9	3.33-12	1.36-10	1.02-11	3.45-8	1.04-10	5.91-8	6.47-8	0	0	0	0	0	0	0	0	0
9	Loss of RCS flow	6.78-7	4.41-11	2.81-10	2.56-11	2.52-6	2.82-9	1.13-6	2.21-8	3.36-8	9.33-11	5.42-8	5.84-8	0	0	0	0	0	0	0	0	0
10	Core power excursion	9.83-14	6.87-18	4.05-17	3.70-18	4.69-15	8.19-19	3.43-17	1.44-17	4.66-15	1.36-17	7.58-15	8.46-15	0	0	0	0	0	0	0	0	0
11a	Turbine trip	7.49-7	5.28-11	3.10-10	2.82-11	3.58-8	6.31-12	2.66-10	1.10-10	4.59-8	1.07-10	6.10-8	1.25-9	0	0	0	0	0	0	0	0	0
11b	Turbine trip, loss of offsite power	1.03-7	5.68-12	1.37-12	4.36-9	1.17-8	6.46-13	7.11-14	8.90-19	1.81-7	9.91-12	2.53-13	3.46-6	0	0	0	0	0	0	0	0	0
11c	Turbine trip, loss of service water	0	0	7.45-7	5.26-9	0	0	3.54-8	2.45-8	0	0	2.81-8	3.56-7	0	0	0	0	0	0	0	0	0
12	Spurious safety injection	7.54-7	5.16-11	1.54-10	4.60-12	2.64-6	3.96-9	2.25-6	3.89-9	3.99-8	1.62-10	1.15-7	1.03-8	0	0	0	0	0	0	0	0	0
13	Reactor trip	0	0	0	0	1.78-9	3.34-12	1.35-10	1.01-11	4.54-8	1.07-10	6.23-8	1.27-9	0	0	0	0	0	0	0	0	0
V	Interfacing-systems LOCA	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

[a] From the Zion PRA (Commonwealth Edison Company, 1981).

[b] Mean values for conditional frequency of entering each plant event sequence category given that a specific initiating event has occurred.

[c] The key to plant event sequence categories is as follows: A, large-LOCA behavior; S, small-LOCA behavior; T, transient behavior; E, early melt; L, late melt; F, fan coolers are operating; C, containment sprays are operating.

[d] Values are presented in abbreviated scientific notation: 1.11-5 = 1.11×10^{-5}.

Table 13-3. Containment-failure modes, their probabilities, and release categories for selected accident sequences[a]

Sequence	Release category 1	2	3	4	5	6	7
AD	$\alpha = .01$		$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$
AFH	$\alpha = .01$	$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$	
AH	$\alpha = .01$		$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$
ACD	$\alpha = .01$	$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$	
S_1D	$\alpha = .01$		$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$
S_1FH	$\alpha = .01$	$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$	
S_1CD	$\alpha = .01$	$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$	
S_1DF	$\alpha = .01$		$\gamma = .2$	$\beta = .0073$		$\varepsilon = .8$	
S_1YD	$\alpha = .01$		$\gamma = .2$		$\beta = .0073$	$\delta = .8$	
S_1H	$\alpha = .01$		$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$
S_2FH	$\alpha = .01$	$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$	
S_2D	$\alpha = .01$		$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$
S_2H	$\alpha = .01$		$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$
S_2CD	$\alpha = .01$	$\gamma = .2$		$\beta = .0073$		$\varepsilon = .8$	
S_2YD	$\alpha = .01$		$\gamma = .2$		$\beta = .0073$	$\delta = .8$	
S_2DF	$\alpha = .01$		$\gamma = .2$	$\beta = .0073$		$\varepsilon = .8$	
S_3H	$\alpha = .0001$		$\gamma = .5$		$\beta = .0073$		$\varepsilon = .5$
S_3FH	$\alpha = .0001$	$\gamma = .5$		$\beta = .0073$		$\varepsilon = .5$	
S_3D	$\alpha = .0001$		$\gamma = .5$		$\beta = .0073$		$\varepsilon = .5$

[a]From Kolb et al. (1981).

Given these results along with the products of a level 1 analysis, the quantitative results of a level 2 PRA are developed as follows:

1. The appropriate containment event trees are combined with each system-event-tree sequence to develop complete accident-sequence descriptions (i.e., combinations of initiating events, system successes and failures, and containment-failure modes).

2. Each accident-sequence frequency is calculated, including the probability of the containment-failure mode.

3. Each accident sequence is assigned to a particular radionuclide-release category.

This may be done either by combining the table of containment-failure modes and release categories with the list of accident sequences and their frequencies or by multiplying the plant matrix and the containment matrix.

The results are generally displayed in a table that groups the accident sequences and their frequencies by radionuclide-release category. An example of such a display, taken from the Reactor Safety Study Methodology

Table 13-4. Containment matrix[a]

Plant state	Release category[b]										
	Z-1	2	2R	Z-3	5R	Z-5	6	7	8A	8B	2RV
SEFC	0	1.000-4	0	1.781-7	1.309-4	8.985-5	0	1.740-4	0	9.996-1	0
SEF	1.781-7	1.899-4	1.309-4	0	0	0	1.740-4	0	9.996-1	0	0
SEC	0	1.000-4	0	1.781-7	1.309-4	8.985-5	0	1.740-4	0	9.996-1	0
SE	2.186-6	1.999-4	9.996-1	0	0	0	1.900-4	0	1.000-4	0	0
SLFC	0	1.000-4	0	9.084-6	1.309-4	8.985-5	0	1.740-4	0	9.995-1	0
SLF	9.084-6	1.899-4	1.309-4	0	0	0	1.740-4	0	9.995-1	0	0
SLC	0	1.000-4	0	9.084-6	1.309-4	8.985-5	0	1.740-4	0	9.995-1	0
SL	2.186-6	1.999-4	9.996-1	0	0	0	1.900-4	0	1.000-4	0	0
TEFC	0	1.000-4	0	9.797-8	1.027-4	9.982-5	0	1.899-4	0	9.995-1	0
TEF	9.797-8	1.999-4	1.027-4	0	0	0	1.899-4	0	9.995-1	0	0
TEC	0	1.000-4	0	9.797-8	1.027-4	9.982-5	0	1.899-4	0	9.995-1	0
TE	2.186-6	1.999-4	9.996-1	0	0	0	1.900-4	0	1.000-4	0	0
AEFC	0	1.000-4	0	1.979-6	1.499-4	1.998-10	0	1.499-4	0	9.996-1	0
AEF	1.979-6	1.000-4	1.499-4	0	0	0	1.499-4	0	9.996-1	0	0
AEC	0	1.000-4	0	1.979-6	1.499-4	1.998-10	0	1.499-4	0	9.996-1	0
AE	1.000	0	0	0	0	0	0	0	0	0	0
ALFC	0	1.000-4	0	1.979-6	9.999-5	2.204-10	0	1.899-4	0	9.996-1	0
ALF	1.979-6	1.000-4	9.999-5	0	0	0	1.899-4	0	9.996-1	0	0
ALC	0	1.000-4	0	1.979-6	9.999-5	2.204-10	0	1.899-4	0	9.996-1	0
AL	1.000	0	0	0	0	0	0	0	0	0	0
VE	0	1.000	0	0	0	0	0	0	0	0	0

[a]From the Zion PRA (Commonwealth Edison Company, 1981).
[b]Values are presented in abbreviated scientific notation: 1.78-7 = 1.78×10^{-7}.

Applications Program, is shown in Figure 13-1. Often only the highest-frequency--that is, the dominant--accident sequences are presented. Sometimes the results are accompanied by a histogram showing release-category frequencies, as in Figure 13-1. Alternatively, the results can be presented in the form of a matrix, as in Table 13-5. In this case, the particular sequences contributing to each release category for a given initiating event, denoted by ϕ, are delineated in the material pertaining to the particular event tree.

Sequence	Release category						
	1	2	3	4	5	6	7
T_2MLU			$\gamma\ 6.0 \times 10^{-7}$		$\beta\ 8.8 \times 10^{-9}$		$\epsilon\ 6.0 \times 10^{-7}$
T_1MLU			$\gamma\ 1.0 \times 10^{-6}$		$\beta\ 1.5 \times 10^{-8}$		$\epsilon\ 1.0 \times 10^{-6}$
V		$V < 4.0 \times 10^{-6}$					
$T_1\ (B_3)\ MLU$			$\gamma\ 1.1 \times 10^{-6}$		$\beta\ 1.6 \times 10^{-8}$		$\epsilon\ 1.1 \times 10^{-6}$
T_2MQ-H			$\gamma\ 5.5 \times 10^{-6}$		$\beta\ 8.0 \times 10^{-8}$		$\epsilon\ 5.5 \times 10^{-6}$
S_3H			$\gamma\ 5.0 \times 10^{-6}$		$\beta\ 7.3 \times 10^{-8}$		$\epsilon\ 5.0 \times 10^{-6}$
S_1D	$\alpha\ 6.7 \times 10^{-8}$		$\gamma\ 1.3 \times 10^{-6}$		$\beta\ 4.9 \times 10^{-8}$		$\epsilon\ 5.4 \times 10^{-6}$
T_2MQ-FH		$\gamma\ 2.5 \times 10^{-6}$		$\beta\ 3.7 \times 10^{-8}$		$\epsilon\ 2.5 \times 10^{-6}$	
S_3FH		$\gamma\ 2.1 \times 10^{-6}$		$\beta\ 3.1 \times 10^{-8}$		$\epsilon\ 2.1 \times 10^{-6}$	
S_2FH	$\alpha\ 1.3 \times 10^{-8}$			$\beta\ 9.5 \times 10^{-9}$		$\epsilon\ 1.0 \times 10^{-6}$	
T_2MLUO			$\gamma\ 4.1 \times 10^{-6}$		$\beta\ 5.9 \times 10^{-8}$		$\epsilon\ 4.1 \times 10^{-6}$
T_2KMU			$\gamma\ 3.9 \times 10^{-6}$		$\beta\ 5.7 \times 10^{-8}$		$\epsilon\ 3.9 \times 10^{-6}$
S_2D	$\alpha\ 2.0 \times 10^{-8}$		$\gamma\ 4.0 \times 10^{-7}$		$\beta\ 1.5 \times 10^{-8}$		$\epsilon\ 1.6 \times 10^{-6}$
S_3D			$\gamma\ 7.0 \times 10^{-7}$		$\beta\ 1.0 \times 10^{-8}$		$\epsilon\ 7.0 \times 10^{-7}$
T_1MLUO			$\gamma\ 2.7 \times 10^{-6}$		$\beta\ 3.9 \times 10^{-8}$		$\epsilon\ 2.7 \times 10^{-6}$
T_3MLUO			$\gamma\ 5.5 \times 10^{-7}$		$\beta\ 8.0 \times 10^{-9}$		$\epsilon\ 5.5 \times 10^{-7}$
T_2MQ-D			$\gamma\ 7.5 \times 10^{-7}$		$\beta\ 1.1 \times 10^{-8}$		$\epsilon\ 7.5 \times 10^{-7}$
Category total	1.1×10^{-7}	1.0×10^{-5}	2.9×10^{-5}	9.7×10^{-8}	4.6×10^{-7}	7.3×10^{-6}	3.5×10^{-5}

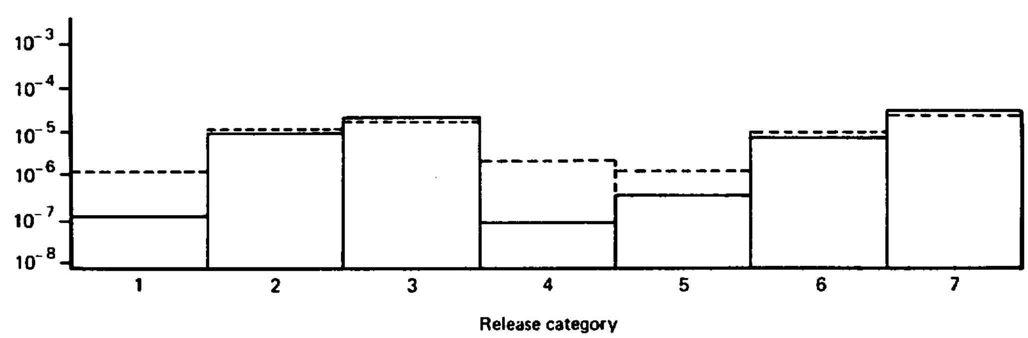

Figure 13-1. Dominant accident sequences with histogram. The category totals given in the tabulation are unsmoothed totals that include the contribution from all the nondominant sequences not shown. From Kolb et al. (1981).

Table 13-5. Release category frequencies for each initiating event

ϕ	ρ Z-1	2	2R	Z-3	5R	Z-5	6	7	8A	8B	2RV
1	3.911-12	5.207-10	5.188-14	1.229-11	6.864-10	1.340-15	8.458-14	1.126-9	4.670-10	6.204-6	0
2	6.880-12	5.331-10	4.880-14	1.055-11	5.551-10	1.166-15	8.143-14	9.945-10	4.462-10	5.328-6	0
3	1.245-14	1.635-9	1.102-9	1.480-10	2.139-9	1.468-9	5.349-13	2.543-9	1.868-9	1.633-5	0
4	1.095-14	2.293-11	5.000-9	2.147-14	2.250-11	2.187-11	9.714-13	4.160-11	1.105-10	2.190-7	0
5	9.475-17	6.020-13	2.791-11	4.144-14	7.594-13	5.423-13	6.001-15	1.048-12	3.979-12	5.953-9	0
6	1.839-16	5.093-13	7.191-11	3.272-14	6.284-13	4.507-13	1.421-14	8.705-13	3.135-12	4.941-9	0
7	1.722-14	4.633-10	7.698-9	2.375-12	5.889-10	4.201-10	1.615-12	8.117-10	8.231-10	4.614-6	0
8	7.043-16	4.075-12	3.175-10	8.195-15	4.128-12	3.994-12	6.547-14	7.600-12	2.706-11	4.004-8	0
9	2.738-14	1.620-10	8.324-9	1.192-11	2.082-10	1.444-10	1.767-12	2.795-10	1.059-9	1.601-6	0
10	4.227-22	3.016-19	1.932-16	1.405-21	3.363-19	2.389-19	3.682-20	4.619-19	5.046-19	2.629-15	0
11a	1.148-14	3.304-10	5.122-9	1.740-12	4.199-10	2.998-10	1.086.12	5.793-10	6.133-10	3.291-6	0
11b	4.362-13	4.159-11	1.995-7	8.202-15	1.935-12	1.634-12	3.791-11	3.129-12	2.089-11	1.703-8	0
11c	7.927-15	1.485-13	3.625-10	4.296-16	9.870-14	6.854-14	6.889-14	1.326-13	3.627-14	7.597-10	0
12	4.263-14	3.711-10	9.025-9	2.835-11	4.800-10	3.324-10	2.179-12	6.430-10	2.654-9	3.686-6	0
13	1.070-14	4.237-11	4.826-9	1.054-13	4.264-11	4.117-11	9.957-13	7.838-11	4.162-10	4.132-7	0
V	0	1.050-7	0	0	0	0	0	0	0	0	0
Total = $\phi\rho$	1.136-11	1.092-7	2.413-7	2.155-10	5.150-9	2.735-9	4.738-11	7.410-9	8.513-9	4.176-5	0

[a]From the Zion PRA (Commonwealth Edison Company, 1981).
[b]Values are presented in abbreviated scientific notation: 3.911-12 = 3.911×10^{-12}.

13.1.3 LEVEL 3 PRA

A level 3 PRA analyzes the transport of radionuclides through the environment and assesses the public-health and economic consequences of the accident in addition to performing the analyses of a level 2 PRA. The quantitative results of this level of PRA integrate results from the systems analysis, the containment analysis, and the consequence analysis. Complementary cumulative distribution functions (CCDFs) are the most common integrated products of these analyses. The results are generally presented in the form of a CCDF accompanied by a table of sequences whose frequencies are grouped by release category.

In calculating a CCDF, magnitudes of health and other effects are predicted for each combination of a weather sequence and an accident sequence. With this combination can be associated a frequency that is the product of the frequency of occurrence predicted for the accident sequence (derived as in Section 13.1.2) and the probability of occurrence for the weather sequence. All combinations of a weather sequence and an accident sequence therefore give a probability distribution on the magnitude of the health or other effects, and this probability distribution can be readily presented in cumulative form, as in Figure 13-2.

The results of a level 3 PRA can also be presented in matrix form. To do so, a site matrix containing the cumulative probability of a given consequence for each release category is developed, as in Table 13-6. When the site matrix is combined with the initiating-event frequencies, the plant matrix, and the containment matrix, the results needed to plot a CCDF are obtained.

Early fatalities and latent-cancer fatalities (shown in Figure 13-2) are probably the most common consequences for which CCDFs are developed. Other possible consequences for which CCDFs can be obtained include--

1. Early illness, which is essentially defined by reference to a whole-body radiation dose large enough to require hospitalization.

2. Genetic effects.

3. Areas requiring decontamination or interdiction.

CCDFs can also be calculated for such quantities as the number of thyroid nodules arising in the affected population, the amount of property damage, or any other consequence that is of interest to the user.

13.2 UNCERTAINTY ANALYSIS

It has been recognized throughout this guide that many sources of uncertainty are associated with each part of the analysis. Consequently, the results discussed in the preceding section can be made more meaningful with some assessment of the associated uncertainties, which arise from the

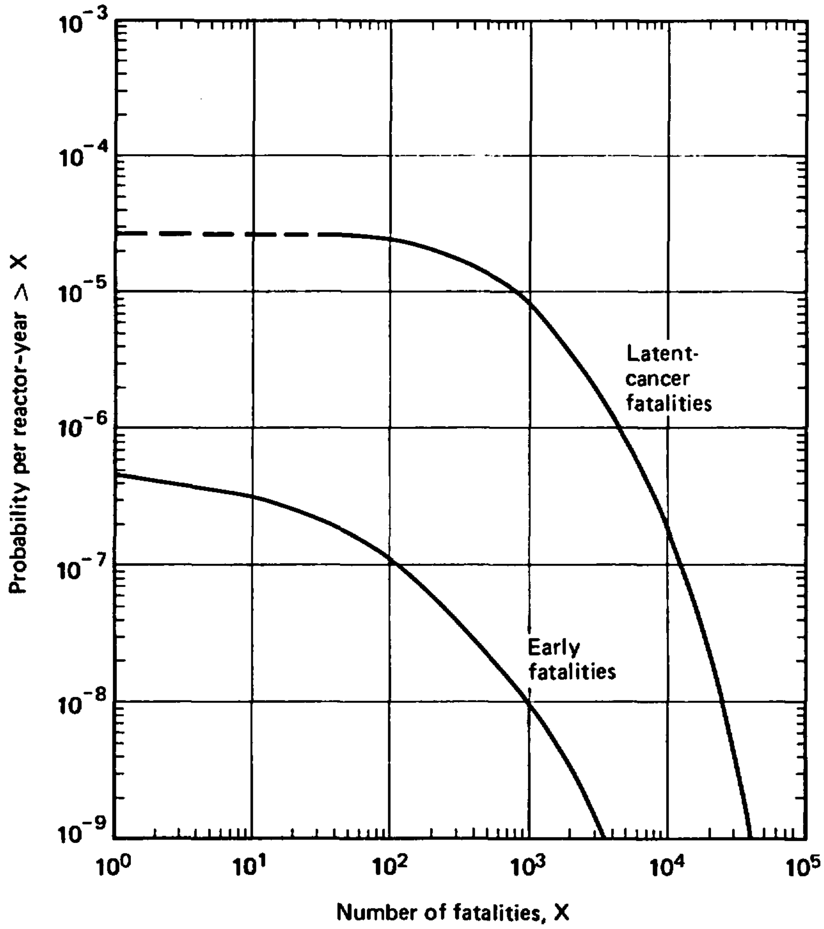

Figure 13-2. Probability distribution for early fatalities and latent-cancer fatalities. From the Reactor Safety Study (USNRC, 1975).

uncertainties identified in each part of the analysis. The assessment may be purely qualitative, but a quantitative assessment would be more informative. This quantitative assessment may be merely at the level of numerical bounds on the results, or it may be a full-scale quantitative assessment extending to probabilistic statements of the confidence-interval type. Chapter 12 discussed some methods that have been used to estimate uncertainties. This section explains which of these methods are applicable for PRAs of various levels of scope and how the uncertainties might be presented.

13.2.1 LEVEL 1 PRA

This level of PRA is the one for which a quantitative uncertainty analysis is best developed. Such an uncertainty analysis involves the estimation of uncertainties in the input parameters of the event- and fault-tree models used to describe plant behavior and the propagation of the uncertainties through the trees. The estimation of input uncertainties is discussed in Chapter 5, and methods for propagation are discussed in Section 12.5.3.

Table 13-6. Point estimate of the site matrix S^T (S transposed) for damage index: early fatalities[a,b]

Damage index	Release category										
	Z-1	2	2R	Z-3	5R	Z-5	6	7	8A	8B	2RV
1.0	1.667-1	3.472-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
2.0	1.563-1	3.472-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
3.0	1.563-1	3.472-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
5.0	1.458-1	3.472-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
7.0	1.458-1	2.778-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
1.0+1	1.354-1	2.778-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
2.0+1	1.146-1	2.431-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
3.0+1	9.375-2	2.431-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
5.0+1	8.333-2	2.431-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
7.0+1	6.250-2	2.083-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
1.0+2	6.250-2	1.389-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
2.0+2	4.167-2	1.389-2	1.042-2	2.083-2	0	6.944-3	0	0	0	0	0
3.0+2	4.167-2	1.389-2	6.944-3	2.083-2	0	6.944-3	0	0	0	0	0
5.0+2	3.125-2	1.389-2	6.944-3	2.083-2	0	6.944-3	0	0	0	0	0
7.0+2	3.125-2	6.944-3	6.944-3	2.083-2	0	6.944-3	0	0	0	0	0
1.0+3	2.083-2	6.944-3	6.944-3	2.083-2	0	3.472-3	0	0	0	0	0
2.0+3	1.000-3	6.944-3	6.944-3	2.083-2	0	3.472-3	0	0	0	0	0
3.0+3	1.000-3	6.944-3	3.472-3	1.042-2	0	3.472-3	0	0	0	0	0
5.0+3	1.000-3	1.000-3	1.000-3	0	0	0	0	0	0	0	0
7.0+3	2.000-4	1.000-3	1.000-3	0	0	0	0	0	0	0	0
1.0+4	2.000-4	1.000-4	1.000-4	0	0	0	0	0	0	0	0
2.0+4	2.000-4	1.000-4	0	0	0	0	0	0	0	0	0
3.0+4	2.000-4	0	0	0	0	0	0	0	0	0	0
5.0+4	0	0	0	0	0	0	0	0	0	0	0
7.0+4	0	0	0	0	0	0	0	0	0	0	0
1.0+5	0	0	0	0	0	0	0	0	0	0	0
2.0+5	0	0	0	0	0	0	0	0	0	0	0

[a]From the Zion PRA (Commonwealth Edison Company, 1981).
[b]Values are presented in abbreviated scientific notation: 1.0+1 = 1.0×10^1.

The results may be displayed as upper and lower bounds on accident-sequence frequencies or as probability distributions on the frequencies, depending on the philosophy adopted for uncertainty analysis. The bounds may be regarded as approximate confidence bounds at an appropriate level of confidence. It should be remembered that, however the uncertainties are displayed, they are conditioned on an assumption of validity and completeness for the fault- and event-tree models, and are therefore a measure of the uncertainty introduced by an imprecise knowledge of the input parameters.

13.2.2 LEVEL 2 PRA

A level 2 PRA involves an evaluation of the containment event tree in addition to the system analysis. The techniques for estimating the uncertainties in the frequencies of the accident sequences in a particular release category are essentially the same as those for a level 1 PRA if the subjectivist approach to uncertainty is adopted. The probabilities on the branches of the containment event tree are, however, based mainly on judgment rather than on data. For instance, one branch on the containment event tree might relate to the likelihood of a steam explosion with attendant containment failure. As discussed in Chapter 7, this likelihood is estimated on the basis of expert judgment. Thus, a full-scale quantitative uncertainty analysis for a level 2 PRA is most easily performed with a subjectivist perspective on uncertainty. Uncertainties in the frequencies in the table or histogram in Figure 13-1 are then displayed as bounds or as a probability distribution (Figure 13-3), as for a level 1 PRA.

An alternative approach would be to tabulate significant sources of uncertainty with a qualitative assessment of their effects.

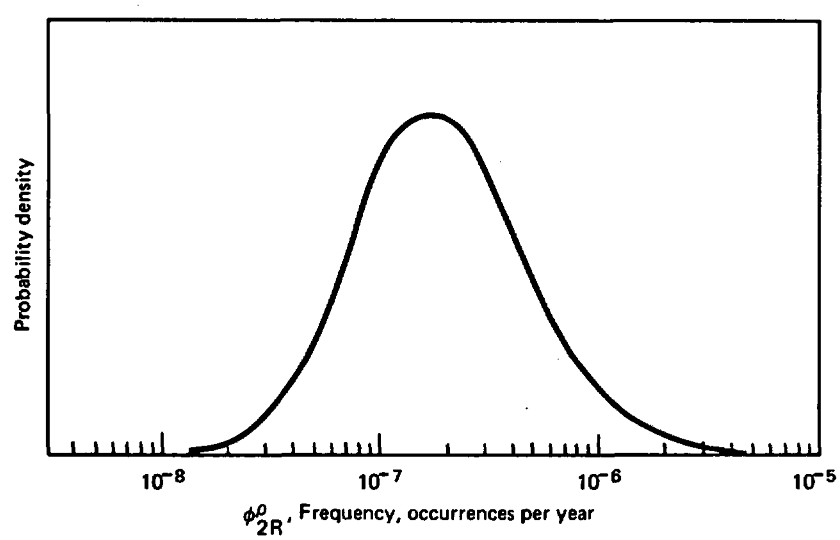

Figure 13-3. Probability distribution for the frequency of release category 2R (internal events only). From the Zion PRA (Commonwealth Edison Company, 1981).

13.2.3 LEVEL 3 PRA

The uncertainty analysis for a level 3 PRA must address the uncertainties from the system analysis, the containment analysis, and the consequence analysis. The origin of these uncertainties and recommendations for their treatment are discussed in the chapters on these topics, and methods for propagating and combining the uncertainties are covered in Chapter 12. To date, the PRA that appears to have the most complete treatment of uncertainties is the Zion study (Commonwealth Edison Company, 1981). A display of its results is illustrated in Figure 13-4. The curve identified by P = .9 is the 90th percentile curve of a probability distribution over sets of CCDFs and represents the CCDF that, in the analysts' judgment, bounds 90 percent of the perceived possible CCDFs.

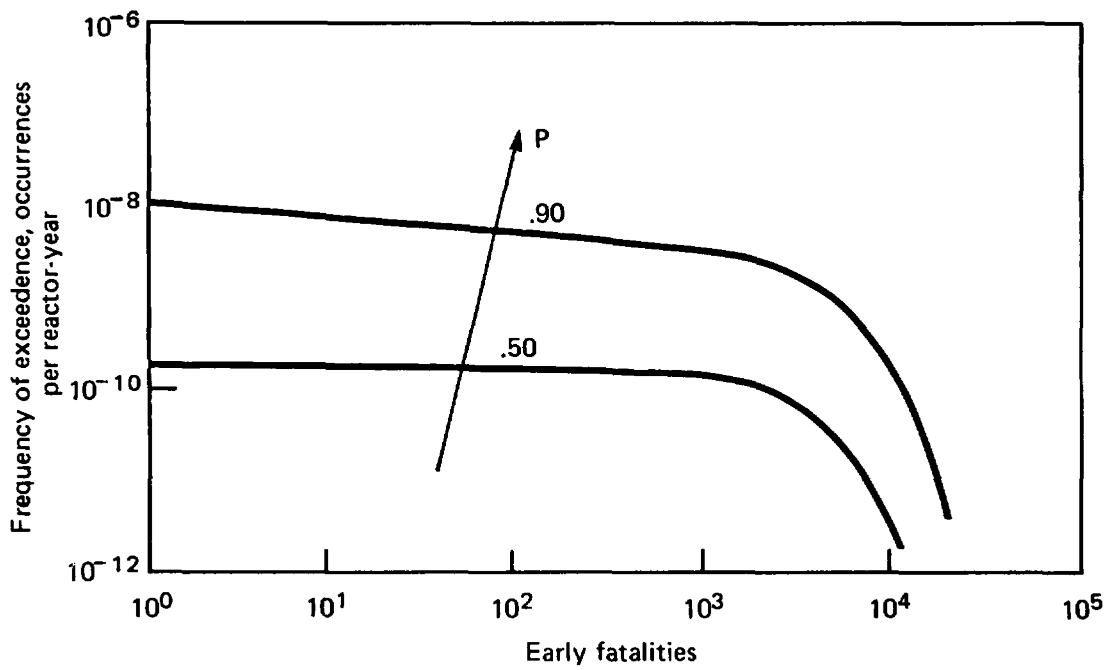

Figure 13-4. Level 2 risk diagram for fatalities: base case, internal and external risk. From the Zion PRA (Commonwealth Edison Company, 1981).

It might be thought preferable not to try to quantify all uncertainties, because the quantification of some uncertainties would of necessity be subjective. An alternative approach is that of the Limerick PRA (Philadelphia Electric Company, 1981), which quantifies uncertainties on the frequencies of accident sequences, but discusses qualitatively uncertainties in the analyses of containment phenomena and the offsite consequences of radionuclide releases. The Limerick approach is illustrated in Table 13-7. Ideally, a table of this type, summarizing the uncertainties addressed and discussing their potential effects, should also accompany the quantitative representation of uncertainty, as typified by the Zion study, to qualify the uncertainties quoted.

Table 13-7. Summary of areas of uncertainty with a moderate effect on the early-fatality CCDF for the Limerick plant[a]

Subject	Assumption used in analysis	Impact[b]
Methodology:		
Incomplete or missing accident sequences	All possible accident sequences are not included. Because of the infinite number of possibilities that accident sequences could take, and because not all these sequences have been included in the quantification effort, it is possible that a sequence with a low probability of occurrence may not be represented.	P
Containment failure leads directly to core melt	Several potential mechanisms connecting containment failure with eventual core melt have been identified. However, this remains an assumption and an area of potential conservatism.	P
Data:		
Meteorological data	A 5-year sample of data (1972-1976) is used to characterize the site weather patterns. Sharp changes in future weather patterns are not included.	C
ADS initiation by operator	For some accident sequences manual depressurization is required. The probability of failure is estimated as 1/500 demands. Because of the uncertainty in the human-error probabilities, this operation is assumed to have a larger uncertainty than typical hardware failures.	P
Equipment:		
Improvement in hardware based on operating experience	Operating problems have resulted in selective improvements in component design. This is the case for diesels, relief valves, scram discharge volume, etc. Some of these improvements are not reflected in the analysis since failure rates are based on the total available data.	P
Containment:	The manner in which the RPV fails is uncertain. The INCORE method, modeled for a PWR, assumes that the RPV ruptures from the stress of the molten core rather than melting through. This model allows the entire bottom head of the vessel to fail at one instant. Other methods assume failure from melting, but the manner of melting is also uncertain.	C

[a]Excerpted from the Limerick PRA (Philadelphia Electric Company, 1981).

[b]Key: P, probability; C, consequences.

13.3 INTERPRETATION OF RESULTS

Quantitative results are not the only results that are important. The qualitative insights derived from analyzing and interpreting the quantitative results are an important product of the analysis. Qualitative insights are developed by analyzing the results of the analysis to identify the plant features that contribute significantly to risk. These insights can be gained in several ways.

One common practice involves an analysis of the most probable cut sets of the dominant accident sequences--that is, the sequences that contribute the most to risk. The most probable cut sets of these sequences represent the most probable ways the sequence can occur. An examination of the cut sets of the dominant accident sequences provides one indication of the plant features that contribute significantly to risk.

If an expression for the combination of failures leading to the accident sequences has been developed, the identification of significant contributors to risk is a straightforward exercise. If the matrix formalism has been used, the same information can be obtained by tracing back through the event trees to identify the sequences that contribute most to a particular plant-damage state, then examining the fault trees for the systems involved in the particular sequence to identify potential cut sets, and finally examining the cause tables to ascertain the most important failure modes.

The results are often analyzed to determine the contribution to risk from classes of events, such as types of initiating events, testing and maintenance, or human errors. The matrix formalism is, perhaps, advantageous for finding the contribution due to particular initiating events. This is done by simply adding the entries in a particular row of the matrix. For classes of primary events, the approach of first generating an equation for the sequence in terms of failure combinations may be advantageous because the contribution of each particular event is shown explicitly.

Further insight can be gained by performing an importance analysis on the results. A variety of importance measures have been developed to obtain different insights into the relative importance of various events (plant features) to the result. These importance measures take into account not only the probability of the event but also the number and probabilities of the cut sets to which the event contributes.

One of the most often used measures is the Fussell-Vesely measure. For a given event, the Fussell-Vesely measure is formed by dividing the total probability of all minimal cut sets containing the event by the sum of all minimal cut sets with or without the event. Several other measures are described in a recent report (Lambert and Davis, 1981). The importance calculations may show that a given event, while not being the most probable event in a given sequence, may be the most significant because it contributes to many different cut sets. "Significant" in this sense generally means those events that have the most potential for changing risk if the probability of the event changes.

Frequently, the study leads to insights into plant design and operational peculiarities. Although these insights may not show up in the dominant accident sequences or as significant contributors to risk, they might still be of value and should be documented in the discussion of results.

Examples of qualitative insights that could be derived from the Reactor Safety Study (USNRC, 1975) include the relative importance to risk of sequences initiated by small-break loss-of-coolant accidents and transient events as well as the importance of human errors, testing, and maintenance to system unavailabilities. Such insights, of course, apply only to the particular plant under study. Caution must be exercised in drawing generic conclusions on the basis of one particular study.

Another important dimension to the interpretation of results is a qualitative discussion of the uncertainties in the answers and the principal sources of these uncertainties. The insights derived from the uncertainty and sensitivity analyses add valuable perspective to the results. This is particularly true if the most significant contributors to risk are accompanied by large uncertainties in assumptions or data.

13.4 CONCLUDING REMARKS

Probabilistic risk assessment techniques are rapidly improving. Ongoing research and development efforts both here and abroad indicate potential advances in the assessment of the radionuclide source term, containment response, human reliability, and external events, for example.

It is not intended that any of the methods or techniques described in this guide be viewed as prescriptive, either individually or collectively. This procedures guide is intended to reflect the current state of the art. Given that state, each of the individual PRA tasks is described, where appropriate, in terms of alternative methods or techniques that have been recognized as being useful. The guide also points to the strengths and limitations of each such method where possible. The users of this guide must be aware of the rapid evolution of the techniques, information, and technology associated with probabilistic risk assessments. In taking advantage of these advances, it will be incumbent upon the users to carefully evaluate each advance and to satisfy themselves as to its validity and usefulness in the context of a given PRA project.

It is in the nature of PRA studies that each such study makes some contribution to the state of the art through the refinement of existing techniques or simply the expansion of collective knowledge. This fortunate circumstance suggests that this guide or any such guide must continue to evolve over time.

The uncertainties in the data should be carried through the analysis where possible, and studies of model sensitivities should be performed where needed. However, it is important to recognize that useful results can be obtained even though the estimates may have large uncertainties. Many of the insights gained in the analysis are not strongly dependent on the uncertainties associated with the analysis. The most important product of the analysis is the framework of engineering logic generated in constructing the models; the numerical estimates of frequencies need only be accurate enough to distinguish risk-significant plant features from those of lesser importance.

The patterns, ranges, and relative behavior that are obtained can be used to develop insights into the design and operation of a plant--insights that can be gained only from an integrated consistent approach like that described in this guide. These insights are applicable to utility and regulatory decisionmaking, although they should not be the sole basis for such decisions. Comparative evaluations can identify the features of the plant that are significant contributors to predicted risk, allowing both the owner and the regulators to focus on them and establish whether they are acceptable. Similarly, the level of regulatory efforts addressed at items with little influence on the predicted risk can be evaluated in a better context. The ordering of dominant accident sequences provides a framework for value-impact analyses of plant modifications. The plant models can be used as a tool for optimizing surveillance intervals and preventive-maintenance programs, improving procedures, and providing perspective to operations personnel on potential multiple-fault events. Employed early, PRA techniques can be used to guide the design process and to establish priorities for quality-assurance activities; if properly developed, they also present a rational method for interpreting operational data.

Thus, PRA techniques can serve as a valuable adjunct to the methods currently used in decisionmaking in both industry and government. Although they are not yet developed to the point where they can be used without caution by decisionmakers, they do provide a framework of integrated engineering logic that can be used to identify and evaluate critical areas that influence the availability or the safety of the plant.

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Appendix A

Charter of the PRA Procedures Guide Project

DEVELOPMENT OF A METHOD FOR
SYSTEMATIC PROBABILISTIC RISK ASSESSMENTS
OF NUCLEAR POWER PLANTS

BACKGROUND

Since the completion of the Reactor Safety Study (WASH-1400), the NRC has been exploring ways to systematically apply probabilistic analysis to nuclear power plants. The NRC, in its Interim Reliability Evaluation Program (IREP) which is now under way, is developing and giving trial use to a procedures guide which could be the basis for systematic analysis of all nuclear power plants, a National Reliability Evaluation Program (NREP). Before settling on any procedures guides for such a broad undertaking the NRC is interested in obtaining the advice and participation of many competent parties, including the nuclear industry and probabilistic analysis experts from within and without the nuclear industry. Thus the NRC seeks to initiate and support a project to develop a procedures guide, a method for systematic probabilistic risk assessments of nuclear power plants.

THE PROJECT

The project envisioned is to develop a Procedures Guide for the systematic application of probabilistic and reliability analysis to nuclear power plants. This Procedures Guide is expected to define the acceptable methodology for performance of such studies. The Procedures Guide is expected to address the following subject areas: (1) system reliability analysis, (2) accident sequence classification, (3) frequency assessment for classes of accident sequences, (4) estimation of radiologic release fractions for core-melt accident sequences, and (5) consequence analysis. For each of these subject areas, the Procedures Guide should delineate (1) acceptable analytic techniques, (2) acceptable assumptions and modeling approximations including the treatment of statistical data, common cause failures and human errors, (3) treatment of uncertainty, (4) acceptable standards for documentation, and (5) quality control. The Procedures Guide is expected to define a practical scope of analysis for such systematic review conducted in the next few years. Thus, the Procedures Guide might recommend omission, simplification, or postponement of some elements of a complete analysis. If it does, the Procedures Guide may or may not include specific guidance on when or how to address these elements later.

The NRC sees this situation as a unique opportunity to use the resources of two technical societies, the Institute of Electrical and Electronics Engineers (IEEE) and the American Nuclear Society (ANS), to develop and review statements of useful PRA methodology and recommend applications. The technical society activities envisioned are two conferences linked by a series of workshops which will prepare material for the conferences. The IEEE is seen as the principal host of the first of these conferences, the Review Conference, because their membership and ability to contribute span not only the nuclear industry but other industries which have used probabilistic and reliability analysis for some time. The ANS is seen as the principal host of the second of these conferences, the Topical Conference, since their membership pervades the nuclear establishment. The ANS is uniquely able to bring the widest range of views with nuclear industry expertise to bear on the matter.

The NRC would work directly with each of the two technical societies supporting and cosponsoring activities specifically related to this project. The societies would be expected to use their resources to obtain the attention and participation of technically qualified parties. The NRC, with Steering Committee advice, may select a time or times in the course of this project to make materials available for general public comment through other channels such as publication in the Federal Register, etc.

POLICY ACTIVITIES

The activity planned to develop a consensus Procedures Guide for probabilistic analysis is premised on the expectation that the use of such a Procedures Guide would be systematically undertaken in the nuclear power industry and that the results of such analyses would be used in regulatory decisionmaking. Neither NRC nor the owners of the nuclear plants can or would delegate their policy setting responsibilities to others. Therefore, the NRC is expected to continue to develop specific policies on the extent and manner in which probabilistic analysis will be used in the regulatory process. The nuclear plant owners are expected to pursue resolution of these policy issues as well, operating individually and through the Atomic Industrial Forum (AIF), through its Policy Committee on Nuclear Regulation and its subordinate committees and subcommittees. The effectiveness of the preparation and use of the Procedures Guide depends heavily on timely policy input to the technical effort. Therefore, it is important that both NRC and the industry pursue resolution of these policy issues through normal channels as well as by dedicating persons to participate in this technical society effort who are significantly involved in resolution of these policy issues.

ORGANIZATION

The organization of this project is intended to enable the NRC and the nuclear industry to work closely with the two technical societies in cosponsoring their activities in a coordinated scheme of action. The project will be directed by a Steering Committee under the joint chairmanship of two representatives of the technical societies, the IEEE and the ANS. The principal work of developing technical documents for the project will be performed by a project Technical Committee. Each of the conferences is expected to have its own conference committee.

The Steering Committee, excluding the two co-chairmen, is drawn from different sources as follows:

Affiliation	Number of Members
NRC	3
IEEE	3
ANS	2
DOE	1
AIF	1
Other Nuclear Industry	4

The Steering Committee will set its final membership. At its discretion, it may include in its number the chairman of the project Technical Committee and the chairmen of the conference committees when they have been chosen by their respective professional societies. The chairman and the members of the Technical Committee will be chosen by the Steering Committee. The Technical Committee is expected to include about seven or eight specialists who have strong technical knowledge of both nuclear power plant analysis and probabilistic and reliability analysis techniques. These experts will be drawn from the nuclear industry, the national laboratories, and the NRC. In addition, as directed by the Steering Committee, the Technical Committee will be augmented from time to time by additional members, drawn from non-nuclear industry and government experts in risk assessment methodologies. They will be assisting the Technical Committee to develop realistic descriptions and evaluations of candidate probabilistic analysis methods as well as reviews of pertinent experience in the use of probabilistic and reliability analysis for consideration by the Steering Committee and the technical society meetings.

It is expected that, under the Steering Committee's direction, the augmented Technical Committee will review the procedures for PRA which have been or are being used in the nuclear and non-nuclear fields and draft the Procedures Guide described above. When the Procedures Guide has been sufficiently developed, it will undergo peer review in the IEEE-sponsored Review Conference. The Review Conference is expected to draw participants from the nuclear industry, from the research community, from professional societies, and from government. The Review Conference is expected to use a suitable choice of format to discuss: (1) status reports of recent PRA activities such as the NRC's IREP, the Zion/Indian Point Study, the Oconee/NSAC review, etc., (2) PRA applications and experience in non-nuclear settings, (3) implications of use of PRA, in the regulatory context, and (4) results of the Technical Committee's work on PRA methodologies with special emphasis on new approaches.

From time to time either before or after the Review Conference the Steering Committee may direct that drafts of the Procedures Guide be circulated to other reviewers for technical comment. Similarly, the NRC may choose to circulate drafts of the Procedures Guide to the general public for information and comment at suitable times.

After the Review Conference the Technical Committee will resume drafting of the Procedures Guide. The Procedures Guide, and the bases for its form and methods, will be reviewed again at workshops and the Topical Conference sponsored by the ANS. It is expected that the Topical Conference will include reports on many PRA projects, technical issues in PRA, and policy issues in PRA, as well as a suitable format for discussion and review of the Procedures Guide. Presumably, the Steering Committee and the Technical Committee will meet again after the Topical Conference to incorporate the comments obtained there. When the Procedures Guide is finished the project will be completed.

SUPPORT

The two professional societies will act as secretariat for or sponsor the activities of this project under separate support agreements with the NRC. In general, the IEEE will sponsor and administer the Review Conference, the IEEE participation in the Steering Committee, and the non-nuclear industry contributions to the work of the Technical Committee. The ANS will sponsor and administer the Topical Conference and provide administrative support for the Steering Committee and the Technical Committee, providing meeting rooms, working facilities, and whatever other physical support services are required. The final division of responsibility will be made by the Steering Committee.

Persons designated to participate in the Steering Committee and the Technical Committee will be expected to make a substantial commitment of their time. It is expected that the Technical Committee will meet for one week every six to eight weeks during the first six months of this project. The nuclear industry and NRC participants will be expected to devote about 20% of their working time to the project. The chairman of the Technical Committee and technical support staff will likely spend about half time on the project. Consultants will work as required.

SCHEDULE

It is a goal that the entire project will be completed in early 1982, about 15 months after the initial meeting of the Steering Committee. The important segments of the schedule include: (1) about five months for initial drafting of the Procedures Guide; (2) an additional five months for review, redrafting, and the Review Conference; and (3) a final five months for a final redrafting, review, the Topical Conference, and final changes. The Steering Committee is expected to set a realistic schedule considering this goal.

The proposed schedule has been established based on the time required to complete the technical effort, assuming that major policy issues which can affect the direction of the work can be resolved in parallel and on a schedule which provides for timely input to the technical effort. It is apparent that this may present difficulties due to the complexity of the issues involved. All parties will dedicate themselves to the principle that such a schedule can be maintained, since it is clear that the proposed schedule is sufficient for both the technical work and the attendant policy discussions.

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Appendix B

List of Participants

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Appendix C

Sources Indexes for Availability and Risk Data

Table C-1. Availability and risk data: source index for summarized sources

Document number and date	Document title	Author or publisher
WASH-1400 (NUREG-75/104), 1975	Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, Appendix III, "Failure Data"; Appendix IV, "Common Mode Failures"	U.S. Nuclear Regulatory Commission
IEEE-STD-500, 1977	IEEE Guide to the Selection and Presentation of Electrical, Electronic and Sensing Component Reliability Data for Nuclear Power Generating Stations	Working Group SC5.3 of Reliability Subcommittee, Nuclear Power Engineering Committee, Institute of Electrical and Electronics Engineers
IEEE-STD-493, 1980	IEEE Recommended Practice for the Design of Reliable Industrial and Commercial Power Systems	Working Group of Reliability Subcommittee, Power Systems Support Committee, Industrial Power Systems Department, Institute of Electrical and Electronics Engineers
NUREG/CR-1635, 1980	Nuclear Plant Reliability Data System 1979, Annual Reports of Cumulative System and Component Reliability	Southeast Research Institute for Subcommittee 58.20 of the American Nuclear Society
GADS, 1981	Ten Year Review, 1970-1979, "Report on Equipment Availability"	National Electric Reliability Council
GADS, 1981	Ten Year Review, 1970-1979, "Component Cause Code Summary Report"	National Electric Reliability Council
NPRD1, 1978	Non-Electronic Parts Reliability Data	Reliability Analysis Center, Rome Air Development Center

Table C-1. Availability and risk data: source index for summarized sources (continued)

Document number and date	Document title	Author or publisher
ORNL/ENG/TM-2, 1976	Nuclear Reliability Assurance Data Source Guide	Oak Ridge National Laboratory
GA-A14839/UC-77	GCR Reliability Data Bank Status Report	General Atomic Company
MIL-HSK-217C, 1979	Military Standardization Handbook: Reliability Prediction of Electronic Equipment	U.S. Department of Defense
NUREG/CR-1278, 1980	Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications	A. D. Swain and H. E. Guttmann, Sandia National Laboratories
EPRI NP-1064, 1979	Analysis of Utility Industry Data Systems	Stone & Webster Engineering Corporation for the Electric Power Research Institute
EPRI NP-1191	Nuclear and Large Fossil Unit Operating Experience	S. M. Stoller Corporation for the Electric Power Research Institute
NRC Memo	Component Failure Rates To Be Used for IREP Quantification	NRC Staff
AD/A-005 657, 1975	Non-Electronic Reliability Notebook	Hughes-Aircraft Company for Rome Air Development Center

Table C-2. Availability and risk data: source index for valves

Document number and date	Document title	Author or publisher
NUREG/CR-1363, 1980	Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants	EG&G Idaho, Inc., for the U.S. Nuclear Regulatory Commission
EPRI NP-241, 1976	Assessment of Industry Valve Problems	MPR Associates, Inc., for the Electric Power Research Institute
ALO-73, 1980	Study of Valve Failure Problems in LWR Power Plants	Burns & Roe, Inc., for Sandia National Laboratories
ALO-75, 1980	Pilot Program To Identify Valve Failures Which Impact the Safety and Operation of Light Water Nuclear Power Plants	Teledyne Engineering Services for Sandia National Laboratories
Nuclear Safety, Vol. 22, No. 2, March-April 1981	Valve Failure Problems in LWR Power Plants	R. J. Reyer and J. W. Riddington, Burns & Roe, Inc.
November 1975	Reliability Report of Dikkers Valves for Use in Nuclear Power Stations	Dikkers Valve Company

Table C-3. Availability and risk data: source index for pumps

Document number and date	Document title	Author or publisher
NUREG/CR-1205, 1980	Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants	EG&G Idaho, Inc., for the U.S. Nuclear Regulatory Commission
EPRI-FP-754, 1978	Survey of Feed Pump Outages	Electric Power Research Institute
EPRI NP-351, 1977	Recirculating Pump Seal Investigation	MPR Associates, Inc., for the Electric Power Research Institute
PVP-PB-032, 1978	Pump Reliability Data Derived from Electricité de France Operating Experience	J. Dorey and B. Gachot, American Society of Mechanical Engineers
EPRI NP-1194	Operation and Design Evaluation of Main Coolant Pumps for PWR and BWR Service	E. Makoy and M. L. Adams, Energy Research and Consultants Corporation, for the Electric Power Research Institute

Table C-4. Availability and risk data: source index for diesel generators

Document number and date	Document title	Author or publisher
NUREG/CR-1362, 1979	Data Summaries of Licensee Event Reports of Diesel Generators at U.S. Commercial Nuclear Power Plants	EG&G Idaho, Inc., for the U.S. Nuclear Regulatory Commission
NUREG/CR-0660, 1979	Enhancement of On-Site Emergency Diesel Generator Reliability	G. L. Boner and H. W. Hammers
OOE-ES-002, 1974	Diesel Generator Experience at Nuclear Power Plants	J. L. Crooks and G. S. Vissing, U.S. Atomic Energy Commission
Nuclear Safety, Vol. 16, No. 2, March-April 1975	"Standby Emergency Power Systems," Part 2, "Later Plants," pp. 162-179	E. W. Hagen
Nuclear Safety, Vol. 14, No. 3, May-June 1973	"Standby Emergency Power Systems," Part 1, "The Early Plants," pp. 206-219	E. W. Hagen
Nuclear Safety, Vol. 20, No. 2, March-April 1979	"Technical Note: Performance of Diesel Generator Units in U.S. Nuclear Power Stations"	E. W. Hagen

Table C-5. Availability and risk data: source index for miscellaneous reports

Document number and date	Document title	Author or publisher
NUREG/CR-1464, 1980	Review of Nuclear Power Plant Off-Site Power Source Reliability and Related Recommended Changes to NRC Rules and Regulations	R. E. Battle et al., Oak Ridge National Laboratory
EPRI NP-2230, 1982	ATWS: A Reappraisal, Part III, "Frequency of Anticipated Transients"	F. L. Leverenz, Jr., et al., Science Applications, Inc., for the Electric Power Research Institute
EPRI NP-2301, 1982	Loss of Off-Site Power at Nuclear Power Plants: Data and Analysis	Electric Power Research Institute
Nuclear Safety, Vol. 19, No. 1, January-February 1978	"A Review of Safety-Related Occurrences in Nuclear Power Plants as Reported in 1976"	R. L. Scott and R. B. Gallaher, Nuclear Safety Information Center
Nuclear Safety, Vol. 20, No. 6, November-December 1979	"Assessment of the Frequency of Failure to Scram in Light-Water Reactors"	G. Apostolakis, S. Kaplan, B. J. Garrick, and W. Dickter
Nuclear Safety, Vol. 19, No. 6, November-December 1978	"Application of Reactor Scram Experience in Reliability Analysis of Shutdown Systems"	G. E. Edison and M. T. Gerstner
ALO-78/SAI-154-79-PA, 1980	Component Failures That Lead to Reactor Scrams	E. T. Burns, R. J. Wilson, and E. Y. Lirn, Science Applications, Inc.
ALO-79/SAI-180-80-PA, 1980	Component Failures That Lead to Financial Shutdowns	Science Applications, Inc.
Nuclear Safety, Vol. 22, No. 2, March-April 1981	"Anticipated Transients Without Scram for Light-Water Reactors: Unresolved Safety Issue TAP A-9"	E. W. Hagen, Oak Ridge National Laboratory

Table C-5. Availability and risk data: source index for miscellaneous reports (continued)

Document number and date	Document title	Author or publisher
NUREG/CR-1331, 1980	Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants	EG&G Idaho, Inc., for the U.S. Nuclear Regulatory Commission
EPRI NP-443	Characteristics of Instrumentation and Control System Failures in Light-Water Reactors	S. L. Dasin, E. T. Burns, V. Cini, and W. S. Loell, Electric Power Research Institute
EPRI NP-1675, 1981	Assessment of Exposure Fire Hazards to Cable Trays	J. S. Newman and J. P. Hill, Factory Mutual Research Corporation, Electric Power Research Institute
CONF-800403, 1980	"The Frequency of Fires in Light-Water Reactor Compartments," in Proceedings ANS/ENS Topical Meeting, April 6-9, 1980, Knoxville, Tennessee	G. Apostolakis and M. Kazarians
Nuclear Safety, Vol. 20, No. 3, May-June 1979	"Nuclear Plant Fire Incident Data File"	A. G. Sideris, R. W. Hockenbury, M. L. Yeater, and W. E. Vesely
Nuclear Safety, Vol. 20, No. 3, May-June 1979	"Review of Fire Protection in Nuclear Facilities of the Atomic Energy Commission, 1947-1975"	W. W. Maybee
IAEA-SM-218/11, 1978	Reliability of Piping in Light Water Reactors	S. H. Bush, International Atomic Energy Agency
EPRI NP-438, 1977	Characteristics of Pipe System Failure in Light Water Reactors	S. L. Basin and E. T. Burns, Electric Power Research Institute

Table C-5. Availability and risk data: source index for miscellaneous reports (continued)

Document number and date	Document title	Author or publisher
NUREG/CR-1730	Data Summaries of Licensee Event Reports of Primary Containment Penetrations at U.S. Commercial Nuclear Power Plants	EG&G Idaho, Inc., for the U.S. Nuclear Regulatory Commission
NUREG/CR-1740	Data Summaries of Licensee Event Reports of Selected Instrumentation and Control Components at U.S. Commercial Nuclear Power Plants	EG&G Idaho, Inc., for the U.S. Nuclear Regulatory Commission

Appendix D

Live Issues in Dispersion and Deposition Calculations

D1 THE GAUSSIAN MODEL AND ITS USE

D1.1 WHY THE GAUSSIAN MODEL?

At the core of any consequence-modeling code is a representation of atmospheric dispersion. Many authors choose to use a solution of the linear diffusion equation

$$\bar{u}(z) \frac{\partial \chi_I}{\partial x} = \frac{\partial K_x}{\partial x} \frac{\partial \chi_I}{\partial x} + \frac{\partial K_y}{\partial y} \frac{\partial \chi_I}{\partial y} + \frac{\partial K_z}{\partial z} \frac{\partial \chi_I}{\partial z} \qquad \text{(D-1)}$$

where

x = the distance downwind (meters).

y = the crosswind distance (meters).

z = the height above the ground (meters).

$\bar{u}(z)$ = the mean wind velocity at height z (m/sec).

$\chi_I(x,y,z)$ = the instantaneous concentration of radioactive material in the air at the point (x,y,z) (Ci/m^3).

K_x, K_y, K_z = the eddy-diffusion coefficients (m^2/sec).

In order to solve the above equation, it is necessary to study the properties of the atmospheric boundary layer--the layer of turbulent air adjacent to the surface. These properties are complicated and not completely understood (Pasquill, 1972). The person who wishes to use his dispersion model in a complete consequence analysis is faced with a bewildering choice of methods of varying degrees of complexity, some of them having an apparently insatiable appetite for computer time.

The most sophisticated of present techniques adopt numerical methods of solution that open up the possibility of simultaneously predicting the behavior of both the radioactive plume and the properties of the atmospheric boundary layer; essentially, they try to calculate the eddy-diffusion coefficients from first principles. These methods generally begin with the basic conservation laws of physics: the continuity equation, or conservation of mass, which states that the rate of change of the mass of air within a small volume in space equals the net flow of mass across the boundaries of the volume; the conservation of momentum, which is Newton's second law; and the conservation of the total quantity of pollutant or radioactive material emitted into the atmosphere. To these conservation equations is added a turbulent-closure assumption. This is an implicit or explicit feature of all models of atmospheric dispersion. It is a simplifying assumption about the action of turbulence in the atmosphere. The resulting differential equations are then solved by the methods of finite differences or finite elements. An example of such a scheme is presented by El Tahry et al. (1981).

These "heavy-duty" numerical models are a foretaste of the kind of techniques that will perhaps become standard features of consequence analyses in a few years. They may indeed be the only techniques that will be sufficiently flexible to adequately handle some of the growing concerns of the present--for example, dispersion at coastal sites, where the properties of the atmospheric boundary layer can be particularly complicated, or the dispersion of a plume containing a spectrum of particle sizes.

For the moment, however, these models are too time consuming for use in complete risk analyses. Furthermore, in spite of the use of these techniques of great complexity, the uncertainties that arise because of the present incomplete understanding of the properties of the atmospheric boundary layer are not necessarily eliminated (Scriven, 1969).

Less complicated models may assume an analytic form for the velocity $\bar{u}(z)$ and the eddy-diffusion coefficients in Equation D-1 (see, for example, Yih, 1951; Smith, 1962). Alternatively, Equation D-1 can be solved numerically if more elaborate representations of $\bar{u}(z)$ or K_x, K_y, and K_z are incorporated (Maul, 1977). These are examples of the "eddy-diffusivity," "gradient transfer," or "K-theory" approach. The use of such techniques in place of the more commonly used Gaussian plume model has recently been discussed by an ad hoc working group in the United Kingdom (Clarke et al., 1979), and it is worth reproducing their conclusion.

> These [eddy-diffusivity] models generally require more computer time to obtain results and have not at present been developed so that the user may easily relate the values of parameters required by these computer models to readily measurable meteorological quantities. Moreover, the results obtained by Barker (1978) and Jones (1979) do <u>not</u> provide evidence that the results of the more complex calculations on their own give either a sufficiently different result, or a greater confidence in the prediction of downwind concentrations, to warrant their additional complexity and cost to users.... There is work being currently undertaken which may enable the more complex transport models to be related to easily measured meteorological parameters (Smith, 1978); however, it is likely that it will be several years before such a scheme has been developed, validated and expressed in readily usable form.

In essence, the use of the eddy-diffusivity or higher-order turbulence-closure models has several disadvantages:

1. Cost in terms of computer time, although this can be reduced if some precomputing is done.

2. The unavailability of the meteorological parameters necessary for the input to the computer model.

3. Evidence that the results, at least when calculated for a flat terrain, do not differ sufficiently from those of simpler models to make their use worthwhile in consequence analyses that require repeated use of the meteorological model.

Hitherto, most authors of consequence codes have found these arguments compelling, and it will be assumed that most readers of this report have at their disposal some version of the basic Gaussian plume model, which is described in Section 9.3. There is no implication, however, that more complicated models should not be used if the reader has sufficient funds and energy. These are the models of the future, and there are already consequence codes that are beginning to take advantage of numerical models; examples are ARANO (Northlund et al., 1979) and CRACIT (Commonwealth Edison Company, 1981; Woodard and Potter, 1979).

D1.2 ACCURACY OF THE GAUSSIAN MODEL

No definitive statement of the accuracy of the Gaussian model is applicable to all circumstances. The American Meteorological Society (1977) has published a brief note on the "Accuracy of Dispersion Models." The question has also been discussed by Clarke et al. (1979). The text that follows addresses three questions:

1. Given the weather conditions, what accuracy can be expected from the Gaussian model?

2. Up to what height is the Gaussian model valid?

3. What sort of accuracy can be expected for quantities that are averaged or cumulated over many uses of the Gaussian model in different weather conditions?

D1.2.1 Gaussian Model in Given Weather Conditions

The discussion that follows relies heavily on the 1977 position paper of the American Meteorological Society. Models used for calculating the near-field (distances of less than 1 km) dispersion of inert pollutants for short averaging times (minutes to hours) have been developed with the aid of various definitive sets of dispersion experiments carried out during the 1950s and 1960s under idealized conditions of uniform terrain, steady weather conditions, and known source terms measured by research-grade instruments. These experiments played an important role in the development of the Pasquill-Gifford dispersion curves. In these ideal circumstances, if the user estimates a certain concentration by these modeling techniques, the observed maximum downwind concentration value should be expected to be within 10 to 20 percent of the calculated value for a surface-level source and within 20 to 40 percent for an elevated source, such as a tall stack.

When dispersion modeling is applied in circumstances that are different from the carefully controlled, idealized situation described above (which is to say, for most applications), accuracy within "a factor of 2" (Islitzer and Slade, 1968) or "a factor of 3" (Clarke et al., 1979) has been estimated in connection with routine applications of dispersion modeling. This estimate is probably realistic for practical modeling applications for which the controlling meteorological parameters are measured from a tower, conditions are reasonably steady and horizontally homogeneous (less than about 50-percent variation from the spatial and temporal average during the experiment), and there are no exceptional circumstances that could affect the atmosphere's dispersive capacity in ways not accounted for by the model.

The American Meteorological Society (1975) has classified several important meteorological circumstances as "exceptional":

1. Aerodynamic wake flows of all kinds, including stack downwash, building wakes, highway-vehicle wakes, and wakes generated by terrain obstacles.

2. Buoyant fluid flows, including plumes from power-plant stacks and accidental releases of heavy, toxic gases.

3. Flows over surfaces markedly different from those represented in the basic experiments, including dispersion over forests, cities, water, and rough terrain.

4. Dispersion in extremely stable and unstable conditions.

5. Dispersion at great downwind distances (more than 10 to 20 km).

The present direct, observational knowledge of dispersion in most of these circumstances amounts to a few special case studies. A recent summary has been given by Draxler (1979), who considers shoreline diffusion; diffusion over rough terrain, complex terrain, forests, and cities; diffusion at low wind speeds; and long-range diffusion. Various sections of this appendix and Chapter 9 are devoted to discussions of how the basic Gaussian model is modified to take account of specific effects, such as plume rise (Section D2), building-wake effects (Section 9.3.1.5), as well as wind shifts and complex terrain (Section D4).

D1.2.2 Height up to Which Gaussian Model Is Valid

As regards validity in height, the generalized scheme of Hosker (1974) has been judged to be valid for release heights of up to about 200 m (Clarke et al., 1979). This is typical of schemes that rely mainly on data collected at ground level. The scheme developed at Juelich in West Germany (Vogt et al., 1978, 1980) is based on measurements taken with emission heights of up to 130 m. Recently, measurements at Karslruhe in West Germany have been extended up to a release height of 195 m (Thomas and Nester, 1980). The influence of this height dependence on consequences has been analyzed by Vogt (1981).

D1.2.3 Many Uses of the Gaussian Model

When calculating complementary cumulative distribution functions, the Gaussian model is used many times to simulate a number of weather conditions. It is intuitively reasonable to expect that there will be a certain element of "swings and roundabouts" in that the calculations in which the consequences are overestimated will compensate for those in which the consequences are underestimated. This has never been proved with scientific rigor, but to the extent that the deviations from the Gaussian model are random, it is a reasonable expectation.

D1.3 METHODS FOR DEFINING STABILITY CATEGORIES

The definition of stability categories is discussed in Section 9.3.1.2, which describes the methods due to Pasquill (1961), the NRC's σ_θ and ΔT methods (USAEC, 1972), and the recent scheme developed by Sandia National Laboratories and to be described by D. J. Alpert and D. C. Aldrich in a report on turbulence-typing schemes.

In general, methods for defining stability categories depend on the measurement of factors that are indirectly related to turbulence intensity, which in essence depends on three physical quantities (Smith, 1979):

1. The upward heat flux H_w from the ground. This is influenced by the insolation conditions and, roughly speaking, determines the enhancement or suppression of turbulence by the action of convection.

2. The mean wind speed $\bar{u}$. This is a measure of the intensity of mechanical turbulence in the atmosphere.

3. The underlying surface roughness z_0.

In neutral conditions (essentially $H_w = 0$), $\bar{u}(z)$ is related to z_0 by the well-known logarithmic law (Slade, 1968)

$$\bar{u}(z) = \frac{u_*}{k} \ln \frac{z}{z_0} \qquad \text{(D-2)}$$

where u_* is the friction velocity and k is Von Kármán's constant ($k \approx 0.4$). It is the quantity u_*^2 that is a measure of the intensity of mechanical turbulence in the atmosphere, and hence z_0 has an important influence on the intensity of the atmospheric turbulence.

Ideally, the definition of stability categories and the parametrization of σ_y and σ_z should reflect the above understanding of the basic physics. In general, a categorizing scheme should contain some quantity or quantities related to H_w and to u_*. The z_0 dependence can then be explicitly incorporated into expressions for σ_y and σ_z.

The foregoing discussion has a direct bearing on a typing scheme recommended by the NRC in Regulatory Guide 1.23 and also used in the Reactor

Safety Study. This scheme directly relates values of the atmospheric temperature gradient dT/dz to stability categories, as shown in Table 9-3. This is an attractive scheme from the user's point of view because, in general, the values of dT/dz can be easily estimated from measurements of the temperature difference ΔT between two points on a meteorological tower, and such measurements are usually made at the reactor site. In view of the fact that measurements of ΔT cannot fully take into account the mechanical component of turbulence as represented by the mean wind speed, it is not surprising that this scheme has attracted some criticism (Weber et al., 1977; Sedefian and Bennett, 1980). Experience also shows that the assignment to stability categories tends to change when the heights at which the sensors are placed are changed. The consequence modelers of the 1980s should be aware of the defects of this method and, if possible, should try to use another. Vogt et al. (1978) have developed a scheme in which both ΔT and the wind speed are used to determine the stability category, and it is possible that this sort of scheme could be used with the data that are usually available for reactor sites. An example of a scheme developed for use at Sandia National Laboratories is given in Table 9-4.

Turner (1969) suggested a variation of Pasquill's scheme in which the incoming solar radiation is classified in terms of measurable quantities: solar elevation angle, cloud amount, and height. He defined seven stability categories, 1 through 7, broadly corresponding to Pasquill categories A through F. Turner's work is the basis of the STAR (Stability Array) program, which has been adopted by the U.S. Environmental Protection Agency (USEPA, 1977) and is probably the most widely used scheme in the United States.

The most up-to-date version of Pasquill's scheme is due to Smith (1972; see also Clarke et al., 1979), who relates stability directly to H_W and $\bar{u}$ and gives nomograms for calculating H_W in terms of the time of year, the time of day, and the amount of cloud cover.

As an example of the uncertainties that can arise simply because of different definitions of stability category, Table D-1 shows an analysis by Sedefian and Bennett (1980) of a year's worth of meteorological data from an instrumented tower on Staten Island, New York, and from the nearby La Guardia Airport. Table D-1 shows the percentage occurrence of each turbulence class as determined by the σ_θ, ΔT, and STAR methods. It can be seen that there is considerable disagreement, and this is amplified when the comparison is reduced to an hourly basis. For example, Sedefian and Bennett show that, of the occasions on which the σ_θ method indicated category A, only 18 percent were predicted to be category A by the ΔT method. The corresponding figures for categories B, C, D, and E were 12, 6, 50, and 55 percent, respectively. The differences in the consequence-analysis results that arise from these uncertainties should in principle be the subject of a sensitivity analysis, though this is rarely, if ever, done in practice. However, Nester (1980) has recently compared five turbulent-diffusion typing schemes with a view to determining which gives the most unequivocal assignment: (1) fluctuations, in the horizontal wind direction; (2) ΔT and wind speed; (3) net radiation and wind speed; (4) a modified Pasquill scheme; and (5) a scheme based on the wind-profile exponent. The first two schemes turn out to be preferable.

Table D-1. Percentage frequency of occurrence of turbulence classes obtained by different methods[a]

Method	Turbulence class A	B	C	D	E	F	G
NRC (σ_θ at 10 m)	7.8	8.5	18.2	42.8	17.9	0.2	4.5
ΔT	11.5	7.3	3.7	38.0	31.3	6.7	1.4
STAR	0.0	2.8	9.1	69.5	18.4		--

[a]From Sedefian and Bennett (1980).

For a comprehensive review of turbulent-diffusion typing schemes the paper by Gifford (1976) is recommended; it also gives parametrizations of σ_y and σ_z. Weber et al. (1977) and Sedefian and Bennett (1980) give critical reviews of the available schemes.

D2 PLUME RISE

The following text draws heavily on a paper presented at the ANS conference on Probabilistic Risk Assessment, Port Chester, New York (Kaiser, 1981). The necessary elements of a plume-rise model can be briefly summarized as follows: (1) the definition of the mode of radionuclide release from the reactor building; (2) the interaction of the buoyant plume with the turbulent wake of the reactor building; (3) the trajectory of the plume; (4) ground-level concentrations under a rising plume; (5) the termination of plume rise; and (6) transition to passive dispersion.

These elements have been discussed elsewhere (Fryer and Kaiser, 1979), and for the present, it is sufficient to focus on just two of them. The first is the "liftoff" problem, or the interaction of the buoyant plume with the turbulent wake; there is new work in this area that is worth discussing. Second, there were large differences between the results of the plume-rise calculations performed by various participants in the Benchmark exercise, and the main reason for this was uncertainty in the predicted height at which plume rise terminates.

D2.1 LIFTOFF

The question of what happens to a buoyant plume if it is first emitted into a turbulent wake has been examined in a series of wind-tunnel experiments (Hall et al., 1980). Typical results are displayed in Figures D-1 and D-2, where the dimensionless ground-level concentration $K = \chi \bar{u} H^2/V$ is plotted as a function of downwind distance, with the results scaled up by a factor of 300 to give an effective building height H of 50 meters. The quantity χ is the ground-level airborne concentration of the effluent (a methane tracer in helium), $\bar{u}$ is the mean wind speed at height H, and V is the volumetric rate of release. Experimental results are presented for several values of a parameter

$$L = gH\Delta\rho/(\rho_a u_*^2) = 28Q/(\bar{u}u_*^2 W) \qquad \text{(D-3)}$$

where Q is the equivalent energy (megawatts) associated with the release, u_* is the friction velocity, W is the width of the turbulent wake, and $\Delta\rho$ is the difference between the density of the air, ρ_a, and that of the plume. The quantity L is a Richardson number that gives a ratio between a

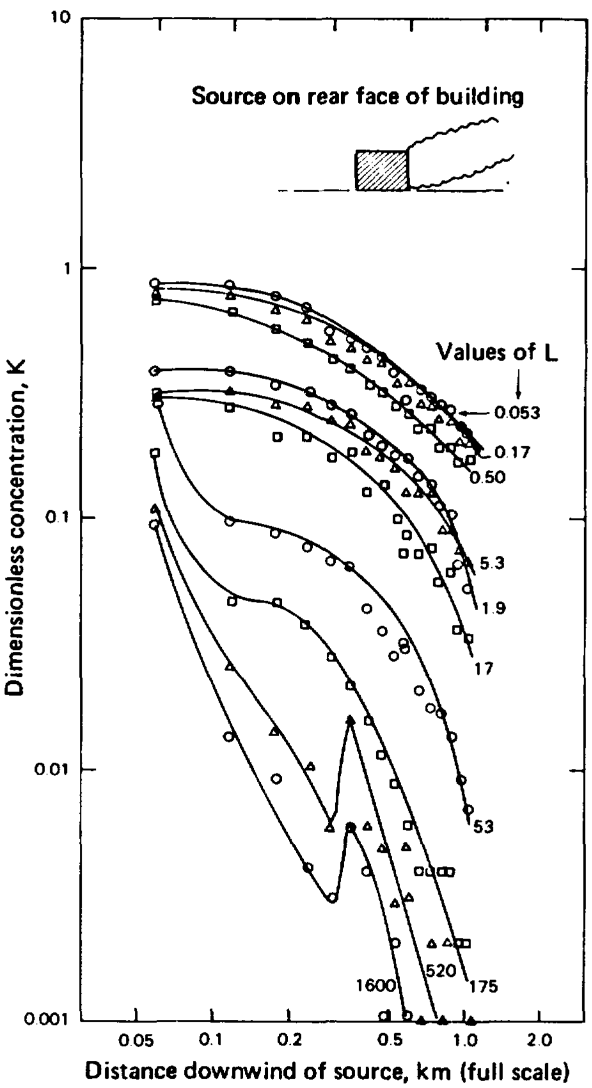

Figure D-1. Plume centerline ground-level concentrations downwind of source. From Hall et al. (1980).

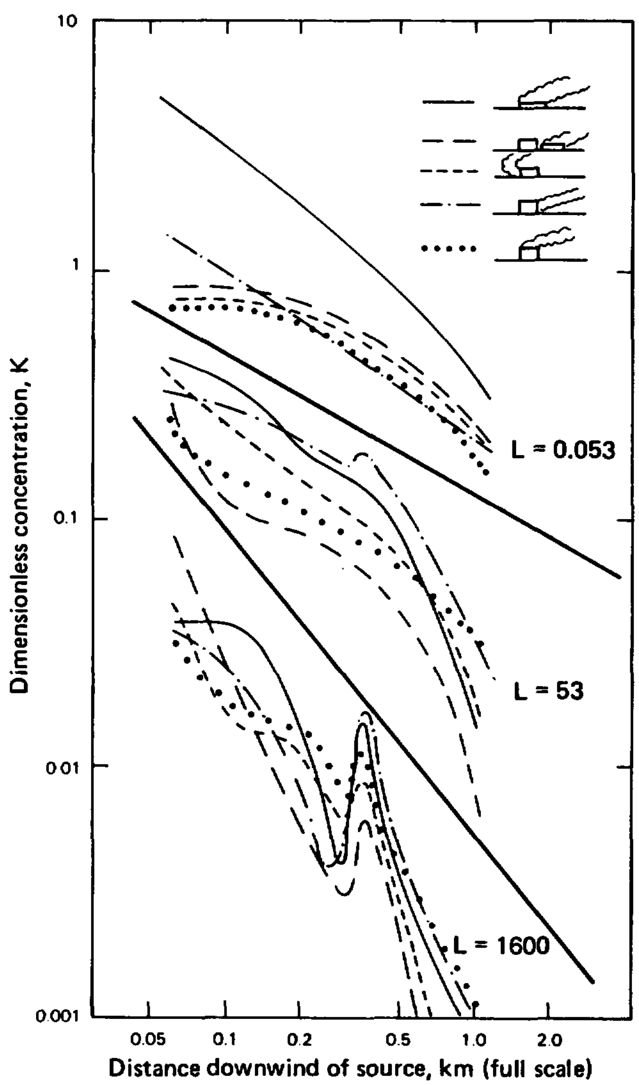

Figure D-2. Plume centerline ground-level concentrations downwind of building for different release conditions and buoyant levels. From Hall et al. (1980).

buoyancy-induced velocity $\sqrt{(g\Delta\rho H)}$ and a velocity u_* characterizing the intensity of turbulence in the atmosphere.

Figure D-1 displays an increasingly effective liftoff as L increases. At the low end of the range, L = 0.053 is characteristic of a virtually passive plume. By contrast, L = 1600 is characteristic of a 1000-MW plume released into a reactor-building wake when the wind speed is 3 m/sec. There seems to be a qualitative change in plume behavior when L lies between 17 and 53. If L = 30 is taken as a critical value in the sense that plumes must achieve this buoyancy in order to lift off, this can be shown to imply that plumes probably need to be as energetic as 10 MW to lift off in a 3-m/sec wind and as energetic as 100 MW to lift off in a 10-m/sec wind.

This has considerable implications for consequence analysis. Of all the release categories considered in the Reactor Safety Study (RSS), only four have predicted associated energy releases of about 10 MW or more (BWR-2, PWR-1a, PWR-1b, and PWR-2) and only one (PWR-1b, 150 MW) has a predicted energy-release rate in excess of 100 MW. In the rebaselining exercise for the PWR (USNRC, 1982), only one sequence, TMLB' (see the RSS for the meaning of symbols), has an energy-release rate (50 MW) sufficient to cause liftoff in most weather conditions. For the rebaselined BWR, a very improbable sequence (AE, with containment failure due to a steam explosion) has Q = 35 MW, while the transient sequence TQUV, with an associated energy-release rate of 4 MW, is the most energetic of the other sequences. A rule of thumb may be appropriate: plume rise will not take place for most accident sequences or categories. Those accident sequences for which plume rise could be important are ones with large predicted releases of radioactivity, but some have such small predicted frequencies of occurrence that they are negligible contributors to public risk. Others, however, such as TMLB', can be shown to contribute significantly to public risk, so that the study of plume rise remains important.

Figure D-2 shows how the wind-tunnel results vary with source configuration. The five different sources illustrated in Figure D-2 are all area sources of the same dimension: (1) on the ground in the absence of a building; (2) on the ground immediately downwind of a building; (3) on the roof of the building; (4) on the upwind face of the building; and (5) on the downwind face of the building. A surprising aspect of these results is that the configuration is not important for buoyant plumes, although it is more so for passive ones. Further experiments planned by the Warren Spring Laboratory in the United Kingdom should provide insight and, possibly, a quantitative method for predicting plume behavior.

D2.2 TERMINATION OF PLUME RISE

The termination of plume rise is one of the major outstanding problems because "the vast majority of plume rise observations show the plume still rising at the greatest distance of observation, except in stable conditions" and "the great unresolved plume rise question is that of final rise or of 'effective stack height' (the rise may never actually terminate) when ambient turbulence is the most effective rise limiting agent. Here, the theoretical solutions offered are many, while adequate data for testing them are practically non-existent" (Briggs, 1975). It follows that the procedures for terminating plume rise in neutral conditions usually involve the postulation of some conservative (in the sense of underestimating plume rise) criterion. Briggs' latest work suggests that the termination of plume rise should be calculated by equating a quantity known as the turbulent-energy dissipation rate (which is essentially the "vigor" of the turbulence) within the plume and outside it in the atmosphere. This prescription pushes the prediction of the final height of rise to the high end of the range of observations or beyond. Indeed, it is more than probable that, in neutral conditions, the rise of such an energetic plume would be terminated by contact with an overhead inversion rather than by the action of the ambient turbulence.

For stable conditions, there is a well-established standard formula:

$$\Delta h = 2.6(F/\bar{u}\beta_T)^{1/3} \qquad \text{(D-4)}$$

where F is the buoyancy parameter ($F \approx 8.9Q\ m^4/sec^3$), $\bar{u}$ is the mean wind speed at the final height of rise or averaged over the depth of the plume, Δh is the final height of plume rise above the source, and β_T is (g/T) $(d\theta/dz)$, g being the acceleration due to gravity, T the temperature of the atmosphere, and $d\theta/dz$ the atmospheric potential temperature gradient.

D2.3 THE IMPACT OF PLUME RISE IN CONSEQUENCE CALCULATIONS

The impact of plume rise on the concentrations of airborne and deposited radionuclides has been treated by Fryer and Kaiser (1980) and will not be repeated here. Instead, this discussion will focus on the impact of plume rise on complementary cumulative distribution functions (CCDFs). Figure D-3 displays a CCDF for early fatalities, conditional on the BMR-1 and BMR-2 releases, which were defined for the Benchmark exercise (Aldrich et al., 1981a) and differ only in that BMR-1 is passive and BMR-2 has an associated energy release of 150 MW. The CCDF is a plot of the conditional probability with which the corresponding number of fatalities is predicted to occur. Note that Figure D-3 gives qualitative examples and does not contain the results of any participant in the Benchmark exercise.

Two key points are to be borne in mind when interpreting these CCDFs. The first is that, in many weather conditions, plume rise is predicted to cause a region of very low concentration immediately downwind of the reactor. This region may extend 10 or more miles downwind, until plume rise terminates and the action of turbulence brings radioactive material back down to ground level. By this time, the plume has essentially "forgotten" about plume rise. The region within 10 miles or so is generally that within which early fatalities are predicted to be confined for a release like BMR-1, except in a few relatively infrequent weather sequences. It follows that, for most weather sequences in which BMR-1 is predicted to cause early fatalities, there will be none for BMR-2. This accounts for the dramatic fall in the predicted conditional probability of occurrence of 10 or more fatalities in Figure D-3, between curves 1 and 3. (The same observation

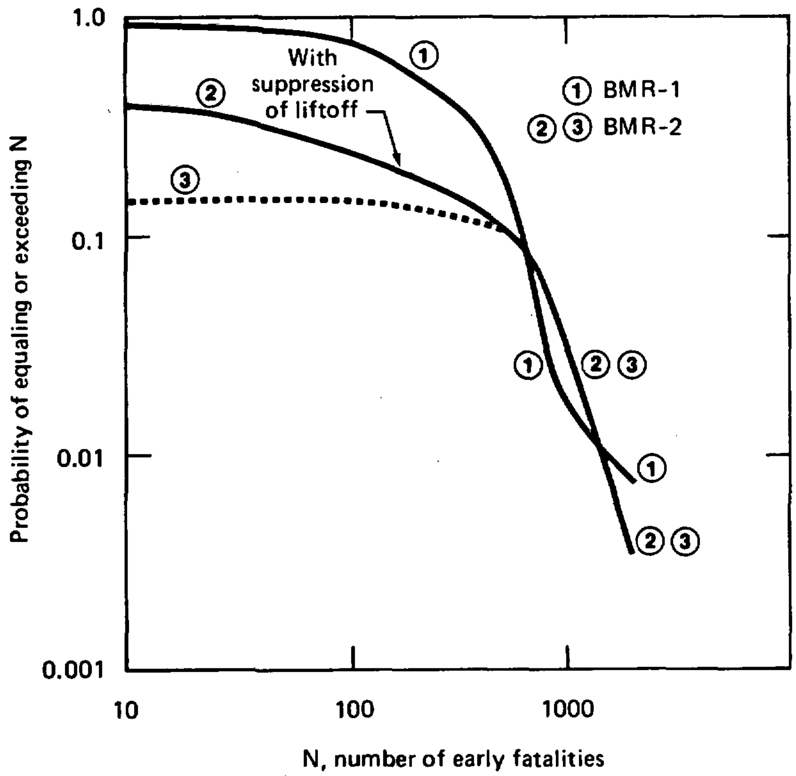

Figure D-3. Complementary cumulative distribution function conditional on BMR-1 or BMR-2 release (Benchmark exercise): uniform population, early fatalities.

would be true if Figure D-3 were extrapolated leftward to one fatality; Figure D-3 begins at 10 fatalities for convenience in presentation.)

The second key point is that there are certain weather sequences in which the plume is predicted to encounter rain beyond 10 miles, possibly over a major center of population. This means that, whether the plume has risen or not, comparable quantities of radioactive material are predicted to be deposited on the ground. It is in these circumstances--a heavy deposit of gamma emitters in a large center of population--that consequence-modeling codes predict peak early fatalities. It follows that plume rise would not be expected to affect to any great degree the predicted peak number of fatalities (or their predicted probability of occurrence, since this probability is simply that of the corresponding weather condition). This aspect of the effect of plume rise is clearly shown in Figure D-3.

Figure D-3 also includes a curve labeled "with suppression of liftoff." The discussion on liftoff earlier in this section indicated that even plumes as energetic as 100 MW or so may not lift off if the wind speed is sufficiently high. Roughly speaking, this means that there are some weather sequences in which the behavior of the BMR-2 plume may not differ from that of the BMR-1 plume, so that some early fatalities would be predicted to occur within 10 miles or so in both cases. This accounts for the fact that, for a buoyant plume with the suppression of liftoff, the predicted frequency of occurrence of 10 or more fatalities is greater than for the case without the suppression of liftoff.

Figure D-4 is similar to Figure D-3, except that it is for latent-cancer fatalities. There is one key point that explains the difference between the BMR-1 and BMR-2 curves in this figure. Consequence-modeling codes generally predict that, whatever the weather conditions, most of the latent-cancer fatalities will occur among large populations several tens of miles downwind. As was explained above, this is the region in which the plume has "forgotten" that plume rise has taken place and is fairly uniformly spread between the ground and the inversion lid. In order to reach this stage, however, the passive plume travels along the ground, whereas, for most of the time, the buoyant plume is well elevated. It follows that the passive plume is predicted to be more effectively depleted by dry deposition, and, at large distances downwind, the buoyant plume is predicted to contain more airborne material and to cause higher ground-level airborne concentrations. Hence the buoyant plume may cause more cancers to develop in the surrounding population, and this is reflected in Figure D-4.

The discussion above contains the essential, albeit somewhat simplified, elements of the effect of plume rise on latent-cancer fatalities and early fatalities. The forthcoming report on the international Benchmark

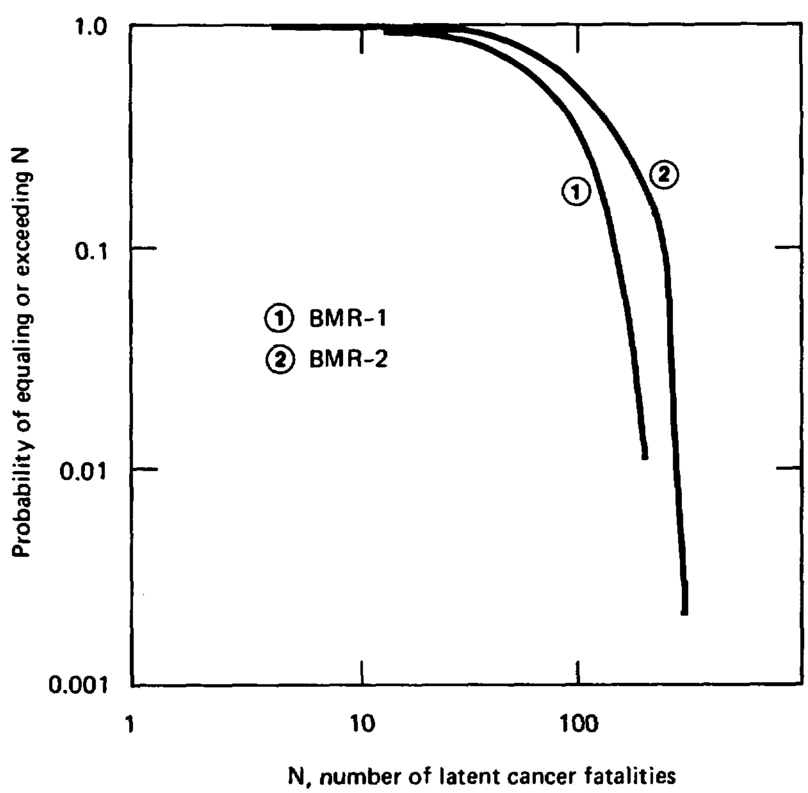

Figure D-4. Complementary cumulative distribution function conditional on BMR-1 or BMR-2 release (Benchmark exercise): uniform population, latent cancer fatalities.

exercise will contain more detail and will also discuss CCDFs for early injuries and for areas of land unacceptably contaminated by deposited radionuclides. Broadly speaking, the CCDFs for early injuries are sensitive to plume rise in much the same way as are those for early fatalities, whereas the CCDFs for contaminated areas are more similar to those for latent cancers.

D3 DRY DEPOSITION

As can be seen from Figure 9-3, the deposition of radioactive material onto the ground is the first step in many of the pathways by which radioactivity can reach people. It is extremely important to be confident that the modeling of the processes that cause deposition is realistic and that any approximations do not introduce wild inaccuracies.

D3.1 DRY-DEPOSITION VELOCITY

A long-established way of dealing with dry deposition is to assume that, if $\chi(x,y,0)$ is the ground-level time-integrated concentration of a radionuclide (in curies per second per cubic meter), the deposited activity is given by

$$\chi_D(x,y) = v_d\chi(x,y,0) \quad \text{Ci/m}^2 \tag{D-5}$$

where v_d is the dry-deposition velocity (Chamberlain and Chadwick, 1953). Implicit in Equation D-5 is the assumption that v_d is measured at a given reference height z_g, which is historically taken to be 1 to 1.5 m over land and 10 to 15 m over water. The values of v_d reported in the literature range from 10^{-3} to 180 cm/sec (Sehmel, 1980), and the choice of a suitable value or values for v_d is one of the trickier inputs to any consequence model. That this is so may seem surprising, since it is nearly 30 years since the concept of a dry-deposition velocity was first introduced, but in fact many of the phenomena that influence the value of v_d are still poorly understood. A single example should suffice to illustrate this point. It is very likely that any particulate matter released from a reactor during an accident will be an aggregate of smaller particles that have been subject to the various aerosol agglomeration processes that are expected to operate within the containment. Regarding aggregates, Sehmel writes in his recent (1980) review that "although attempts have been made to describe the settling velocities of aggregates, no general method exists to predict their settling velocities. Studies have shown that the settling velocities of aggregates can be as small as 0.01 of that for a solid particle of equal mass (Sehmel, 1956). Although the settling velocities are small, the deposition velocities of aggregates have not been quantified." It is important for the reader to bear in mind that the difficulty in assigning a value to v_d is the source of some of the greatest uncertainties in consequence modeling, precisely because the deposition process is a key factor in such a large number of the pathways to people.

D3.1.1 Dry-Deposition Velocity of Particulate Matter

The physical and chemical phenomena that influence v_d are many and complex. Sehmel (1980) divides these influences into three broad groups--meteorological variables, the properties of the depositing materials, and surface variables--and lists a total of about 80 factors that affect the dry-deposition removal rate for particles. One of the parameters to which v_d is most sensitive is the particle diameter d, as can be seen from Figure D-5, which shows experimental results for the velocity of dry deposition onto various surfaces and for a number of values of the friction velocity u_* (Slinn, 1978). The high values of v_d at the right-hand end of the figure are due to the dominance of the gravitational-settling velocity v_g. At the left-hand end, the increase in v_d as the particle diameter decreases is due to the increasing effectiveness of Brownian motion as a means of transporting particles to the surface in question. The minimum of v_d at a diameter of 1 to 10 μm is characteristic of most experimental and theoretical treatments of dry deposition (Sehmel, 1980; Slinn, 1977, 1978; Caporaloni et al., 1975) and corresponds to those particle sizes for which both Brownian motion and gravitational settling are relatively ineffective.

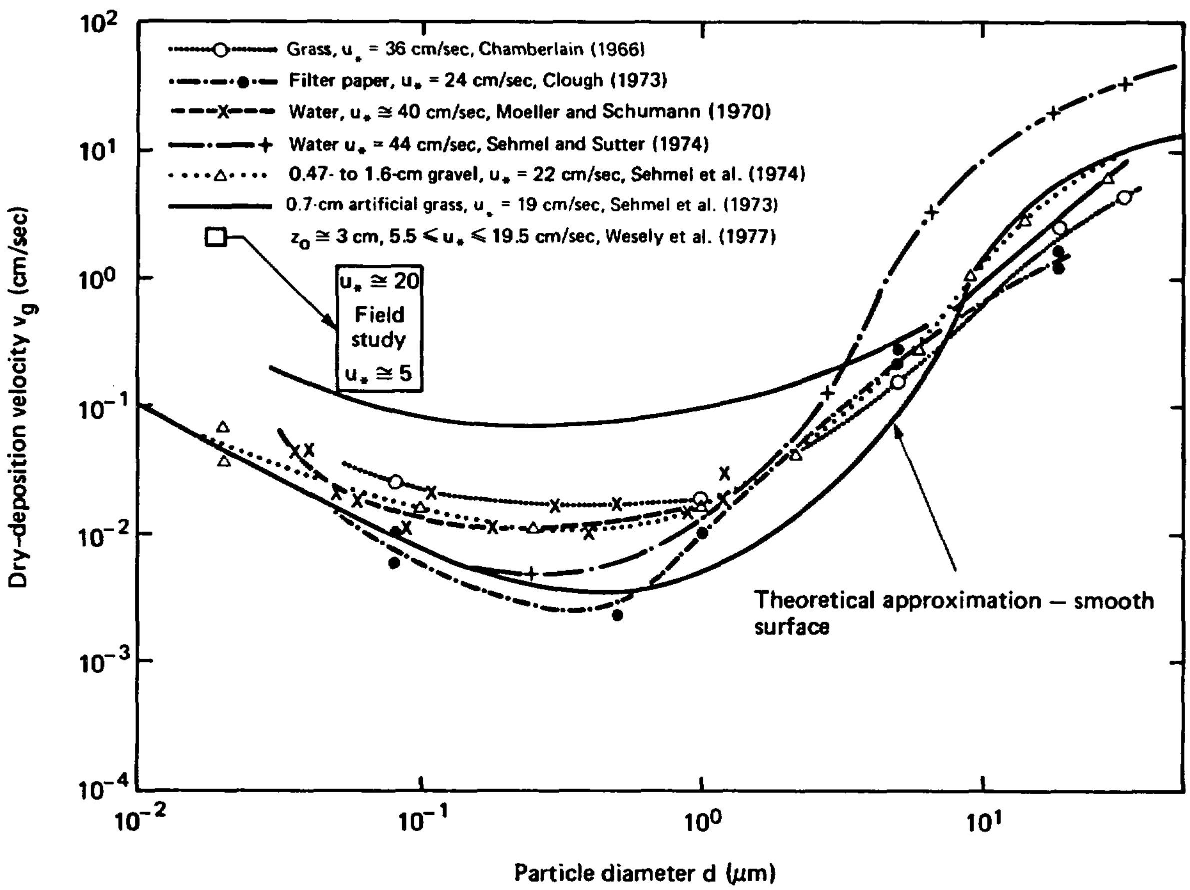

Figure D-5. Dry-deposition velocity as a function of particle size for several substrates, from experimental data reported by various authors. From Slinn (1978).

The experimental results shown in Figure D-5 are all for surfaces that are very smooth in the meteorological sense--that is, with effective values of the meteorological roughness length z_0 that are very much less than 1 cm, except perhaps in the case of the field study. Theoretical studies indicate that there is a strong dependence on z_0. This is to be expected because a rougher surface enhances mechanical turbulence in the atmosphere and increases the rate of particle diffusion toward that surface. Figure D-6 gives a typical example of Sehmel's theoretical predictions for v_d as a function of d for various roughness lengths and particle densities. These predictions are based on correlations derived from wind-tunnel data for the surface mass-transfer resistance for depositing particles. Also shown are some examples of the effect of density. This figure clearly shows that, for particle diameters of 1 to 10 μm, the dry-deposition velocity is a sensitive function of z_0.

In reality, however, matters are even more complicated than is implied by Figure D-6. The value of v_d is also influenced by the nature of the roughness elements--for example, whether they are smooth or "sticky." If there is vegetation on the surface, parameters that could conceivably influence the value of v_d because they determine the surface area available for deposition are the total biomass per unit volume, B; a typical length scale λ_s, which might be, for example, the radius of individual fibers in the vegetation; the height of the vegetative canopy, H_V; and $\bar{e}$, the average mass density of the foliage. Slinn (1977) introduces a parameter

$$\gamma = H_V B/\lambda_s \bar{e}$$

and has developed a theory of the dependence of v_d on γ. Typical results of this theory are shown in Figure D-7.

Figures D-5, D-6, and D-7 illustrate very effectively the difficulties in assigning a value to the dry-deposition velocity of particulate matter released from a reactor during an accident. One of the first requirements is to assign a particle diameter or spectrum of diameters. At present, this is usually done in an ad hoc manner, as illustrated by scoping calculations reported in a recent NRC publication (USNRC, 1981), in which it was assumed that in the containment there is an initial concentration of 1.0 kg/m^3 of aerosol with a mean radius of 0.1 μm. It was shown that it takes about 64 sec to produce an aerosol with a mean radius of 1.0 μm. A subsequent doubling in radius takes 450 sec, and a further doubling to 4 μm takes 3600 sec. With regard to the time scale of a typical severe core-melt accident (the Reactor Safety Study (USNRC, 1975) indicates release durations of 0.5 to 4.0 hr for release categories PWR 1 through 5 and BWR 1 through 4), it seems reasonable to assume radii of one to a few micrometers. This was the assumption made in the Reactor Safety Study and, for the present, seems to be the best available estimate, although improvements that are beginning to be made in the modeling of aerosols within the reactor-coolant system and containment--incorporating, for example, gravitational agglomeration--may change this.

From Figures D-5, D-6, and D-7, it is seen that the deposition velocities for particles 1 to 10 μm in diameter vary from 0.005 to 20 cm/sec. For surfaces of plausible roughness (in general, the smoothest land that would

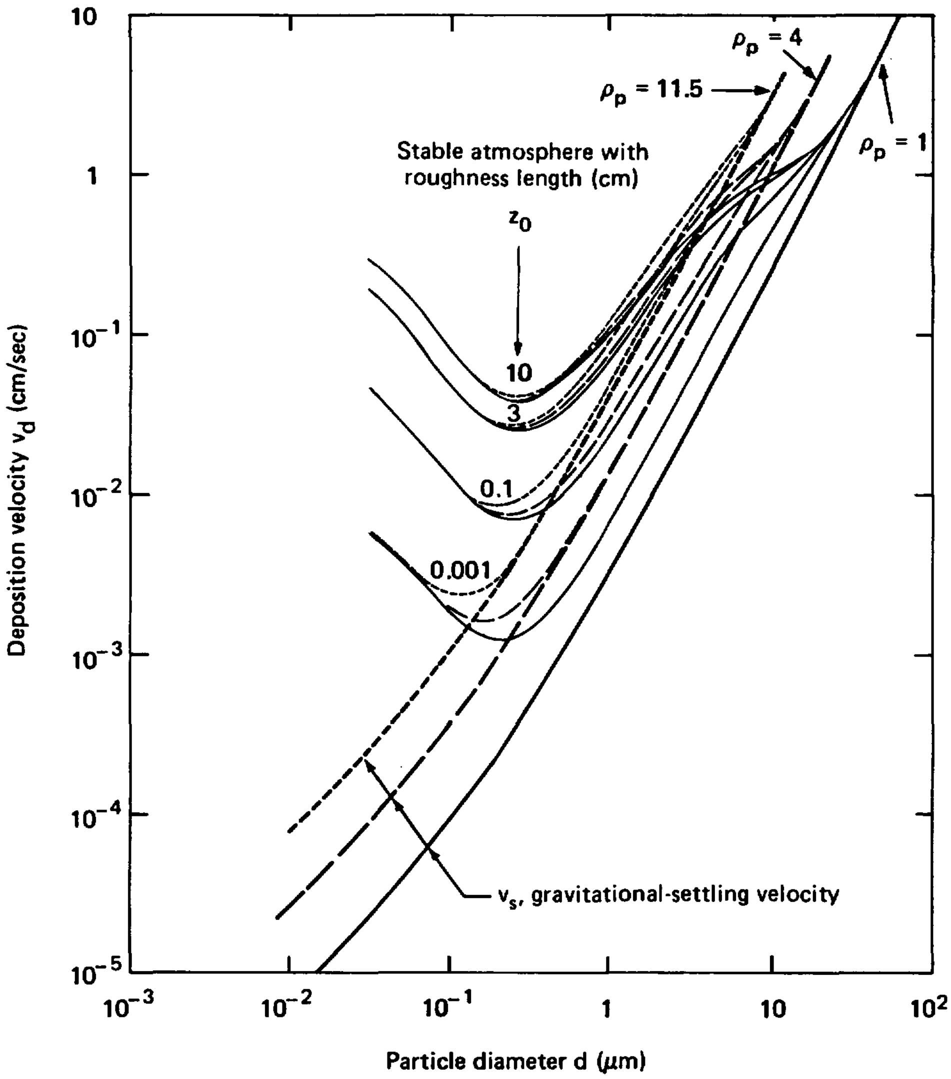

Figure D-6. Effect of the meteorological roughness length z_0 and particle density ρ_p on deposition velocity. From Sehmel (1980).

be of interest would be (say) desert with $z_0 \approx 1$ cm), the range of v_d spreads from 0.05 to 20 cm/sec. In the Reactor Safety Study, the dry-deposition velocity was judged to lie in the range 0.1 to 10 cm/sec, with 1 cm/sec taken as the expected average. In the light of the foregoing discussion, the recent reviews by Slinn and Sehmel have produced no reason for changing this estimate.

It is pertinent to remark that, in the aftermath of the Windscale accident of 1957, Chamberlain (1959) studied the pattern of iodine-131 deposition in England. He deduced deposition velocities in the range 0.24 to 0.52 cm/sec from grass analysis and gamma surveys in the north of England (the Preston, Burnley, Leeds, Lancashire, and Sheffield areas). From measurements in the south of England (the Harwell area), the estimated deposition velocity was found to be about three times less than that in Lancashire and Yorkshire. It is thought that the differences between the measurements in the north and the south of England were due to differences in wind conditions: when the radioactive cloud arrived in the Harwell area, the wind was very light, and hence there was negligible contribution to v_d from eddy diffusion. These measurements were the basis for the British "tradition" of taking $v_d = 0.3$ cm/sec (Beattie and Bryant, 1970).

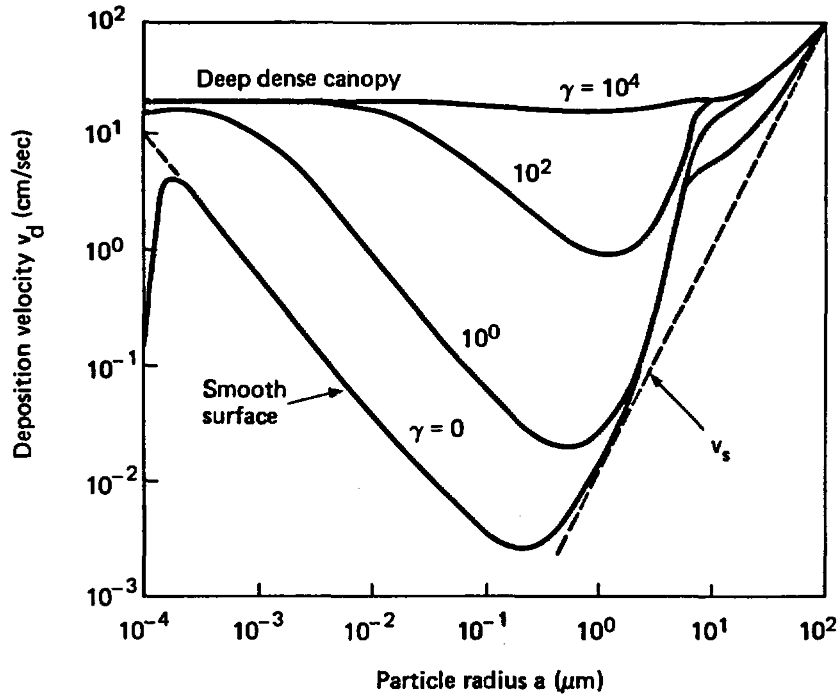

Figure D-7. Deposition velocity as a function of particle radius for a smooth surface and ground covered with a deep dense canopy of vegetation. From Slinn (1977).

In conclusion, for the consequence models that use a single deposition velocity for particulate matter released during a reactor accident, it is reasonable to assume that v_d is in the range 0.1 to 1 cm/sec. Over rough or heavily vegetated surfaces, deposition velocities of up to 10 cm/sec may be appropriate.

D3.1.2 Dry-Deposition Velocity of Gases and Vapors

The important gaseous radionuclides that are emitted from a reactor during an accident are isotopes of the noble gases xenon and krypton. Because of their inert nature, very little deposition is expected. This has been verified by tests on xenon, carried out by Nebeker et al. (1971), and on krypton by Voilleque et al. (1970). It is therefore recommended that the deposition velocities for the noble gases be taken to be zero.

In the past, safety studies have assumed that the radiologically important iodine isotopes would escape from the containment as elemental-iodine vapor. Iodine is highly reactive in this state and would be expected to have a high deposition velocity. Because it is such an important radionuclide, there have been numerous studies of the value of v_d for elemental iodine, and Sehmel (1980) lists 20 experiments that give results in the range 0.02 to 26 cm/sec. These results are so scattered that the deposition velocity for elemental-iodine vapor cannot be predicted more accurately than can that for particulate matter.

Recent studies suggest that the emphasis on elemental iodine may be misplaced, however. The available data suggest that the iodine that escapes from fuel is most likely to be a metallic iodide (Campbell et al., 1981). There are strong indications that the metal is cesium, since thermodynamics arguments show that cesium iodide is very stable. If this is so, iodine should be treated on the same footing as particulate matter, since cesium iodide would be expected to condense onto particles.

Another form of iodine which has given rise to concern and which would emerge into the atmosphere as a vapor is methyl iodide (CH_3I). This is a highly unreactive compound, which is precisely why it causes concern: it is difficult to trap and requires specially impregnated charcoal filters to remove it from a gas flow. This same lack of reactivity means that its deposition velocity is low, however, and Sehmel (1980) lists five experiments that give v_d values in the range 10^{-4} to 10^{-2} cm/sec. In general, the deposition of methyl iodide is not a significant problem in consequence analysis. Indeed, the experience gained in the Reactor Safety Study has led to the judgment that methyl iodide can be neglected, and many recent applications of CRAC ignore it altogether.

Two conclusions can be deduced from the foregoing. In general, there is no need to treat iodine as a vapor rather than particulate matter, and the deposition velocity for noble gases should be assumed to be zero.

D3.1.3 Possible Future Developments in Defining v_d

The current interest in radionuclide source terms--an interest that is demonstrated by the May 1981 issue of Nuclear Technology, which contains nine papers about "Realistic Estimates of the Consequences of Nuclear Accidents"--means that, during the next few years, considerable attention is likely to be devoted to the processes of aerosol agglomeration, which, on the one hand, sustain the expectations of several authors that a significant reduction in source terms can be demonstrated and, on the other hand, ought also to give information on the size distribution of particles released into the atmosphere. It follows that future generations of consequence models may well need to treat a spectrum of deposition velocities since aerosol-agglomeration processes invariably lead to a range of particle sizes.

The reviews of Sehmel (1980) and Slinn (1977, 1978) show that theoretical developments are in hand that take into account the effect on deposition velocity, friction velocity, surface roughness, vegetative cover, and so on. Hence, future consequence models may make provision for values of v_d that change as the surface changes--from forest, to farmland, to an urban area, and to water, for example--and as meteorological conditions change.

D3.2 CALCULATION OF DEPOSITED QUANTITIES OF RADIOACTIVITY

D3.2.1 Modifications of the Gaussian Model: Source-Term Depletion

Once a value of v_d has been chosen, Equation D-1 can be used to determine the necessary modifications to the standard Gaussian formula, which is given in Equation 9-1. One of the simplest procedures is to assume that, as material is deposited on the ground, it is replenished from above at such a rate that the Gaussian profile in the vertical is maintained; that is, depletion occurs throughout the plume. It can then be shown that Equation 9-1 should be modified by replacing the total emitted activity Q by Q(x), the activity remaining at a distance x downwind, where

$$\frac{Q(x)}{Q} = \exp\left\{\left(\frac{2}{\pi}\right)^{1/2} \frac{v_d}{\bar{u}} \int_0^x \frac{dx'}{\sigma_z(x')} \exp\left[-\frac{h^2}{2\sigma_z^2(x')}\right]\right\} \tag{D-7}$$

The proof of this result has been given by Van der Hoven (1968).

A word of warning is pertinent here. Many authors (see, for example, USNRC, 1975; Kaiser, 1976) establish a computational grid, often spaced in roughly equal intervals in ln(x). Equation D-7 is then calculated from interval to interval, using fairly gross approximations. For example, the Reactor Safety Study assumes that, over a spatial interval extending from x_i to x_{i+1}, the quantity

$$\bar{z}_i = \left(\frac{\pi}{2}\right)^{1/2} \sigma_z \exp\left(\frac{h^2}{2\sigma_z^2}\right) \tag{D-8}$$

is constant, and $\bar{z}_i$ is evaluated at the midpoint of the interval. Equation D-7 then becomes

$$\frac{Q(x_{i+1})}{Q(x_i)} = \exp\left(\frac{v_d t_i}{\bar{z}_i}\right) \tag{D-9}$$

where the quantity t_i is the time it takes to cross the spatial interval, $t_i = (x_{i+1} - x_i)/\bar{u}$. A further approximation is made--namely, that the exponent on the right-hand side of Equation D-9 is small, so that

$$\frac{Q(x_{i+1})}{Q(x_i)} = 1 - \frac{v_d t_i}{\bar{z}_i} \tag{D-10}$$

and

$$\frac{Q(x_i)}{Q} = \prod_{j=1}^{i} \left(1 - \frac{v_d t_j}{\bar{z}_j}\right) \tag{D-11}$$

Care should be taken to make the intervals small enough to ensure that the rounding errors introduced by this process do not build up excessively.

The accuracy of Equation D-7 has been studied in a number of publications (Draxler and Elliott, 1977; Horst, 1977; Corbett, 1980). The concern is that its derivation depends on the assumption that the Gaussian profile is maintained in the vertical. Comparisons of Equation D-7 with numerical models show that, for values of $v_d/\bar{u}$ on the order of 10^{-3}, agreement is good to within 10 percent out to 100 km from the source. Indeed, Corbett (1980) shows that, with "reasonable" values of the deposition velocity and wind speed, the source-depletion method is valid for distances out to 100 km from the source in all weather conditions except the most stable (Pasquill categories F and G). Even for category F, with a wind speed of 2 m/sec, the method is valid for deposition velocities of up to 0.003 m/sec.

The implications of the foregoing discussion are that, for models incorporating time-independent weather conditions, the source-depletion method may break down when the plume is assumed to travel large distances in stable weather conditions. For models incorporating time-varying weather conditions (see Section D4), this is unlikely to cause a significant problem, because it is very rare that stable weather conditions persist long enough for the plume to travel many tens of kilometers. Hence, Equation D-7 is a sufficiently accurate representation of the state of the art for its continuing use in consequence analysis to be recommended. Note, however, that this judgment is based on a comparison between models and not on experiment.

Finally, Equation D-7 should be modified to account for the presence of the inversion lid. Referring to Equation 9-11, it can readily be shown that, in order to take account of multiple reflections at the lid, the term $\exp(-h^2/2\sigma_z^2)/\sigma_z$ in Equation D-7 should be replaced by

$$\exp\left(-\frac{h^2}{2\sigma_z^2}\right) + \exp\left[-\frac{(2\ell + h)^2}{2\sigma_z^2}\right] + \exp\left[-\frac{(2\ell - h)^2}{2\sigma_z^2}\right] + \cdots \qquad \text{(D-12)}$$

When the radioactivity becomes uniformly mixed between the lid and the ground, as described in Equation 9-12, Equation D-7 becomes

$$\frac{Q(x)}{Q(x_\ell)} = \exp\left[-\frac{v_d}{\bar{u}}\frac{(x - x_\ell)}{\ell}\right] \quad \text{for } x > x_\ell \qquad \text{(D-13)}$$

where x_ℓ is the distance from the source at which Equation 9-12 becomes valid and $Q(x_\ell)$ is the amount of radioactivity that remains airborne at that distance.

D3.2.2 Alternative Approaches to the Modeling of Dry Deposition

A useful review of dry-deposition models has been prepared by Kaul (1981).

Further Modifications of the Gaussian Model

At the simplest but, as has been seen, quite adequate level, there is the modification to the Gaussian model described in Equation D-5. At the next level, there are somewhat more elaborate modifications designed to try to overcome the reservations about the source-depletion model--namely, that the Gaussian profile in the vertical is maintained no matter what the rate of depletion.

Overcamp (1976) has modified the Gaussian plume model to account for gravitational settling by replacing the fixed height of emission h by $h' = h - v_g x/\bar{u}$; that is, the centerline of the plume moves down at a rate equal to the gravitational-settling velocity. In addition, only the image source is depleted--that is, $\exp\left[-(z + h)^2/2\sigma_z^2(x)\right]$ in Equation 9-1 is modified by a multiplicative factor $\alpha(x)$ that depends on the gravitational-settling velocity, the dry-deposition velocity, and a quantity known as the turbulent-diffusion velocity, which is the "average speed of diffusion away from the centerline of the plume."

Horst (1977) has developed an exact solution for a Gaussian plume modified by Equation D-5 without assuming that the Gaussian profile is maintained in the vertical. The solution is a numerical one and turns out to be very expensive in computer time. Further discussion of modified Gaussian models for dry deposition has been given by Kaul (1981). At present, there seems to be no compelling reason for recommending any of them in preference to Equation D-6, at least in the case where gravitational settling can be neglected.

Gradient-Transfer Methods

At the next level of sophistication, these methods rely on the solution of the linear diffusion equation, Equation D-1, with the boundary condition

$$\left(K_z \frac{\partial \chi}{\partial z}\right)_{z=z_g} = v_d \chi \tag{D-14}$$

where all the quantities are evaluated at the reference height z_g. Numerical methods of solution are required, and, as discussed in Section D1, these models have the disadvantage that they require more computer time and that they have not at present been developed so that the user can easily relate the required values of parameters to readily measurable meteorological quantities. Hence, they cannot be recommended for general use in the present generation of consequence models.

Examples of the use of such a numerical model in a consequence analysis have been given by Nordlund et al. (1979) and will appear in the forthcoming report of the International Benchmark Comparison of Consequence Models.

Particle-in-Cell Models

Particle-in-cell models like ADPIC (Atmospheric Dispersion Particle-in-Cell; Lange, 1978) solve the three-dimensional linear diffusion equation by finite-difference methods with a given nondivergent wind field such as that provided by the code MATHEW (Sherman, 1978). The concentration is represented by a very large number of Lagrangian marker particles transported through a network of grid cells by a "pseudovelocity" field. Such velocity fields are composed of the sum of the advection velocity and the "diffusion velocity," which is derived from the solution of the advection-diffusion equation. Dry deposition is handled simply by vectorially adding the deposition velocity to the pseudovelocity in the ground-level grid cells. This method is far too expensive in computer time to contemplate using it in a complete risk assessment, but in principle it does point the way toward the next but one or two generations of consequence models.

D3.2.3 Gravitational Settling--Future Trends

Implicit in the foregoing discussion of deposition velocities and models has been the assumption that gravitational settling is relatively unimportant. This is because, as assumed in the Reactor Safety Study, "the effect of sedimentation on particle deposition rates becomes negligible when the fall (or settling) velocity of the particle is much lower than the particle velocity controlled by vertical turbulence and mean air motions. This occurs when the fall velocity is lower than about 1 cm/sec" (for particles smaller than 15 μm). As described in Section D3.1.1, typical estimates of the particle diameters likely to be seen in the aftermath of a reactor accident give values of a few micrometers. Hence, gravitational settling has generally been neglected in consequence modeling.

This neglect has recently been questioned by Kaul (1981), who has carried out scoping studies for the international Benchmark exercise. It is possible that the closer attention to source terms, discussed in Section D3.1.3, will provide proof of particle diameters so great that gravitational settling cannot be neglected, for some accident sequences.

D4 CHANGING WEATHER CONDITIONS

One of the most difficult problems to manage in a comprehensive consequence model is that of changing weather conditions. Indeed, even today, many consequence-analysis codes do not allow the weather conditions to change once a release of radioactivity has been assumed to take place. Most of the codes in the international Benchmark study are of this type. This problem is treated at progressively higher levels of sophistication in the models discussed below.

Constant-weather codes. The British code TIRION is an example of a constant-weather code (Kaiser, 1976; Fryer and Kaiser, 1979). The use of such codes is not acceptable in the United States for complete risk assessments, since there is a long tradition of the use of changing weather conditions, although there are applications for which certain parts of such codes can be useful.

Changing weather conditions but no wind-direction change, weather conditions determined everywhere by onsite data. CRAC and CRAC2 are examples of this sort of code and represent the state of the art in the sense that they embody the method that is most readily available to most consequence modelers in the United States.

Changing wind direction and weather conditions, determined by onsite data. To the author's knowledge, CRACIT (Commonwealth Edison Company, 1981) is the only code that is both capable of doing this and has been used in a consequence analysis in the United States. The German code UFOMOD has an option capable of modeling changing wind directions during releases of prolonged duration (Aldrich et al., 1979; Schueckler et al., 1979).

Changing wind direction and weather conditions, determined by data collected at the site and at a number of surrounding meteorological stations. In principle, the wind direction can also be modeled by taking account of topographical features, using, for example, potential flow theory. This represents the limit of what is being attempted in consequence analyses, and CRACIT appears to be the only code that attempts to be this ambitious.

A discussion of this hierarchy of models follows.

D4.1 CHANGING WEATHER CONDITIONS BUT NOT WIND DIRECTIONS

D4.1.1 An Example--CRAC

CRAC is typical of codes that attempt to take into account changing weather conditions but not wind directions. It takes a number of assumed accident starting times throughout the year and, for each of these, models the movement of a puff downwind, allowing for hourly changes in weather conditions. Figures D-8 and D-9, taken from a draft of the CRAC user's manual (scheduled to be published in 1982 by the NRC), give instructive summaries of the kind of considerations that are necessary in order to implement this model.

Figure D-8 shows how the width of the plume grows as atmospheric stability undergoes two changes. At time $t = 0$, the plume is released in stable conditions at the source O. It has a finite width OA-OB because of the reactor-building wake and behaves as if it is emerging from a virtual point source O'. When it has reached a distance 2 downwind, the weather conditions become less stable and the plume begins to grow more rapidly, as if from a virtual source located at 2'. When the plume reaches position 4, the weather conditions revert to stable and the rate of growth of plume width is less rapid, as if from a virtual source at 4'.

The treatment of the vertical dispersion is more complicated. Figure D-9 follows the growth in height of the plume through three stability changes. The weather is initially stable, and the plume has a nonzero height because of the finite dimensions of the reactor-building wake. It grows as if emerging from a point source at O'. When the vertical standard deviation σ_z becomes equal to $0.465\ell_1$, where ℓ_1 is the stable mixing height, σ_z is assumed to grow linearly. This occurs at a distance X_A downwind. This assumption is a peculiarity of CRAC, originally due to Turner (1969), and it is of no great importance in this context.

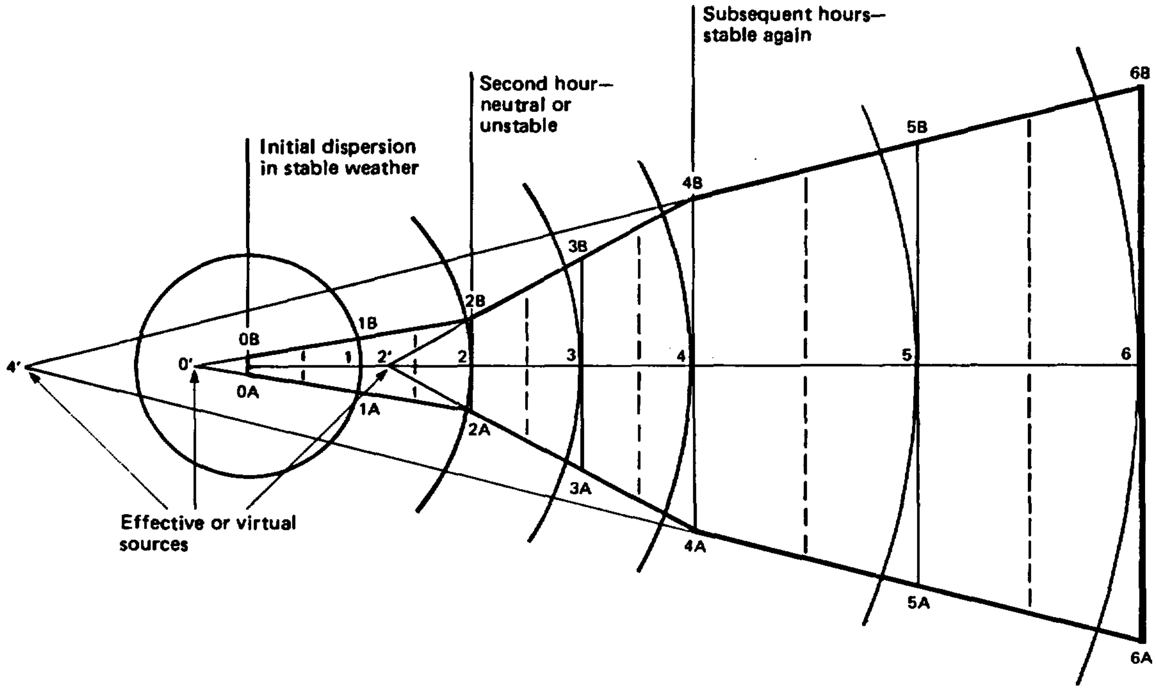

Figure D-8. Plan view of the growth of plume width with changing stability conditions. From a draft of the CRAC user's manual.

At a distance X_2 downwind, the stability changes to unstable, with a mixing height ℓ_2. Since $\sigma_z > 0.465\ell_2$ at X_2, it continues to grow linearly, but at a greater rate until, at X_3, the stability changes again and the mixing height falls back to ℓ_1. Since $\sigma_z > \ell_1$, it is assumed that the plume cannot grow any further and σ_z remains constant until the plume reaches X_4, when the weather once again becomes unstable. The growth of σ_z is then resumed and continues linearly until terminated at $\sigma_z = 0.8\ell_2$. This sort of procedure, whereby the relative magnitude of σ_z and the mixing height must be continually monitored, is essential in this changing-weather analysis. For the case where a buoyant plume may or may not penetrate an inversion lid, the monitoring procedure can become elaborate.

For example, in CRACIT the following cases are considered:

1. Neutral and unstable atmospheres--plume levels off well below the inversion layer.

2. Plume penetration of elevated stable layer.

3. Plume dispersion for fumigation and trapping under a lid, including--

   a. A plume that is trapped under the lid but has not yet reached the ground.

   b. A plume that is trapped and has reached the ground by the middle of a spatial interval but not by the end of the previous spatial interval.

c. A plume that is trapped and has not reached the ground by the middle of a spatial interval but has reached the ground by the end of the spatial interval.

d. A plume that has fumigated in a given spatial interval.

e. A plume that has been trapped and reached the ground, or has fumigated to the ground, and is subsequently treated as a ground-level release.

4. Growth of dispersion coefficients in spatial intervals downwind of the fumigation or trapping interval.

A particular case in which the inversion lid can be important is that of an onshore breeze at coastal sites. The air coming in from the sea is relatively stable. As it begins to travel over land, mechanical turbulence is generated at ground level, and the layer of turbulent air that is produced increases in height with distance inland. This can cause fumigation of the plume. An instructive discussion of this effect can be found in the description of CRACIT in the Zion study (Commonwealth Edison Company, 1981).

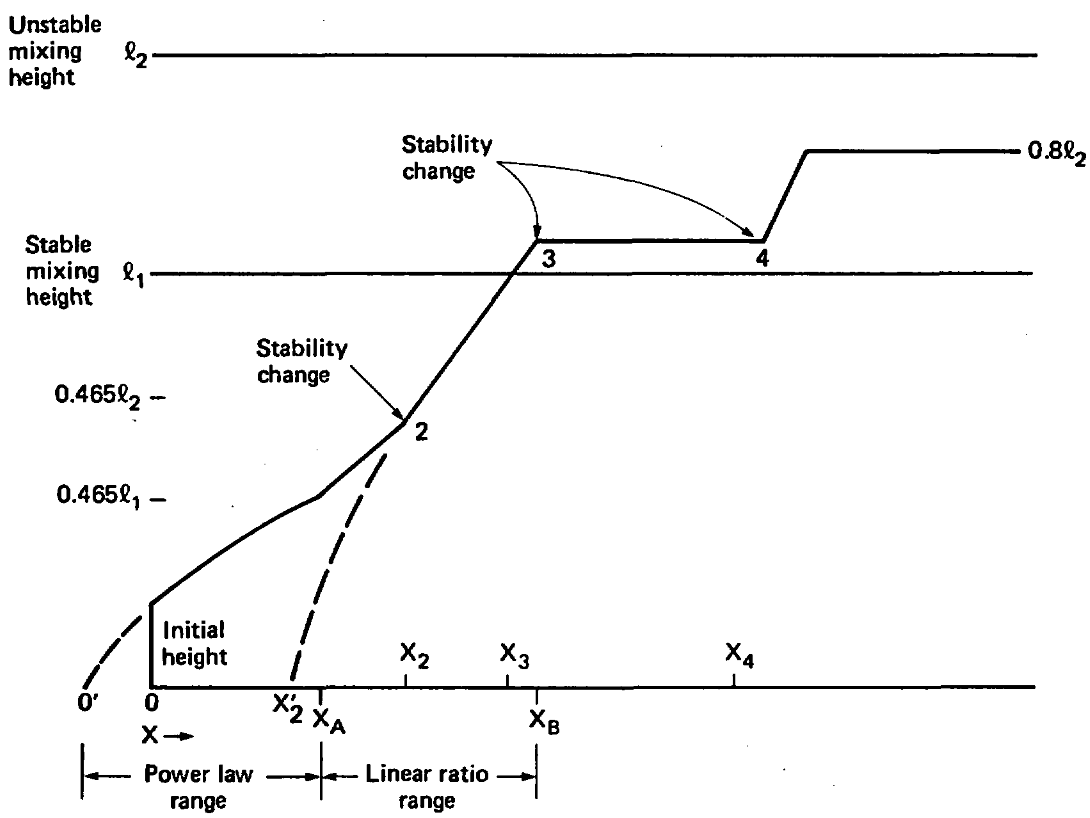

Figure D-9. Side view of the vertical growth of a plume. From a draft of the CRACuser's manual.

D4.1.2 Sampling

One of the problems that has, in the past, led to questions about the use of CRAC is doubt about the adequacy with which the code covers the range of possible weather conditions. In the course of a year, there are 8760 possible starting times, assuming that the meteorological data are available on an hourly basis. Conventionally, CRAC is used with 91 starting times. It is usual to rotate each of these sequences through the 16 sectors, making use of a wind rose to weight the answers so obtained. In principle, this then covers 91 x 16 = 1456 sequences, but it relies on the assumption that, if a certain sequence of stability and rainfall changes occurs with the wind initially blowing toward (say) the north, it will occur for the wind blowing in all other directions.

This procedure can lead to considerable uncertainties in the CCDFs calculated in the consequence analysis (Ritchie et al., 1981; see also Figure 9-17), because it is very likely that the weather sequences not sampled will include some that contribute significantly to the CCDF. Peak values of consequences can be underestimated by as much as a factor of 10. Furthermore, if a particularly adverse sequence is selected as one of the 91, it will be assigned a frequency of 1/91, whereas if it only occurred once in the year, the correct frequency should be 1/8760.

These uncertainties have been addressed in CRAC2, in which the entire year's worth of weather data is first assigned to groups of sequences with given characteristics--for example, that rain begins at a certain distance from the source, that the wind speed drops at a certain distance from the source, or that certain stability categories occur. In total, 29 "weather bins" are defined and the wind rose is worked out for each. Subsequently, the sampling procedure is operated so as to ensure that each weather bin is taken into account, removing the possibility that important weather sequences are omitted or given excessive weights. Ritchie et al. (1981) show that the uncertainties on CCDFs are much reduced by this procedure.

In CRACIT, a slightly different approach is adopted. First, a number of sequences are sampled at random from the year's worth of data, 288 such sequences being a representative number. Second, the meteorological data file is searched for the sequences that could contribute to the tails of CCDFs--for example, the sequences in which the plume would encounter rain over a center of population. These sequences are subsequently run through CRACIT.

D4.2 CHANGING WEATHER CONDITIONS AND WIND DIRECTION: ONSITE DATA

The straight-line model adopted in CRAC clearly lacks realism, should there be a change in wind direction, and, within the last year or two, there has been a move to develop models that take this into account. The simplest assumption is to approximate the radioactive release by a single puff, which might follow a path like that shown in Figure D-10. The wind direction is assumed to change everywhere when it changes at the source. The code CRACIT is able to simulate this kind of release, although it is usually operated at a higher level of sophistication (see Section D4.3). A similar scheme has been discussed by Vogt et al. (1980).

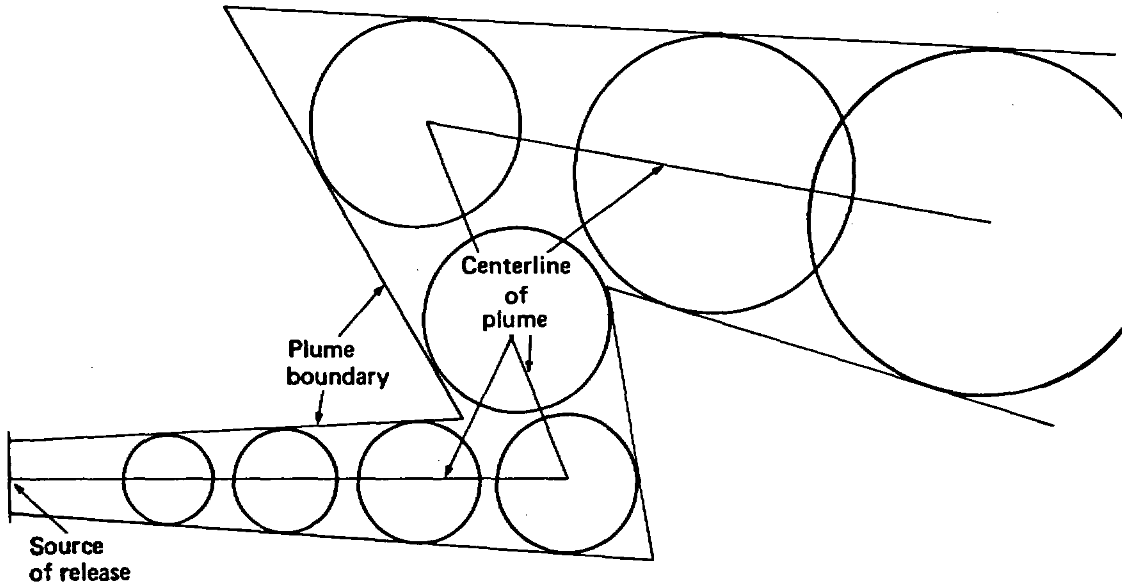

Figure D-10. Representation of plume: the growth of the puff as wind direction changes twice. From the Commonwealth Edison Company (1981).

On comparing Figures D-8 and D-10, it is clear that the new trajectory model forces the abandoning of symmetry about the initial wind direction, which leads to considerable complications in the computational procedure. Recognizing the desirability of retaining this symmetry, the authors of the German Risk Study proposed a wind-shift model (Aldrich et al., 1979) in which the concentration profiles are rotated with each change in wind direction (Figure D-11). This procedure works reasonably well for small changes in wind direction, but breaks down if the wind direction reverses.

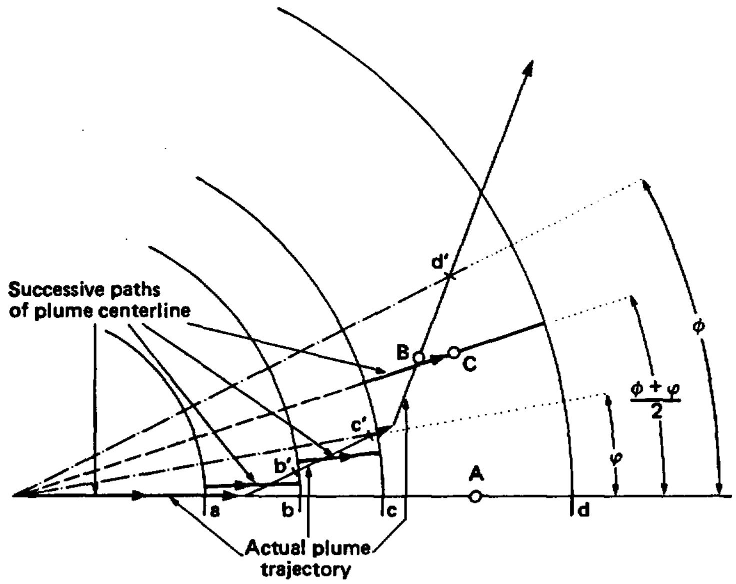

Figure D-11. Schematic showing the calculation of rotation angle for wind-shift model. From Aldrich et al. (1979).

Figure D-10 represents the path of a single puff. For prolonged releases, this is not a realistic description of the path of the plume. Figure D-12 shows what happens for a release lasting 2 hours. Initially, the plume is emitted from the source and follows wind direction 1. After an hour, it roughly covers the area WXYZ. If the wind direction then changes to direction 2, the whole of the plume will be transported sideways. This can be plausibly modeled as a finite number of puffs, following trajectories AA'A", BB'B", etc. Interpolation between these puffs is in principle necessary to obtain a realistic concentration profile. Vogt et al. (1980) discuss such a scheme, and CRACIT is currently being developed to run in a similar way. Meanwhile, if the release is continuing, a further plume W'X'Y'Z' develops. If the wind direction changes again, each of the puffs from the first hour follows a path like that of the puff in Figure D-10, while the plume from the second hour can be represented by four further puffs. An elaborate scheme of this nature has apparently not been attempted in a full consequence analysis. The reason is the complexity in the dose-calculation grid and the cost in computer time.

D4.3 CHANGING WEATHER CONDITIONS AND WIND DIRECTION: MANY SOURCES OF DATA

Another feature of CRAC that sometimes attracts adverse criticism is the use of onsite weather data to determine weather conditions as many as 500 miles away. Elaborate schemes have been developed by which weather data simultaneously gathered at a large number of weather stations can be processed in order to predict the path of a plume. Such procedures can be expensive in computer time. For example, the use of the code ADPIC (Lange, 1978) in the Benchmark study is restricted to the analysis of one month's

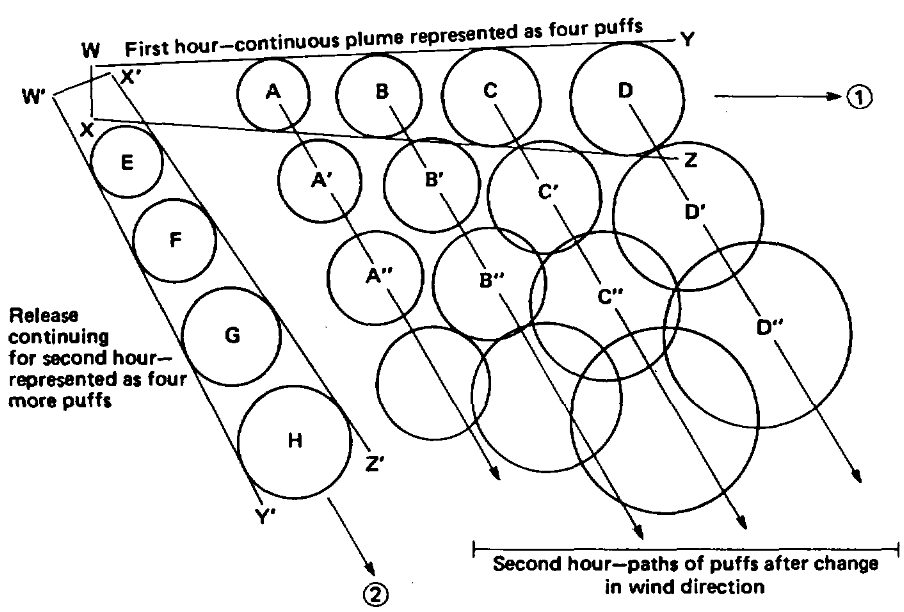

Figure D-12. Continuous plume represented as a series of puffs.

worth of weather data to minimize costs; it is certainly out of the question to use it in a comprehensive risk analysis at present.

The CRACIT code divides the region through which the plume travels into smaller areas within which the weather conditions are determined from data collected at a nearby weather station. CRACIT also contains a potential flow model for determining the effect of terrain features on wind speed and direction. The reader is referred to the CRACIT manual for details.

D4.4 COMMENTS

The impression that emerges from the foregoing discussion is that consequence analyses that take account of shifting wind direction can become very elaborate, especially if a multipuff treatment is adopted. It is natural to ask, Is it worth it? The answer is that it is worth it if it can be demonstrated that the procedure is manifestly more realistic than simpler ones and that uncertainties are reduced or, at the very least, not increased. If it is asked whether the wind-shift models achieve these objectives, the answer is that the case is not yet proven.

As far as the degree of realism is concerned, the fact that changes in wind direction are taken into account at all is clearly a step forward. On comparing Figures D-10 and D-12, however, it is apparent that a single-puff model may well predict airborne concentrations and deposited levels of radioactivity that are as far from being realistic as those given by the straight-line model. The multipuff model of Equation D-13, which, on the face of it, might be more realistic, has not, to the authors' knowledge, been used in a complete published risk assessment, although an example should become available when the Indian Point study is published.

The question of the degree of uncertainty also remains to be resolved. As already discussed, one of the difficulties with CRAC, when used in the Reactor Safety Study, was to ensure that the sampling of weather sequences was adequate. With these more elaborate models, in which the treatment of a single weather sequence can be much more costly than in CRAC, the problem is enhanced and the question of uncertainty has not yet been resolved.

The above comments about realism and uncertainty apply if site risk in a broad sense is being examined. There may, however, be certain site-specific applications for which a trajectory or multipuff model could give some useful insights. One such application could be the attempt to model evacuation together with plumes that change direction (see Appendix E3.1).

The use of wind-shift models may be unimportant for sites with steady winds for 80 to 90 percent of the time. Some U.S. valley sites have low wind speeds for 30 percent or so of the time, however, and this causes plume-meander problems that cannot be realistically modeled with straight-line plumes.

It is pertinent to remark that a comparison of CRAC, CRAC2, CRACIT, and NUCRAC has been carried out, assuming a large release of radioactivity at a river-valley site (Aldrich et al., 1981a,b).* Figures D-13 and D-14 show the results of the calculated CCDFs for early fatalities for two population distributions, one uniform and one nonuniform, with no emergency response. Experience with CRAC and CRAC2 shows that differences on the order of those shown on Figure D-14 can arise simply because of the different techniques employed to sample the weather data (see, for example, Figure 9-17). In any event, it can be concluded that, at least in this particular example, the results of the CRAC2 and CRACIT calculations are surprisingly close.

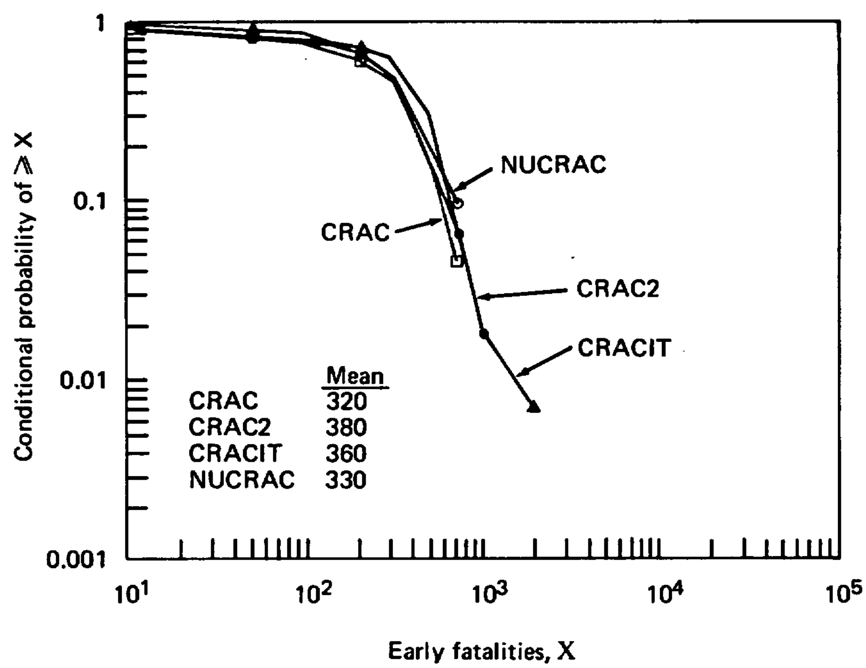

Figure D-13. Early-fatality CCDF, river-valley site. Uniform population, no emergency response. From Aldrich et al. (1981b).

Figure D-15 shows that, for CRAC, CRACIT, and CRAC2, the CCDFs for latent-cancer fatalities lie close together. Differences in the probabilities of peak events are almost certainly due to different meteorological sampling techniques. The NUCRAC results are somewhat higher because NUCRAC does not make use of the "central estimate" (see Section 9.4.8.4). The

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*Using the same nuclide groups as in the Reactor Safety Study, the release fractions are as follows: Xe-Kr, 1.0; I_2, 0.3; Cs-Rb, 0.3; Te-Sb, 0.3; Ba-Sr, 0.03; Ru, 0.03; La, 0.003. The time of release and duration of release were chosen to be 1 hr. This is the BMR-1 release already mentioned in Section D2.

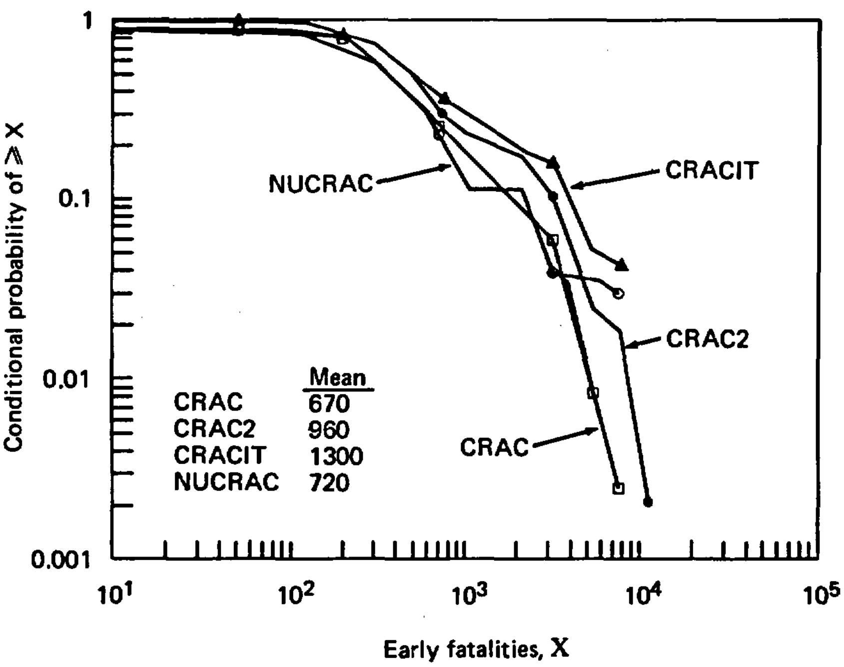

Figure D-14. Early-fatality CCDF, river-valley site. Realistic population, no emergency response. From Aldrich et al. (1981b).

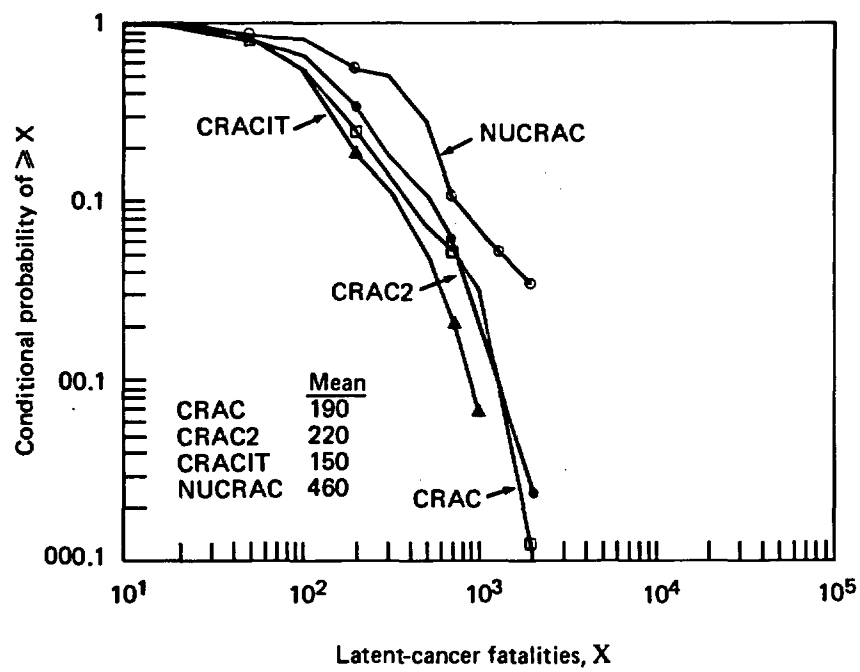

Figure D-15. CCDF for latent-cancer fatalities, river-valley site, realistic population. From Aldrich et al. (1981b).

closeness of these CCDFs is not surprising: it is by now well established that it is the magnitude of the source term that, in given weather conditions, controls the predicted number of latent-cancer fatalities.

For the present, then, it is judged that the time is not yet ripe for a universal move toward the use of wind-shift models in consequence analyses. The issues of realism and uncertainty are still a matter for research and debate. The organizations that see advantage in such techniques, however, are nonetheless encouraged to implement them in consequence analyses. Finally, it seems to the authors that the focus of the debate about wind-shift models is likely to move away from a simple comparison of the relative merits of CRAC and CRACIT (the forthcoming Benchmark report will discuss this; see also Aldrich et al., 1981a,b). After all, as can be seen from Figures D-8, D-10, and D-12, neither the straight-line model nor the puff-trajectory model is wholly realistic. The important question to ask is, Is it worth going to the expense of a full multipuff treatment?

The kind of consequence model that may well emerge in the future could be a hybrid. There is a strong incentive to use relatively inexpensive, straight-line models, which may well be of adequate accuracy for the calculation of many consequences. In order to treat the remaining cases--that is, those weather sequences for which wind shift cannot be ignored--it may be necessary to have a multipuff model on hand; this, however, should be used as sparingly as possible.

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Appendix E

Evacuation and Sheltering

E1 DESCRIPTION OF MODELS IN CRAC AND CRAC2

As stated in Section 9.2.1.6, it is in the selection of evacuation and sheltering strategies that the user of consequence-modeling codes can have a considerable influence on the results of his analysis of early fatalities and injuries. It is therefore pertinent to review the available models in some detail and to discuss in considerable depth the input-data requirements.

E1.1 THE RSS EVACUATION AND SHELTERING MODEL

In the Reactor Safety Study (RSS--USNRC, 1975), it was assumed that people are evacuated radially from a "keyhole"-shaped area such as is defined by a circle of radius r_e, a sector of angular width θ, and a further circle of radius $r_1 > r_e$ (see Figure E-1). In the RSS, r_e and r_1 were taken to be 5 and 25 miles, respectively, and a value of 45° was assigned to θ.

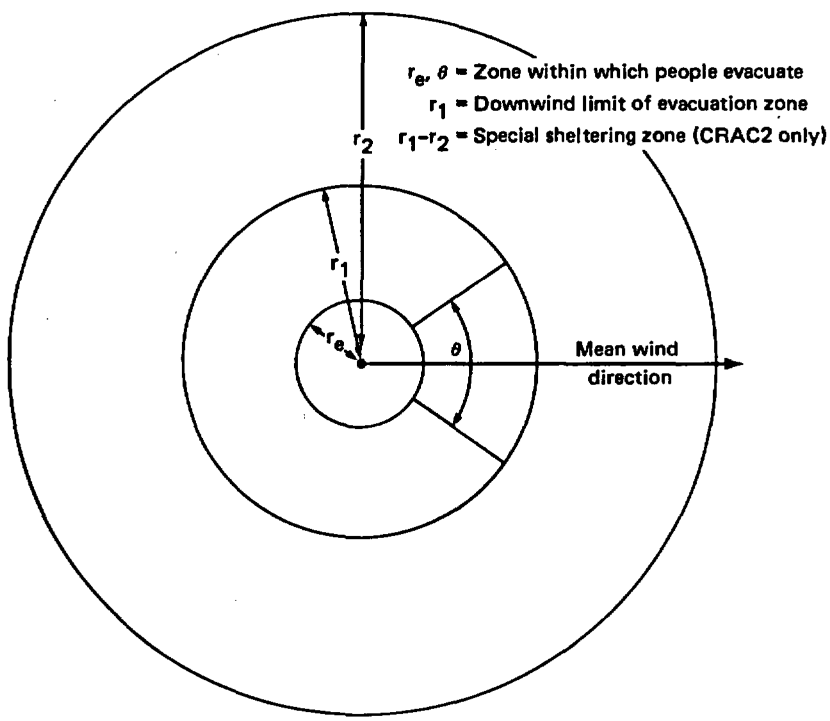


Figure E-1. Zones in CRAC and CRAC2 evacuation models.

In order to determine a suitable effective radial evacuation speed, reference was made to evacuation experience in the United States for the period 1959-1973, which is summarized in a report (Hans and Sell, 1974) published by the Environmental Protection Agency (EPA). The statistical analysis of the EPA data and its suitability as a foundation on which to base the modeling of evacuations in response to reactor accidents are discussed in Appendix VI of the RSS.

The RSS evacuation model postulates that evacuated persons will move radially away from the reactor at a constant "effective" speed immediately on warning by nuclear plant personnel of the impending release. No specific delay time is assumed for the notification of responsible authorities, the decision to evacuate, the time required by officials to notify people to evacuate, and the time required by people to mobilize and get underway.

Representative effective evacuation speeds were derived from the EPA data by dividing the recorded evacuated distances by the corresponding total time required to complete the evacuation--that is, essentially the time taken for the last person to leave the evacuation zone. This total time includes the delays mentioned above. Thus the effective speed is lower than the speed at which people would actually travel once they begin to move away from the reactor.

The statistical analysis of the EPA data performed in the RSS showed that (1) a lognormal distribution can be suitably used to describe the distribution of effective evacuation speeds; (2) the likely effective speeds are small; (3) the range of likely effective speeds is large; and (4) the number of persons evacuated had no statistically significant effect on the effective speed of evacuation. Because there is a large variation in effective evacuation speeds, the use of one "representative" speed was considered inappropriate. The distribution of evacuation speeds chosen to represent the EPA data was made up of three velocities--0, 1.2, and 7.0 mph, with probabilities of 30, 40, and 30 percent, respectively--and the population in the sector of width θ within 25 miles was assumed to move away radially at each of these three speeds in turn.

During the evacuation, the people are assumed to be unshielded from exposure to airborne radioactive material both externally and through inhalation. They are shielded from exposure to ground contamination by surface-roughness elements and the use of automobiles. If the evacuating people are overtaken by the cloud of radioactive material, it is assumed that they inhale the radioactive cloud and are exposed to cloudshine as if they remained stationary at the point at which the cloud reaches them. Subsequently, they are assumed to accumulate a radiation dose from gamma rays emitted by deposited radionuclides at the same point. It is assumed that, after 4 hours, the people will have moved outside the contaminated area and the accumulation of external exposure ceases.

People who do not evacuate (i.e., those beyond 25 miles) are assumed to be relocated after 7 days. However, if the dose accumulated within the first 7 days from exposure to contaminated ground exceeds 200 rads, then the people are assumed to be relocated within 1 day. Shielding factors for people beyond 25 miles are assumed to be typical of those for "normal activity," as described in Section E2.6.

E1.2 REVISED EVACUATION MODEL

There are a number of imperfections in the RSS model, among them the following:

1. The EPA data on which the RSS model is based are susceptible to more than one interpretation, as will shortly become clear.

2. Calculations that use effective evacuation speeds without an initial delay time do not provide realistic descriptions of the spatial or temporal movements of evacuating persons.

3. The assumption that evacuating persons overtaken by the radioactive cloud are exposed to the cloud for the entire duration of its passage and to ground contamination at a constant rate for 4 hours is also unrealistic.

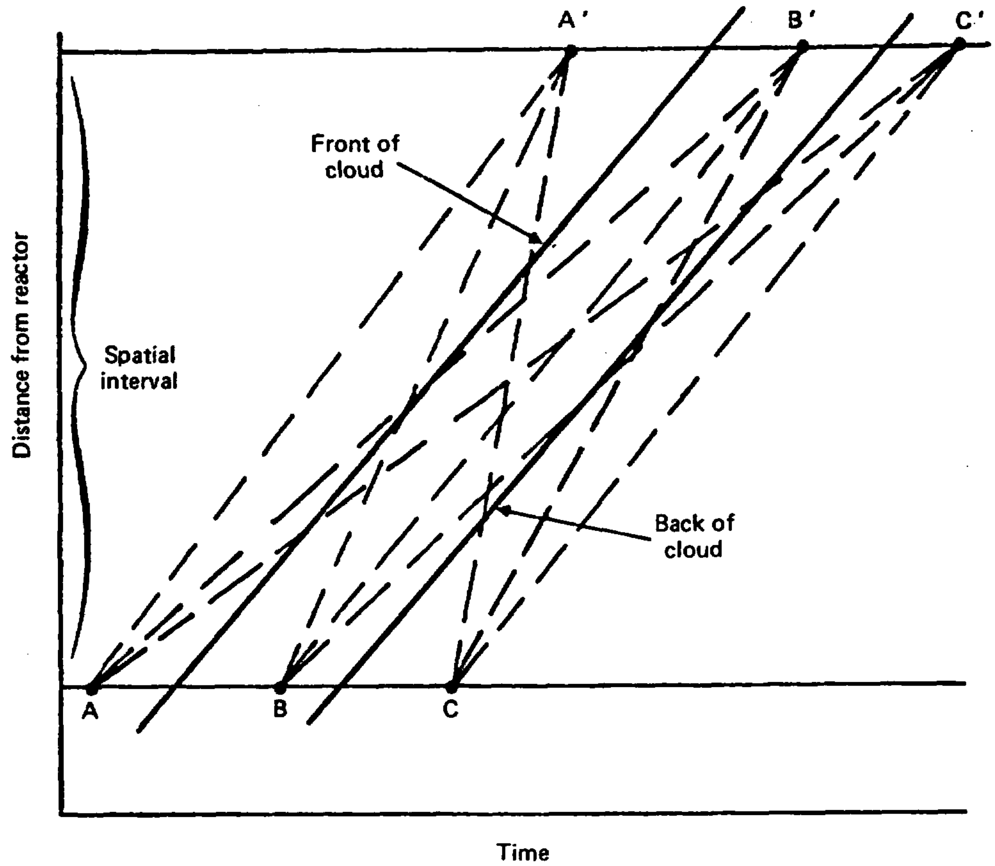
4. Shielding factors and breathing rates may differ markedly during delay and transit times.

In view of the foregoing, a revised evacuation model has been developed and included in CRAC2 (Aldrich et al., 1978a, b; Aldrich, Blond, and Jones, 1978). It is intended to be used with a delay time followed by evacuation radially away from the reactor at higher constant speeds than those used in the RSS. Different shielding factors and breathing rates are used while persons are stationary or in transit. All persons in the new model travel a designated distance from the evacuated area and are then considered to be no longer at hazard (removed from the problem).

A new feature of the revised evacuation model is the incorporation of a special sheltering zone lying between the radii r_1 and r_2 in Figure E-1. It is assumed that people in this area are instructed to take special precautions, such as retiring to the basements of their houses (if there are any) or to other buildings that provide effective attenuation of the gamma rays emitted by the passing cloud or by deposited radionuclides.

The new model also allows for the fact that the passing plume is of a finite length that depends on the wind speed and the duration of release. For simplicity, the cloud is assumed to be of constant length after the release, and the concentration of radioactive material is assumed to be uniform over the length of the cloud at any given time. The radial position of evacuating persons, while stationary and in transit, is compared to both the front and the back of the cloud as a function of time to determine a more realistic period of exposure to airborne radionuclides. Thus, people traveling rapidly enough may escape the cloud altogether. Others may be overtaken by the cloud or overtake it. In all, there are nine possibilities, as shown in Figure E-2.

The revised treatment also calculates the periods of time during which people are exposed to radionuclides on the ground while they are stationary and during evacuation. Because radionuclides would be deposited continually from the cloud as it passes a given location, a person while under the cloud would be exposed to a ground contamination that is less heavy than



(A, A'): People travel in front of cloud.
(A, B'): Cloud overtakes people.
(A, C'): Cloud overtakes and passes people.
(B, A'): People escape from under cloud.
(B, B'): People travel under cloud.
(B, C'): Cloud passes people.
(C, A'): People overtake and pass cloud.
(C, B'): People overtake cloud.
(C, C'): People travel behind cloud.

Figure E-2. Relative paths of evacuating population and plume. From Aldrich, Blond, and Jones (1978).

would be the case once the cloud has passed. To account for this in a simple way, the new model assumes that persons are exposed to (1) the total ground contamination calculated to exist after the passage of the cloud, when behind the cloud; (2) one-half the calculated concentration when anywhere under the cloud; and (3) no concentration when in front of the cloud.

E2 INPUT DATA--CRAC2

Since, as has been remarked previously, the user can considerably influence the output of a consequence analysis by his choice of input parameters for the evacuation and sheltering model, it is instructive to review in some depth how this input can be derived. The discussion is focused on the revised model in CRAC2 for ease of presentation; this should not be construed as a recommendation for the use of CRAC2 in preference to other codes.

E2.1 MAXIMUM EVACUATION DISTANCE AND RADIUS OF SHELTERING ZONE

The radius r_1 is the maximum distance downwind to which evacuation takes place within a sector whose centerline is the mean wind direction. In the RSS, as has been seen, this figure was taken to be 25 miles. Since the RSS was written, however, the NRC has provided guidance on the size of emergency-planning zones (EPZs) in NUREG-0654 (USNRC, 1981). Two EPZs have been defined, one to mitigate the consequences arising from the plume-exposure pathway and one to mitigate the consequences arising from the ingestion pathway. The NRC/EPA Task Force on Emergency Planning (USNRC, 1981) selected a radius of about 10 miles for the plume-exposure EPZ, for the following reasons:

1. Estimated doses from the traditional design-basis accidents do not exceed Protective Action Guide (PAG) levels (USEPA, 1975) outside this zone.

2. Estimated doses from most core-melt sequences do not exceed PAG levels outside this zone.

3. For the worst-case core-melt sequences, immediate life-threatening doses would not generally occur outside this zone.

4. Detailed planning within 10 miles would provide a substantial base for expanding response efforts should this prove to be necessary.

It is within this radius of 10 miles that detailed evacuation plans must be made in order for a new power plant to be licensed. Hence, it is to be expected that evacuation, if implemented, will be particularly effective within 10 miles. The wording of the NRC guidance leaves no doubt, however, that emergency-response procedures should be implemented beyond 10 miles if need be, but these may naturally take longer to implement and not be as effective as those within 10 miles.

A suitable way of taking this into account would be to take 10 miles for the maximum downwind distance of evacuation r_1 (although the existence of an EPZ of radius 10 miles should not be taken to mean that there is no other choice for r_1) and to simulate preventive countermeasures farther out by making use of a special sheltering zone lying between the radii r_1 and r_2. Studies have shown that the use of a hybrid shelter/evacuation scheme with $r_1 = 10$ miles and $r_2 = 25$ miles produces complementary cumulative distribution functions (CCDFs) for early fatalities that are much the same as those for studies in which evacuation was taken out to 25 miles (Aldrich et al., 1978a; 1979), at least when the sheltering was assumed to be in houses with shielding properties characteristic of those in the Northeastern United States. A key parameter is the length of time for which people shelter before leaving the sheltering zone. Aldrich et al. (1978a) take 6 hours as a reasonable figure and, in the absence of guidance from experience, this is as reasonable an assumption as any. It must be emphasized that the predictions of early fatalities and early injuries are very sensitive to this parameter, and the user should take care to be as realistic as possible.

E2.2 RADIUS AND ANGULAR WIDTH OF KEYHOLE-SHAPED SECTOR

In the RSS, r_e and θ were taken to be 5 miles and 45°, respectively. It is likely that guidance can be found in the emergency plans or in associated literature. To take an example at random, for the Palo Verde site in Arizona, r_e is 2 miles and θ is 67.5° (three sectors) (County of Maricopa, 1981). By contrast, some other emergency plans envisage the evacuation of the full EPZ, that is, r_e = 10 miles and θ = 360°. The value should be considerably in excess of the width of one sector (22.5°) to allow for fluctuations in the wind direction. Predicted numbers of casualties are insensitive to it so long as it is wide enough to cover the whole plume. The bigger θ is, the larger the predicted costs, since more people are evacuated, but this is generally a relatively small part of the total economic cost of an accident.

E2.3 DELAY TIME AND EVACUATION SPEED

Aldrich et al. (1978a) have examined the EPA evacuation data on which the RSS model was based and have concluded that it is consistent with an assumed evacuation speed of 10 mph and a spectrum of delay times with a mean of 3 hours and 15- and 85-percent confidence limits of 1 and 5 hours, respectively. These results can be approximated by assuming that any one of the delay times 1, 3, or 5 hours may occur with relative probabilities of 30, 40, and 30 percent, respectively. (CRAC2 allows the implementation of up to six evacuation strategies.)

If there are site-specific studies of evacuation strategies, these can in principle be used in order to estimate delay time and evacuation speed. Typical results of such a study are summarized in Figure E-3, taken from the report of the NRC/EPA Task Force on Emergency Planning (USNRC, 1981).

This example shows the importance of modeling a spectrum of delay times and/or effective evacuation speeds. Other sources of evacuation delay times and speeds can be found in the literature (Urbanik et al., 1980). The Pacific Northwest Laboratory has developed a computer code (EVACC) for estimating evacuation times (Moeller and Desrosiers, 1981). This code was prepared when the NRC increased the size of the plume-exposure EPZ to 10 miles. There was then a need for a model that could calculate time estimates by accurately representing the road network, population distribution (permanent, transient, and special facilities), weather conditions, warning times, response times, and delay times for each site. The EVACC code satisfies these requirements. It calculates the population distribution within the EPZ as a function of time and distance from the reactor. Another source of information on evacuation times around plants has been prepared by Urbanik (1980) in response to a request from the NRC; this is a summary of evacuation times for 52 nuclear power plants. A convenient review has been given by Urbanik et al. (1980).

The importance of as realistic an estimate of delay time as possible cannot be overestimated. Aldrich et al. (1979) indicate that, for delay times of 3 hours or more, the CCDFs for early fatalities are insensitive to evacuation speed over a range from 5 to 40 mph; that is, the exposure received by individuals during the 3-hour delay period is considerably larger than that received while in transit. By contrast, if the delay time is as little as 1 hour, the CCDF is drastically reduced in frequency at all levels of consequence, assuming a 10-mph effective evacuation speed (see Figure 9-7). These results apply to CCDFs calculated assuming that it is the spectrum of release categories PWR 1-4, defined in the RSS, that are being considered. It is pertinent to remark that it is the difference between the warning time (Section 9.4.2.3) and the delay time that is the true measure of the time available for evacuation before release takes place. The four PWR categories above have warning times in the range of 1 to 2 hours (see Table 9-1). For very short or very long warning times, the above conclusions would have to be modified.

E2.4 MAXIMUM DISTANCE OF TRAVEL DURING EVACUATION, r_{ev}

In the revised evacuation model, people travel radially to a fixed distance beyond the maximum evacuation distance r_1 and then are removed from the problem. This treatment accounts for the fact that, after traveling outward for some distance, people would be expected to learn their position relative to the cloud and be able to avoid it. There is no definitive guidance on assigning a value to this distance, but if r_1 is 10 miles, it seems reasonable to take r_{ev} as 15 miles.

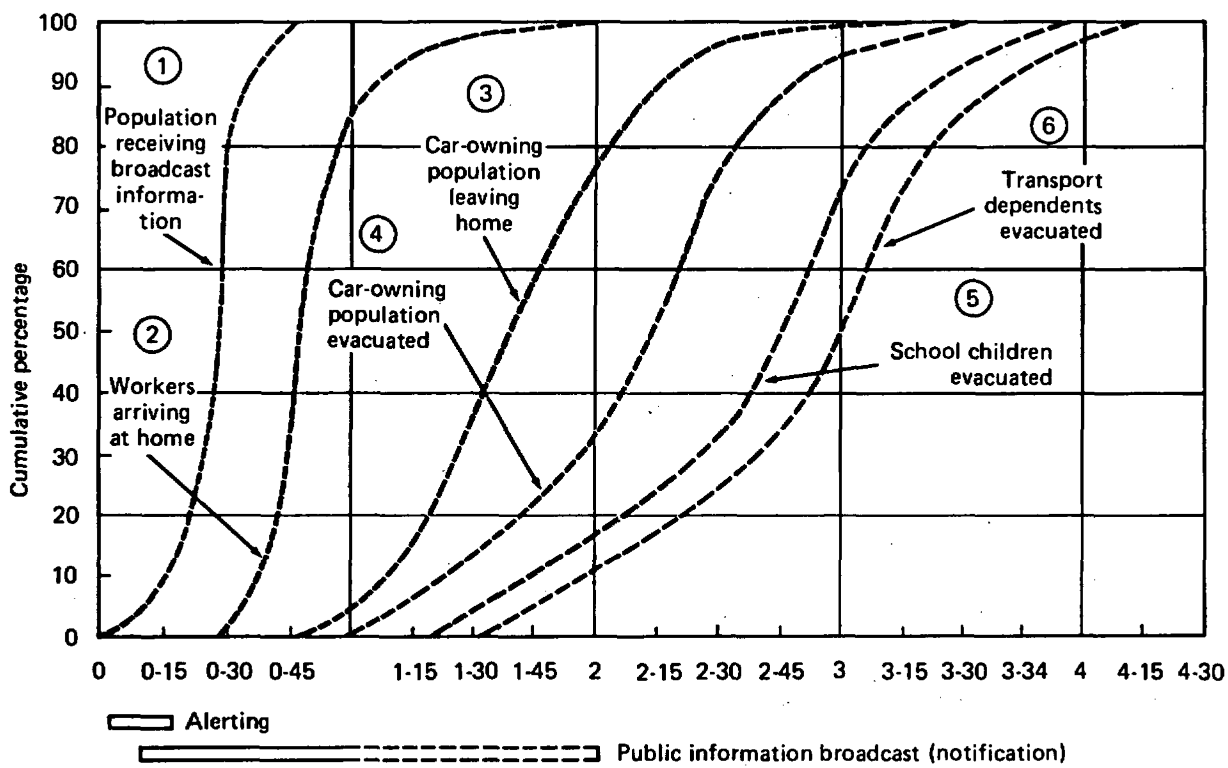


Figure E-3. Time estimates for population evacuation: typical response curves. These curves are suggestive of a hypothetical 10-mile-radius EPZ. Similar curves can be developed for subareas of the entire EPZ. The horizontal displacement of these curves along the time axis as well as the slope of the curves will vary with the characteristics of the EPZ or its subareas. From USNRC (1981).

This approach is preferable to that in the RSS model, in which people traveled downwind until they were assumed to leave the CRAC grid (500 miles downwind) or, if the cloud were to catch them, they would inhale the cloud at the point where they were overtaken and would be exposed to cloudshine and then to groundshine for 4 hours before being removed from the problem.

E2.5 CRITERION OF DURATION OF RELEASE FOR EVACUATION

If the duration of release is longer than this criterion, everybody within a radius r_1 is evacuated to allow for a possible change in wind direction. Three hours is the figure that has been enshrined in the CRAC standard data bank, and it is probably as good as any. Evacuating extra people in this way contributes to the cost of the accident but does not increase or decrease the predicted number of health effects. As already mentioned, some emergency plans envisage that the whole EPZ will be evacuated as soon as a warning has been given, in which case this time should be set to zero.

E2.6 SHIELDING FACTORS

The effectiveness of a structure in shielding the occupants from cloudshine or groundshine is well understood and is discussed in Appendix VI of the RSS and elsewhere (Aldrich et al., 1977; Burson and Profio, 1975). Table E-1 lists representative shielding factors derived by Aldrich et al. (1977) for various buildings or activities (note that these factors lie within a range of possibilities and are not unique).

In principle, CRAC2 requires cloud and surface shielding factors for four circumstances: (1) during evacuation; (2) while waiting to evacuate; (3) for people obeying special sheltering instructions in the region between radii r_1 and r_2 in Figure E-1; and (4) people behaving "normally" beyond r_2.

Table E-1. Representative shielding factors[a]

Type of structure	Representative shielding factor: Cloud	Representative shielding factor: Ground
Wooden house, no basement	0.9	0.4
Wooden house, basement	0.6	0.05
Brick house, no basement	0.6	0.2
Brick house, basement	0.4	0.05
Large office or industrial building	0.2	0.02
Outside	1.0	0.7
Commuting	1.0	0.7

[a]From Aldrich et al. (1977). See this reference for the range of factors that these single values represent.

E2.6.1 Shielding Factors During Evacuation

Shielding factors during evacuation can be taken as identical with those for "commuting" in Table E-1; that is, the cloud shielding factor is 1.0 and the ground shielding factor is 0.7.

E2.6.2 Shielding Factors While Awaiting Evacuation

In principle, these factors could be different from those used during evacuation. For example, people could be sheltering in their houses while waiting for transport. On the other hand, some people would be driving home to pick up their families and there would be no additional shielding. The choice of these shielding factors is therefore somewhat arbitrary; one of the easiest assumptions to make is that they are the same as those for evacuation, but there is no reason why the user should not choose something somewhat smaller.

E2.6.3 Shielding Factors in the Special Sheltering Zone

It is assumed that people within the sheltering zone take refuge in the most effective way possible. How effective this is depends on the nature of the structures in the neighborhood of the reactor. Guidance on this comes from the RSS, particularly Figure VI 11-9 of Appendix VI, which gives the percentage of brick-built houses in each state of the contiguous United States. The U.S. Department of Commerce (1972) has also published a census of detailed housing characteristics. For example, in Arizona 80 to 90 percent of houses are built of brick, but only 4 percent have basements; in Pennsylvania 60 to 70 percent of houses are of brick, the remainder being of wood, and 80 percent of all houses there have basements. Thus, in Pennsylvania an effective sheltering strategy would be to order all people to retire to houses with basements, where the cloud shielding factor would be 0.4 and the ground shielding factor would be 0.05. In Arizona, where there are no basements, sheltering in brick-built houses would give cloud shielding factors of 0.6 and ground shielding factors of 0.2.

People near large office buildings could make their way there and obtain the benefit of the cloudshine shielding factor of 0.2 and the groundshine shielding factor of 0.02 shown in Table E-1. CRAC2 is not detailed enough to be able to distinguish between people sheltering in structures with different shielding factors, however.

As mentioned before, people sheltering in this way are assumed to move quickly away after a time for which the value 6 hours has been suggested.

E2.6.4 Normal Activity

People beyond r_2 (Figure E-1), whose behavior is unaltered by the escape of radioactive material, will still be exposed to the passing cloud and to deposited gamma emitters. Robinson and Converse (1966) have examined the typical use that people make of their time and have derived Table E-2, to which representative shielding factors have been added.

Average shielding factors for all of the activities and places listed in Table E-2 can simply be derived by weighting the shielding factors by given percentages and summing:

$$\text{Cloud shielding factor} = [69.2(0.6) + 19.6(0.2) + 5.0(1.0) + 6.2(1.0)]/100 = 0.57$$

$$\text{Ground shielding factor} = [69.2(0.2) + 19.6(0.02) + 5.0(0.7) + 6.2(0.7)]/100 = 0.22$$

In the RSS, people were assumed to be exposed to cloudshine and subsequently to groundshine for 7 days before being relocated. If the whole-body radiation dose accumulated over this period was predicted to be greater than 200 rem, however, the people were to be relocated in one day. Some such limitation on accumulated dose is clearly realistic.

Table E-2. Typical use of time with examples of representative shielding factors

Place or activity	Hours per day	Fraction of total time (%)	Representative shielding factor[a] Cloud	Ground
Home (brick house)	16.6	69.2	0.6	0.2
School or work[b]	4.7	19.6	0.2	0.02
Commuting	1.2	5.0	1.0	0.7
Outdoors	1.5	6.2	1.0	0.7

[a]See Table E-1.
[b]Assumed to be a large building.

E2.6.5 Shielding Factors--Discussion

The use of shielding factors is sometimes criticized on the grounds that the results of the calculations can be distorted by the use of an average shielding factor, rather than a distribution of shielding factors, which would be expected in practice.

The calculation of latent health effects is based on the population dose (integral of population times dose, usually expressed as man-rem) rather than individual doses. It follows that latent health effects estimated from an average shielding factor should be identical with the results calculated from the appropriate distribution of shielding factors if a linear dose-response model is used.*

Early fatalities and early illnesses, on the other hand, are threshold effects and are calculated on the basis of the dose delivered to individual persons. It is possible, for example, that doses calculated by using an average shielding factor may fall below a threshold, whereas, in practice, some members of the population who happen to have been less well shielded may have received doses exceeding this threshold. For this reason, the use of average shielding factors may introduce some error into the calculation of early effects.

Aldrich et al. (1977) have carried out some scoping calculations and conclude that, in most instances, the use of average shielding factors will contribute only small errors compared to the overall uncertainties determined by all the other factors required in the consequence assessment. Occasionally, it is possible that the combination of a given accident sequence and weather sequence will lead to circumstances in which the predicted radiation dose in a large center of population is near a threshold. In this case, the predicted number of early fatalities or injuries obtained by using an average shielding factor could differ significantly from the number obtained by using a distribution of shielding factors.

This combination is likely to be a relatively rare occurrence. Furthermore, since the CCDFs that are the products of a consequence analysis consist of contributions from many accident and weather sequences, it is judged that the use of average shielding factors does not introduce significant errors into these results. However, if the user is concerned about this averaging in any particular application, there is no reason why he should not consider a spectrum of shielding factors.

E2.7 BREATHING RATES

CRAC2 requires as input the breathing rate b_r since the quantity of radioactive material inhaled is directly proportional to this quantity. In principle, different breathing rates can be input for people during the delay time, during evacuation, while sheltering, and for "normal" activity.

The International Commission on Radiological Protection (ICRP) has given guidance on an average figure for "normal" activity (ICRP, 1975). This is the figure used in the RSS: $b_r = 2.66 \times 10^{-4}$ m^3/sec. For convenience, this can be used for the breathing rates while waiting to evacuate and while evacuating. Further discussion of this topic is given in Section 9.3.3.1.

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*Aldrich et al. (1977) indicate that the latent health effects estimated by using an average shielding factor are nearly identical with the results calculated by using the appropriate distribution of shielding factors, even when nonlinear dose-response models are assumed.

The breathing rate for people taking sheltering precautions can be reduced to simulate the effectiveness of buildings at filtering radioactive particulate matter. In Section 9.3.4.5 it was suggested that for people sheltering in basements a reduction of the predicted inhalation doses by a factor of 2 is plausible. Such people might also take the precaution of breathing through a mask or a wet towel or some such, as discussed in Section 9.3.4.5. Hence an effective breathing rate ($b_r = 1.33 \times 10^{-4}$ m^3/sec) could well be justifiable for people who are assumed to be taking special sheltering precautions.

E2.8 SUMMARY

Table E-3 contains a summary of the data that are typically required as input to the CRAC2 evacuation model. It must be emphasized that these are examples only. They should not be taken over directly for any specific application of the consequence-modeling code.

Some points worth reemphasizing are as follows:

1. As far as possible, site-specific input information should be used, based on emergency plans and local studies.

2. The existing body of U.S. evacuation experience has been summarized by Hans and Sell (1974) and interpreted by Aldrich, Blond, and Jones (1978) with a spectrum of delay times (1, 3, and 5 hours) and an effective speed of 10 mph. If the parameters derived from the plans mentioned above are greatly at variance with the experience, they should be regarded with suspicion and, if need be, revised.

3. In general, a spectrum of parameters such as delay times and sheltering factors should be considered. In particular, attention should be given to a fraction of the population who will not or cannot evacuate. This group of people may well dominate the early fatalities.

E3 CRACIT EVACUATION MODEL

The evacuation models discussed above have in common the assumption that evacuees move radially outward from the reactor. In some cases, terrain features like river valleys force people to travel in certain directions, perhaps even toward the reactor for a time rather than away from it. CRACIT (Commonwealth Edison Company, 1981) is an example of a code that can take this into account. It does so by matching a plume path like that shown in Figure D-10 with a fine grid (see Figure E-4).

The fine grid used in CRACIT has 64 sectors, 400-m spacing out to 20 miles, and increasingly larger spacing out to 2000 miles. The code calculates ground, cloud, and inhalation doses for each element on this grid, at

Table E-3. Summary of representative input data for the CRAC2 evacuation model[a]

Parameter	Value		Comment
Radius of evacuation zone, r_1	10 miles		NRC guidance on radius of plume-exposure EPZ; r_1 may take on larger values if required
Radius of special sheltering zone, r_2	25 miles		Arbitrary value customarily assumed in applications of CRAC2
Radius of circle within which everyone is evacuated, r_e	2 miles		Depends on county emergency plan
Angular width of downwind sector for evacuation	67.5°		
Time delay before evacuation	1, 3, 5 hours[b]		Interpretation of EPA study of data on U.S evacuation experience
Evacuation speed	10 mph		
Maximum distance of travel while evacuating	15 miles from reactor		Arbitrary but sensible value
Duration of exposure to groundshine			
Within special sheltering zone	6 hours[c]		Suggested by Aldrich et al. (1977)
Beyond r_2	7 days[d]		As in the Reactor Safety Study (USNRC, 1975)
Shielding factors	Cloud	Ground	
Waiting to evacuate	1.0	0.7	
Evacuating	1.0	0.7	
Sheltering	0.4	0.05	Characteristic of brick houses with basements
Beyond r_2	0.57	0.22	
Breathing rate			
While sheltering	1.33×10^{-4} m^3/sec		Takes account of effectiveness of building in filtering aerosols
Otherwise	2.66×10^{-4} m^3/sec		Breathing rate of ICRP reference man (ICRP, 1975)

[a]The data included here are given as examples only. They should not be used directly for specific applications of CRAC2. Site-specific features should always be taken into account.

[b]With probabilities of .3, .4, and .3, respectively.

[c]Longer sheltering periods are also plausible.

[d]Duration of exposure would be 1 day if the predicted radiation dose exceeded 200 rem.

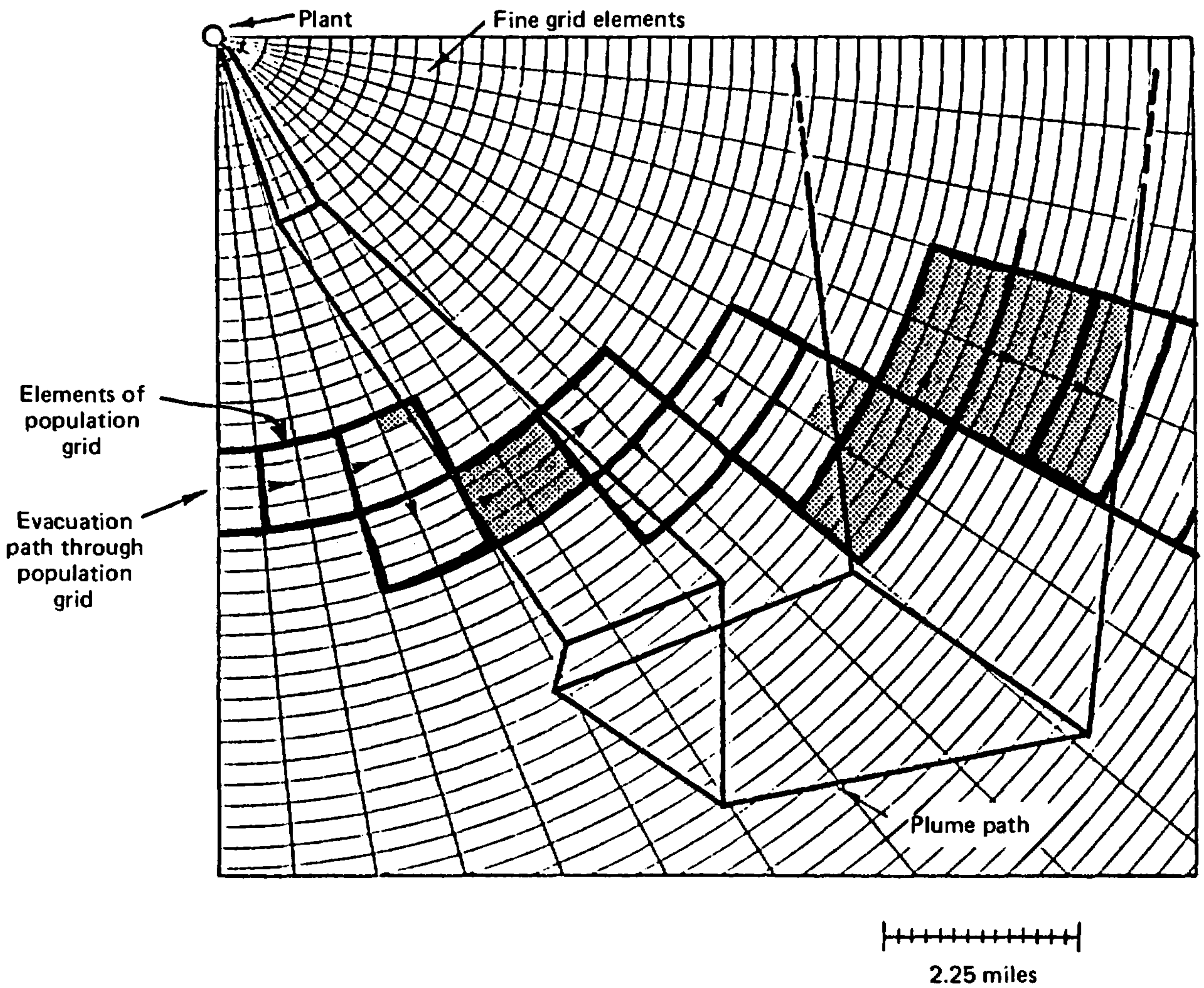

Figure E-4. Illustration of plume and evacuation paths on fine grid (dose calculations made by CRACIT in shaded fine-grid areas). From Commonwealth Edison Company (1981).

first assuming that people do not move from that element. In subsequent computations, these doses are adjusted by scaling and interpolation to obtain doses for people moving across the grid.

The information on doses must be matched by information on populations and the movement of people while evacuating. The people are assigned to a population grid that is somewhat coarser than the fine grid, having 32 sectors and larger radial intervals. Typical elements of the population grid are superimposed on Figure E-4.

Evacuation data for each population grid element are fed in. For each element, CRACIT requires (1) the distance a resident of the grid element travels to leave it; (2) the sector and segment identifiers of the next population grid element in the evacuation path; and (3) the distance traveled while crossing the next grid element. Once an evacuee enters a new grid element, he is assumed to follow the path of the evacuee who started from that element. Clearly, the input required for this evacuation model must be obtained from a study of the road network in the vicinity of the reactor in question and is more elaborate than that required by CRAC.

The evacuation paths in CRACIT can be simulated out to any desired distance, generally 10 miles. Evacuees stop at this distance and are assumed to remain there for 4 hours.

CRACIT can also accept different evacuation speeds for each population grid element, one set for each evacuation scenario. Speeds set for each element allow for the simulation of bottlenecks along the route.

The calculation of the exposure of an evacuee at any point is a matter of timing. Evacuee arrival and departure times for every population grid element along the evacuation path are computed from the evacuation travel distances and speeds discussed above, after taking warning times and delay times into account. Similarly, cloud-front arrival times and cloud departure times are known as a function of distance along the cloud's path. If the evacuee leaves a population grid element before the cloud arrives, he receives no exposure. If he arrives in a population grid element after the cloud has departed, he receives only ground exposure. If the cloud and the evacuee are simultaneously present in one population grid element, the evacuee receives both ground exposure and cloud exposure. Ground-exposure time is computed simply from time spent on contaminated ground. Cloud exposure is estimated as some fraction of the full-cloud exposure determined by the ratio of time spent in the presence of the cloud to the time required for full passage of the cloud.

In essence, once CRACIT has calculated doses at each element of the fine grid, it undertakes an elaborate bookkeeping and adding-up procedure that is really not all that complicated but requires so many operations that only a computer can do it. Readers are referred to the report of the Zion study (Commonwealth Edison Company, 1981) for details of the calculation.

E4 DISCUSSION

No doubt the potential user of consequence-modeling codes would like to be told that there is a particular evacuation model that is clearly the best and therefore should be used in preference to all others. This is not possible, however.

At first sight, models that attempt to simulate the road network around the site and the movement of evacuees along those roads might be thought to be more realistic. The problem with this assessment is that there are uncertainties in parameters, such as delay time, that tend to swamp the increased accuracy one might expect from the road-network models. Thus, for example, as described in Section E1.2, Aldrich et al.

(1979), working with release categories PWR 1-4 as defined in the RSS, have shown that, for delay times of 3 hours or more, the CCDFs for early fatalities are insensitive to evacuation speed over a range from 5 to 40 mph. This is because most of the radiation dose received by individuals is predicted to accumulate during the delay time. By contrast, for delay times of an hour or less, provided that the chosen evacuation speed is not ridiculously low, people can generally be expected to remove themselves from the path of the radioactive plume before it reaches them, and, again, a sophisticated road-network model would give no advantage. For intermediate delay times, the details of the evacuation speeds and routes could be important, however. It is pertinent to remark that the evacuation modeling does not in general affect the peak of the early-fatality or early-injury CCDFs. This is because these peaks are in general predicted to occur when rain deposits radionuclides in a large center of population beyond the evacuation zone. If this population center is in the sheltering zone, it is the shielding assumptions that are important. If the center is beyond the sheltering zone, consequence-modeling codes generally assume nothing more sophisticated than relocation after a day or a few days.

The main advantage of a road-network model is likely to be for those sites where there are bottlenecks. The model can then help identify potential problems. The network model is manifestly more realistic in certain cases, such as for seashore evacuation, where CRAC2 would send people radially out to sea. As was concluded in the discussion of wind-shift models in Appendix D4, however, consequence modelers are still debating the pros and cons of road-network evacuation modeling.

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Appendix F

Liquid-Pathway Consequence Analysis

F1 INTRODUCTION

It is difficult, given the variety of potential release situations, to define a clear set of procedures for a water-pathways consequence analysis. The transport of radionuclides in hydrospheric systems is affected by natural processes that are difficult to model. Any given reactor site is unique in many important surface and subsurface hydrospheric characteristics, and hence each site will have unique modeling needs. The definition of these needs, and the selection of appropriate analysis techniques, requires a great deal of judgment on the part of the risk analyst. Therefore, rather than define a specific set of procedures, this appendix provides a general introduction to the problem of modeling water pathways and describes an approach for analyzing the consequences of radionuclide releases into the water pathway. It discusses the site-specific characteristics that might influence risk through this pathway and, wherever possible, recommends modeling approaches.

F2 OVERVIEW

F2.1 SCOPE OF THE WATER-PATHWAYS PROBLEM

The potential consequences resulting from accidental releases of radioactive material to water pathways have not been examined with the same degree of detail as those resulting from releases to the atmosphere. Risks from the atmospheric pathway are generally considered to be dominant for two interrelated reasons. First, the time that radioactive contaminants would take to first reach the human population would probably be shorter for the atmospheric pathway. Delays in the hydrospheric transport of contaminants would allow for significant radioactive decay. Second, initial atmospheric exposure would usually be involuntary, whereas, in most cases, exposure to hydrospheric contamination could be largely avoided by the implementation of appropriate protective measures. As a result, individual doses resulting from the water pathway would probably be small, and early health effects would be unlikely.

Consequences resulting from radionuclide releases to the hydrosphere could be influenced by several factors: the type and characteristics of the release; characteristics of the local surface and subsurface hydrologic system; exposure pathways to the human population; interactions of the human population with encountered activity; and possible mitigating actions to reduce or prevent consequences.

A nuclear reactor accident could result in different types of release to the hydrosphere. Releases directly to groundwater are possible after a core-melt accident, provided that the melt penetrates the containment basemat. Other water-pathway sources could result indirectly from the atmospheric releases of radioactivity (e.g., rainout of contaminants onto surface-water systems).

The extent and rate of radionuclide transport and dispersion will depend on the characteristics of the hydrospheric system. Radioactive materials released into a groundwater aquifer can be transported to nearby surface-water bodies like lakes, estuaries, oceans, rivers, or reservoirs. The net velocity at which contaminants can move in a hydrospheric system will be affected by their interactions with soil and particles of sediment.

The impact of a reactor accident would depend on the amount of radioactive material that reached the human population. The primary exposure pathways would most likely be the ingestion of contaminated water and the ingestion of contaminated foods. Consequences could include economic, latent somatic, and genetic effects. Economic costs would result from measures taken, if any, to reduce radiation exposure.

F2.2 GENERIC LIQUID-PATHWAY STUDIES

Since the Reactor Safety Study (RSS), three generic studies dealing with the effects of radioactive releases to the water pathway have been performed (Offshore Power Systems, 1977; USNRC, 1978; Niemczyk et al., 1981). Two of these studies, conducted by the NRC and Offshore Power Systems, compared the potential environmental impacts of accidental releases to the hydrosphere from floating nuclear plants with those from land-based nuclear plants.

The third of these studies, which was performed at Sandia National Laboratories, evaluated the consequences that could result from accidental releases to the hydrosphere and compared them with those for releases to the atmosphere. These consequences were evaluated for each of four generic hydrospheric systems: large lakes, estuaries, oceans, and rivers. Only releases to the hydrosphere that would result from a molten reactor core penetrating the containment basemat were considered.

The Sandia study demonstrated that the water pathway can contribute significantly to reactor risk, if dose-mitigating actions are not taken. The total population doses for all the generic water bodies except the ocean were found to be approximately equivalent, given similar releases to the water bodies and given that the water bodies are considered in isolation. The radiation doses for the ocean system were found to be approximately one order of magnitude lower than those for reactors at the other sites. Table F-1 shows the Sandia results for the generic water bodies. The relative importance of several major exposure pathways was also evaluated. Table F-2 contains the relative ranking of the drinking water, aquatic food, and shoreline exposure pathways for each of the generic water bodies. The drinking-water pathway was found to be the largest contributor to the total population dose at all freshwater sites, whereas the aquatic food pathway is the major contributor at all saltwater sites.

Table F-1. Estimated population doses for the isolated generic sites[a,b]

Water body	Estimated population dose (man-rem)[c,d]		
	Melt debris	Sumpwater	Total
Large lake	4×10^4 to 6×10^6	1×10^3 to 1×10^7	4×10^4 to 2×10^7
Lake nearshore	2×10^4 to 7×10^5	1×10^3 to 2×10^6	2×10^4 to 2×10^6
Small estuary	2×10^6 to 3×10^7	1×10^5 to 4×10^7	2×10^6 to 7×10^7
Large estuary	1×10^6 to 2×10^7	2×10^4 to 2×10^7	1×10^6 to 3×10^7
Ocean	1×10^5 to 2×10^6	1×10^4 to 1×10^6	1×10^5 to 4×10^6
Ocean nearshore	1×10^5 to 2×10^6	1×10^4 to 1×10^6	1×10^5 to 4×10^6
Free-flowing river	3×10^5 to 1×10^7	1×10^4 to 5×10^7	3×10^5 to 6×10^7
Dammed river	1×10^6 to 3×10^7	1×10^5 to 5×10^7	1×10^6 to 8×10^7

[a]From Niemczyk et al. (1981).

[b]The releases considered are for a PWR-7 accident and are assumed to occur instantly into the groundwater.

[c]Upper ends of the indicated ranges represent groundwater travel times of approximately 100 days or less; lower ends represent times of 1000 days. Doses for longer travel times are much smaller.

[d]No dose-mitigating procedures were assumed.

Table F-2. Relative importance of exposure pathways at each of the generic sites[a]

Water body	Drinking water	Aquatic food	Shoreline usage
Importance with respect to population dose			
Isolated aquifer	1	--	--
Large lake	1	2	3
Estuary	--	1	2
Ocean	--	1	2
River system	1	2	3
Importance with respect to average individual dose			
Isolated aquifer	1	--	--
Large lake	2	1	3
Estuary	--	2	1
Ocean	--	1	2
River system	2	1	3

[a]From Niemczyk et al. (1981).

F3 APPROACH TO WATER-PATHWAY ANALYSIS

A site-specific water-pathway study is governed by the understanding of the processes involved. There is no single water-pathway consequence code that is applicable for all situations. Individual reactor sites will have pathway characteristics that are unique. Therefore, careful judgment is required to select and apply liquid-pathways models during a consequence analysis.

The initial objective of a water-pathway analysis should be to determine whether releases to the water pathway are important relative to releases into the atmosphere. In general, an analysis should be performed using simple models and conservative assumptions (e.g., neglecting dose-mitigating measures) to assess latent somatic effects. Care should be taken to select simple models that reasonably approximate the dynamics of the contaminants in the hydrospheric system and, to the extent possible, incorporate the most important pathway characteristics. Sensitivity analyses should be performed to assess the impact on predicted consequences of uncertainty in the most important hydrologic parameters. If it is found that the water pathway is not important for reactor risk, then the water-pathway consequence analysis is complete.

If the water pathway is found to be important, then additional analyses should be performed to assess the effect of less conservative assumptions on the liquid-pathway risk. Potential reductions due to mitigating actions might be evaluated.

A water-pathway consequence analysis can be divided into several tasks:

1. Acquisition of background information.
2. Selection of models.
3. Gathering and processing of data.
4. Exercising the models and interpreting the results.

These tasks are not necessarily independent. For example, the detail, availability, and uncertainty of data will affect the selection of appropriate models. These four tasks will be discussed individually in the sections that follow. A short discussion of exposure-mitigating actions will also be included.

F3.1 ACQUISITION OF BACKGROUND INFORMATION

Before beginning a water-pathways risk assessment, the analyst should familiarize himself with the physical processes and pathway characteristics that would be considered, including (1) the possible source terms and their characteristics, (2) the dispersion of contaminants in the hydrosphere and the physical processes that would affect it, (3) the possible interactions between the human population and the contaminated hydrosphere, and (4) the individual and societal risks that could result. Both the NRC and the Sandia liquid-pathway reports provide a good introduction to the entire liquid-pathways modeling problem. Additional sources of information include a review by Onishi et al. (1981) and The Water Encyclopedia (Todd, 1970).

F3.1.1 Determination of the Source

Sources of contamination can result from radionuclides released directly to the hydrosphere or indirectly from atmospheric releases. The most important direct releases would result from a reactor core melting through a containment basemat. Three types of direct releases into groundwater due to core melt could occur (USNRC, 1978; Niemczyk et al., 1981): the leaching of contaminants from the core-melt debris; the flow of contaminated sumpwater into the ground; and the injection of contaminants into the soil during depressurization.

A fourth type of release is also possible: the escape of sumpwater into surface water along a route other than through the core-melt hole. The first release would generally occur rather slowly; the other three could take place relatively quickly. The magnitude and the probability of these releases would depend on the reactor design and the accident scenario. Tables F-3, F-4, and F-5 compare the RSS releases with those that were analyzed in the Sandia study.

Other direct sources of contamination could result from accidents within the design basis. Such accidents could lead to the release of contaminated effluents directly to a surface-water body. However, such releases would not be expected to significantly affect reactor risk since the resultant doses would not be large.

Sources of hydrospheric contamination that would result indirectly from the atmospheric pathway have generally not been considered during consequence calculations. The most important sources would include direct deposition (e.g., rainout) of airborne contaminants onto surface-water bodies; erosion and washoff of ground-deposited radionuclides to water pathways; and leaching of ground-deposited contaminants into a groundwater aquifer. Since the deposition and washoff releases would be directly to accessible surface-water bodies (i.e., no delays caused by groundwater transport), these releases could dominate liquid-pathway risk. In general, indirect sources could occur for any accident scenario leading to an atmospheric release of radioactive material.

F3.1.2 Site Characteristics

The analyst should next determine the important site characteristics. This would include defining the hydrospheric system through which the contaminants would be transported and determining the interactions between the human population and the contaminated hydrosphere.

After a core-melt accident, the initial movement of contaminants would usually be by groundwater transport. The rate and importance of this

Table F-3. Airborne release fractions

Release category	Xe-Kr	I[a]	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR-1	0.9	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR-2	0.9	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR-3	0.8	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR-4	0.6	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-3}
PWR-5	0.3	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-4}
PWR-6	0.3	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR-7	6×10^{-3}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
BWR-1	1.0	0.40	0.40	0.70	0.05	0.5	5×10^{-3}
BWR-2	1.0	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR-3	1.0	0.10	0.10	0.0	0.01	0.02	4×10^{-3}
BWR-4	0.6	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1×10^{-4}

[a]Organic iodine is combined with elemental iodines in the calculations. Any error is negligible since the release fraction of organic iodine is relatively small for all large-release categories.

[b]Includes Ru, Rh, Co, Mo, Tc.

[c]Includes Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm.

Table F-4. Leach release fractions[a]

Element	PWR 2-7; BWR 3-4			PWR-1			BWR-1			BWR-2		
	Min	Best	Max	Min	Best	Max	Min	Best	Max	Min	Best	Max
Xe	0	0	0	0	0	0	0	0	0	0		0
I	0	0	0.010	0	0	0.010	0	0	0.010	0		0.010
Cs	0	0	0.050	0	0	0.050	0	0	0.050	0		0.050
Te	0	0	0.010	0	0	0.010	0	0	0.010	0		0.010
Sr	0.760	0.890	0.980	0.76	0.890	0.950	0.76	0.890	0.950	0.760	0.89	0.900
Ru	0.675	0.920	0.980	0.60	0.600	0.600	0.50	0.500	0.500	0.675	0.92	0.970
La	0.940	0.987	0.998	0.94	0.987	0.997	0.94	0.987	0.995	0.940	0.98	0.996

[a]From Niemczyk et al. (1981).

Table F-5. Sumpwater release fractions for some release categories[a]

Element	PWR-1			PWR-3			BWR-5			BWR-7		
	Min	Best	Max	Min	Best	Max	Min	Best	Max	Min	Best	Max
Xe	0	0	0	0	0	0	0	0	0	0	0	0
I	0.29	0.30	0.30	0.79	0.80	0.800	0.96	0.97	0.970	0.990	1.000	1.000
Cs	0.55	0.60	0.60	0.75	0.80	0.800	0.94	0.99	0.990	0.950	1.000	1.000
Te	0.59	0.60	0.60	0.69	0.70	0.700	0.98	0.99	0.990	0.990	1.000	1.000
Sr	0	0.06	0.19	0	0.09	0.220	0.02	0.11	0.240	0.020	0.110	0.240
Ru	0	0	0	0	0.05	0.295	0.02	0.08	0.325	0.020	0.080	0.325
La	0	0.01	0.06	0	0.01	0.060	0	0.01	0.060	0.002	0.013	0.060

[a]From Niemczyk et al. (1981).

transport are determined by (1) whether or not the groundwater aquifer flows to an accessible water body, (2) distances to surface-water bodies, (3) the effective groundwater-flow velocity, and (4) the physical and chemical composition of the soil.

The rate of radionuclide movement within a hydrospheric system would depend primarily on the general advective and convective characteristics of the system. These characteristics should be determined for the hydrospheric system under consideration. In addition, physical processes that could significantly influence radionuclide movement should be identified (e.g., radionuclide interactions with sediments and soils).

After defining the hydrospheric system, the analyst should determine what interactions occur between this system and the human population. Important considerations would include (1) fishing industries, (2) use of the system as a source of drinking or irrigation water, and (3) commercial and recreational uses of the system. These interactions will determine the possible exposure pathways for the human population.

F3.1.3 Individual and Societal Risks

The risk resulting from the liquid pathway would depend on the quantity of radioactive material reaching the human population. Pathway consequences could include any combination of latent-cancer, genetic, and economic effects. Economic effects would result from any dose-mitigating actions that might be taken; for example, the interdiction of a city's drinking-water source could lead to significant economic costs.

F3.2 SELECTION OF MODELS

The selection of pathway models is the next task that would need to be completed for a liquid-pathways consequence analysis. Pathway models should be capable of reasonably approximating the dynamics of the contaminants within the liquid pathway. A good axiom to follow in selecting models is, "simple is good, simpler is better," so long as they answer the question posed within the desired degree of accuracy (Gloyna, 1977). The degree of realism inherent in each model depends on the ability of that model to account for the physical processes that are involved. As a general rule, complex models are capable of yielding more realistic results. However, a realistic model requires realistic input data. Little is gained by using highly sophisticated models when the input parameters are ill-defined. In addition, detailed site-specific data are often not available.

The sections that follow briefly discuss the analysis of hydrospheric transport, human exposure pathways, and consequences. Reviews of computer codes that can be used for assessing radionuclide releases to the environment have been presented by Hoffman et al. (1977) and Strenge et al. (1976).

F3.2.1 Hydrospheric Transport

The transport of radionuclides in a hydrospheric system can be affected by various mechanisms. These include advective-mass transport, dispersive-diffusive transport, and sediment transport.

The general movement of contaminants in a system can be described by

$$\frac{dc}{dt} = -U_x \frac{dc}{dx} - U_y \frac{dc}{dy} - U_z \frac{dc}{dz} + \frac{d}{dx}\left(D_x \frac{dc}{dx}\right) + \frac{d}{dy}\left(D_y \frac{dc}{dy}\right) + \frac{d}{dz}\left(D_z \frac{dc}{dz}\right) - \lambda c + S(t)$$

where c is the concentration of the material in the water; U_x, U_y, and U_z are the x, y, and z components of the water velocity; D_x, D_y, and D_z are the dispersion coefficients for the x, y, and z directions; λ is the decay constant of the radionuclide; and S is the rate of release (and/or removal) of the material into (and/or from) the water. The relative significance of these transport mechanisms will depend on the transport problem being addressed. The source and/or sink terms can represent various mechanisms in the transport process, such as sedimentation (resuspension and deposition), sorption, and biological uptake.

Methods recommended by the NRC for treating transport processes are presented in Regulatory Guide 1.113 (USNRC, 1977). This document describes some acceptable methods for predicting the transport of radionuclides for rivers, open coasts, estuaries, and impoundments. The transport models discussed in this guide do not explicitly include sediment-uptake processes, although sediment transport is recommended as a consideration. Under certain conditions, adsorption and absorption on bottom sediments can be important (USNRC, 1978; Niemczyk et al., 1981; Onishi et al., 1976). The ability of suspended and bottom sediments to adsorb and absorb radioactive nuclides from solution may create a significant pathway to people. However, the sorption of radionuclides is also an important mechanism for reducing the area of influence of accidental releases.

Hydrologic modeling is most easily discussed by considering each of several basic receiving water systems. A brief discussion of critical characteristics and applicable radionuclide-transport models will be included for each system. Table F-6, which is taken from a workshop proceedings (Gloyna, 1977), provides a general summary of applicable hydrologic transport models. An excellent review of transport models is presented by Onishi et al. (1981).

Rivers

Advection and dispersion processes have been more thoroughly investigated and validated in rivers than in any other aquatic environment (Gloyna, 1977). Presently available river models can adequately simulate the transport of radionuclides (e.g., Sr-90 and Cr-51) that are transported mostly in a nonsorbed form. Some recent models can reasonably describe the transport, dispersion, and resuspension processes for radionuclides (e.g., Cs-137, Co-60, and Mn-54) that are easily sorbed by suspended or bottom sediments. Effects of the direct uptake and/or desorption of nuclides by

Table F-6. Summary of hydrologic models[a]

Author	Applicable water body[b]	Advection	Diffusion dispersion	Biotic uptake	Sedimentation	Number of dimensions: One	Number of dimensions: Two	Number of dimensions: Three	Steady state	Dynamic	Analytic solution	Numerical solution technique[c]	Presently includes radionuclides	Verified with in situ radionuclide data
Armstrong and Gloyna, 1968	B	X	X	X	X	X				(d)		FD	X	(e)
Bramati et al., 1973	B, D	X	X			X			X		X		X	
Dailey and Harleman, 1972	B, D	X	X			X				X		FD		
Daniels et al., 1970	D	X	X			X				X		FD	X	
Fletcher and Dotson, 1971	B, D	X	X			X			X		X		X	
Harleman et al., 1976	B, D	X	X			X				X		FD		
Hydroscience, 1968	B, D	X	X			X			X		X			
Leendertse, 1970	C, D	X	X				X			X		FD		
Martin et al., 1976	B	X				X			X		X		X	
Onishi et al., 1976	B, C, D	X	X		(f)		X (X-Y)			X		FE	X	
Onishi, 1977	B, E	X	X		(f)		X (X-Z)			X		FE	X	X
Pagenkopf et al., 1976	C	X	X				X			X		FE		
Ryan and Harleman, 1973	E	X	X			X				X		FD		
Shih and Gloyna, 1967	B	X	X	X	X	X				(d)	X		X	(e)
Shirazi and Davis, 1974	A	X	X					X	X		(g)			
Shull and Gloyna, 1968	B	X	X	X	X	X				(d)	X		X	
Stolzenbach et al., 1972	A	X	X					X		X		FD		
Ward, 1973	D	X	X				X			X		FD		
Waldrop and Farmer, 1974	A	X	X					X		X		FD		
Water Resources Engs., 1973	B	X	X	X	X	X			X			FD	X	
Water Resources Engs., 1974	D	X	X			(h)				X		FD		
Water Resources Engs., 1976	B	X	X			(h)				X		FD		
Watts, 1976	A	X	X			X			X		X		X	
Yotsukura and Sayre, 1976	B	X	X				X		X		X	FD		(i)

[a]From Gloyna (1977).
[b]Key: A, initial dilution; B, river systems; C, coastal systems and Great Lakes; D, estuarine systems; E, impoundments.
[c]Key: FD, finite difference; FE, finite element.
[d]Constant input parameters.
[e]Laboratory verification.
[f]Computes inventory of sediment and radionuclides in the bed.
[g]Integral model (three-dimensional equations reduced to one dimension and integrated via marching solution).
[h]Quasi two-dimensional.
[i]Verified against tracer data.

aquatic biota are less well known than those involving sediments, and very few attempts have been made to include them. The critical parameters to be considered in river modeling include (1) flow characteristics, such as the river discharge rate; (2) sediment characteristics (e.g., diameter and mineral composition of sediment); and (3) such radionuclide characteristics as adsorption and desorption rates. The relative importance of these characteristics will depend on the chemistry of the radionuclides being considered.

Various transport models, ranging from simple to complex, are available. Simple models would include those that represent advection-diffusion processes in an algebraic form (e.g., Fletcher and Dotson, 1971; Bramati et al., 1973; Soldat et al., 1974; and Martin et al., 1976). These models employ such factors as the mixing ratio and average transit time in quantifying advection and diffusion effects and are generally applicable only for chronic (routine) releases of radionuclides. More complicated models, based on the solution of the advection-diffusion equation, are needed to approximate the transport of dynamic accidental releases. One-dimensional analytical solutions of the equation can be obtained for the transport of radionuclides, as was done for the NRC liquid-pathway study (USNRC, 1978) and the Sandia liquid-pathway study (Niemczyk et al., 1981) and by Shih and Gloyna (1967), Shull and Gloyna (1968), and NRC Regulatory Guide 1.113 (USNRC, 1977). Numerical schemes, such as the finite-difference method, can be employed to approximate solutions to the one-dimensional advection-diffusion equation, as was done by Armstrong and Gloyna (1968) and White and Gloyna (1969). These models are generally capable of treating instantaneous, continuous, or time-varying releases of radionuclides into receiving waters.

A significant increase in sophistication occurs when the solution of the two-dimensional form of the advection-diffusion equation is attempted. Solutions of the two-dimensional equation have been developed and applied by Onishi (1977). Such models may be used for analyzing the transport of radionuclides in vertically well-mixed streams that cannot be assumed to be homogeneous across the flow field.

Estuaries

The mathematical models for estuaries can be used to predict distributions for radionuclide concentrations under both freshwater and reversing tidal flow conditions. The presently available models that can be used to approximate radionuclide concentrations in estuaries can be broken down into two major categories: tidally averaged and tidal-transient models. For tidally averaged models, the advective effect of the tidal cycle is taken into account by the diffusion term as an effective dispersion. This approach has the advantages of simplicity and can often give reasonable results. In general, the dispersion coefficients are the critical parameters for all simple models.

The previously cited models by Bramati et al. (1973) and Fletcher and Dotson (1971) can also be used to model steady-state releases into estuaries. Although estimates of the mixing ratio and transit time are subject to error, these simplified models can provide adequate answers if conservative assumptions are employed. One-dimensional tidally averaged models were used for both the NRC and the Sandia liquid-pathway studies. These models are based on the solution of the advection-diffusion equation with the assumption that tidally induced advection can be sufficiently described by longitudinal dispersion. Regulatory Guide 1.113 (USNRC, 1977) contains additional solutions to the advection-diffusion equation.

River models, such as that by White and Gloyna (1969), may also be applicable for estuaries. In these models, advective transport is considered to be a result of spatially averaged stream velocity. The advective effects of the estuarine tidal cycle are accounted for in the diffusion term. An alternative to this approach is a model that lumps the tidal hydraulics into the advective transport term. Such a model, which in effect is a transient estuary model applicable to instantaneous or accidental releases, was developed as part of the radionuclide studies for the Columbia River estuary (Daniels et al., 1970).

Coastal Systems and Great Lakes

The effect of sediment transport on radionuclide migration will generally be less important in the sea than in rivers or estuaries. Most radionuclides would be expected to remain in the dissolved phase. Near the shore, coastal dispersion is complex and may require more complicated models. Coastal currents, irregular bathymetry, and tidal oscillation complicate the flow field and invalidate simple models that are applicable further off the shore. In the open sea, the phenomenological patch-spreading model based on the dispersion of tracers has been widely used and appears to be adequate for instantaneous releases. Some numerical models (see, for example, Eraslan, 1975) are useful close to shore. Semianalytical models like the MIT transit-plume model (Adams et al., 1975) are appropriate if the receiving-water geometry is sufficiently open. Coastal models for the Great Lakes are similar in many respects to oceanic models, especially in the nearshore zone and for the spreading of instantaneously released patches. Contamination can be spread throughout the entire lake on the order of weeks. Because Great Lakes flushing times are on the order of years, mixed-tank models may be useful for time scales longer than a month. Although the time scale for sedimentation is long, sedimentation in lakes is recognized as an important mechanism for the removal of some elements, such as cesium. Stratification can limit the depth of effective mixing in both oceanic and Great Lakes models (Gloyna, 1977). Seasonal turnovers, upwelling, and other stratification phenomena can complicate the analysis.

Critical parameters for modeling coastal systems include (1) coastal current patterns with regard to winds and tides, (2) turbulent-transport coefficients, (3) diffusion-transport coefficients, and (4) sediment-transport coefficients. Parameters important for Great Lakes include (1) current fields, (2) turbulent-transport coefficients, (3) diffusion-transport coefficients, and (4) sediment-transport coefficients.

Two-dimensional solutions of the advection-diffusion equation were used to approximate the nearshore transport of radionuclides for the NRC and the Sandia liquid-pathway studies. Analytical solutions were obtained by assuming that the nearshore region has a constant depth d, a straight shoreline, and a constant alongshore water velocity. Both studies also used mixed-tank models to approximate the long-term concentration of contaminants in the Great Lakes. These models included the effects of radionuclide removal from the water column by interaction with sediments.

More details on applicable coastal and Great Lakes models are available in NRC Regulatory Guide 1.113 (USNRC, 1977). This document gives solutions of the advection-diffusion equation for steady-state and transient releases. Other models that can be applied to coastal systems include those by Onishi et al. (1976), Leendertse (1970), and Pagenkopf et al. (1976). These models are based on the numerical solution of a two-dimensional form of the advection-diffusion equation and are applicable to dynamic releases.

Impoundments

Impoundments include dammed rivers, small lakes, and offstream cooling ponds. In general, the present understanding of impoundment behavior is sufficient for liquid-pathways consequence analyses. Because of the likely dynamics of releases to impoundments, completely mixed tank models used with conservative assumptions can give concentrations adequate for dose calculations. Parameters that are important for impoundment modeling include (1) inflow and outflow rates, (2) sedimentation and sediment-transport processes, (3) mixing of stream inflows with the rest of the impoundment, (4) sorption and desorption mechanics, (5) diffusion coefficients, and (6) time scales for horizontal and vertical mixing.

Applicable mixed-tank models would include those derived for both the NRC and the Sandia liquid-pathway studies and those discussed in NRC Regulatory Guide 1.113. Compartment models (e.g., Helton and Kaestner, 1981) may be applicable to radionuclide transport in a series of surface-water systems. Such models treat individual areas (i.e., compartments) of the surface hydrosphere as perfectly mixed tanks. Radionuclides in different areas are placed in different compartments, and the radionuclide distribution that results from movements between these is then determined.

Groundwater

Numerous models, ranging from simple to complex, are available for evaluating the dispersion and transport of radionuclides in groundwater. The extent of complexity that is needed in the model will depend on the modeling situation. For example, if human exposure occurs directly from contaminated aquifer water, then the concentration of contaminants at the point of withdrawal is important. However, if exposure is from contaminated surface water, then the quantity of contaminants entering the surface-water body is relevant. In general, multidimensional models will be required for the former, while one-dimensional models will be adequate for the latter. Parameters that are important for calculating the dispersion of radionuclides in a groundwater aquifer include (1) the effective groundwater velocity, (2) diffusion coefficients, (3) retention of the radionuclides by the substratum under consideration, and (4) radionuclide decay.

Multidimensional point concentration and one-dimensional flux models were derived and used for both the NRC and the Sandia liquid-pathway studies (USNRC, 1978; Niemczyk et al., 1981). These models, which are based on the solution of the advection-diffusion equation, are applicable for most simple modeling situations. The transport of contaminants in groundwater is also treated in computer programs by Ahlstrom and Foote (1976), Reeves and Duguid (1975), and Campbell et al. (1980).

F3.2.2 Exposure Pathways to People

When predicting consequences associated with accidental releases of radionuclides, the analyst should consider the potential pathways by which radioactive material might move to people. In practice, it is generally found that the total dose for a given release will be dominated by a limited group of pathways. Most of the alternative pathways are found to be relatively insignificant (e.g., contaminated fishing gear). The dominant exposure pathways are generally expected to include (1) the ingestion of drinking water, (2) the ingestion of aquatic foods, (3) the ingestion of irrigated crops and related animal products, (4) external exposure from contaminated shorelines, and (5) external exposure from immersion in contaminated water.

Exposure-pathway models for estimating radiation doses largely follow from models developed for the evaluation of chronic releases (e.g., Ng et al., 1968; Soldat et al., 1973; and Lyon, 1976). The dose received by members of the population depends on the integrated pathway exposure. In the evaluation of chronic releases, the environment is usually assumed to be in equilibrium, with the period of exposure taken to be a year. The predicted dose rate (mrem/yr) can be compared to radiation guides. However, doses resulting from accidental releases will be determined for exposure periods that range from a few days to years. Pathway radionuclide concentrations will be time dependent because steady-state conditions may not be present. Therefore, doses should be computed as the time integral of the pathway-concentration functions.

The general expression used to calculate individual doses for both the NRC and the Sandia liquid-pathway studies can be written as

$$\text{dose}_p = U_p \sum_i d_{ip} f_{ip} \exp(-\lambda_i t_p) \int_0^t C_{ip}(t')\, dt'$$

where

t = the time after the accident.
p = the exposure pathway.
U_p = the individual usage rate.
d_{ip} = the dose factor for isotope i and pathway p.
f_{ip} = the removal or modification factor for isotope i and pathway p.
λ_i = the radioactive-decay constant for isotope i.
t_p = the delay time for pathway p.
$C_{ip}(t)$ = the concentration of isotope i in pathway p.

The factor $\exp(-\lambda_i t_p)$ accounts for radioactive decay during the period after removal from the water body and before exposure. However, all individuals affected by a given pathway do not receive the same exposure. Differences in exposure can be caused both by differences in individual usage and by spatial and temporal variations in radionuclide concentrations. Therefore, a pathway population can be regarded as being composed of subgroups differing either in usage patterns, in concentrations encountered, or in both. The general expression used to calculate population doses for both the NRC and the Sandia liquid-pathway studies can be written as

$$\text{dose}_p = \sum_{p'} P_{pp'} U_{p'} \sum_i d_{ip} f_{ip} \exp(-\lambda_i t_p) \int_0^t C_{ipp'}(t')\, dt'$$

where $P_{pp'}$ is the population size of subgroup p' in pathway p and all other symbols are as previously defined.

In general, the concentration function, C_{ip}, of radionuclide i in pathway p must be evaluated for each radionuclide released to the environment and for each pathway of concern. This function will in many cases be the water concentration or some fractional multiple of the water concentration (e.g., drinking water or immersion pathways). In other cases, this function can depend on the water concentration and other physical and biological processes (e.g., aquatic food, irrigated-food, and shoreline-exposure pathways). These latter three concentrations can be treated with models ranging from simple to complex. For example, the concentrations of radionuclides in aquatic organisms can be treated with various levels of models (USNRC, 1978; Niemczyk, 1980; Niemczyk et al., 1981; Marietta et al., 1980). Complex models can generally provide better approximations of the pathway concentrations; however, they are limited in use and capabilities by the large quantities of site-specific data that are needed. Simple models can provide adequate approximations for many applications of liquid-pathway modeling, especially for slowly changing water concentrations of contaminants.

F3.2.3 Dosimetry and Health Effects

The dosimetry and health-effects models that are used for the atmospheric consequence analysis should also be used for the liquid-pathways consequence analysis. Using the same models will facilitate the comparison of airborne- and water-pathway consequences. Section 9.3 discusses the models that are applicable for the atmospheric pathway.

F3.3 GATHERING AND PROCESSING DATA

It is generally the analyst's responsibility to collect and process data in some or all of the following areas:

1. Radionuclide release data (e.g., magnitudes, durations, rates, probabilities).

2. Transport data (e.g., diffusion coefficients, distribution coefficients for suspended and bottom sediments, flow rates).

3. Population and exposure-pathway data (e.g., pathway populations, rates of drinking-water consumption, water-use characteristics, radionuclide-concentration factors for aquatic foods).

4. Dosimetry and health-effects data.

5. Economic data.

6. Interdiction criteria.

In general, the selection of input data can significantly influence whether the consequence-analysis results are meaningful or not.

Radionuclide-transport and exposure data will vary from one hydrospheric system to another, and within a hydrospheric system, for various reasons: (1) differences in hydrosphere type and characteristics (e.g., small streams versus large rivers); (2) varying water-use characteristics; (3) differences in aquatic biota; (4) differences in the physical and chemical characteristics of soils and sediments; and (5) differences in such water characteristics as temperature, chemistry, and salinity (Yousef et al., 1970). Because of these variations, a liquid-pathway consequence analysis should use data that are specific to, or are representative of, the hydrosphere under consideration. Detailed discussions of the types of data that would be needed for a liquid-pathway study are contained in both the NRC and the Sandia liquid-pathway reports (USNRC, 1978; Niemczyk et al., 1981). More important, these reports contain references to sources and bodies of data that can be used for site-specific risk evaluations. A source of detailed absorption/desorption data is the report by Onishi et al. (1981).

F3.4 USE OF MODELS AND INTERPRETATION OF RESULTS

The initial objective of a liquid-pathway consequence analysis is to determine whether or not the liquid pathway is important for reactor-risk calculations. This could be accomplished by initially performing a conservative bounding analysis. Dose-mitigating measures would not be assumed for such an analysis.

Sensitivity analyses should be performed for the most important hydrologic parameters to determine their effect on consequence predictions. If it is decided that the bounded risks (i.e., latent somatic risks like latent cancers and thyroid nodules) are not significant in comparison with those expected from the atmospheric pathway, then the liquid-pathway consequence analysis is essentially finished.

However, if the liquid pathway is found to be important for reactor risk, then additional analyses should be performed. The impact of dose-mitigating actions on the consequences should be determined. Such actions would include the interdiction of the pathway (e.g., confiscation of contaminated foodstuffs) and isolation of the pathway source (e.g., constructing a grout curtain around the melted reactor core). Since these actions can result in large societal and economic costs, liquid-pathway costs should also be evaluated and possibly compared with those resulting from the atmospheric pathway.

F3.5 DOSE-MITIGATING ACTIONS

Dose-mitigating actions can be employed either close to the accident site or along the various exposure pathways farther away. Actions of the former type are called source-interdiction procedures and are possible only for direct releases to an aquifer and other subsurface water bodies. Actions of the latter type are called pathway-interdiction procedures and are generally the only recourse if source-interdiction procedures are not possible. In principle, a complete elimination of all biological consequences is possible by successfully implementing source- and pathway-interdiction measures. However, the potential benefits would need to be weighed against the possible costs: disruption of people's lives (e.g., loss of livelihood or recreational facilities); disturbance of the ecosystem, which might not be too adversely affected by the radiation itself; and possibly substantial monetary costs.

F3.5.1 Source Interdiction

Source-interdiction measures would likely be feasible only for a reactor core melt, although there is some possibility that such procedures could also work for isolating sumpwater and depressurization releases. The feasibility of isolating "prompt" releases (i.e., sumpwater and depressurization releases) is dependent on the characteristics of the stratum under the containment.

Possible methods of containing radioactive contaminants within the reactor area after a core-melt accident include the following:

1. Injection or withdrawal of water.
2. Lowering of the water table.
3. Installation of a grout curtain.
4. Installation of a slurry wall.

The first three methods were suggested during the NRC liquid-pathway study (USNRC, 1978) and were considered further by Niemczyk et al. (1981). The fourth method is discussed by Harris et al. (1981).

Wells can be drilled in the area directly surrounding the contaminated source, either to withdraw contaminated water for treatment or to inject uncontaminated water as a barrier. In the former method, the contaminated groundwater can be isolated by drilling a number of wells across the hydraulic gradient at positions downgradient from the radioactive source. These wells can then be used for withdrawing the contaminated water for treatment and eventual disposal or reinjection into the aquifer. In the latter method, the injection of uncontaminated water into wells drilled across the aquifer gradient can stop or reverse localized groundwater flow. Both of these measures are illustrated in Figures F-1 and F-2. Such methods could take months to implement and would generally be considered to be short-term methods (Niemczyk et al., 1981).

Pumping can be used to lower the water table in the immediate area of the reactor and therefore could be an effective interdiction method for core-melt leaching. If the water table can be sufficiently lowered, then leaching of the melted core could be eliminated. Figure F-3 illustrates how such a method might work. These measures could be implemented in a few months (Niemczyk et al., 1981). The advantages of drawdown pumping are that such a procedure would allow additional time for (1) a more permanent solution (e.g., grouting) to be carried out and (2) decay to reduce the radioactive content of the material. However, water pumped from the wells could be contaminated and therefore may need treatment.

A waterproof barrier might be formed by surrounding the melt debris with grouting. Grouting materials would be injected into the adjacent rock or soil to seal all voids, cracks, and seams. The total time required for grouting procedures is estimated to range from 15 months to 3 years (Niemczyk et al., 1981). Figure F-4 shows a possible grouting configuration for a core melt. The installation of a grout curtain would be

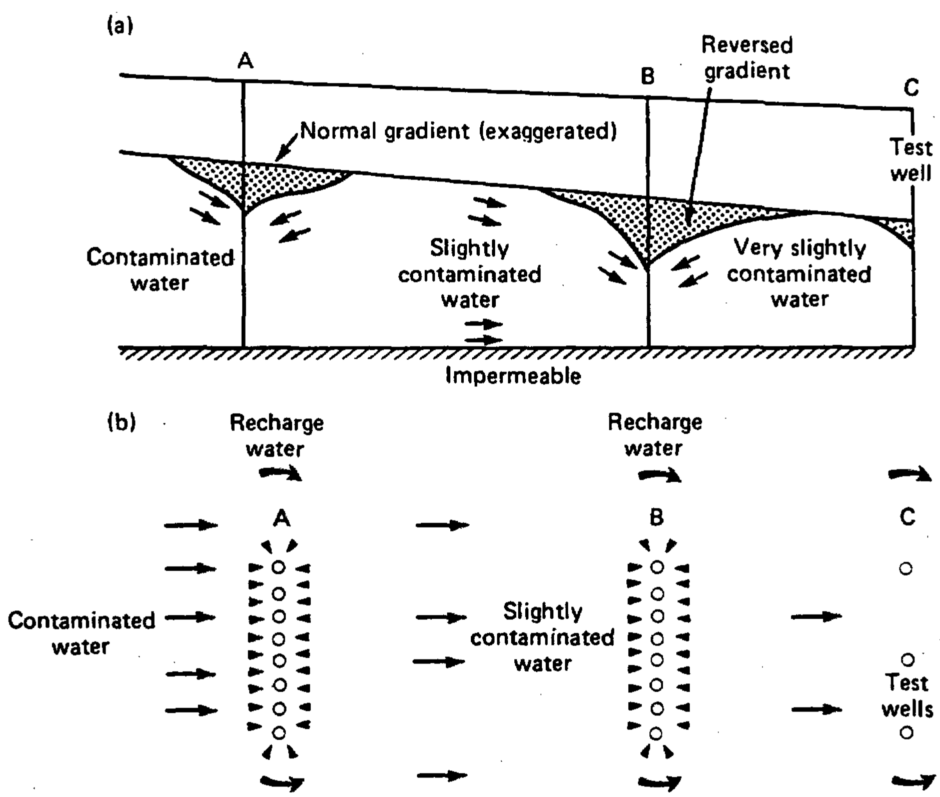

Figure F-1. Pumping wells downgradient from the source of contamination. From Niemczyk et al. (1981).

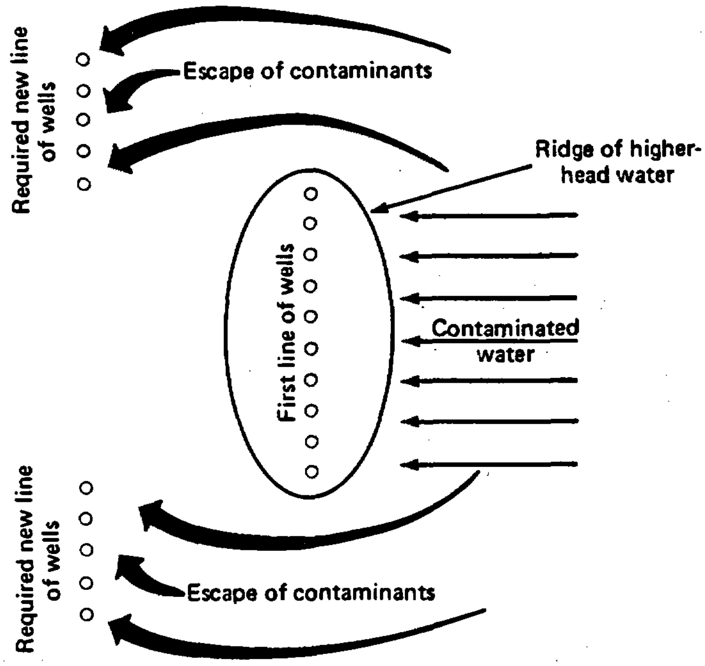

Figure F-2. Installation of a recharging field in an aquifer without lateral confinement. From Niemczyk et al. (1981).

considered as being a long-term solution and, in general, would nearly always be possible. However, if the task consumed too much time, then some release might occur before the job is finished. In addition, dewatering of the isolated area might be needed to account for water seepage.

A properly constructed slurry wall could provide a continuous low-permeability barrier around the melt debris. The technique would involve continuous excavation of a trench to tie-in strata. A slurry, consisting primarily of bentonite clay and water, would be used to maintain the trench open with vertical sides, even below the water table. The trench would be backfilled with either soil mixed with bentonite slurry or with a cement-bentonite mixture. As with the grout curtain, dewatering of the isolated area might be needed.

F3.5.2 Pathway Interdiction

Given that source-interdiction measures are not feasible or successful, the only recourse will be pathway interdiction. The specific pathway measures employed at a given site would depend on both the characteristics of the affected water body and the sizes of the populations at hazard.

The procedures for pathway interdiction can be divided into two types (USNRC, 1978; Niemczyk et al., 1981): those involving interruption of the flow of contaminants along a pathway and those involving the removal of contaminants from the environment. The interruptive procedures include the confiscation of polluted food, substitution or treatment of drinking water,

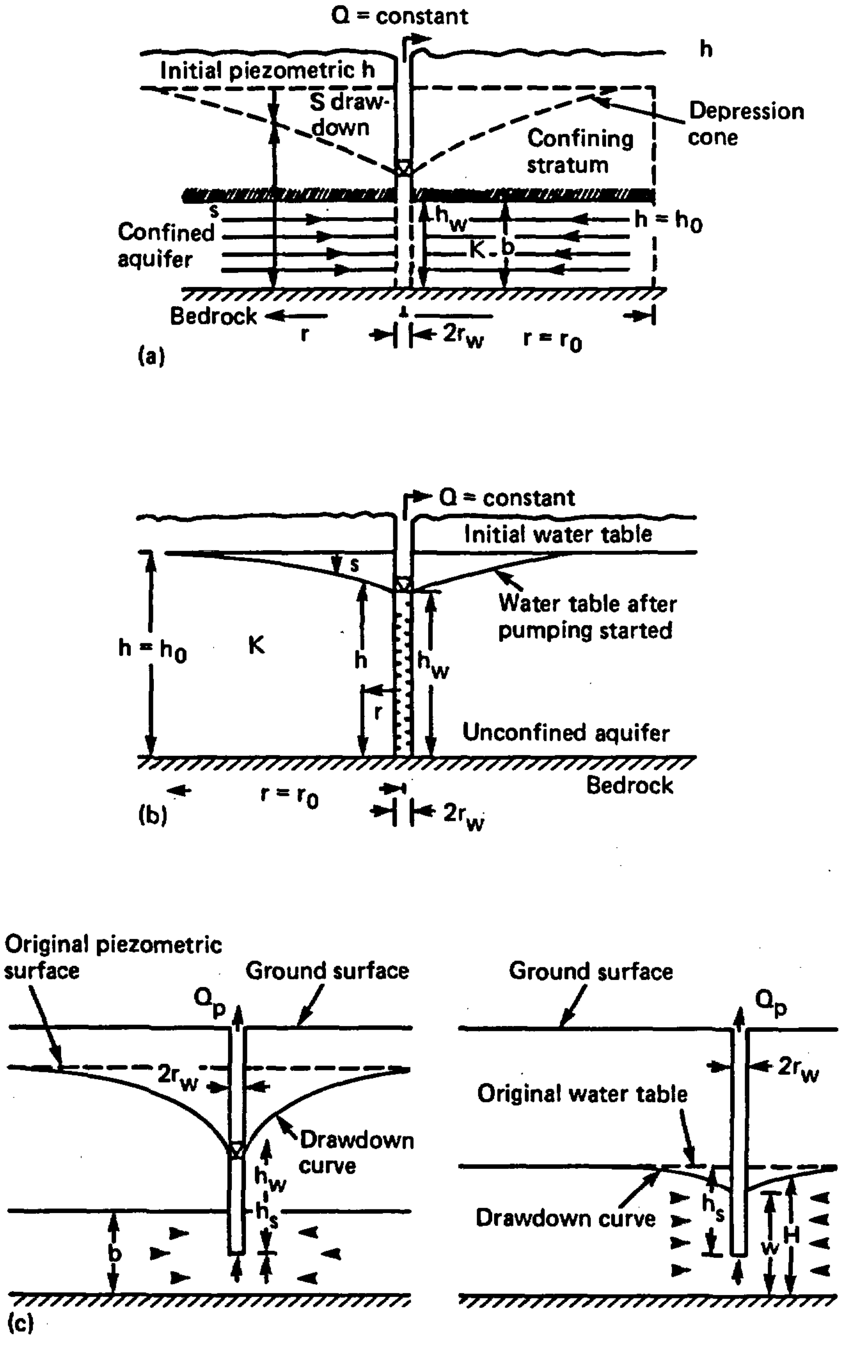

Figure F-3. **Radial flow to (a) a well completely penetrating a confined aquifer, (b) a well completely penetrating an unconfined aquifer, and (c) an incompletely penetrating well in a confined (left) and unconfined (right) aquifer. From Niemczyk et al. (1981).**

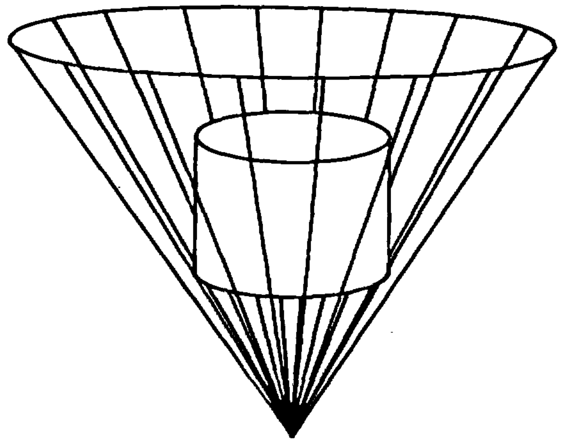

Figure F-4. Possible grouting configuration for the melt. From Niemczyk et al. (1981).

and denial of access to polluted areas. Restorative procedures include the treatment of polluted water in a water body, the dredging of contaminated sediment, and the decontamination of land. Monitoring would be an integral part of all pathway-interdiction procedures.

The feasibility of pathway interdiction at any site depends not only on the pertinent dominant pathways but also on the scale of the overall effort. The characteristics of the contaminated water bodies that determine the effectiveness of pathway interdiction include type of water, flushing times, and sedimentation properties. In general, the magnitude of effort would be much less for interruptive procedures than for restorative ones.

The costs of pathway interdiction can be either direct or indirect. Direct costs would include both monetary outlays and adverse social impacts. Examples include (1) the outlay for pathway monitoring; (2) the outlay for interrupting a given pathway; (3) the value of the output of interrupted industries; (4) the outlay for providing alternative sources of supply (e.g., drinking water); and (5) the loss of jobs in severely impacted industries. Indirect costs can also be either economic or social. For example, the disruption of an industry in one sector of the economy can generate indirect losses in another.

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Appendix G

Radionuclide Releases to the Ground: Treatment in the Reactor Safety Study

In degraded-core accidents, radionuclides can be released either to the air or to the ground. Methods for evaluating releases to the air are described in Chapter 8. The state of the art in evaluating releases to the ground is not as far advanced. This appendix summarizes the treatment used in the Reactor Safety Study.*

There are three ways in which radionuclides can be released to the ground or the groundwater during an accident in a light-water reactor:

1. Leaching of radionuclides from the core melt by groundwater after it has penetrated the concrete basemat.

2. Spillage of contaminated plant water. The water could come from the primary system or from containment sprays. Spillage may occur through any suitable opening in the containment--for example, through the core-melt hole in the concrete basemat. Surface water could also be contaminated by water spills from the containment.

3. Depressurization of the containment through the core-melt hole in the concrete basemat. Any radionuclides entrained in the reactor atmosphere will be carried into the ground and groundwater.

The Reactor Safety Study considered two different ground-release cases: (1) early radionuclide release by depressurization through the concrete basemat and (2) a delayed release by leaching from the solidified core mass. For the depressurization release, the analysis conservatively assumed that all the radionuclides dissolved rapidly and completely in groundwater.

For each radionuclide, the fraction of the reactor-core inventory released to the groundwater by depressurization was calculated by using depressurization release fractions from the CORRAL code. Release rates to the groundwater were then calculated. In the case of the leach release, the radionuclide inventory in the core mass 1 year after core meltdown was used as a basis for the calculations. It was assumed that all isotopes of the noble gases, halogens, alkali metals, and elements of the tellurium group had already been removed from the core mass by other processes. An empirical leach-rate expression was used to evaluate elution curves for several of the more important radionuclides.

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*U.S. Nuclear Regulatory Commission, 1975. Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400 (NUREG-75/014), Washington, D.C.

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This procedures guide describes methods for performing probabilistic risk assessments (PRAs) for nuclear power plants at three levels of scope: (1) systems analysis; (2) systems and containment analysis; and (3) systems, containment, and consequence analysis. After reviewing its objectives and limitations, this document describes the organization and management of a PRA project and then presents procedures for accident-sequence definition and systems modeling, human-reliability analysis, the development of a data base, and the quantification of accident sequences. Procedures for evaluating the physical processes of core meltdown are presented next, followed by guidance on the evaluation of radionuclide releases from the containment as well as the analysis of environmental transport and offsite consequences. The analysis of external hazards is discussed next, including procedures for seismic, fire, and flood analyses. The guide concludes with suggestions for the development and interpretation of results and the performance of uncertainty analyses.

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