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 decision-making 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		Major assumptions	Impact	Remarks
Segment	Task			

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

$$\text{var}(x) = \int_R f(x)(x - \mu)^2 \, dx \qquad (12\text{-}1)$$

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

$$s = [\text{var}(x)]^{1/2} \tag{12-3}$$

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

$$\langle P_{i1}, x_{i1} \rangle \langle P_{i2}, x_{i2} \rangle \ldots \langle P_{ij}, x_{ij} \rangle \ldots \langle P_{im}, x_{im} \rangle \quad (i = 1, \ldots, n;\ j = 1, \ldots, m)$$

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

$$z = \phi(x_1, x_2, \ldots, x_i, \ldots, x_n) \qquad (12\text{-}6)$$

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

$$E[z] = \mu_z = \int x_1 \int x_2 \cdots \int x_n \, \phi(x_1,x_2,\ldots,x_n)$$

$$f(x_1,x_2,\ldots x_n)\, dx_1\, dx_2 \ldots dx_n \tag{12-8}$$

$$V(z) = \sigma_z^2 = E[z^2] - \mu_z^2 \tag{12-9}$$

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

$$\mu_z \cong \bar{z} + \frac{1}{2} \sum_{i=1}^{n} c_{ii} \sigma_i^2 + \sum_{i=1}^{n-1} \sum_{j=i+1}^{n} c_{ij} \, \mathrm{cov}(x_i, x_j)$$

and

$$V(z) \cong \sum_i \sum_j b_i b_j \, \mathrm{cov}(x_i, x_j)$$

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. Lognormal. 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

$$\text{var}(Q^*) = \sum h_i^2 \, \text{var}(\theta_i^*) + \sum_i \sum_j h_i h_j \, \text{cov}(\theta_i^*, \theta_j^*)$$

where $\text{cov}(\theta_i^*, \theta_j^*)$ 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

$$\text{var}(Q^*) = \sum_{i=1}^{k} (h_i^*)^2 \, \text{var}(\theta_i^*)$$

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 $\text{var}(\theta_i^*)$. 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

$$T^* = \frac{\lambda_E}{\text{var}^*(\lambda_E^*)}$$

$$f^* = T^* \lambda_E^*$$

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

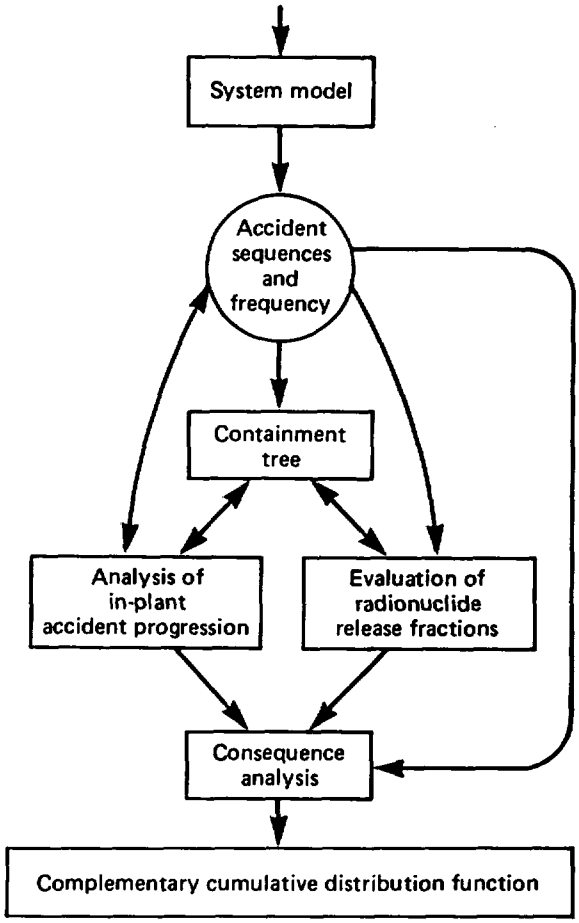


Figure 12-3. PRA information flow.

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,...,n), the overall uncertainty factor UF and the median M are given by

$$UF = \exp\left\{\left[\sum_{i=1}^{n} (\ell n\ E_i)^2\right]^{1/2}\right\}$$

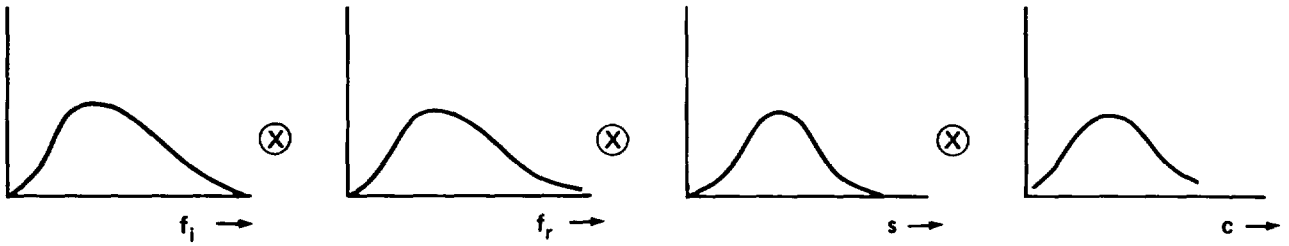
$$M = \prod_{i=1}^{n} m_i$$

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.

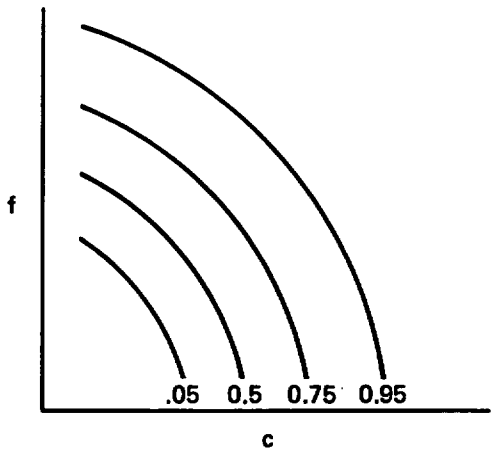

Figure 12-5. Display of uncertainties in a complementary cumulative distribution function (an f/c curve, where f is frequency and c is consequence).

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.

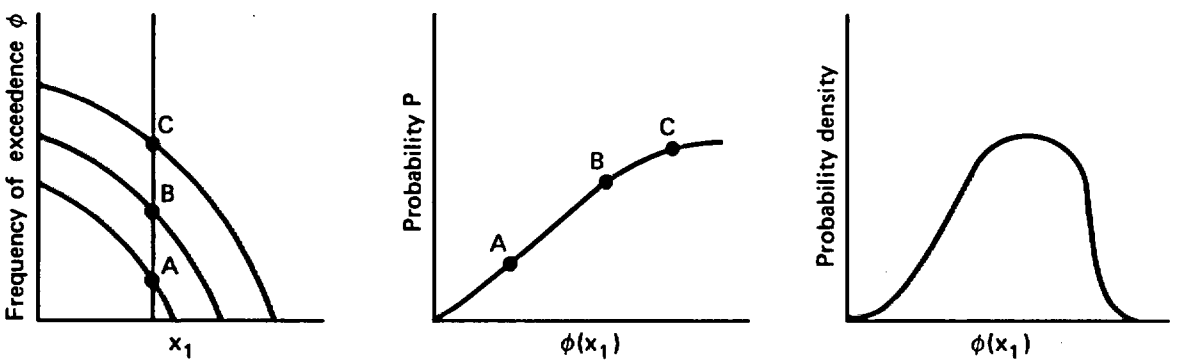

Figure 12-6. Development of cut curves from a family of risk curves.

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]

	Initiating event	Plant event sequence category[c,d]																			
		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 x 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}

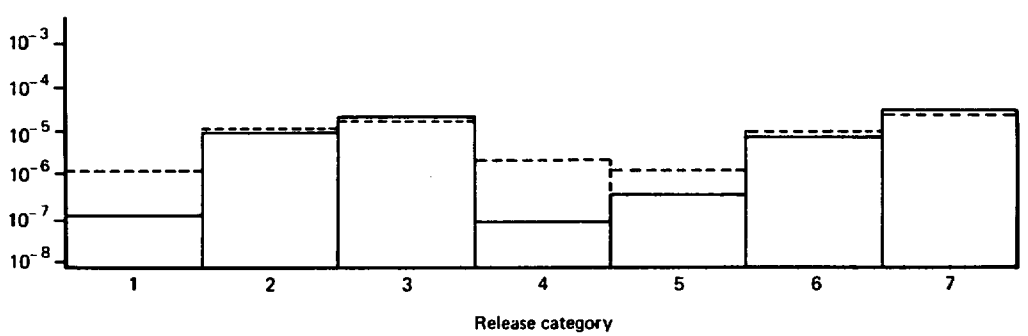

Figure 13-1. Dominant accident sequences with histogram. The category totals given in the tabulation are unsmoothed totals that include the contribution from all the nondominant sequences not shown. From Kolb et al. (1981).

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

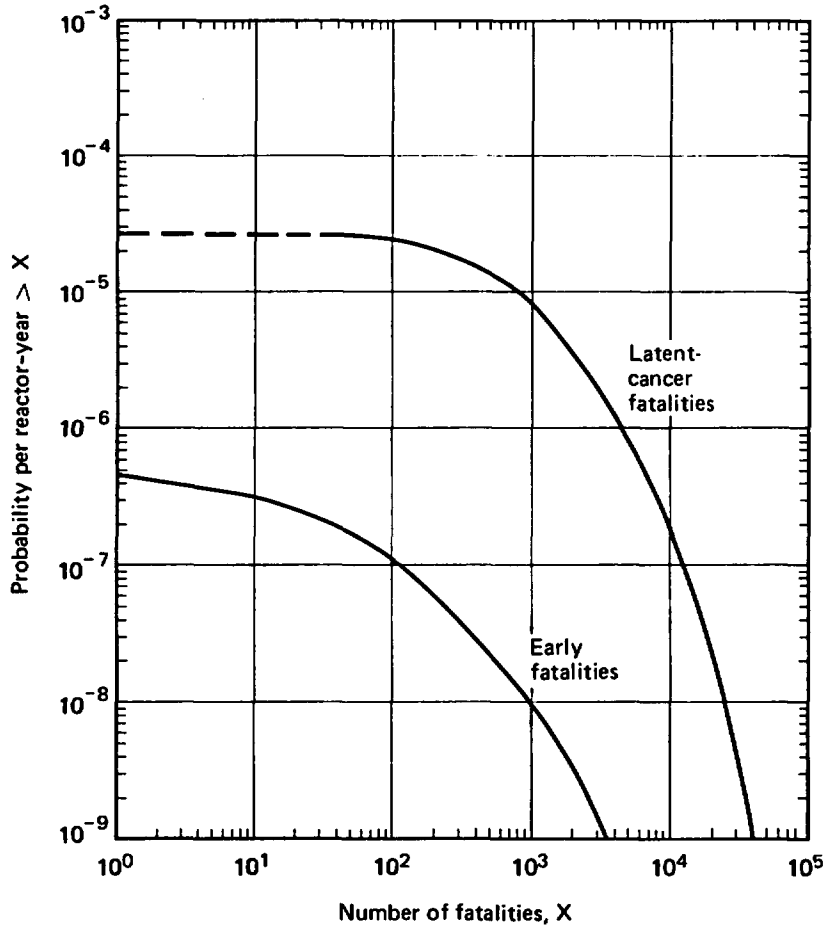

Figure 13-2. Probability distribution for early fatalities and latent-cancer fatalities. From the Reactor Safety Study (USNRC, 1975).

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.

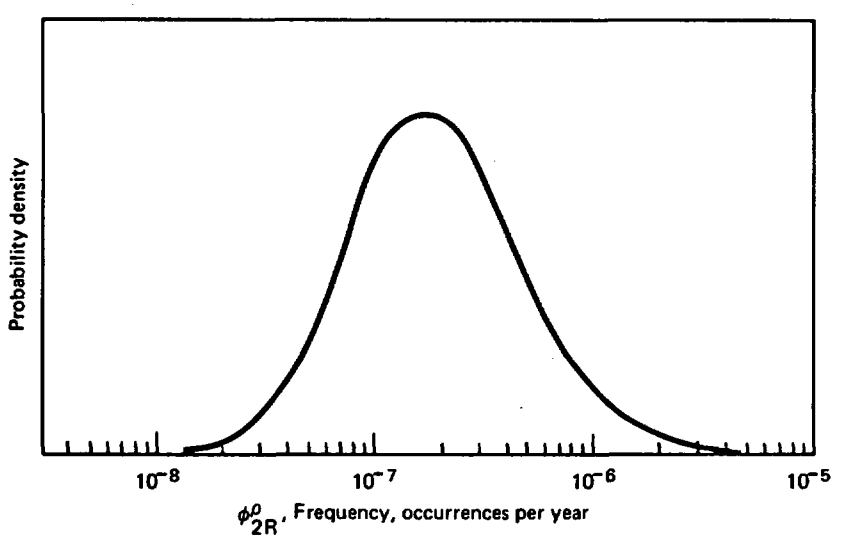

Figure 13-3. Probability distribution for the frequency of release category 2R (internal events only). From the Zion PRA (Commonwealth Edison Company, 1981).

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.

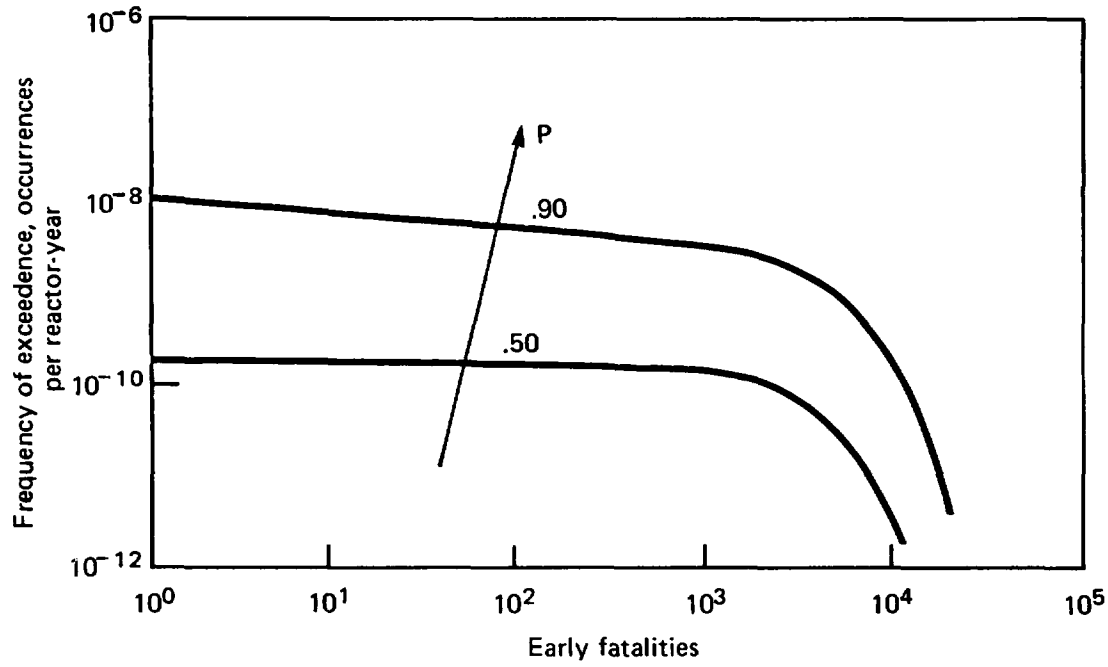

Figure 13-4. Level 2 risk diagram for fatalities: base case, internal and external risk. From the Zion PRA (Commonwealth Edison Company, 1981).

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
00E-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

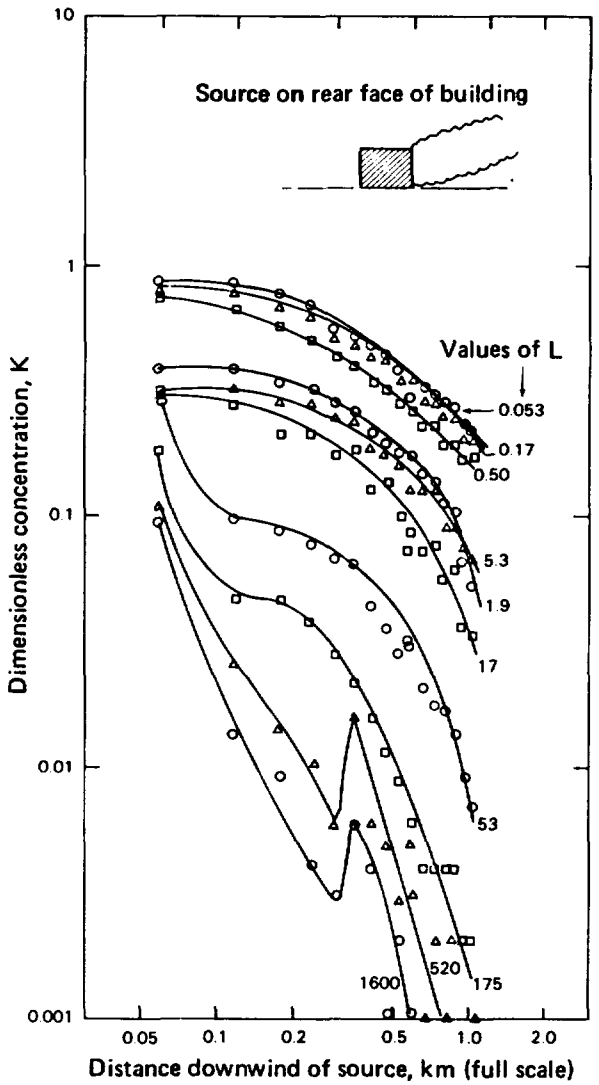

Figure D-1. Plume centerline ground-level concentrations downwind of source. From Hall et al. (1980).

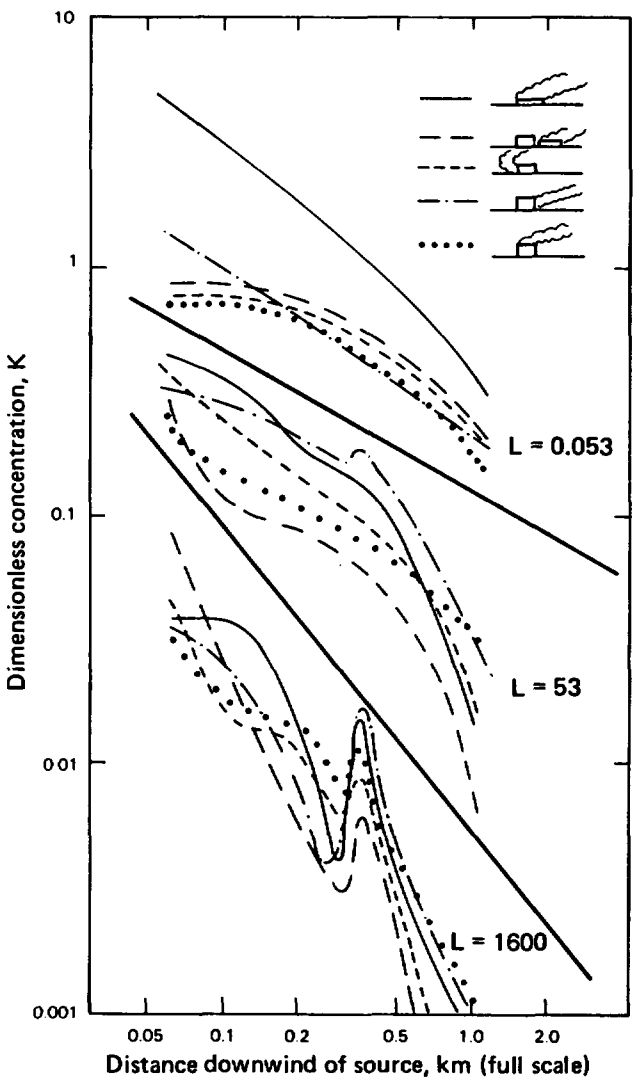

Figure D-2. Plume centerline ground-level concentrations downwind of building for different release conditions and buoyant levels. From Hall et al. (1980).

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

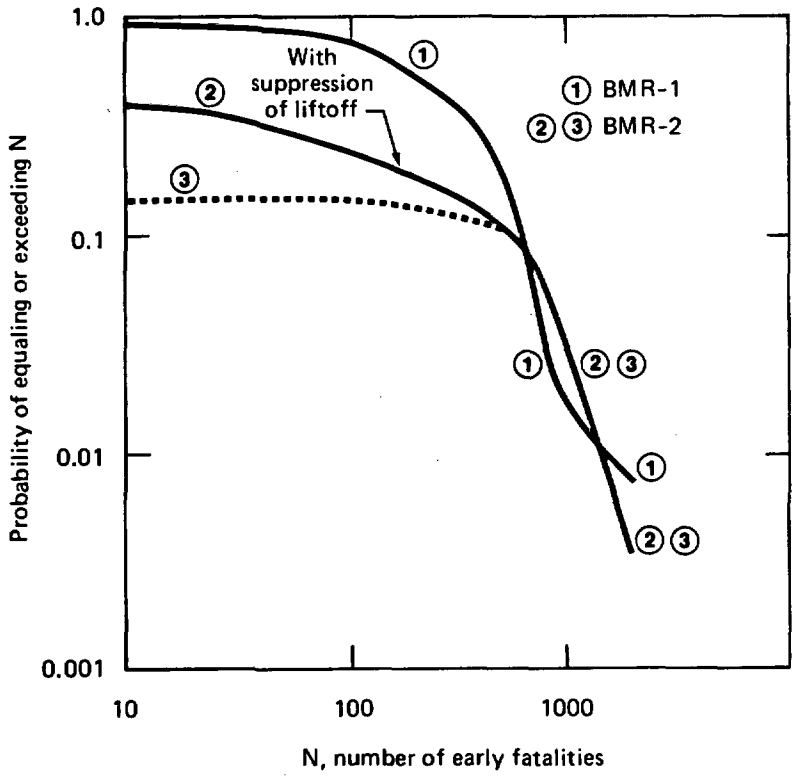

Figure D-3. Complementary cumulative distribution function conditional on BMR-1 or BMR-2 release (Benchmark exercise): uniform population, early fatalities.

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

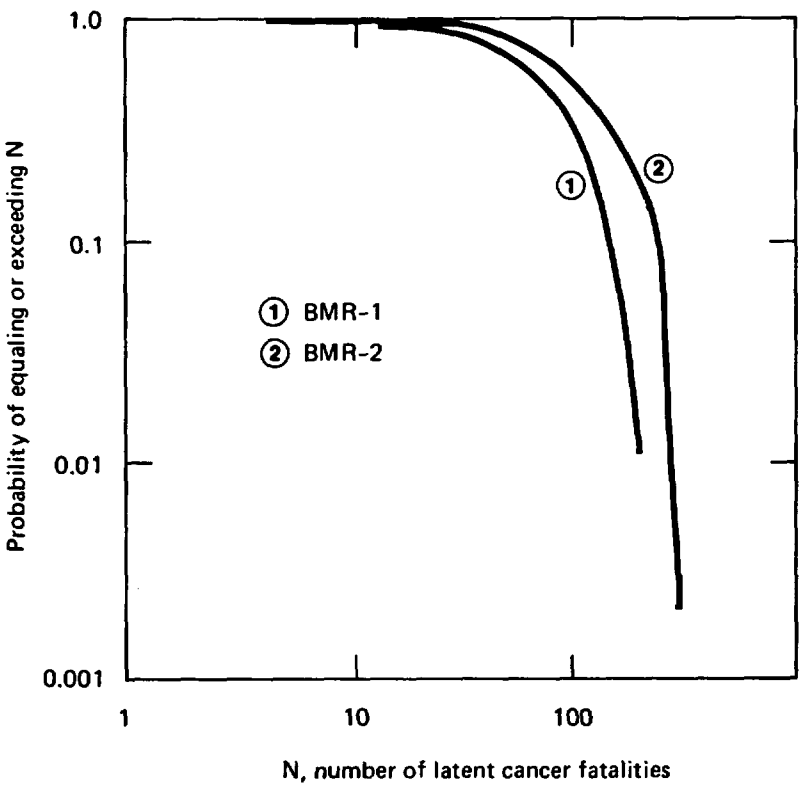

Figure D-4. Complementary cumulative distribution function conditional on BMR-1 or BMR-2 release (Benchmark exercise): uniform population, latent cancer fatalities.

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

$$\chi_D(x,y) = v_d\chi(x,y,0) \quad Ci/m^2 \qquad (D-5)$$

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.

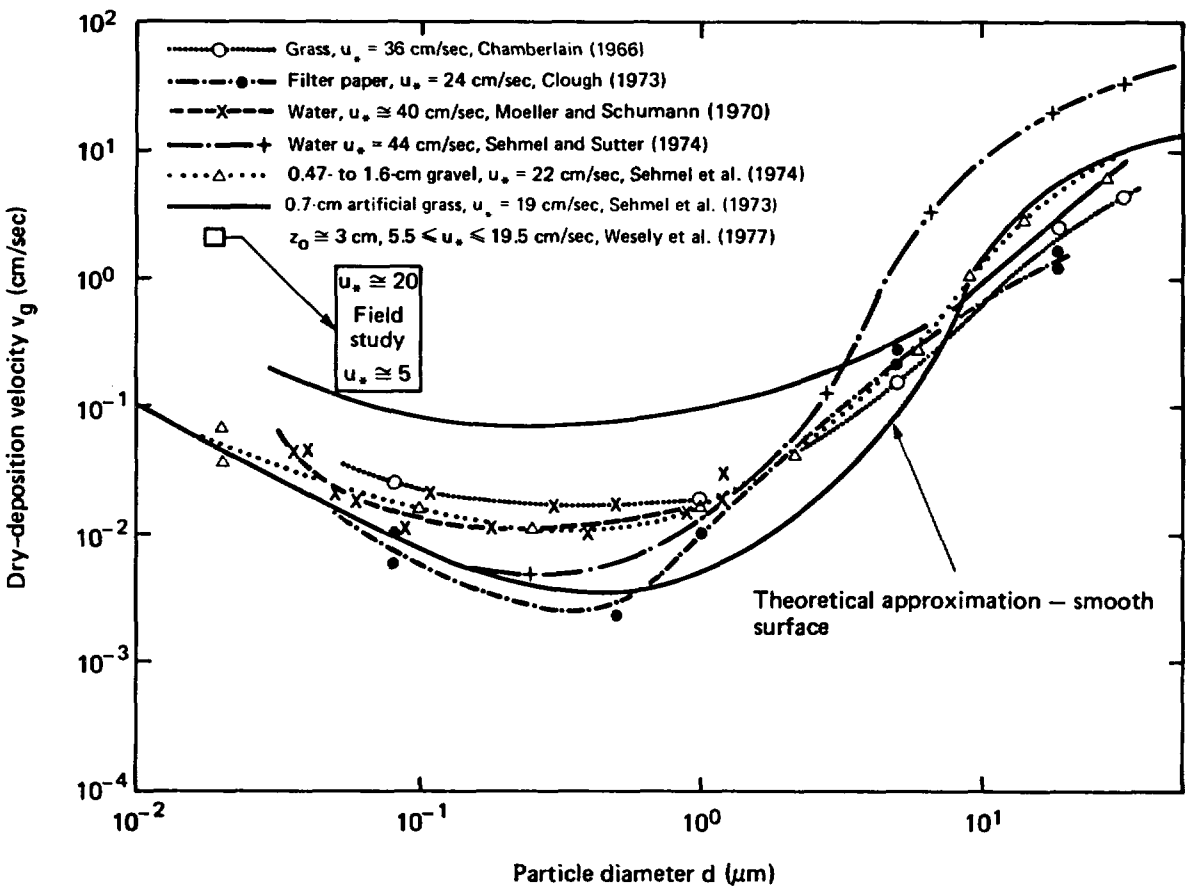

Figure D-5. Dry-deposition velocity as a function of particle size for several substrates, from experimental data reported by various authors. From Slinn (1978).

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

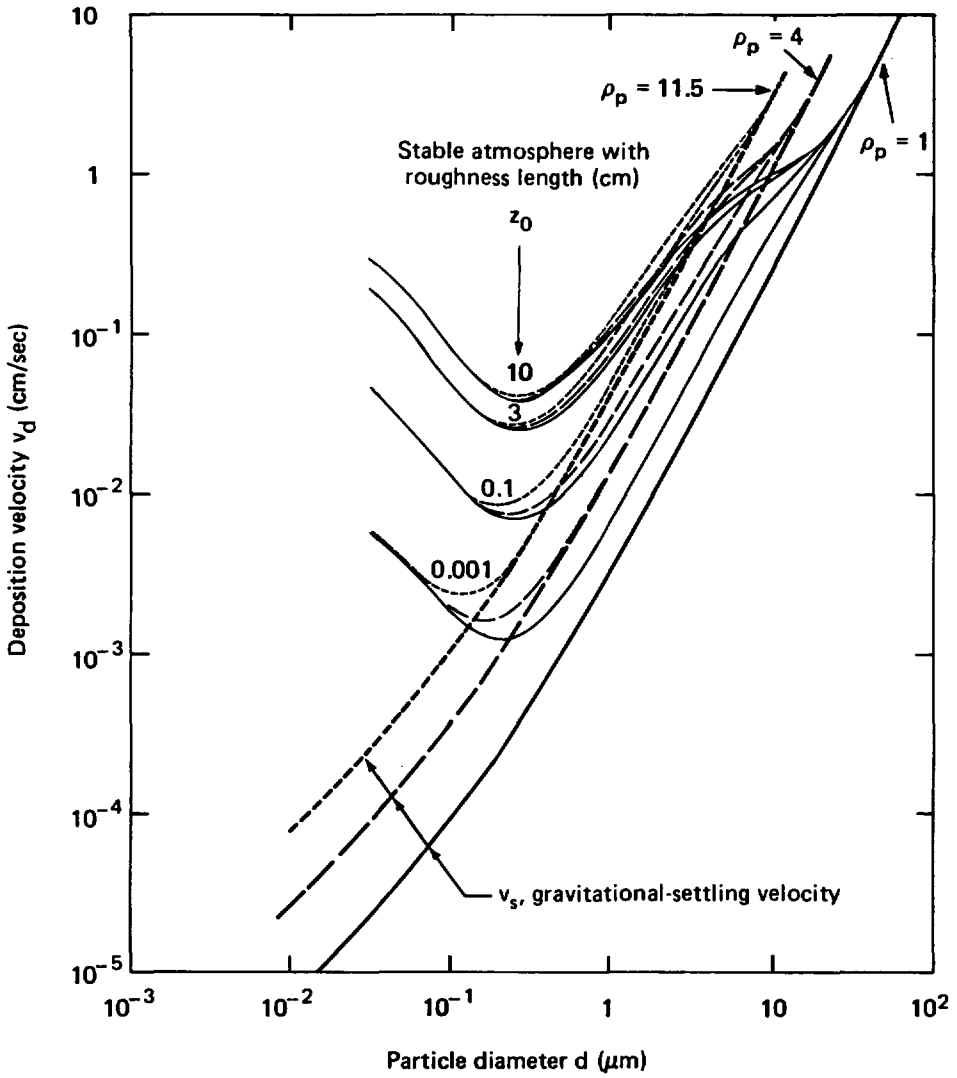

Figure D-6. Effect of the meteorological roughness length z_0 and particle density ρ_p on deposition velocity. From Sehmel (1980).

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).

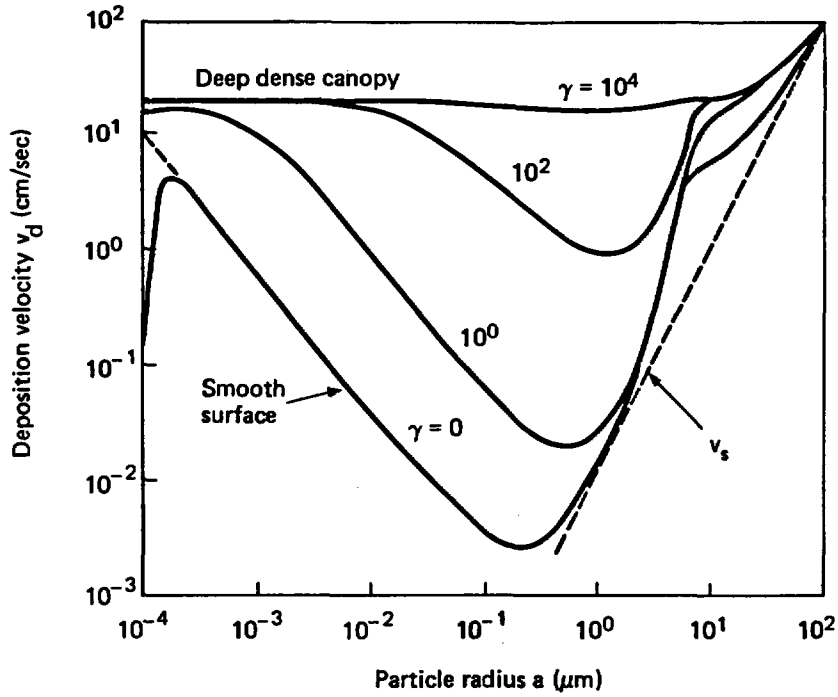

Figure D-7. Deposition velocity as a function of particle radius for a smooth surface and ground covered with a deep dense canopy of vegetation. From Slinn (1977).

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

$$\frac{Q(x)}{Q} = \exp\left\{\left(\frac{2}{\pi}\right)^{1/2} \frac{v_d}{\bar{u}} \int_0^x \frac{dx'}{\sigma_z(x')} \exp\left[-\frac{h^2}{2\sigma_z^2(x')}\right]\right\} \qquad \text{(D-7)}$$

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

$$\bar{z}_i = \left(\frac{\pi}{2}\right)^{1/2} \sigma_z \exp\left(\frac{h^2}{2\sigma_z^2}\right) \qquad \text{(D-8)}$$

is constant, and $\bar{z}_i$ is evaluated at the midpoint of the interval. Equation D-7 then becomes

$$\frac{Q(x_{i+1})}{Q(x_i)} = \exp\left(\frac{v_d t_i}{\bar{z}_i}\right) \qquad \text{(D-9)}$$

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

$$\frac{Q(x_{i+1})}{Q(x_i)} = 1 - \frac{v_d t_i}{\bar{z}_i} \qquad \text{(D-10)}$$

and

$$\frac{Q(x_i)}{Q} = \prod_{j=1}^{i} \left(1 - \frac{v_d t_j}{\bar{z}_j}\right) \qquad \text{(D-11)}$$

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, $\exp[-(z + h)^2/2\sigma_z^2(x)]$ 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

$$\left(K_z \frac{\partial \chi}{\partial z}\right)_{z=z_g} = v_d \chi \qquad \text{(D-14)}$$

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.

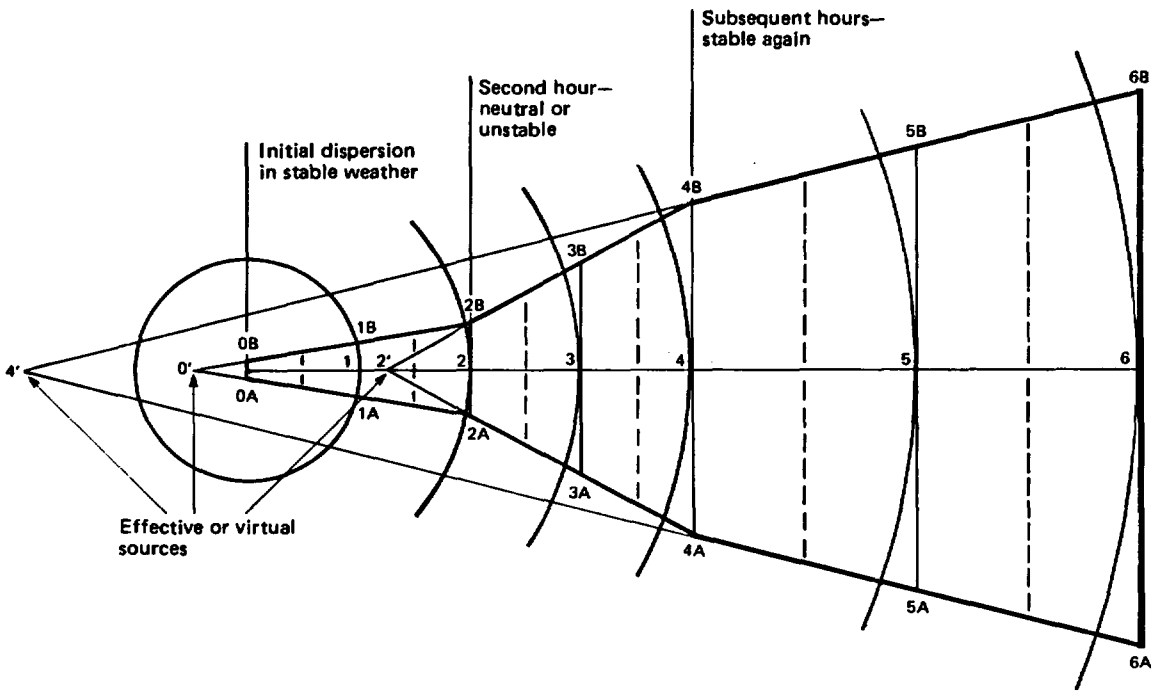

Figure D-8. Plan view of the growth of plume width with changing stability conditions. From a draft of the CRAC user's manual.

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).

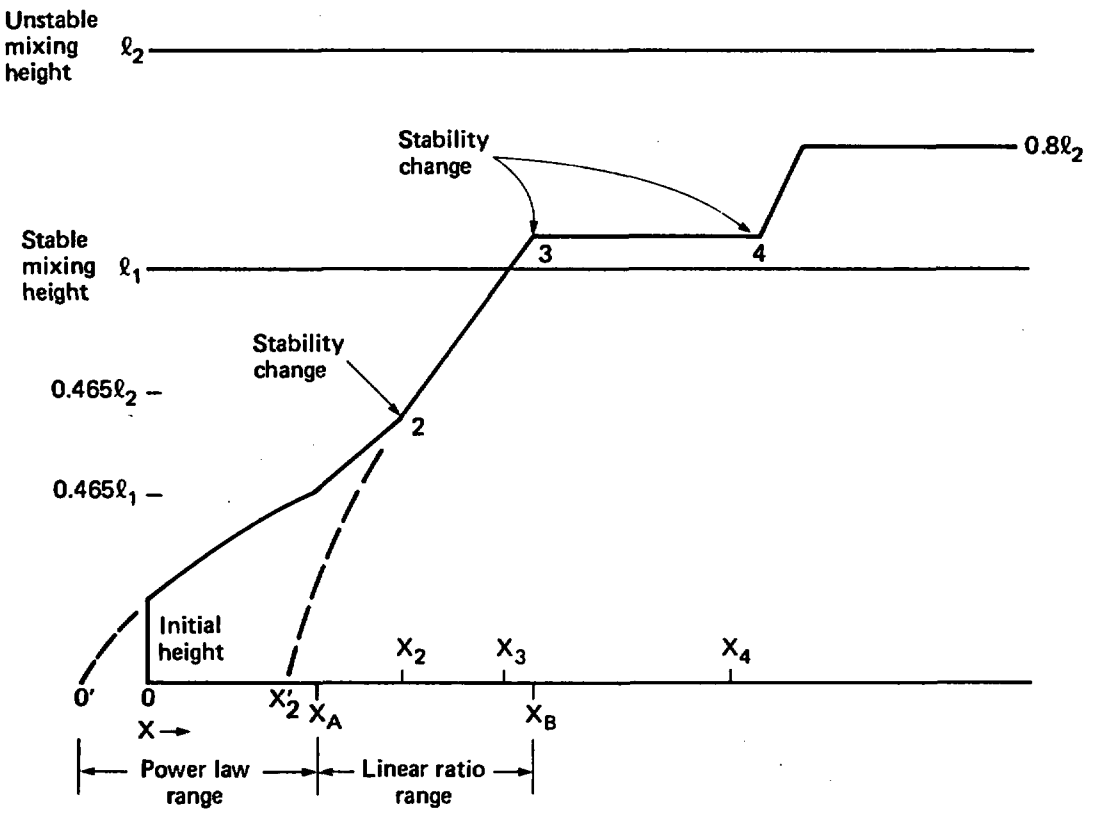

Figure D-9. Side view of the vertical growth of a plume. From a draft of the CRAC user's manual.

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).

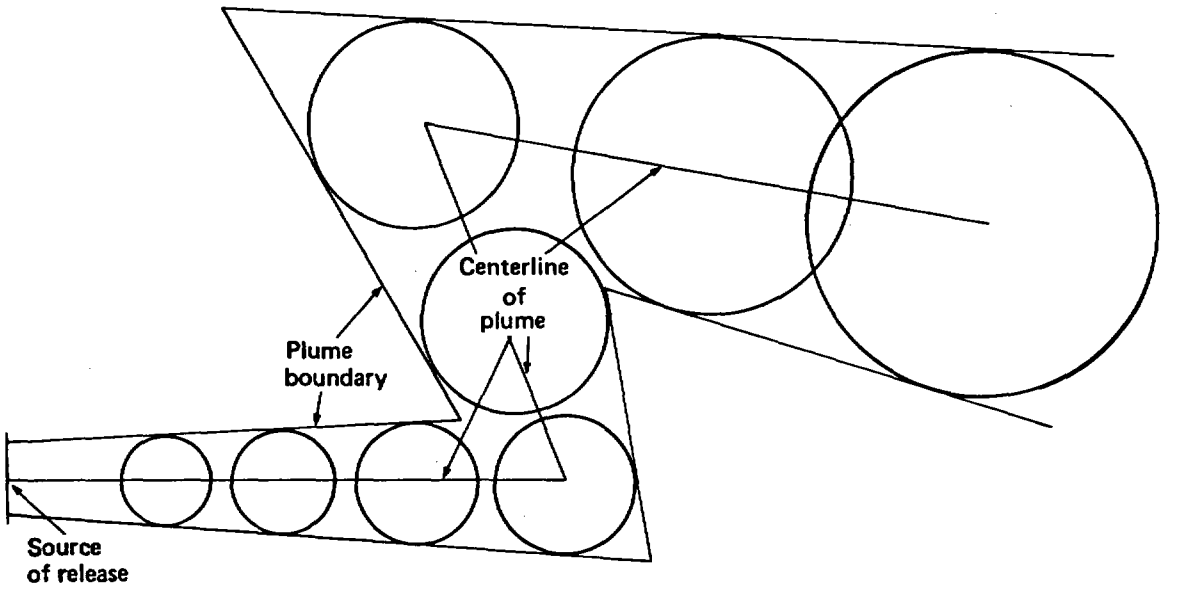

Figure D-10. Representation of plume: the growth of the puff as wind direction changes twice. From the Commonwealth Edison Company (1981).

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.

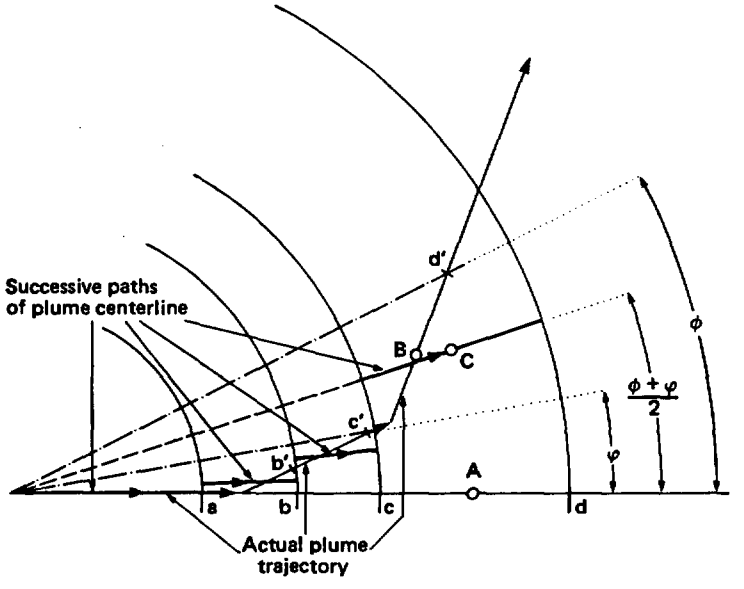

Figure D-11. Schematic showing the calculation of rotation angle for wind-shift model. From Aldrich et al. (1979).

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

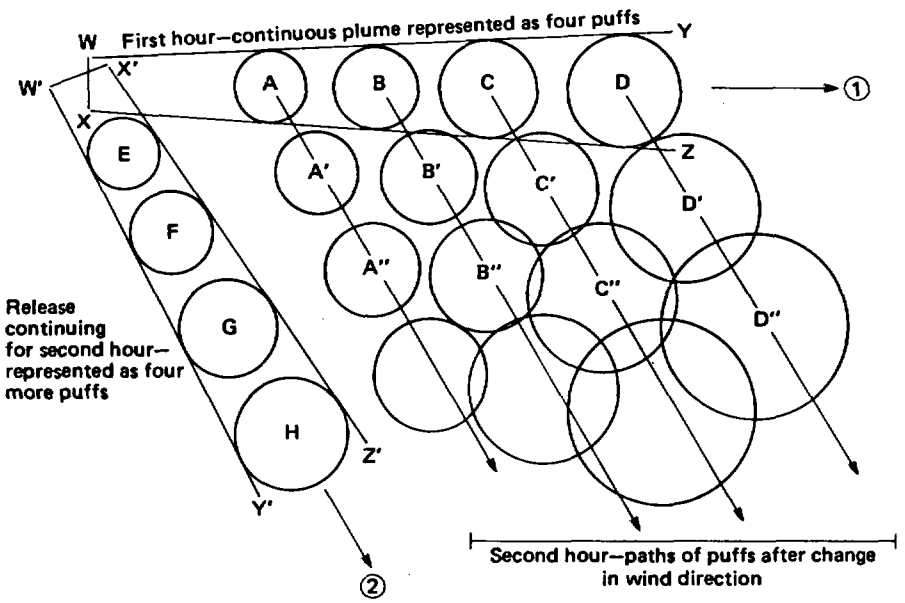

Figure D-12. Continuous plume represented as a series of puffs.

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.

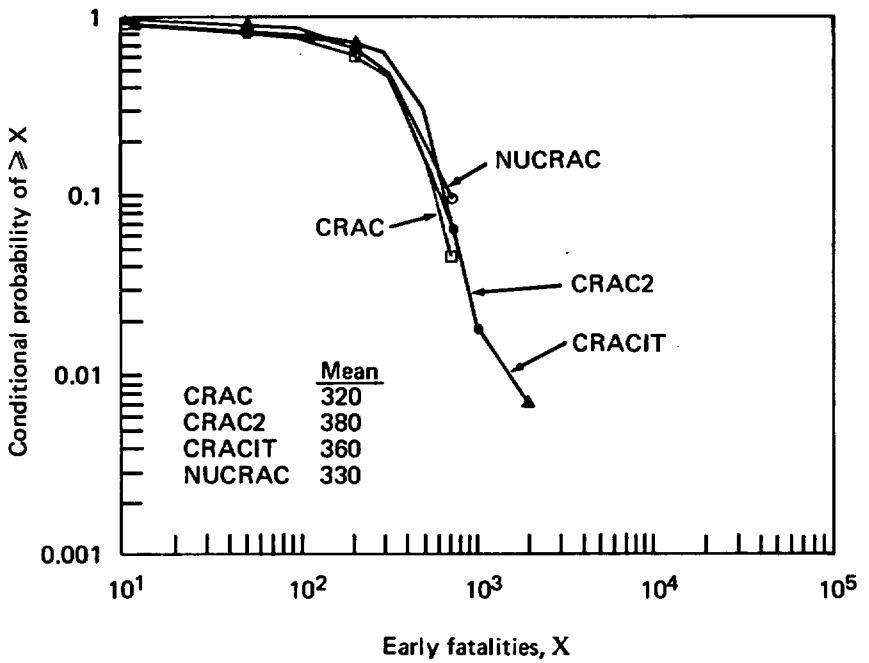

Figure D-13. Early-fatality CCDF, river-valley site. Uniform population, no emergency response. From Aldrich et al. (1981b).

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.

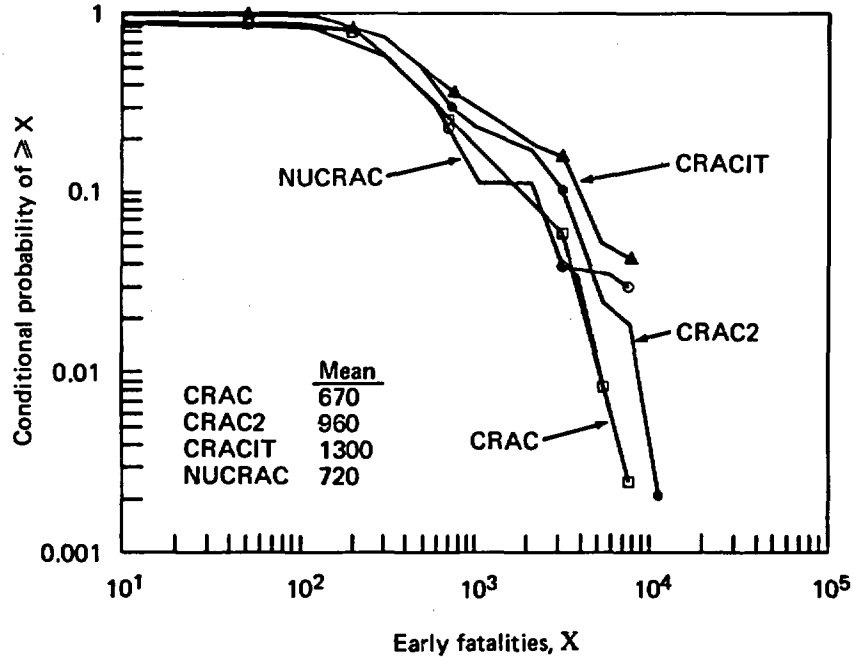

Figure D-14. Early-fatality CCDF, river-valley site. Realistic population, no emergency response. From Aldrich et al. (1981b).

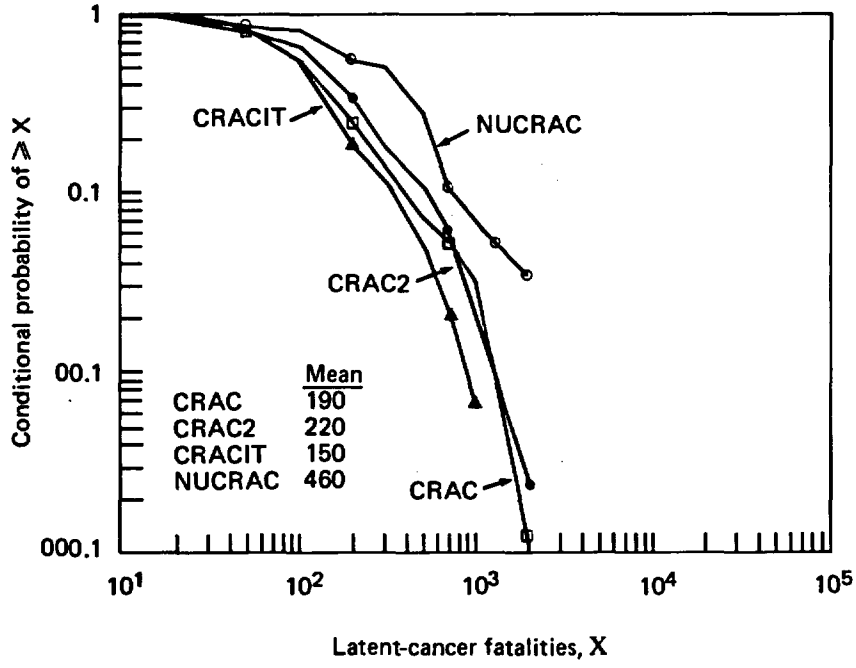

Figure D-15. CCDF for latent-cancer fatalities, river-valley site, realistic population. From Aldrich et al. (1981b).

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.

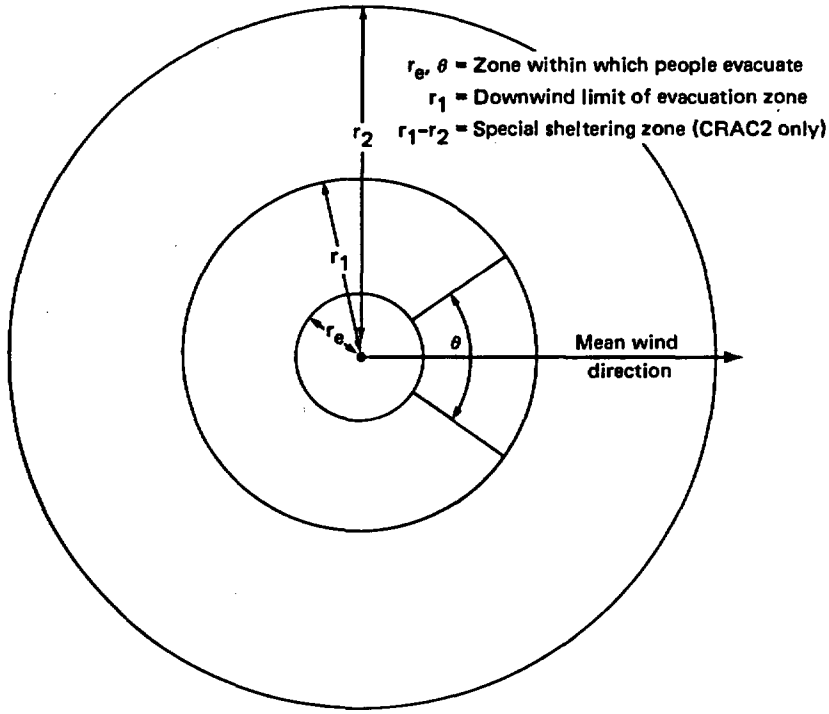


Figure E-1. Zones in CRAC and CRAC2 evacuation models.

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.

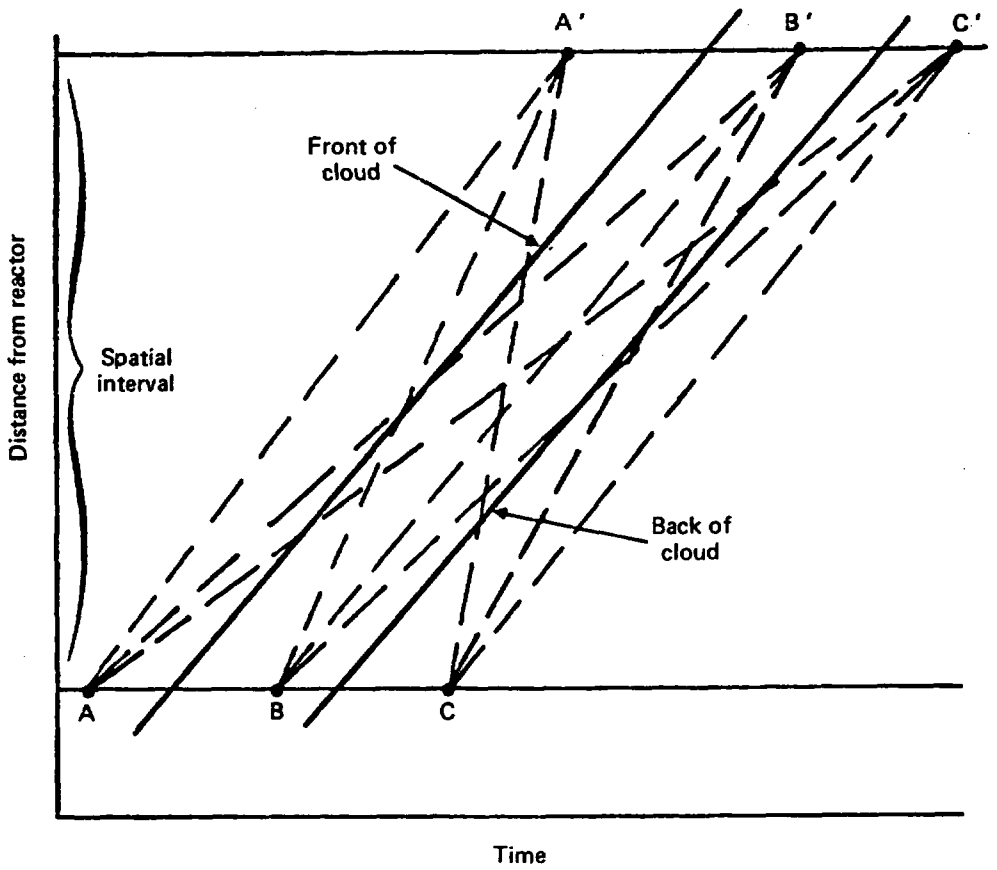
4. Shielding factors and breathing rates may differ markedly during delay and transit times.

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.

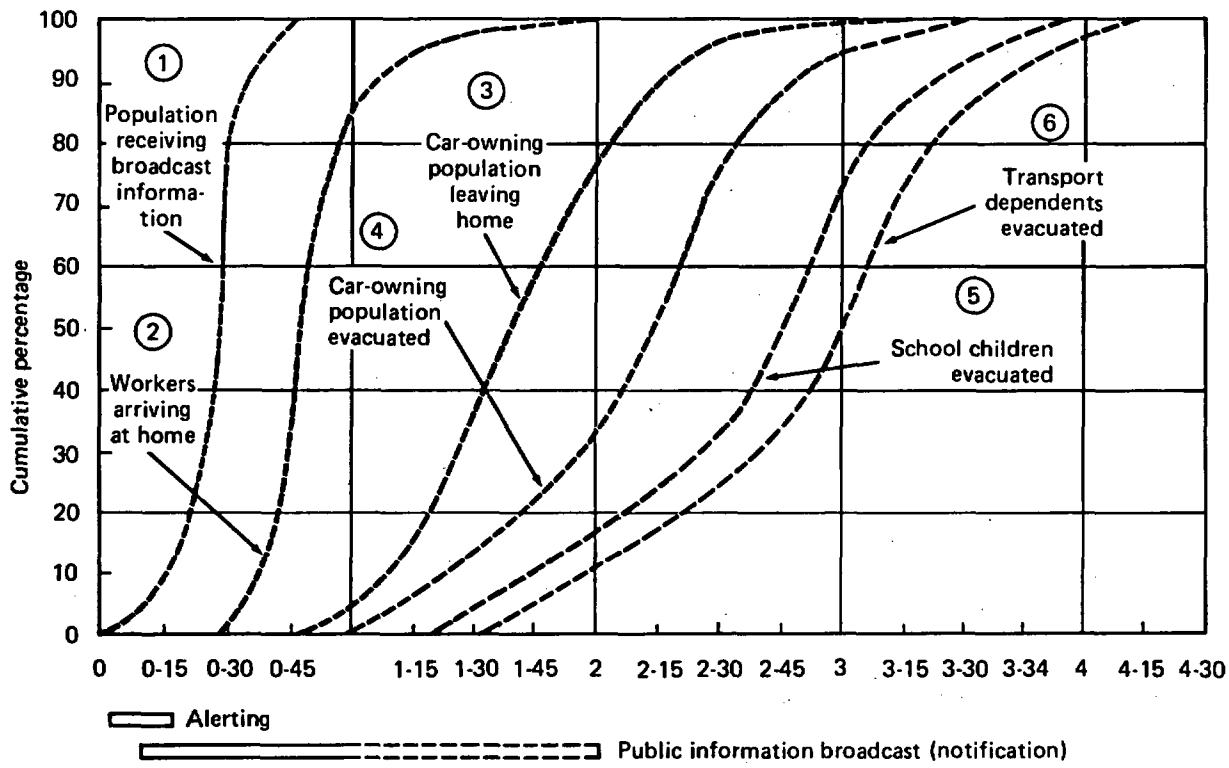


Figure E-3. Time estimates for population evacuation: typical response curves. These curves are suggestive of a hypothetical 10-mile-radius EPZ. Similar curves can be developed for subareas of the entire EPZ. The horizontal displacement of these curves along the time axis as well as the slope of the 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	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.

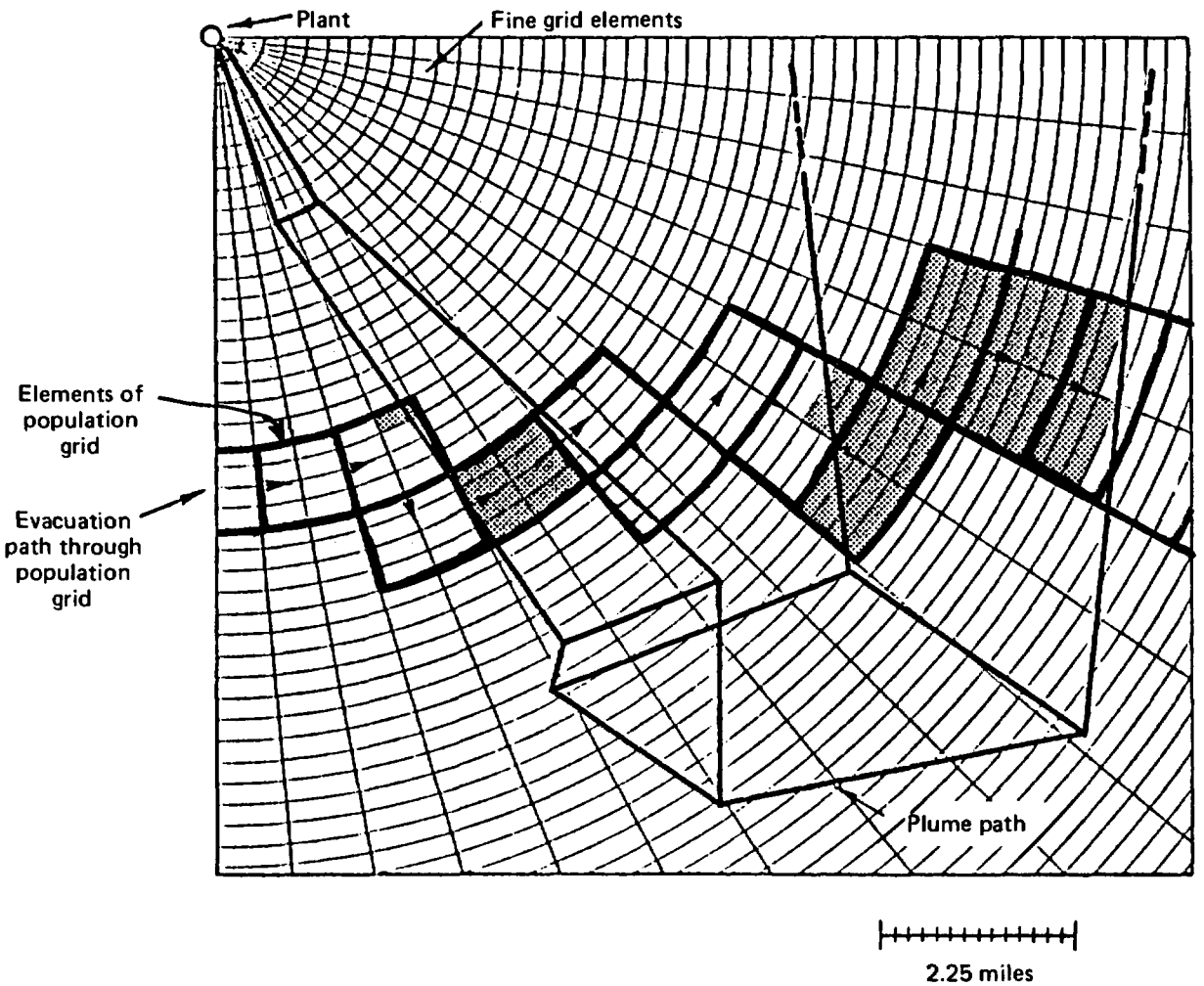

Figure E-4. Illustration of plume and evacuation paths on fine grid (dose calculations made by CRACIT in shaded fine-grid areas). From Commonwealth Edison Company (1981).

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	Two	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		FE	X	X
							(X-Z)							
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

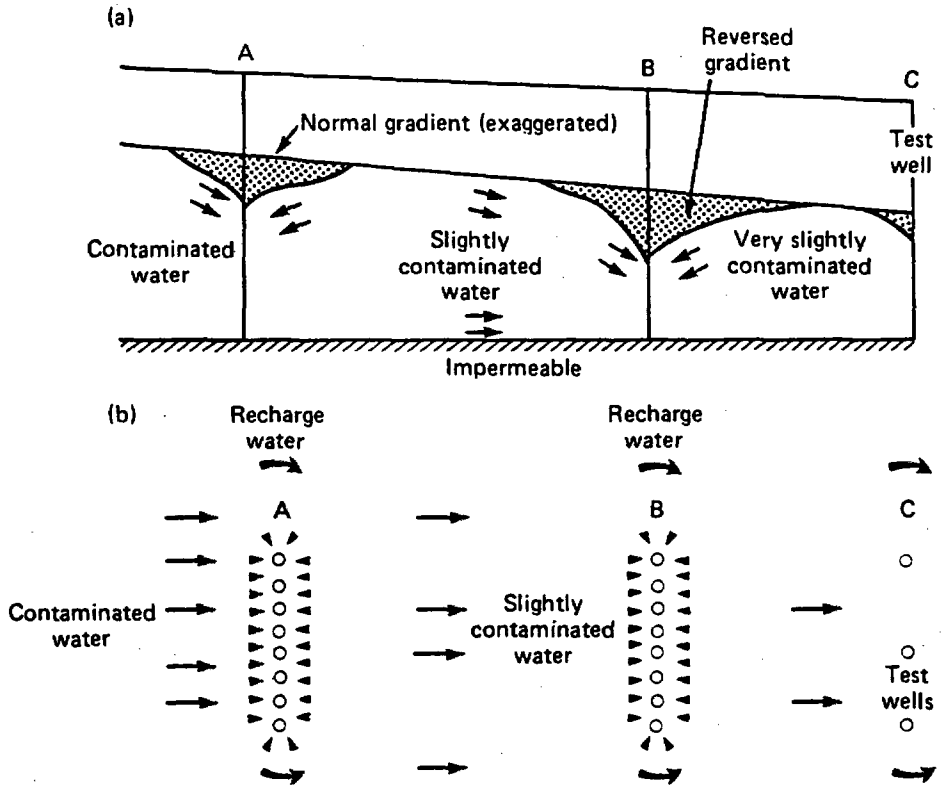

Figure F-1. Pumping wells downgradient from the source of contamination. From Niemczyk et al. (1981).

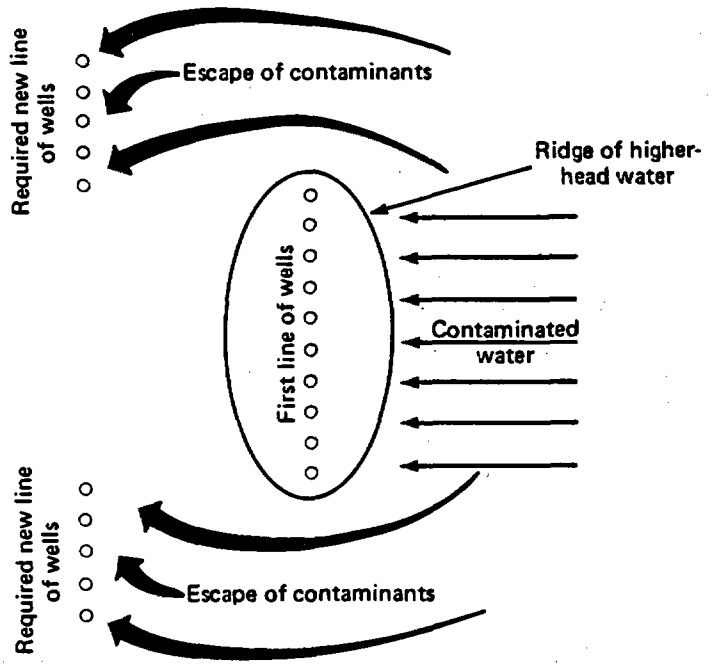

Figure F-2. Installation of a recharging field in an aquifer without lateral confinement. From Niemczyk et al. (1981).

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,

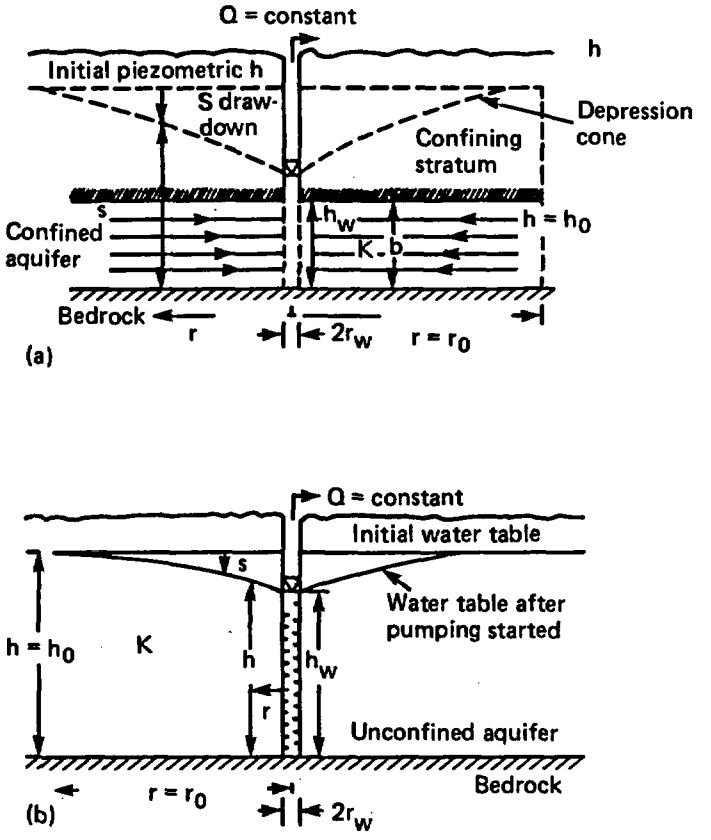

Figure F-3. Radial flow to (a) a well completely penetrating a confined aquifer, (b) a well completely penetrating an unconfined aquifer, and (c) an incompletely penetrating well in a confined (left) and unconfined (right) aquifer. From Niemczyk et al. (1981).

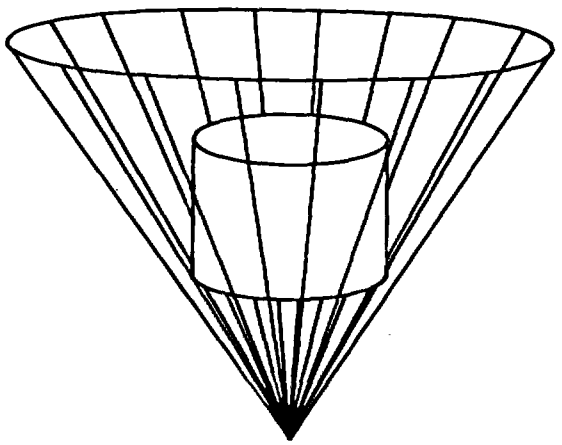

Figure F-4. Possible grouting configuration for the melt. From Niemczyk et al. (1981).

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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