Chapter 6

Accident-Sequence Quantification

6.1 OVERVIEW

6.1.1 INTRODUCTION

This chapter describes the process of quantifying accident sequences, which, as described in Section 3.4, are developed by an event-tree procedure that considers both initiating events and the success or failure of the relevant systems (or functions) in succession. The quantification may address each individual sequence or each of several groups of sequences, called "plant-damage bins" (PDBs), formed by combining sequences with certain similarities. Depending on the purpose and the intended use of the study, it may be desirable to estimate the uncertainty in the analysis that results from uncertainties in estimating the frequencies of initiating events and the probabilities of primary events. As defined in Chapter 3, primary events include basic events, undeveloped events, developed events, and external events. Both primary and secondary failures may be modeled by primary events.

The quantification task uses combinations of primary-event probabilities and the Boolean expressions developed in Chapter 3 to calculate a sequence frequency. Two approaches to this task are outlined. One approach is fault-tree linking, which determines the minimal cut sets for an accident sequence or a plant-damage bin. The minimal cut sets of an accident sequence are subsequently quantified to produce an estimated frequency for an accident sequence or a plant-damage bin. The other approach, which uses event trees with boundary conditions, quantifies system models under various conditions and multiplies system-failure probabilities by initiating-event frequencies to estimate an accident-sequence frequency.

Each accident sequence contains an initiating event and the subsequent failure of one or more safety systems. The system failures can represent combinations of faults undetected before the initiating event, failures of components or the operator to act on demand, failures of components to operate throughout a specified interval, or component unavailabilities due to testing or maintenance. In each case the component is functionally ineffective and unable or unavailable to carry out its mission. The probability of any of these faults is termed "failure probability." Thus, as used here, "failure probability" incorporates failure to start and/or failure to operate. The primary-event types and models for their treatment, including means or other distribution parameters, are described in Chapters 4 and 5. If the accident-sequence frequency is to be expressed as a point value, point-value estimates for primary events will be required. Possible point-value estimates include the mean, median, maximum-likelihood, and engineering estimates. Whichever point-value estimate is selected, the basis and the rationale for its selection should be given.

The results of the accident-sequence quantification task may or may not be the last task of the PRA study. In a level 1 PRA (see Chapter 2), where the objective is the quantification of core-melt sequences, the final product is the estimated frequency of the accident sequences, and there may be no need to distinguish these sequences in more detail than by the occurrence of core melt. In the more general case, in which containment failure, radionuclide release, or offsite consequences are to be analyzed (level 2 or 3 PRA), the results of the accident-sequence quantification are used as input to the containment analysis described in Chapter 7. In this case, the containment analyst will provide guidance as to which sequences are to be aggregated into the various plant-damage bins.

Besides the results generated for use in the risk assessment, the fault trees, event trees, and logic models can provide great insight into design and operation. They can be used to obtain both quantitative and qualitative information about systems. Quantitative techniques are available for using fault-tree models to derive reliability parameters (Henley and Kumanoto, 1981) and importance measures (Barlow and Proschan, 1975; Lambert and Gilman, 1977) for systems and components appearing in the accident-sequence models. Qualitatively, the models have a number of uses, from determining the minimal cut sets for a system to using variable transformations to analyze common-cause events (Worrell and Stack, 1981).

The remainder of this overview section discusses general approaches to accident-sequence quantification. Section 6.2 identifies the inputs to event-tree quantification: initiating events, component-failure values, dependent failures, and system fault trees. Section 6.3 covers the steps in accident-sequence quantification; it also discusses the treatment of multiple sequences, which can be combined into plant-damage bins, and presents the logic for reducing the quantification effort through screening and truncating. Section 6.4 describes methods for treating uncertainties and tracing their propagation through the accident sequences. Section 6.5 discusses some modeling considerations for accident-sequence quantification.

Section 6.6 discusses the computer codes that can be used for event- and fault-tree quantification or searches to identify potential dependent failures. Finally, Sections 6.7 and 6.8 summarize requirements for documentation and the assurance of technical quality, respectively.

6.1.2 APPROACHES TO ACCIDENT-SEQUENCE QUANTIFICATION

This section describes ways of evaluating the frequency of accident sequences from the initiating event, fault-tree models of the systems, and event-tree descriptions of the system failures making up that sequence. The fault trees are the logic models for combining faults (primary events) within a system or sequence; they are a set of Boolean expressions that can be reduced to minimal cut sets via Boolean algebra. These minimal cut sets represent the smallest sets of primary events that must exist simultaneously for the system failure (or sequence) to occur. A probability expression for the top event of the system failure or sequence can be determined from the minimal cut sets and used to quantify the probability of the top event.

Chapter 3 describes how the fault trees are developed to obtain Boolean expressions for each sequence. Additional information regarding fault-tree development and reduction as well as Boolean algebra can be found in the Fault Tree Handbook (Vesely et al., 1981a) and Applied Boolean Algebra (Hahn, 1966). In the discussion that follows, it is assumed the reader is familiar with these concepts.

At least two distinct approaches have so far been used to quantify the frequencies of accident sequences. One consists of combining system and component failures that are not necessarily independent; the other consists of combining event-tree tops that are all independent. The first approach automatically takes into account intersystem dependences within a sequence; the second method involves two steps--the quantification of each independent top event and the multiplication of the probabilities of those top events to get a sequence frequency. The quantification method chosen should correspond to the method used to create event- and fault-tree models.

Fault-Tree Linking

This approach combines (links) the fault trees for the event-tree tops (system headings) with an AND gate to form a new top event that is the accident sequence. Furthermore, if accident sequences with the same initiating event are combined in the same plant-damage bin, an OR gate may be used to combine the accident-sequence fault trees into a single model. Since initiating events are assumed to be mutually exclusive, the estimated frequencies for sequences with different initiating events can be summed to produce an estimated frequency of the plant-damage bin. The assumptions and ramifications of ORing the sequences in a bin are discussed in Section 6.3. A fault-tree-reduction code is then used to find the minimal cut sets of this new top event. Any dependences in the way of shared components or support systems are thus automatically accounted for in the Boolean reduction process, provided that unique identifiers have been assigned to these components across the respective fault trees. With this process the quantification takes place on the overall sequence cut sets as opposed to the individual systems or subsystems.

Event Trees with Boundary Conditions

In this approach, dependences like those between a support system and two or more front-line systems are explicitly displayed in the event tree. A front-line system is a system that directly performs a safety function, an example being the high-pressure injection system. A support system is a system that is needed for a front-line system to perform its safety function; an example is the ac electric power system. Each system is quantified for every set of boundary conditions that have a unique effect on system-failure probability, where the boundary conditions are a given set of component and system states that affect the system being quantified. The quantification involves the calculation of conditional probabilities since specific component and system states are assumed. Events are combined within the event tree by multiplication to obtain estimated frequencies or approximate frequency distributions for each sequence. The estimated sequence frequencies within each plant-damage bin are then summed to obtain a total estimated frequency for each bin.

6.2 INPUTS TO ACCIDENT-SEQUENCE QUANTIFICATION

The plant logic model consists of the event trees and fault trees developed in Chapter 3. It is composed of various primary events, which may be initiating events, component failures, unavailabilities due to testing or maintenance, recovery failures, dependent failures (beta factors), human errors, or external events. All primary-event values are expressed as probabilities except for the values of initiating events, which are frequencies, and the values of external events, which can be frequencies or probabilities. Table 6-1 lists the sources for primary-event values in this document. Solution of the plant logic model yields combinations of initiating events and system failures that are evaluated to yield accident-sequence frequencies.

Table 6-1. Sources of primary-event values

Primary event	Source (section)
Initiating event	5.3.4
Component failure	
Failure on demand	5.3.1.1.2
Failure in time (standby)	5.3.1.1.2
Failure in time (annunciated)	5.3.1.1.2
Failure in time after successful start	5.3.1.1.2
Test unavailability	5.3.2
Maintenance unavailability	5.3.3
Recovery (i.e., nonrecovery)	5.4
Dependent failures (beta factors)	5.4
Human errors	4
External events	10

Figure 6-1 shows the important elements in the quantification procedure. This particular diagram illustrates the steps in the fault-tree-linking approach. When event trees with boundary conditions are used, the steps are somewhat different, as will be described in Section 6.3; however, the inputs, as indicated within the broken lines, are similar.

Before quantification begins, the logic models are defined and developed as described in Chapter 3. Consistent event trees and the fault-tree models required by their top events (headings) are developed by the systems analyst(s). These logic models go hand in hand regardless of whether fault-tree linking or event trees with boundary conditions are used in quantification. The fault-tree logic models identify the primary events in sufficient detail for the data analyst to interpret them and provide models and the associated parameters for their quantification, including undependabilities and frequencies, and their distribution parameters if uncertainty is to be propagated.

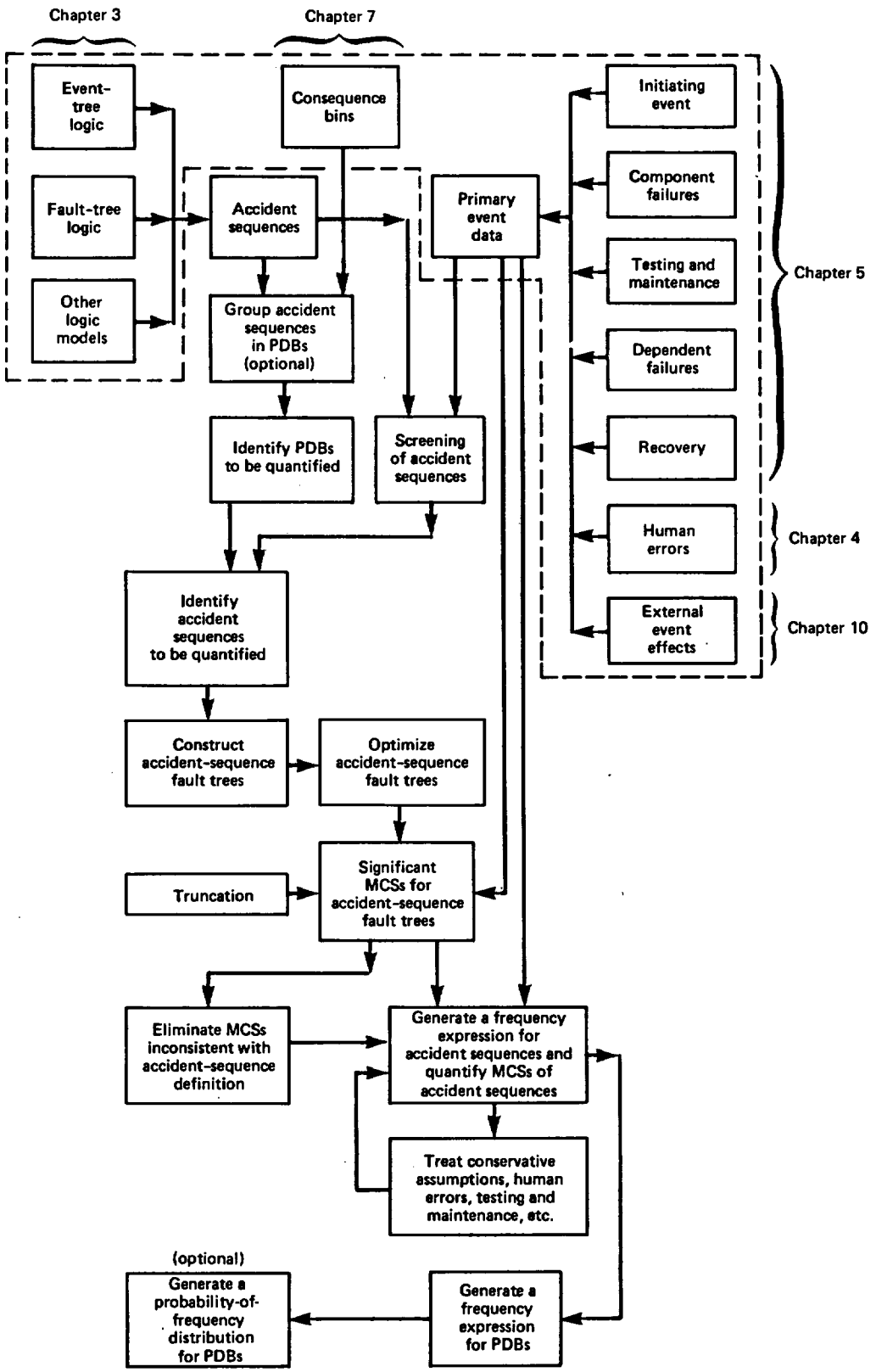


Figure 6-1. Inputs and steps for quantification.

The data base (models and parameters) for hardware failures and for test and maintenance activities is provided in Chapter 5, while that for human errors either in maintenance or during post-accident recovery procedures is provided in Chapter 4. The analysis of external hazards is conducted as described in Chapter 10, if that option is part of the study scope.

The event trees for each initiating event define the accident sequences to be evaluated, including the definition of the set of system faults that are included in each. The fault-tree models indicate the primary-event faults and fault combinations that cause these system faults to occur.

The consequences of accident sequences are then evaluated by the process described in Chapter 7. This process may or may not group the accident sequences into plant-damage bins. However, because of the similarities among certain accident sequences and the amount of work involved in their analysis, the accident sequences are usually so grouped. For our purposes a PDB can contain one accident sequence (in which case the PDB and the accident sequence are synonymous) or many accident sequences if the results of the containment analysis so specify. Basically, the binning process provides some ability to combine and reduce the total number of sequences in quantification, but binning is not a requirement for quantification.

6.3 QUANTIFICATION OF ACCIDENT SEQUENCES

6.3.1 GENERAL PROCEDURE

Accident-sequence analysis begins with the identification of the accident sequences to be analyzed, usually followed by a grouping of accident sequences into plant-damage bins, as defined by the consequence analysis. Sequences that cause similar physical responses in the plant are grouped into the same bin. The selection criteria for a PDB can be as coarse as "core melt" or "no core melt" or so fine as to require a unique bin for each accident sequence. The accident sequences in each bin may then be screened to eliminate those that will not contribute significantly to the total frequency of the bin.

Once the accident sequences to be quantified have been screened, a probability expression for each sequence is created from the solution of the plant logic model and then used to combine the estimated values for initiating and primary events. The two methods described in this section (i.e., fault-tree linking and event trees with boundary conditions) differ in this part of the process. In fault-tree linking, the accident sequence is represented by a fault tree whose top event is an AND gate with inputs representing the top gates of the system fault trees for each system depicted in the accident sequence. System dependences are explicitly treated in the fault-tree logic. The resultant sequence fault tree can then be analyzed by a number of available fault-tree-reduction techniques. The result of this step is a set of accident-sequence minimal cut sets (discussed in Chapter 3)

whose frequency estimates dominate the frequency of the accident sequence. The cut sets are then used to develop a probability expression for determining the sequence or bin frequency. This frequency can be characterized as a distribution or as an estimate.

If the analyst elects to use the other quantification method--event trees with boundary conditions--each branch of the event tree is evaluated, with the appropriate boundary conditions reflecting the various states of the support systems appearing in the path of the accident sequence. Thus these support dependences are treated within the event tree. Once all branch-point probabilities have been quantified, the accident-sequence frequency is obtained by simply multiplying the probabilities of the branch points in the accident sequence. If uncertainty calculations are to be made, the uncertainty for each branch point is derived and propagated through the accident sequence.

Both fault-tree linking and the use of event trees with boundary conditions result in point estimates for accident sequences or plant-damage bins. Section 3.7, "Analysis of Dependent Failures," explains the basic premise of both approaches and the fact that both methods, when rigorously applied, will result in equivalent solutions. However, since both methods apply some approximations and assumptions in practice, the final results for any given solution may vary if the assumptions used are not carefully examined. Fault-tree linking and event trees with boundary conditions are described below in Sections 6.3.2 and 6.3.3, respectively.

6.3.2 FAULT-TREE-LINKING METHOD

This approach involves constructing accident-sequence fault trees, solving these fault trees for dominant cut sets, generating a probability expression from the accident-sequence dominant cut sets, and combining the probability expressions for each accident sequence into an expression for the entire plant-damage bin.

6.3.2.1 Identification of Accident Sequences To Be Quantified

The first step is the identification of the accident sequences to be quantified. Of some help in this is the concept of plant-damage bins (PDBs), which are generated during the consequence analysis when accident sequences use the same mapping to release categories. When a PDB contains more than one sequence, the probabilities of the accident sequences can be summed to yield a PDB frequency that is mapped with the release categories. However, if the mutual exclusivity of the accident sequences has been lost (i.e., success states were not modeled in the accident logic), a conservative result may be obtained when the algebraic sum is used. This potential problem can be reduced by using a logical OR to combine accident sequences that are not mutually exclusive, thereby eliminating cut sets or sequences that subsume others within the PDB.

Valuable information can be gained by examining the release-category mapping for each PDB. This information can be used to establish the relative effects of each PDB on the analysis results and determine which ones will have the greatest effect on the results of the consequence analysis. This will allow the analyst to ensure that low-probability/high-consequence sequences will not be left out of the analysis. Some analysts choose sequences without regard to the PDBs, whereas others rely on the PDBs for guidance in choosing sequences for quantification. If accident sequences are chosen without regard to PDBs, a small-probability cutoff is used to eliminate sequences from consideration. If that cutoff results in the elimination of all sequences from the more-severe PDBs, the truncation value is lowered until sequences in the most-severe PDBs appear. Care must be taken to ensure that all significant contributions to bin frequency are taken into account, including the contributions of large numbers of low-frequency sequences.

If PDBs are used to group sequences, a number of approaches for eliminating accident sequences are available. If the frequency of a particular PDB can be shown to be less than the frequency of other, more-severe, PDBs, the entire PDB and its sequences can be eliminated.

Within a particular PDB chosen for quantification, some sequences can be discarded for any one of several reasons. Boolean reduction, at the system level, can eliminate several sequences in a PDB. It may be possible to estimate the frequency of some of the sequences and to eliminate those that do not significantly contribute to the PDB frequency. Finally, sequences within a PDB that are identical except for their initiating events can be modeled as one sequence, with a single initiating-event frequency representing the combined frequencies of the initiating events.

Boolean manipulation of the accident sequences in a PDB can generate a subset of sequences that can replace the original set of sequences. For example, given accident sequences TABC and $T\bar{A}BC$, where A, B, and C are system fault trees identical for both sequences, the Boolean properties of consensus and subsuming terms allow us to perform the following:

$$TABC + T\bar{A}BC = TBC$$

thereby replacing two more complex sequences with one simple sequence. A practical amount of sequence reduction at the sequence level can decrease the number of sequences that must be analyzed.

Some systems represented in the accident sequences are sufficiently similar to systems analyzed elsewhere that a reasonable failure-probability estimate can be used. In some cases, it may be possible to estimate the probability of a sequence, keeping in mind the potential systems interactions. If an accident sequence has a very low probability estimate, in comparison with the other sequences in its PDB, it can be eliminated from further analysis.

6.3.2.2 Construction of Accident-Sequence Fault Trees

The accident-sequence fault tree has an AND gate as its top gate. The inputs to the top gate are as follows:

1. The initiating event.

2. The system fault trees for the system failures depicted in the accident sequence. The minimal cut sets of the Boolean intersection of the system fault trees will be called the "system-failure minimal cut sets."

3. The dual fault trees for the system successes depicted in the accident sequence. (A dual fault tree is a success tree--the complement of the normal fault tree.)

Inclusion of the dual fault trees used to model system successes will eliminate system-failure minimal cut sets, which cause the failure of a system defined to be in a state of success. This will prevent system-failure cut sets from appearing in multiple accident sequences. The inclusion of these dual fault trees can greatly complicate the analysis and may not be required to obtain the desired result. The elimination of accident-sequence cut sets violating system-success states defined in accident sequences is discussed later in this section.

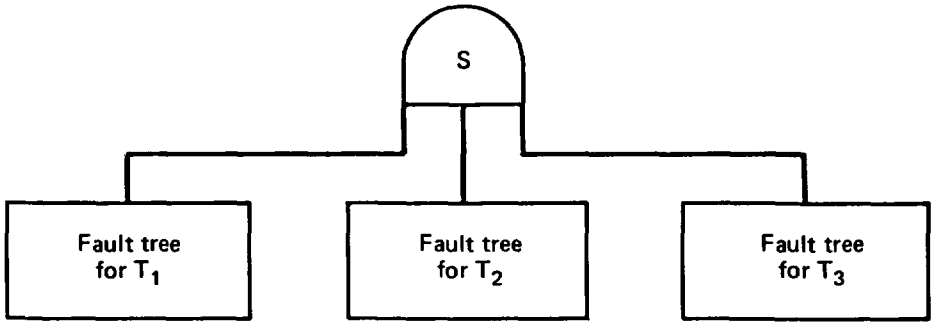
In fault-tree linking, there is a potential for the problem of the so-called circular-logic loop. When a number of fault trees are linked together, certain types of dependences can result in a situation where the failure of system A causes the failure of system B and the failure of system B causes the failure of system A. Any attempt to combine the two fault trees for these systems will meet with difficulties unless one branch of the circular logic is artificially cut off. Such problems would be revealed by the fault-tree processing code. Should this situation arise, it should be brought to the attention of the fault-tree modeler(s), who should modify the logic in one of the fault-tree logic models accordingly.

6.3.2.3 Optimization of Fault Trees

The number of events in the system fault trees can be substantially reduced by defining an equivalent system fault tree in which independent subtrees (modules) are replaced by developed events. The independent subtrees must be independent with respect to all of the systems represented in the accident sequence, including the initiating-event fault tree.

The concept of independent subtrees is relative to the top event of the fault tree being evaluated. In a particular event-tree sequence, such as $S = T_1 * T_2 * T_3$, a subtree of T_1 may be independent in fault tree T_1 but may contain events that also appear in T_2. This implies that the subtree is not independent with respect to the event-tree sequence S. The subtrees

that are independent with respect to S are found by identifying the independent subtrees of the following accident-sequence fault tree:



However, if this approach is taken for each event-tree sequence, then the independent subtrees will have to be identified for each event-tree sequence. A more efficient approach is to identify the independent subtrees relative to the intersection of all system failures represented in the event tree. Then the independent subtrees will be independent for any sequence of system failures and system successes. Some of the advantages of this approach are the following:

1. Once the independent subtrees have been identified, they can be used for any event-tree sequence.

2. Quantification and evaluation of the independent subtrees need to be done only once and will apply to all event-tree sequences.

3. The fault-tree analyst who wishes to verify that the reduced fault trees are equivalent to the original fault trees needs to become familiar with only one set of independent subtrees that applies to all event-tree sequences.

The concept of independent subtrees is very powerful, and their use is almost always beneficial. The fault-tree analyst can frequently create independent subtrees while coding his tree for computer analysis, although care should be taken to ensure that only events appearing as a group are included in the module. Because of the relative simplicity of the independent subtrees, a number of more-sophisticated analysis techniques, such as an analysis of time-dependent failures, can be performed on the independent subtree and the results included in the accident-sequence fault tree as a primary event with an associated probability.

6.3.2.4 Determination of Significant Minimal Cut Sets for an Accident Sequence

The accident-sequence minimal cut sets can represent the solution to a very large fault tree because the accident-sequence fault tree is formed by combining, under an AND gate, several system fault trees. Consequently, there may be millions or even billions of minimal cut sets for a particular

accident sequence. In order to minimize the number of minimal cut sets at this stage of the analysis, the dual fault trees representing system successes are typically not included at this time. (Although their inclusion would eliminate some system-failure minimal cut sets, it greatly increases the size and complexity of the fault-tree model being analyzed.) To reduce the number of minimal cut sets, a truncation value can be used to eliminate cut sets that make a negligible contribution to the total sequence probability. Note that truncation eliminates minimal cut sets that do belong to the set of the minimal cut sets for the accident sequence, whereas not including system-success states may leave in, for the time being, sets of primary events that appear to be the minimal cut sets of the accident sequence but, in fact, are not.

The truncation process eliminates minimal cut sets from the set of minimal cut sets for the accident sequence and thus is nonconservative. If a suitably low truncation value is used, the effect on the total accident-sequence probability is slight. Since this process is nonconservative, care must be taken to ensure that an appropriate truncation value is used. The truncation value should be constant or increasing throughout the solution process. The effect of the truncation value used should be bounded and shown to be insignificant. If at any time after truncation the point estimate of a primary event is increased, the truncation is not valid, and the process must be repeated with the new value.

In truncation the primary events are treated as if they are statistically independent and any dependences have been incorporated into the primary-event probabilities. A problem that may lead to a nonconservative result can arise when the cut set includes components whose failure probability was derived from pooled information under the assumption that they are identical. When the primary-event probabilities for those components are multiplied together, the result may be smaller than the result obtained by a cut-set evaluation that uses a single distribution or confidence bound to represent both components. If the uncertainty in the point estimate is relatively small, the error thus introduced is not significant. However, this error can become important when uncertainties are large. Unfortunately, at present there is no automated approach to this problem, and therefore analysts should use care to minimize this effect.

If the remaining cut sets are processed in any manner that serves to increase their likelihood (i.e., the addition of beta factors or common-cause human-error events), the truncation process is invalid, because some of the truncated cut sets could be increased in value above the truncation point. When truncation is used, all primary-event commonalities must be explicitly represented in the fault tree.

The cut sets that survive the truncation must then be examined to eliminate those that are inconsistent with the accident-sequence definition. The cut sets are inspected and modified to remove overly conservative assumptions about primary-event data. The minimal cut sets can be inconsistent with the accident-sequence definition for the following reasons:

1. The cut set may violate the system-success states in the sequence.

2. Cut sets may contain mutually exclusive primary events.

These minimal cut sets can be eliminated either manually or through the use of fault-tree models and computer codes.

The system-success states appearing in the accident-sequence fault tree can be accounted for by directly inspecting the minimal cut sets when there are few significant minimal cut sets. Unfortunately, a large number of significant minimal cut sets may prohibit direct inspection. It is then necessary to use a computer code, which can be done by either of two methods: list matching or the dual-fault-tree approach.

The list-matching approach is based on the fact that, if a minimal cut set for the system-failure portion of an accident sequence will also imply the failure of a system required to be in a success state, this minimal cut set should be deleted. List matching is performed by matching the failure cut sets of the accident sequence with the fault tree for system success. Any sequence-failure minimal cut set that is an implicant of a minimal cut set for a system success can be eliminated.

The dual-fault-tree approach involves obtaining the set of minimal cut sets of the dual fault tree and forming the Boolean conjunction of this set with the set of minimal cut sets for the accident-sequence failures. This process eliminates any terms that are products of an event and its complement because $P * \bar{P} = \phi$. To get a result of the same form as that obtained with list matching, the complemented events are deleted, and the resulting cut sets are simplified and reduced. Care must be taken to ensure that primary-event definitions are consistent for the system fault trees and dual fault trees.

When multiple failure modes of components appear in the system fault trees (e.g., switch A fails open and switch A fails closed) without modeling the mutually exclusive nature of these events, it is possible for minimal cut sets to contain mutually exclusive primary events. Such minimal cut sets should be eliminated, either through direct inspection or by using a computer code and a transformation of variables to explicitly model the mutually exclusive failure modes. Applying the identity $P * \bar{P} = \phi$ will accomplish this. These techniques have been used in modeling operational procedures such as technical specifications and plant management policy.

If conservative assumptions have been made about component recovery from failure or conservative probability estimates have been used in screening cut sets, it may be desirable to treat these conservatisms in a more realistic manner. Two methods are used to this end:

1. Some accident-sequence cut sets may contain events where repair or recovery may occur before the time the component is required to perform its function. This stems from a desire to simplify the system models or the fact that recovery from equipment failures can be dependent on other failures that have occurred. When the system analyst has determined that credit for a recovery act can be taken, that recovery act can be appended to the cut set as a primary event with an associated probability, thereby reducing the probability of the cut set.

2. If a conservative probability estimate has been used in the cut-set-screening process, more realistic estimates may be used if such data are available.

If conservatism in the screening process has been excessive, then a relaxation of the conservatisms may lower cut-set probabilities drastically. This may necessitate additional fault-tree quantification since cut sets previously excluded in the truncation process may become relatively significant.

6.3.2.5 Quantification of Accident-Sequence Cut Sets

The quantification of accident-sequence cut sets begins with the generation of a probability expression for the sequence minimal cut sets. This expression is then used to quantify the accident sequence by using the estimated values for primary-event probabilities and estimated initiating-event frequencies to yield a best-estimate value for the frequency of the event sequence. The probability expression is also used as the basis for the uncertainty analysis described in Section 6.4.

A number of techniques are available to generate probability expressions for the minimal-cut-set representation of accident sequences (Barlow and Proschan, 1975). They range from generating an upper bound by means of the sum-of-products rare-event approximation (often adequate in nuclear plant risk analysis because of the small numerical magnitude of the core-melt risk frequencies), using bounding techniques that generate both upper and lower bounds, or generating the exact probability expression for the top event. Where the sum-of-products method yields an overconservative expression for the accident-sequence frequency (e.g., system-success probabilities and the failure probabilities of less reliable individual systems would have cut sets that do not fit the rare-event approximation), one of the bounding techniques or an approximation of the exact expression can be used.

When an approximation other than the sum of products is used, it is usually done by eliminating cut-set intersections that do not contribute to the final result probabilistically. The approximation can be conservative or nonconservative, but the effects on the final result must be shown.

Generation of the probability expressions can be extremely difficult: most computer codes that generate an exact probability expression are generally unable to handle more than a few hundred minimal cut sets because of the required computer time and storage. However, a new method (Corynan, 1982) may significantly increase this number.

Two techniques are used to generate, for accident sequences, Boolean expressions that can be more efficiently quantified: the creation of independent subtrees (modularization) and the creation of mutually exclusive sets of cut sets. These techniques are described below.

The use of independent subtrees, or modules, allows the analyst to replace a portion of the fault tree with a single event. A probability expression for the independent subtree is then generated, and a probability

for the single event is created. This process combines a number of cut sets and greatly reduces the work of constructing the probability expression. Care must be taken, however, to ensure that an independent subtree is truly independent with respect to all logic within an accident sequence. The probabilities (e.g., unavailabilities) calculated for the independent subtree may also be time dependent for nonrepairable failures (see the discussion of nonrepairable failures in Chapter 5).

If a plant-damage bin containing multiple accident sequences is to be quantified, the probability expressions for the accident sequences within the bin must be combined. If the sequences have retained their mutual exclusivity, the probability expressions can simply be summed. However, the fault-tree-linking method usually does not yield mutually exclusive accident-sequence cut sets. The sequence-probability expressions can be summed to yield a conservative result in this case. If more exact results are desired, advantage can be taken of the fact that initiating events are generally treated as mutually exclusive. The PDB cut sets can be put into mutually exclusive groups by sorting the accident-sequence cut sets by initiating event. These groups are mutually exclusive, and their probability expressions can be summed to yield an expression for the entire PDB. Care must be taken to ensure that the combined initiating events are identically defined.

Once a probability expression for a plant-damage bin has been developed, the frequency for the bin can be obtained by replacing the variables in the probability expression with their best estimates and evaluating the equations. If a distribution for the frequency of the bin is desired, the uncertainty analysis described in Section 6.4 can be used to propagate primary-event distributions to obtain a probability-of-frequency distribution.

6.3.2.6 Evaluation of Common-Cause Events and Dependences

Fault-tree linking provides a structure that can be used to perform the common-cause analysis described in Section 3.7. The dependent-failure approach and the qualitative common-cause search can be applied to the fault tree directly or to the minimal cut sets of the accident-sequence fault tree. The approach taken depends primarily on the number of minimal cut sets generated by the accident-sequence fault tree since the solution and enumeration of large numbers of cut sets are impractical.

If the dependent-failure approach is to be used for quantifying common-cause events, there are at least two distinct methods for applying it. Typically with small fault-tree models generating hundreds of cut sets, the beta-factor method can be applied on a cut-set basis. This approach requires that all the minimal cut sets for the fault tree be generated (i.e., no probability truncation) and that each cut set be individually examined to determine whether a dependent-failure probability should be applied to increase the cut-set frequency or probability. Since all the cut sets must be generated and examined, there is a limitation on the total number of cut sets that can be analyzed. While it may prove to be impractical to apply dependent-failure probabilities to all the cut sets of the accident

sequence, it may be possible to apply them to the cut sets of independent subtrees within the accident-sequence fault tree, since the independent subtrees are quantified individually and replaced by primary events within the accident-sequence fault tree. If the fault tree has been modularized, care must be taken that dependences between modules are calculated and included.

For accident-sequence fault trees that generate too many minimal cut sets for using dependent-failure probabilities on an individual basis, Section 3.7 describes a method for introducing dependent-failure probabilities as primary events in the system fault trees. This method uses solutions at intermediate gates of the accident-sequence fault tree to analyze portions of systems and derive dependent-failure probabilities from those solutions. The accident-sequence fault tree is then modified to include new primary events representing the dependent-failure probabilities, at the appropriate places. The modified fault trees are then solved in a normal typical fashion (including truncation) to yield a result with dependent-failure probabilities included.

Similarly, qualitative searches can be made for common-cause events on the accident-sequence cut sets (Burdick et al., 1976; Rooney and Fussell, 1978; Worrell and Stack, 1981). As already discussed, if any cut sets were eliminated during the fault-tree solution, the common-cause analysis is not complete, and the results of common-cause searches may not include all significant common-cause events. One way around this problem is to break the accident-sequence fault tree into subtrees for which all the cut sets can be obtained. The cut sets for each subtree are then searched for common-cause modes within that subtree and the results are propagated to the top of the accident-sequence fault tree (Wagner et al., 1977). In this manner all the cut sets can be analyzed.

Another approach to the common-cause search is to use a transformation-of-variables technique to change the fault tree to a form reflecting the effects of common-cause events; it has been described by Rasmuson et al. (1979), Putney (1981), and Worrell and Stack (1981). Once the fault tree has been transformed, it can be solved to yield minimal cut sets containing one or more common-cause events, combinations of common-cause events, or cut sets containing common-cause events. Combining multiple common-cause events and combining common-cause events with random-failure events have been shown to be important in Fire Related Accident Sequences at CRBRP (Science Applications, Inc., 1978).

6.3.3 EVENT TREES WITH BOUNDARY CONDITIONS

When the method of event trees with boundary conditions is used, algebraic expressions are (usually) implicitly developed for each PDB by a stepwise process. This development process is implicit because, unlike in the fault-tree-linking method, no single Boolean expression at the component level is defined for each bin--it is merely implied. However, after an optional initial screening for dominant sequences, either method can be used to combine distributions in an identical way. The key differences between the methods lie in how the dominant sequences are defined and how the frequency for each plant-damage bin is arrived at. The main steps in this

approach are outlined below, followed by a discussion of means to limit event-tree size.

As described in Section 3.7.3.3, the method of event trees with boundary conditions uses more detailed event trees and therefore simpler fault trees than does the fault-tree-linking approach. In particular, the support systems found to be important are included explicitly as top events in the event trees. In this approach, then, "systems" or "top events" are narrowly defined. Thus, important dependences between top events are shown explicitly in the event tree rather than being contained in the fault trees underlying the top events. In this approach, separate fault trees or system models are, in effect, also written for each branch point of the event tree. These fault trees then explicitly recognize the states of the systems or top events upstream on the path leading to that branch point.* When such a fault tree is quantified, it yields the split fractions--that is, the frequencies of the events that make up the sequence--for that specific branch point. To be more specific, it yields the split fraction for that top event conditional upon the path through the event tree by which that top event is reached.

Consider as a simple example the event tree in Figure 6-2.

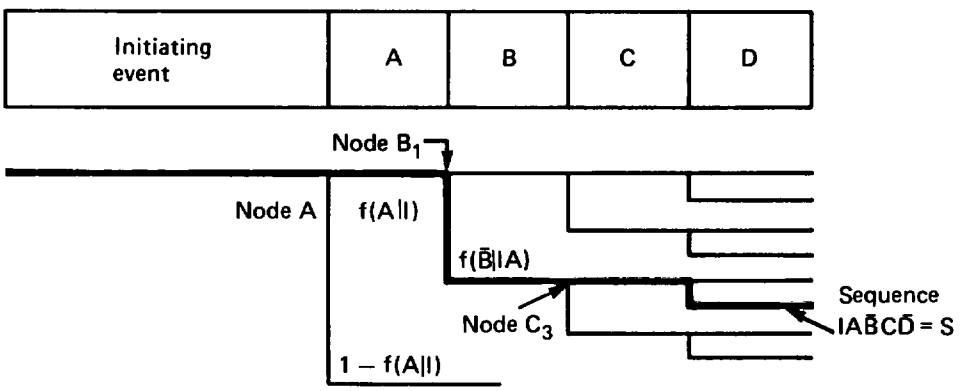


Figure 6-2. Sample event tree.

Each path through this event tree (i.e., each accident sequence) is characterized by the particular initiating event (or entry state to the tree) and by the failed and partially failed systems in the path. Consider, for example, the path

$$S = IA\bar{B}C\bar{D}$$

This sequence, consisting of initiating event I followed by the success of systems A and C and the failure of subsystems B and D, is represented by the

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*This recognition can also be thought of as boundary conditions on the system fault tree--hence the term "event trees with boundary conditions."

darkened line in the diagram and is designated by a bar over the symbol in the name of the sequence.

The likelihood of a sequence is quantified by reference to a "thought experiment" in which the reactor in question is imagined to be operated for many, many billions or trillions of years. We then ask ourselves, "In this experiment, how frequently, in times per operating year, does this accident sequence occur?" This frequency is referred to as the "sequence frequency," or, if the sequence is represented by a path in an event tree, it could be called the "path frequency."

Since we have not, in fact, done this experiment, we cannot, of course, say what this sequence frequency is with complete certainty. However, we can logically infer some things about this frequency from the frequencies of the "elemental" events that make up the sequence (i.e., the split fractions).

These elemental frequencies are themselves known only within a certain degree of accuracy, which can be expressed by giving a probability curve for each elemental frequency. These elemental probability curves can then be combined or "propagated" appropriately to develop probability curves for the frequencies of the accident sequences, if desired.

In the thought experiment, let $\phi(I)$ be the frequency per plant-year with which the initiating event I occurs. This is then the frequency of the left end, or "trunk," of the tree in Figure 6-2. It is then split up into the frequencies of the various branches. Thus, now consider all the instances in our thought experiment when event I occurred and let $f(A|I)$ be the fraction of those instances in which system A succeeded (i.e., was available). Then $f(A|I)$ is the fraction of those sequences entering node A that emerge through the upper branch at the right of node A.

In our thought experiment, then, $\phi(I)\ f(A|I)$ is the number of sequences, per plant-year, that enter node B_1. Out of all those sequences, let $f(\bar{B}|IA)$ be the fraction that emerges from B_1 along the lower branch. The term $f(\bar{B}|IA)$ is then the split fraction at node B_1.

Proceeding in this way, we can finally express the frequency of sequence S, in our thought experiment, in terms of $\phi(I)$ and the split fractions along the path. Thus,

$$\phi(S) = \phi(I)|f(A|I)\ f(\bar{B}|IA)\ f(C|IA\bar{B})\ f(\bar{D}|IA\bar{B}C)$$

where

$\phi(S)$ = the frequency of accident sequence S

$\phi(I)$ = the frequency of initiating event I

$f(A|I)$ = the frequency of success for system A, given that I has happened (i.e., the split fraction at node A)

$f(\bar{B}|IA)$ = the frequency of failure for system B, given that I has happened and A has succeeded (the split fraction at node B_1)

$f(C|IA\bar{B})$ = the frequency of success for system C, given that I has happened, A has succeeded, and B has failed

$f(\bar{D}|IA\bar{B}C)$ = the frequency of failure for system D, given I, A, $\bar{B}$, and C

From this equation, therefore, we can calculate the frequency of sequence S from $\phi(I)$, which comes directly from data analysis (see Chapter 5), and from the split fractions that come from system fault trees.

Note that these fault trees must be specialized to each branch point. Thus, for example, suppose A and B were support systems. Then $f(C|IA\bar{B})$, the split fraction at node C_3, must be calculated from the system model for system C with the recognition (or "boundary condition") that support system A is working and support system B is not.*

The next section elaborates on the development of event trees and the computation of the split fractions. After that, we generalize the example of Figure 6-2 and discuss the calculation of PDB frequencies.

6.3.3.1 Event-Tree Development and the Determination of Split Fractions

The first step is to develop event trees displaying all the significant intersystem dependences between the front-line systems whose performance is pertinent for the initiating event of interest. These result from common support systems and any other dependences (human error, environmental) judged to be important. The event trees include these support-system operability states as well as those of the front-line systems. Section 3.7.3 illustrates the event-tree development. Note that the pertinent dependences between support systems are to be identified and displayed in the event tree. In addition, multiple branches (reflecting partial success) rather than just binary (success or fail) branches are used where this more appropriately describes the support-system states and facilitates the quantification of the front-line system. For example, for the electric power heading of the event tree with, say, two buses supplying the safety systems, four branches would be included in the event tree to describe the availability of electric power. These branches would represent "both buses working," "bus 1 working and bus 2 failed," "bus 1 failed and bus 2 working," and "both buses failed."

When the event trees have been completed, the split fractions in the event trees are determined from logic models for the system or top event under the conditions represented by the particular branch point or node in question. The system logic models are usually in the form of fault trees, but they can be reliability block diagrams, GO models, subevent trees, FMEA models, or any other kind of model, all of these forms, if properly done, being logically equivalent.

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*This can often be conveniently accomplished as suggested in Section 3.7.3.3 by writing a single fault tree for system C in which the states of systems A and B are regarded as "house events." It is not necessary to do this, however.

Simple fault trees are then written to relate the state of the top-event system to the states of its components. From the minimal cut sets of these trees, we can obtain the necessary condition for system failure in terms of sets of component failures. That is, the system does not fail unless at least one cut set of components fails.

The question then devolves upon what could cause the failure of one of these cut sets. The answers to this question are recorded and systematized through the use of a cause table (see Figure 6-3 for an abbreviated example). In this table, all possible causes ("candidate" causes) are listed in the left column. Each cause is then evaluated as part of the system analysis. The components that would fail from this cause are listed in column 3. If those components constitute a cut set, thus failing the system, this is noted in column 4. If a particular cause does result in system failure, the frequency* of that failure is recorded in column 2. (More specifically, what is recorded here is the fraction of times in our thought experiment that the system fails at the branch point in question as a result of this particular cause.)

The sum of the entries in column 2 (i.e., the sum of all frequencies of system-failure causes) is the split fraction for system failure at the branch point in question. The bottom of the cause table can be used to accommodate the contribution from "other" causes (i.e., from all causes not otherwise called out in the table). If such entries are used, the analyst should be careful to list all contributors to "other causes."

If the system should fail as a result of a particular cause, we then ask whether that same cause might also result in some other system failing or in an initiating event. If so, then it is a potential "common" cause and needs to be called out for special treatment in the analysis. Columns 5 and 6 in the cause table are used to call attention to such situations. Because split fractions are simply multiplied together, the identification of dependent failures in the cause table and subsequently in the event tree is critical and should be given a great deal of attention.

6.3.3.2 Computation of PDB Frequencies

Event trees are not limited as in Figure 6-2 to nodes with two branches. Therefore, to generalize the notation, let f_{nb} denote the split fraction at node n that goes with branch b. With these quantities established for each branch point, one can calculate the frequency of each accident-sequence path as

$$\phi(S) = \phi(I) f_{1b,1} f_{2b,2} \cdots f_{nb,n} \cdots \qquad (6\text{-}1)$$

$$= \phi(I)\ f(S)$$

where b_n is the branch chosen by the path at node n.

*These, along with the $\phi(I)$, are examples of elemental frequencies.

Cause	Failure frequency	Effect			
		Components	System	Other systems	Initiating events
Coincident hardware failures	4.5×10^{-6}	Mainly pumps	Fails	No effect	No effect
Testing	1.0×10^{-10}	Pumps	No effect	No effect	No effect
Maintenance and hardware failure	2.0×10^{-4}	Pumps or MV-8700A, B	Fails	No effect	No effect
Human error and hardware failure	8.2×10^{-9}	MOV-8809A, B closed failure on other side	Fails	No effect	No effect
Other	4.6×10^{-5}	Valves or pumps	Fails	No effect	No effect
Total	3.0×10^{-4}				

Dominant contributor = maintenance combined with hardware failure.

Figure 6-3. Example of format for a cause table for double failures (buses available).

The term f(S) on the right-hand side, the product of split fractions along a given path, thus has the meaning of "conditional frequency"; that is, for all the times initiating event I occurs, f(S) is the fraction of times in which accident sequence S results. In this way one can compute the conditional frequency for each path in the tree. These numbers thus characterize the tree itself, without reference to the frequency of the incoming entry state. Each sequence or path culminates in an exit state (i.e., a particular state of operability-functionability with respect to front-line systems).

Now let us focus attention on a particular exit state, say y_j, and let S_{ih} denote a particular accident sequence going from entry state i to exit state y_j. By summing over all such sequences, we obtain

$$m_{ij} = \sum_h f(S_{ih}) \tag{6-2}$$

The quantity m_{ij} is thus the conditional frequency of occurrence of exit state y_j given that initiating event i has occurred. That is, out of all the times entry state i occurs, m_{ij} is the fraction of times that exit state j occurs.

If we now let $\phi(I_i)$ be the frequency of initiating event i, then

$$\phi(I_i)m \tag{6-3}$$

is the frequency of occurrence of exit state y_j as a result of initiating event I_i. Moreover,

$$\sum_i \phi(I_i)m_{ij} \tag{6-4}$$

is the frequency of occurrence of exit state y_j as a result of all initiating events.

Equation 6-2 can now be recognized in essence as a matrix multiply operation. Thus, if we assemble the m_{ij} into a plant matrix M and the $\phi(I_i)$ into an initiating-event row vector ϕ^I, then

$$\phi^Y = \phi^I M \tag{6-5}$$

where ϕ^Y is a row vector containing the frequencies $\phi(Y_j)$ of the various plant-damage states Y_j.

The process of Equations 6-1 through 6-5 is carried through by first using point estimates (essentially mean values) of all the frequencies and split fractions to obtain point estimates for the frequencies $\phi(Y_j)$. These point estimates can then be used to eliminate from the uncertainty analysis those sequences whose point estimates do not contribute to the point estimate of the result. When point estimates are used, the analyst should ensure that the failure-rate dependences among systems containing components assumed to be identical will not cause a nondominant sequence to become a

contributor to the PDB frequency. To determine probability distributions for the $\phi(Y_j)$, we "propagate" the uncertainties in the elemental cause and initiating frequencies through the cause table and through Equations 6-1 through 6-5. In this operation, as in all probabilistic operations, attention must be paid to dependences between probability distributions. Also, as in all arithmetic, minor quantities in the calculation need not be treated with high accuracy; they can be approximated, upper bounded, or rounded off as appropriate, but such shortcuts should be well documented. Such shortcuts are especially useful in the computation of probability curves to avoid unnecessary computational labor.

6.3.4 APPROACHES TO REDUCING EVENT-TREE COMPLEXITY AND PROCESSING EFFORT

In order to keep the event trees manageable in size and the analysis practical, the analyst will need to make some assumptions and approximations that permit the omission of certain dependences from the event tree. In addition, some iteration is to be expected between logic-model development and quantification; that is, to some extent the event tree may have to be modified as quantification proceeds. Techniques available to assist the analyst include screening, bounding, and the use of impact vectors.

6.3.4.1 Bounding

To simplify the event trees and the quantification task, a conservative assumption can be used, perhaps by not taking full credit for the provided redundancy. For example, in the case of two highly reliable actuation signals, each of which initiates both of two safeguards systems, it may be useful to assign and restrict one signal to each system, thus eliminating the need to explicitly include actuation as a common support system.

A second example would be the assumption that all valve motor control centers connected to a vital electric-power bus are in effect a part of that bus. Such an assumption would be made to avoid the necessity of multiple additional electric-power states when the elements of a distribution system could potentially be common to valves in two front-line systems. This particular example is related to the discretization of support states considered in the event tree.

If no dominant impact results from making conservative assumptions, as often happens, the assumption can be accepted. However, should such an assumption artificially yield a dominant impact, it may be necessary to reexamine and refine the event-tree model to reduce the impact.

6.3.4.2 Screening

A study or an analysis can be made to examine the necessity of including a support system. If it can be shown that the support system is

extremely reliable, it may be possible to leave it out because it will have a negligible impact. The basis for such an assumption should obviously be documented.

6.3.4.3 Use of Impact Vectors

This technique, illustrated by an example in Section 3.7.3.3, can be a powerful logical approach. The support-system event tree is developed separately from that for the front-line systems. Then the impact of each support-system state (success/failure combination) on the front-line systems is developed in the form of an "impact vector" that describes the front-line systems that fail as a result of support-subsystem failures. The sequences can be collapsed down to the unique impacts that serve as the boundary conditions for evaluating the front-line systems. This variant of the event tree-boundary condition approach uses the quantification of the intermediate support-system states. Since both frequencies and damage-level information are available, it is possible to determine the risk-dominant support-system states before quantifying the front-line trees. Support-system states not significant to risk can be "pruned" at this step.

6.3.5 COMMENTS ON DIFFERENCES IN SEQUENCE-QUANTIFICATION APPROACHES

Two approaches to accident-sequence quantification--fault-tree linking and event trees with boundary conditions--have been described. Both make use of event trees in conjunction with fault trees. Both approaches require some assumptions and approximations to be practical--for example, the truncation of cut sets or the elimination of some dependences by making use of approximations. In the fault-tree-linking technique, the event trees have been constructed at a high level in terms of the function or system success or failure definition: it is necessary to display only the front-line functions or systems. The dependences on support systems and subsystems are accommodated entirely within the fault trees. The resultant linked fault trees are thus large and complex. When the fault trees and event trees are large, the existence of automated and efficient computer reduction techniques makes analysis by this approach possible in spite of the many cut sets that can be generated for quantification.

In the other quantification method, which uses event trees with boundary conditions, the more elaborate event trees are broken down to explicitly display the significant dependences. The resultant fault trees (or reliability block diagrams) for the event-tree top events are thus simpler and independent, and can be analyzed by hand without resorting to computer-assisted fault-tree reduction. Heavy reliance is placed on the analyst to identify and separate the dependences in the event-tree modeling. Considerable care must therefore be taken to ensure that the significant dependences in a sequence have either been identified and included as top events in the event tree or are otherwise accounted for in generating the split fractions along an accident-sequence path.

It should be noted that the use of event trees with boundary conditions generally yields many more sequences because of its evaluation for the various mutually exclusive support-system states. Several such sequences would combine to result in the same front-line-system configuration as that identified in fault-tree linking.

Overall, the basic conceptual difference between the methods is where in the process quantification (conversion from symbolic representation to numerical results) takes place: stepwise throughout the process (for event trees with boundary conditions) or as a single step near the end (for fault-tree linking). Both methods can be successfully employed and have been used in major studies performed to date. An advantage of stepwise quantification is a reduction in the need to carry through algebraic terms, so that quantification can be performed manually. An advantage of quantification as the last step is that the symbolic representation allows computer searches for dependences as the last step before quantification and the presentation of results in terms of cut sets for dominant accident sequences.

6.4 TREATMENT OF UNCERTAINTY

The probability or frequency estimates that are obtained by analyzing fault trees or event trees are generally associated with considerable uncertainty. The uncertainty comes from the following principal sources:

1. The specified models are incorrect. Basic assumptions about the accident sequences, system-failure modes, and the application of the quantification formulas may not be correct.

2. Important failure modes have been overlooked (completeness problem). The scope of the risk assessment may preclude the analysis of all initiating events, the analyst may not have all the required information, or the quantification process may have truncated large numbers of low-probability events that sum to a significant probability.

3. The values of the input parameters are not exactly known. Data limitations or uncertainties in component-failure rates require the use of probability distributions or interval estimates to model frequencies for initiating events and probabilities for system failures.

Although it may be possible to quantify the contribution to total uncertainty made by each of these sources, in practice it is very difficult to develop credible quantitative measures for all the sources of uncertainty in the analysis. It is usually more practical to perform additional analyses to ensure that the modeling is correct than to try estimating a particular quantitative uncertainty. This section discusses these uncertainty sources and describes a method for evaluating their contribution to total uncertainty in the analysis.

6.4.1 SOURCES OF UNCERTAINTY

Table 6-2 lists the uncertainties that can affect the estimates of accident-sequence frequencies as well as the sections of this guide that discuss these uncertainties. The major sources of uncertainty that are directly related to accident-sequence quantification are truncation schemes that eliminate accident sequences or accident-sequence cut sets that are determined to be insignificant. The errors they produce are nonconservative. Another source of error in quantification is the rare-event approximation used to develop a probability expression for the accident sequences; it produces conservative errors. Accident-sequence quantification provides the opportunity for assessing the effect of uncertainties in the input data on the calculated frequencies of accident sequences.

Table 6-2. Contributors to uncertainty in estimates of accident-sequence frequency

Uncertainty type	Source of uncertainty	PRA Procedures Guide section
Model uncertainties	Event- and fault-tree models do not correctly account for time-dependent component failures, component dependences, etc.	3.9
	Failure modes improperly defined	3.9
	Component-failure models may not be correct (i.e., exponential failure model)	5.7
	Approximations are used to sum large numbers of cut sets (i.e., rare-event approximation)	6.4.1
	Human errors	4
	External events	10.4, 11.2, 11.3, 11.4
Completeness	Event- and fault-tree models do not contain important failure modes	3.9
	Data base may not include all pertinent failures or experience	5.7
	Large numbers of low-probability accident sequences and cut sets may have been eliminated through truncation	6.4.1
Input-parameter uncertainty	Mission time for the operation of various systems may not be known exactly	3.9
	There are uncertainties in the frequencies of initiating events, component-failure rates, and test and maintenance parameters	5.7, 6.4.1

6.4.2 SOME PROCEDURES FOR UNCERTAINTY AND SENSITIVITY ANALYSIS

The uncertainty introduced through Boolean manipulations, truncations, and screenings should be small in comparison with that in the accident-sequence logic models and the data base. However, significant uncertainty can be introduced through the elimination of large numbers of low-frequency cut sets or accident sequences whose sum contributes significantly to the PDB frequency. In order to quantify this contribution, the cut sets must be generated and quantified. Unfortunately, most truncation schemes used in fault-tree analysis have no capability for estimating this contribution.

One way to estimate the total contribution of many low-frequency events is to use a direct-quantification code like WAM-BAM (see Section 6.6). The direct-quantification codes are very efficient and can use a much lower truncation value because they do not have to perform cut-set manipulations. Moreover, WAM-BAM has the capability to estimate an upper bound on the sum total of the truncated terms. By comparing the direct-quantification result obtained with a lower truncation value against the result of the cut-set solution, the analyst can determine whether a lower truncation value would significantly affect the result. In addition, the WAM-BAM output can be examined to determine the upper bound probability of the terms eliminated during the direct quantification. If the value is small, the use of truncation can be shown to have a small effect on the cut-set solution process.

When trying to evaluate the contribution to system-failure probability from variations in input parameters, the analyst can either perform a probabilistic importance analysis to get a qualitative feel for the effect of input parameters on the results or derive probability distributions or interval estimates for the result.

Probabilistic importance measures are a means of estimating the contribution of a primary event to the accident-sequence frequency. There are three principal types of measure: the Barlow-Proschan (Barlow and Proschan, 1975), the Fussell-Vesely (Fussell, 1975), and the Birnbaum (1969) measures; they have been defined and described by Lambert and Gilman (1977). The Barlow-Proschan and the Fussell-Vesely measures are more closely related to each other than to the Birnbaum measure. The exact nature of the relationships among these and other measures is discussed by Engelbrecht-Wiggans and Strip (1981).

The Barlow-Proschan and the Fussell-Vesely measures compute the probability that a primary event is contributing to the failure of a system and therefore provide information on which primary events, if made more failure-resistant through improved quality or redundancy, will most decrease the probability of a system failure.

The Barlow-Proschan measure of the importance of a primary event i is the probability of the system failing because a minimal cut set containing i fails, with primary event i failing last. By this definition, the most important primary event in a system is the most unlikely primary event in the most likely minimal cut set.

The Fussell-Vesely measure of the importance of a primary event is the probability primary event i is contributing to system failure, given the

system has failed. It is estimated by dividing the sum of the failure probabilities of the minimal cut sets that contain primary event i by the failure probability of the system. The most important primary event in the system according to this definition is the primary event in the most likely group of minimal cut sets. Thus, this definition gives some measure of the probability that the recovery of a primary event will restore the system.

The Birnbaum measure indicates the sensitivity of the overall system-failure probability to the probability of an individual primary event. Thus, it measures the rate of change in system-failure probability to change in primary-event probability. The upgrading function, which is closely related to the Birnbaum measure, can be used in many circumstances to help decide which primary events would contribute most to reducing system-failure probability.

As described by Engelbrecht-Wiggans and Strip (1981), these measures are intimately linked, and their differences are quite subtle. It is therefore difficult to recommend which measures are appropriate in different situations. The choice between the Barlow-Proschan/Fussell-Vesely and the Birnbaum measures is difficult because they measure slightly different aspects of system-failure probability, although frequently the former measures are more appropriate for measuring system improvement. However, Lambert (1975) demonstrates the use of the upgrading function (a variant of the Birnbaum measure) for selecting primary events for change to improve system-failure probability.

Chapter 12 discusses various methods for performing sensitivity studies and for propagating probability distribution and interval estimates based on the simplified equation for the frequency. Section 6.6 discusses the computer codes (e.g., SAMPLE) that can be used in the actual propagation. The manner in which the propagation is performed should be consistent with the data used in the analysis.

A consideration in the propagation of primary-event uncertainty through a top-event probability expression is the method of treating the uncertainty distribution or interval estimates of two primary-event probabilities derived from components assumed to be identical. Their uncertainty parameters are considered to be correlated. In evaluating the probability expression, only one distribution should be used to represent uncertainty for every primary event whose probability is derived from components assumed to be identical. Consider, for example, the probability expression

$$\begin{aligned} P(\text{top}) = {} & P(\text{pump A}) * P(\text{pump B}) \\ & + P(\text{pump A}) * P(\text{control B}) \\ & + P(\text{pump B}) * P(\text{control A}) \\ & + P(\text{control B}) * P(\text{control B}) \end{aligned}$$

If pumps A and B along with controls A and B are assumed to have identical failure rates, the probability expression should be changed to the form

$$P(\text{top}) = [P(\text{pump})]^2 + 2[P(\text{pump})\ P(\text{control})] + [P(\text{control})]^2$$

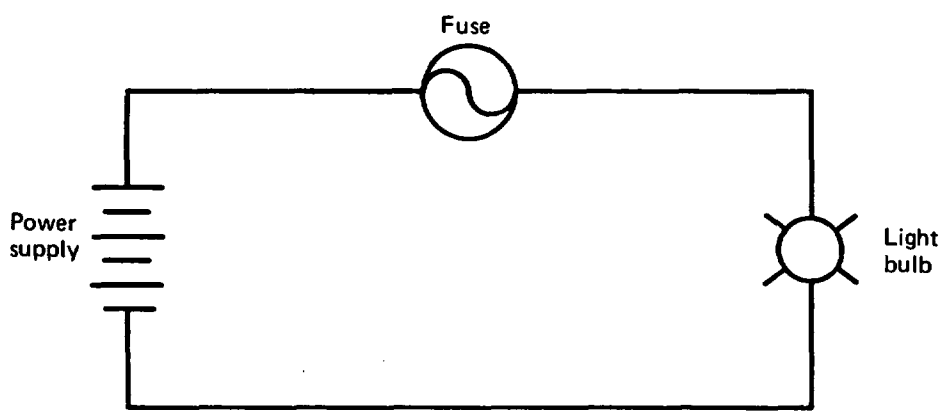
In this way the assumption that the primary events are identical can be correctly evaluated. With independent primary events and distributions, the sums or products of the means of the distributions for the individual primary events will yield the correct mean for the top event. The potential cause for error in assuming that components are identical has been discussed by Apostolakis and Kaplan (1981). In practice, the propagation of uncertainty in primary-event probability may be very difficult to perform by methods other than Monte Carlo for large numbers of independent modules containing similar components.

6.5 SOME MODELING CONSIDERATIONS FOR ACCIDENT SEQUENCES

While performing the general quantitative procedures it is important to note problems that can give erroneous results if the quantification analyst indiscriminately plugs primary-event probabilities into a fault-tree logic model. These problems relate to (1) repair when a secondary fault exists and (2) the potential for simultaneous testing and maintenance. Both cases can be resolved by requantifying a new primary event with a slight modification to the fault tree or pertinent cut set.

6.5.1 QUANTITATIVE ANALYSIS OF FAULT TREES THAT DO NOT REPRESENT REPAIR TREES

Probabilistic risk assessment uses fault trees to model the system failures represented in event trees. In quantifying these fault trees, all methods used in PRA computer programs assume the system fault trees also represent the system repair trees; that is, if component A fails and causes the system to fail, then repairing component A repairs the system. All system fault trees containing secondary failures of components, however, do not represent system repair trees, and the system-failure probabilities calculated by means of these fault trees and standard methods are underestimated. (Secondary failures are causes of malfunction for which the component itself is not accountable.)



Consider, for example, the system shown on the preceding page and the loss of light resulting from fuse-opening failures only. An analysis finds two sources of fuse-opening failures: (1) the fuse opens because of fuse defects ($\lambda_1 = 10^{-6}$ hr^{-1} and $\mu_1 = 10^{-2}$ hr^{-1}) and (2) the power supply surges, causing the fuse to open ($\lambda_2 = 10^{-2}$ hr^{-1} and $\mu_2 = 1$ hr^{-1}). Here the failure rate λ is the probability the component fails in time t to t + dt. The repair rate, μ dt, is the probability a component is repaired in time t to t + dt given the component is failed at time t. Using the standard methods and assuming the time-to-failure and time-to-repair distributions are exponential, the steady-state unavailability of the light from fuse-opening failures only is found to be

$$\bar{\Delta}_{fuse} = \bar{\Delta}_1 + \bar{\Delta}_2 - \bar{\Delta}_1\bar{\Delta}_2$$

$$\approx \frac{\lambda_1}{\mu_1 + \lambda_1} + \frac{\lambda_2}{\mu_2 + \lambda_2}$$

$$= \frac{10^{-6}}{10^{-2} + 10^{-6}} + \frac{10^{-2}}{1 + 10^{-2}}$$

$$= 0.01$$

This calculation assumes that the fuse (and thus the system) is repaired when the power supply is repaired if it caused the failure. Repair in this case, however, requires fixing both the power supply and the fuse. Thus, regardless of the cause of failure, the fuse must be repaired in order for the light to be available. For this example, the steady-state unavailability of the light from fuse-opening failures only is approximately 0.5. (The fuse fails at a rate of approximately 10^{-2} hr^{-1} and is repaired at a rate of approximately 10^{-2} hr^{-1}. The method for determining the repair rate is given on page 6-30.)

The error results from treating the failure logic for the component malfunction (the fuse failing open in this case) as the repair logic when, in fact, it is not. The malfunction occurs because the component is defective <u>or</u> because a secondary cause of failure arises; the malfunction is repaired only when the component <u>and</u> the secondary cause of failure are repaired. In such cases, appropriate measures must be taken to account for this difference in failure and repair logic. One method of eliminating this problem is to include all secondary causes of component failure in a single new secondary-failure primary event for that component. This must be implemented before the minimal cut sets are obtained. The data for this new secondary-failure primary event should reflect the rate at which the component fails from any secondary failure (neglecting those accounted for in the common-cause analysis) and the rate at which it and any secondary-failure causes are repaired. The failure and repair characteristics of the new secondary-failure primary event can be estimated directly from failure and repair data (Chapter 5) or can be synthesized from the failure logic and the failure and repair characteristics of the secondary causes of failure.

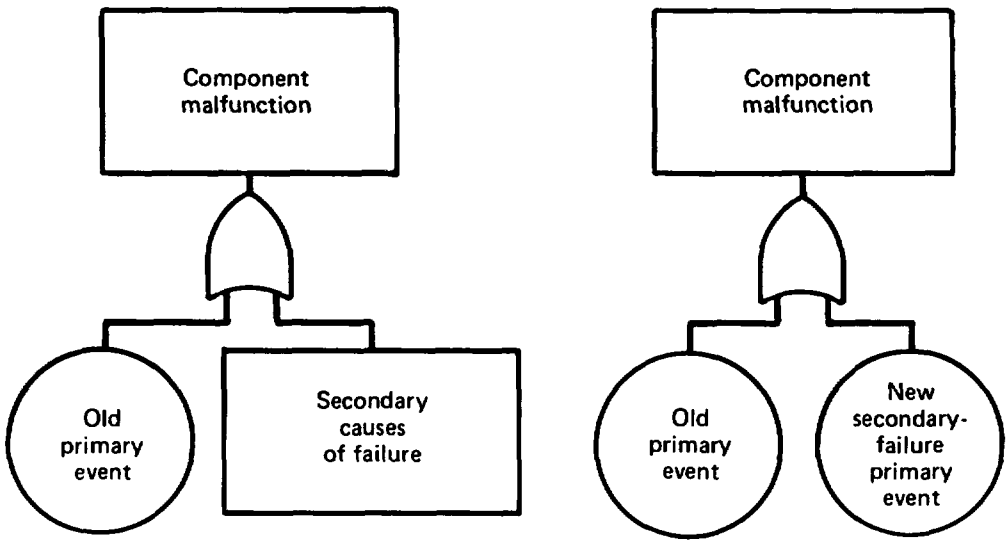


Figure 6-4. Procedure for synthesizing the failure and repair characteristics of a new primary event.

The procedure for synthesizing the failure and repair characteristics of this new primary event is as follows (see Figure 6-4):

1. Identify each primary event in a system fault tree that is affected by secondary failures. This requires examining each primary event in a system fault tree to determine whether it will fail as a result of other primary-event failures in the fault tree. In general, these secondary causes of failure are logically OR'ed with the primary event.

2. Calculate by standard methods a failure rate (λ) and a repair rate (μ) for a fault event that represents the secondary failure. This fault event can then be treated as a new secondary-failure primary event.

3. Calculate for the new secondary-failure primary event an adjusted repair rate that accounts for the repair of the secondary failures and the old primary event. Plant repair policy determines how to calculate the adjusted repair rate for the new secondary-failure primary event. For example, if the old primary event and secondary failures are simultaneously repaired, then the repair rate for the new secondary-failure primary event is given by

$$\mu_{npe} = \min(\mu_{ope}, \mu_{sf})$$

where the subscripts npe, ope, and sf stand for new secondary-failure primary event, old primary event, and secondary failures, respectively. If the old primary event is repaired after the simultaneous repair of secondary failures, then

$$\mu_{npe} = \left(\frac{1}{\mu_{ope}} + \frac{1}{\mu_{sf}}\right)^{-1}$$

Other repair policies may dictate other appropriate methods for calculating the repair rate for the new secondary-failure primary event.

After all the identified primary events have been transformed by this procedure, the system fault tree will represent the system repair tree. The primary events in this system fault tree, however, are not necessarily independent. This dependence among primary events results from common events in the development of secondary causes of component failure for the various malfunctions appearing in the logic model. To ensure correctness, these dependences must be accounted for if they occur.

6.5.2 TEST AND MAINTENANCE

Before quantification, accident-sequence cut sets are screened to eliminate those that are inconsistent with the accident-sequence definition. Thus, cut sets containing two or more test and maintenance primary events considered mutually exclusive because of noncoincident testing schedules or technical specifications are eliminated from the list of accident-sequence cut sets in a plant-damage bin. For the remaining cut sets that contain test and maintenance primary events, these events are assumed to be random and independent. If a cut set contains two or more test and maintenance primary events, however, the probability that these primary events occur simultaneously will often be greater than the value calculated by treating them as random and independent. In this case, the cut-set frequency can significantly increase because of the simultaneous occurrence of these primary events.

Simultaneous testing and maintenance can occur for any of several reasons. Components in separate systems may unknowingly be tested at the same time because of coincident testing schedules. For example, a pump in system A tested every 8000 hours and a pump in system B tested every 6000 hours might be simultaneously tested every 24,000 hours after the first simultaneous test. Human error that results in simultaneous testing and maintenance in violation of technical specifications is another cause. These and any other causes of simultaneous testing and maintenance must be accounted for to avoid underestimating the frequency of an accident sequence.

To illustrate the significance of simultaneous testing and maintenance, suppose a cut set contains two test primary events for two diesel generators. If testing is monthly and requires an hour, then the estimated testing unavailability of each diesel generator is 1.4×10^{-3}, and the unavailability contribution of the pair, assuming random and independent testing, is 1.9×10^{-6}. If, however, the two diesel generators are simultaneously tested once every 10 years in violation of technical specifications, then the simultaneous-testing unavailability of the pair is 1.1×10^{-5}.

The following procedure can be used to account for the effect of simultaneous testing and maintenance:

1. Identify the cut sets in a plant-damage bin that contain two or more test and maintenance primary events.

2. For each of these cut sets, replace the test and maintenance primary events with a single new test and maintenance primary event that represents the unavailability due to simultaneous testing and maintenance.

The probability to be used for the new simultaneous test and maintenance primary event is the fraction of time the replaced test and maintenance primary events occur simultaneously. If the simultaneous testing and maintenance results are in violation of technical specifications, then the probability for this new primary event is given by the product of the probability of violating technical specifications through simultaneous testing and maintenance and the probability of the replaced primary event with the shortest average test and maintenance time. For example, if the probability of violating technical specifications is .01 and the probabilities of three replaced test and maintenance primary events are .001, .0001, and .00001, then the probability for the single new test and maintenance primary event is 10^{-6}.

If the simultaneous testing and maintenance is due to coincident test schedules, then the probability of this new primary event is given by

$$\bar{A}_{npe} = \frac{\min\left(T_{av,i}\right)}{T_{period}} P$$

where $T_{av,i}$ is the average amount of time required to perform testing and maintenance on each replaced primary event i in the cut set, T_{period} is the time between coincident tests, and P is the probability the replaced primary events are tested and maintained at the same time during the test period.

Consider, for example, the coincident test schedules of the two pumps described earlier. If the average test and maintenance time for either pump is 4 hours and testing is to be performed within a 72-hour period, then the probability of the new simultaneous test and maintenance primary event for these two pumps is given by

$$\bar{A}_{npe} = \frac{4}{24,000} \frac{4}{72} = 9.3 \times 10^{6}$$

This assumes random testing of the pumps during the 72-hour period.

6.6 COMPUTER CODES

This section describes a number of computer codes currently available for the qualitative and quantitative evaluation of system or plant logic models. It is difficult to recommend a specific code for use in evaluating plant or system logic models. A great many codes or code packages are available, each code having some particular objective toward aiding or

improving the solution of complex models. This document does not endorse any of the computer codes described here.

Even for a particular function, it is difficult to reach a consensus on a given code because many different factors--such as available computer facilities, staff expertise, and the specific objectives of the analysis--affect the selection of computer assistance. Moreover, not all existing codes are described here--only those which are not proprietary and for which sufficient literature is available; also included are some codes whose owners provided related material and documents.

Some comments can be made, however, on the basis of experience with several of these codes. The code SETS, developed at Sandia National Laboratories, has wide applications in solving fault- and event-tree models as well as in searching for dependent failures. Being relatively sophisticated, it may require a considerable amount of computer time and knowledge of the code if its substantial capabilities are fully exercised. The WAM series, whose development was sponsored by the Electric Power Research Institute, also has broad applications and is readily usable.

The codes described here are divided into five groups by general function. Groups 1 through 4 are summarized in Tables 6-3 through 6-6. Group 1 consists of codes that perform the qualitative evaluation of a fault tree (i.e., codes that compute minimal cut and/or path sets). Group 2 contains codes for quantitative analyses. This group includes codes that require as input the structural information embodied in the cut sets and those that are designed to perform direct numerical evaluations of a system without computing cut sets as a necessary intermediate step. It also contains several codes that have special applications in quantitative analysis. The codes in group 3 have been developed to aid in the identification or analysis of dependent failures. Group 4 consists of codes that can perform uncertainty analyses through the input cut sets, system function, or fault-tree structure (i.e., provide confidence intervals for point estimates). Finally, group 5 contains all codes developed to aid data and other analyses. Because of their diversified functions, the codes in this group are not being presented in tabular form. Besides these five groups of codes, there is a group of codes that are proprietary and therefore not discussed in this guide.

6.6.1 COMPUTER CODES FOR THE QUALITATIVE ANALYSIS OF FAULT TREES

This section deals with codes that compute the minimal cut and/or path sets of a fault tree or perform Boolean reduction for the fault trees. Minimal cut sets give all the unique combinations of primary-events that cause system failure; minimal path sets give the smallest group of primary-event failures that must not occur in order for the system failure not to occur.

Minimal cut sets are used by some codes to evaluate fault trees. In particular, minimal cut sets are used by some codes (e.g., KITT and SUPER-POCUS) to calculate the probability, unavailability, or unreliability of the

top event. Some codes, such as SAMPLE and SPASM, use minimal cut sets for sensitivity or uncertainty analyses. Minimal cut sets are also used in some codes to perform importance calculations (e.g., IMPORTANCE). Finally, in dependent-failure analysis, some codes (e.g., COMCAN) use minimal cut sets for common-cause searching. The minimal cut sets themselves provide much useful information about the design weaknesses of the system. Furthermore, minimal cut sets can be compared with the original tree to identify possible errors in the fault-tree logic.

Two methods of calculating minimal cut sets are used in the codes. One is deterministic; the other is a Monte Carlo approach. The deterministic method uses Boolean-algebra principles to sort through the fault-tree structure, which must first be encoded in a suitable format. Although accurate and rigorous, this method can be slow for large fault trees. However, modern approaches like fault-tree modularization have made its use for large fault trees very feasible. The Monte Carlo approach randomly selects the events in the fault tree and combines them to test whether the fault-tree logic is satisfied. If an event combination is selected that does satisfy the logic, a cut set has been established. This method is less accurate but sometimes faster than the deterministic method. Both methods can be streamlined for more economical use by limiting the size of the fault tree to be examined or by setting a limit on the size of the output minimal cut sets. Further details about the methods for determining minimal cut sets can be obtained from the Fault Tree Handbook (Vesely et al., 1981a).

One disadvantage of the minimal-cut-set codes is that the storage and computer time for processing even medium-size trees can become quite prohibitive. The number of cut sets can increase drastically with a slight increase in the number of gates or primary events. For example, one tree with 299 primary events and 324 gates had more than 67 million cut sets. However, the number of gates and primary events is not the only indicator of the complexity of the tree, whereas the configuration of the gates and primary events is an important contributor to its complexity. Therefore, a fault tree with fewer gates and primary events than another tree can contain more cut sets, simply because it has a different logical or structural configuration. Thus, it is often difficult to predict the storage requirements and running time for a given tree.

Several methods can be used to overcome or at least alleviate the problem of obtaining all the minimal cut sets. The most common is to eliminate, during the processing, cut sets whose order (number of events in the cut set) is larger than a preselected number or whose probability is less than a specified value. In the Reactor Safety Study (USNRC, 1975), for example, only single- and double-event cut sets were retained; the higher-order cut sets were analyzed only for common-cause-failure potentials. Another method of alleviating the problem is to reduce and simplify the fault tree before generating cut sets; for example, the WAMCUT-II method substantially reduces the number of cut sets. Finally, tree modularization is sometimes used as an alternative method of reducing the number of cut sets (e.g., the PL-MOD method).

Presented below are brief descriptions of the qualitative-evaluation codes, including purpose, method, input and output, language and type of

computer, and special features. It should be noted that a number of the qualitative-evaluation codes have limited quantitative capabilities. These capabilities are also discussed. A summary of these codes is presented in Table 6-3.

ALLCUTS

ALLCUTS finds minimal cut sets from fault trees with AND and OR gates (Van Slyke and Griffing, 1975).

ALLCUTS uses a top-down successive Boolean substitution algorithm similar to that of MOCUS. An auxiliary program, BRANCH, can be used to check the input and cross reference the gates and input primary events.

Required input consists of control information and a description of the fault tree, but the code allows the option of entering primary-event probabilities. If these data are input, ALLCUTS can compute the top-event probability. Output from ALLCUTS can be printed. The cut sets are sorted, and up to 1000 minimal cut sets in descending order of probability can be generated and printed. Cut sets within a specified probability range can be obtained and printed.

The limited number of cut sets that ALLCUTS generates restricts its use, especially for large fault trees. ALLCUTS handles up to 175 primary events and 415 gate events. The code is very similar to MOCUS. If requested, ALLCUTS performs a limited search of minimal cut sets to identify common manufacturer, common susceptibility to secondary failure causes, and close proximity of primary events. ALLCUTS is written in Fortran IV for the CDC 7600 computer.

FATRAM

FATRAM is used to find minimal cut sets from fault trees with AND and OR gates (Rasmuson and Marshall, 1978).

The FATRAM algorithm is very similar to that of MOCUS, but it uses less core and less computation time. The reduction in core requirements is achieved by (1) resolving OR gates with gate inputs and AND gates as early as possible, (2) handling replicated primary events, (3) postponing until last the resolution of OR gates with only primary-event inputs, (4) writing out cut sets without expanding in core, and (5) eliminating supersets at very early stages.

Required input consists of control information and a description of the fault tree. Eight-character alphanumeric names can be used for the fault-tree events. The output consists of minimal cut sets up to a level specified by the user.

Most of the characteristics of FATRAM are similar to those of MOCUS. However, because of the improved methodology, larger fault trees can be handled with more efficiency. FATRAM has an input-error-checking procedure and is written in Fortran IV for the CDC Cyber 76 computer.

Table 6-3. Computer codes for qualitative analysis[a]

Code	Input	Limit on number of gates or events	Types of gates	Limit on number or size of cut sets[b]	Method of generating cut sets[a]	Other outputs	Fault-tree truncation	Other features	Type of computer, language, and availability
ALLCUTS	8-character alphanumeric names, control information, primary-event probability, fault-tree description	175 primary events and 425 gates	AND OR	Up to 1000 cut sets can be generated	Top-down successive Boolean substitution	Cut sets in specified probability range, cut set and top-event probability	Minimal cut sets, probability	Fault-tree plotting option	IBM 360/370 CDC 7600 Fortran IV
FATRAM	8-character alphanumeric names, control information, fault-tree description	None	AND OR	None	Top-down successive substitution with gate-coalescing option	Minimal cut sets up to specified order	Minimal cut sets	--	CDC Cyber 76 Fortran IV Available from EG&G Idaho, Inc.
FTAP	8-character alphanumeric names, control information, fault-tree description	None; computer memory is limiting factor	AND OR K-of-N NOT	Minimal cut sets of up to order 10 can be generated	Top-down, bottom-up, and Nelson method (prime implicants)	Minimal cut sets and prime implicants	Minimal cut sets	Independent subtrees automatically found and replaced by module	IBM 360/370 CDC 6600-7600 Fortran IV Available from Operations Research Center, University of California, Berkeley
MOCUS	8-character alphanumeric names, control information, fault-tree description	None	AND OR INHIBIT	Minimal cut sets of up to order 20 can be generated	Top-down successive Boolean substitution	Path sets	Minimal cut sets	Cut sets can be automatically punched on cards or on-line data sets for use by KITT or SUPERPOCUS	IBM 360/370 CDC 7600 Fortran IV Available from Argonne Software Center
PL-MOD	79-character alphanumeric names, control information, fault-tree description, failure data	None; computer memory is limiting factor	AND OR NOT K-of-N	None	Bottom-up modularization and decomposition of fault tree into best modular representation	Probability of top event, time-dependent characteristics of top event, minimal cut sets, uncertainty for top event	Minimal cut sets	Option of not generating minimal cut sets for quantifying fault tree	IBM 360/370 PL/I Available from Argonne Software Center
PREP	8-character alphanumeric names, control information, fault-tree description	2000 primary events and 2000 gates	AND OR INHIBIT	Minimal cut sets of up to order 10 can be generated	Combinatorial testing	--	No	Minimal cut sets can be automatically punched on cards or on-line data sets for use in KITT or SUPERPOCUS	IBM 360/370 CDC 7600 Fortran IV Available from Argonne Software Center
SETS	16-character alphanumeric names, user's program, failure data, fault-tree description	8000 events (gates and primary events together)	AND OR INHIBIT PRIORITY Exclusive or special	None	Top-down Boolean substitution, but user's program can be designed for any other method	Probability of minimal cut sets, prime implicants	Yes, based on both cut-set order and probability	Automatic fault-tree merging and plotting; on-line data sets can be stored on tapes for use in other runs; independent subtrees can be obtained to simplify cut-set generation	CDC 7600 Fortran IV Available from Argonne Software Center
SIFTA	10-character alphanumeric names, control information, failure data, fault-tree description	None; computer memory is limiting factor	AND OR K-of-N	No cut sets generated	Pattern-recognition technique to reduce structure of tree; numerical simulation to calculate probabilities	New structure of tree after reduction; probability of top event	Independent branches of tree with small probability	Handles trees with multiple top events; merging of fault trees possible; fault trees can be plotted	HP-1000 Available from Atomic Energy Control Board, Ottawa, Canada

Table 6-3. Computer codes for qualitative analysis[a] (continued)

Code	Input	Limit on number of gates or events	Types of gates	Limit on number or size of cut sets[b]	Method of generating cut sets[a]	Other outputs	Fault-tree truncation	Other features	Type of computer, language, and availability
TREEL and MICSUP	8-character alphanumeric names, control information, fault-tree description	None; computer memory is limiting factor	AND OR INHIBIT	Minimal cut sets of up to order 10 are generated	Top-down successive Boolean substitution	Path sets	Minimal cut sets	Can determine minimal sets of intermediate gates	CDC 6400 Fortran IV Available from Operations Research Center, University of California, Berkeley
WAMCUT, WAMCUT II	10-character alphanumeric names, control information, failure data, fault-tree description	1500 primary events and 1500 gates	AND OR NOT NOR NAND ANOT ONOT K-of-N	Up to 2000 minimal cut sets of any order can be generated	Bottom-up Boolean substitution; WAMCUT-II finds independent subtrees, replaces them by pseudo-component, then uses top-down Boolean substitution	Probabilities of minimal cut sets and top event; first and second moments of minimal cut sets and top event	Yes, based on both cut-set order and probability	Plot option; can generate minimal cut sets of intermediate gates	CDC 7600, IBM 370 Extended Fortran IV Available from EPRI Code Center

[a]All the codes listed here have routines for checking input errors. These routines are very extensive in the codes FTAP, MOCUS, PREP, SETS, SIFTA, TREEL-MICSUP, and WAMCUT. ALLCUTS uses the auxiliary code BRANCH for checking input errors.

[b]Or prime implicants.

FTAP

FTAP, used to obtain minimal cut and path sets (Willia, 1978), determines minimal cut sets of any order for fault trees with AND, OR, K-of-N, and NOT gates.

The FTAP algorithm is based on one of three methods selected by the user: top-down, bottom-up, and Nelson. The top-down and the bottom-up approaches are basically akin to the methods used in MOCUS and MICSUP, respectively; the Nelson method is a prime-implicant algorithm that is applied to trees containing complement events and uses a combination of top-down and bottom-up techniques. FTAP uses two basic techniques to reduce the number of nonminimal cut sets, thereby increasing the code's efficiency. The first technique, used in the bottom-up and Nelson methods, is modularizing independent portions of the tree; it is somewhat similar to the SETS algorithm. The second technique, used in the top-down and Nelson methods, is called the "dual algorithm." In this algorithm, the product of sums is transferred to the sum of products whose dual is then taken by using a special method. It is claimed that the nonminimal sets appearing during the construction of the dual "will always be less than the number of such sets in [the original product of sums], usually many times less."

The input information required by FTAP consists of control information and a description of the fault tree. Eight-character alphanumeric names are used for the events in the fault tree. The output, which can be printed, includes the list of minimal path and cut sets and, where applicable, the list of prime implicants.

The code is able to generate cut sets of high order with high efficiency. Flexibility in the use of one of the three algorithms provides a tool to more efficiently evaluate large fault trees. FTAP has an extensive error-checking procedure. Written in Fortran IV and assembly language, FTAP can be used with the CDC 6600/7600 and the IBM 360/370 computers.

MOCUS

MOCUS is used to find minimal cut or path sets from fault trees with AND, OR, and INHIBIT gates (Fussell et al., 1974). Written to replace PREP as a minimal-cut-set generator for the KITT codes, it can determine minimal cut sets of up to order 20 (maximum length specified by the user).

The MOCUS algorithm uses successive Boolean substitution, starting from the top event and working down the tree until all gates have been replaced by primary events. If the tree contains no replicated events, the end result of the substitution is minimal cut or path sets; otherwise Boolean identification should be applied to minimize the cut or path sets.

Required input consists of control information and a description of the fault tree. Optional input includes eight-character alphanumeric names for primary events, primary-event failure rates, and primary-event repair times.

MOCUS output can be printed, punched on cards, or written to either a temporary or permanent on-line set. The list of minimal cut sets can be

passed to the quantitative codes SUPERPOCUS or KITT-1 and KITT-2 from the punched cards or from on-line data sets.

The computer time required by MOCUS is approximately a linear function of the order of the cut sets desired. However, large fault trees require a prohibitive amount of time to generate cut sets of high order (say four events or more). Since it does not handle NOT gates and special gates, MOCUS is somewhat limited in use, but it is very efficient in determining low-order cut sets. In addition to the top event, the cut sets of intermediate gates can be obtained. MOCUS is written in Fortran IV for the IBM 360/370 and the CDC 7600 computers. It has an extensive routine for checking input errors and requires no external routine.

PL-MOD

PL-MOD directly obtains modular minimal cut sets of any length for fault trees developed with AND, OR, NOT, and K-of-N gates (Olmos and Wolf, 1977). Modular minimal cut sets make fault-tree quantification very simple.

The PL-MOD algorithm is based on fault-tree decomposition and modularization. A module is a collection of primary events that are independent of the result of the tree and can be replaced by a "supercomponent" (i.e., the module). PL-MOD separates all replicated events from the rest of the tree, modularizes the independent portion of the tree, and then finds Boolean relations between the replicated events and the modules. The Boolean relation is reduced and presented in the disjunctive normal form that is the modular minimal cut set. The code MODCUT is used (Modarres et al., 1980) to expand the modular minimal cut sets to obtain the minimal cut sets. MODCUT determines minimal cut sets of any length (maximum length specified by the user).

Required input is control information and a description of the fault tree; optional input includes up to 79-character alphanumeric names for the primary events and the gates of the fault tree, primary-event failure rates, repair rates, and average test length. The input is free in format. Output from PL-MOD or MODCUT can be printed; it includes the list of modular minimal cut sets and minimal cut sets as well as the probabilities of primary events, modules, and gates.

An option of time-dependent analysis (the PL-MODT code) calculates system unavailability at different times. Fussell-Vesely importance calculations can be performed for all primary events, modules, and modular cut sets, and a Monte Carlo simulation option is available for uncertainty analysis. The Monte Carlo simulation subroutine is similar to SAMPLE, but, because PL-MOD uses modular cut sets for quantification, the calculations are more efficient. Because PL-MOD is written in PL/I language, it has the disadvantage of machine dependence (PL/I is not available in many computer systems) and lack of familiarity with PL/I language among scientific users. However, the use of the code is very simple and straightforward.

PREP

PREP obtains minimal cut or path sets from fault trees with AND, OR, and INHIBIT gates (Vesely and Narum, 1970). It can determine minimal cut sets up to order 10 (maximum length specified by the user).

PREP consists of two parts: TREBIL and MINSET. TREBIL (for "tree build") takes the user's input description of a fault tree and builds a Fortran subroutine of the Boolean equations representing the fault tree. MINSET then uses the TREBIL-produced subroutine to determine cut and/or path sets. MINSET uses combinatorial testing to find the minimal sets. For example, it systematically fails all single primary events, pairs of primary events, groups of three primary events, etc., to find which combinations cause the top event of the fault tree to occur. The time required for the analysis is an exponentially increasing function of the average length of the desired cut sets.

Required input consists of control information and a description of the fault tree. Optional input includes eight-character alphanumeric names for primary events, primary-event failure rates, and primary-event repair times. Most of the PREP input is identical with the input to MOCUS. Output can be printed, punched on cards, or written to either a temporary or a permanent on-line data set. The list of minimal sets can be passed to the quantitative codes SUPERPOCUS, KITT-1, or KITT-2 from the punched cards or from on-line data sets.

The main disadvantage of PREP is that it requires a prohibitive amount of computer time for large-order cut sets (more than, say, three events). Moreover, PREP does not handle NOT gates and special gates, which makes the use of PREP somewhat limited. However, for obtaining cut sets of low order (up to, say, three events), PREP is very efficient. The primary events are assumed to be independent; unlimited replicated events are allowed; cut and path sets of intermediate gates cannot be generated. Written in Fortran IV for the IBM 360/370 and the CDC 7600 computers, PREP has an extensive routine for checking input errors and requires no external routine.

SETS

The SETS (Set Equation Transformation System) code, developed by Sandia National Laboratories, is a general program for the manipulation of Boolean equations to find minimal cut or path sets (Worrell and Stack, 1978). It finds cut sets of any length (the maximum length can be specified by the user) for fault trees with AND, OR, NOT, or special gates (specified by the corresponding Boolean equation).

SETS is run by using a user's program designed by the user. The user's program must be so set up that the fault tree is evaluated efficiently, and it largely determines the evaluation algorithm. In general, two major algorithms are used. The first substitutes the Boolean equation of each for the top to the lowest branches of the tree. The second identifies independent subtrees, replaces them by a module, and then performs a simple substitution of the Boolean equation from top to bottom. By manipulating the user's program, these two algorithms can be applied first to intermediate gates and then to higher-order gates, which causes a bottom-up solution of the tree.

Required input consists of the SETS user's program and a description of the fault tree. Input events (gates and primary events) can have up to 16-character alphanumeric names. The input is free in format, which makes its preparation very simple. The output can be printed or stored on tape or

disk for further use. For example, if cut sets are obtained for an intermediate gate and stored on tape, in another SETS run one can read the equation from this tape to solve the Boolean equation of higher-order gates. The list of minimal sets can be passed to the SEP code (Olman, 1981) for uncertainty or importance analysis. The SEP code gives a powerful quantitative capability to SETS.

SETS has an option of logical merging for fault trees. This is very useful when systems in the event trees (i.e., front-line systems) must be merged with their support systems. Steady-state probability calculations are performed by SETS and make it possible to truncate the Boolean equation by probability or cut-set order. SETS can handle up to 8000 events (gates and primary events), which makes it capable of handling very large fault trees. Its main disadvantage is that an efficient fault-tree evaluation is highly dependent on a right setup of the user's program, which requires extensive knowledge and experience on the part of the user. SETS has been used in several PRA studies conducted in the Interim Reliability Evaluation Program and the Reactor Safety Study Methodology Applications Program. A plot code can be used to plot the fault trees on a Calcomp plotter from the input fault-tree description of the SETS output tape. SETS is written in Fortran IV for the CDC 7600 computer. An extensive routine for input-error checking is available.

SIFTA

SIFTA (Simple Fault Tree Analysis) performs logical restructuring and reduction, and probability calculation on fault trees with AND, OR, and K-of-N gates (Waddington and Wild, 1981).

The algorithm for evaluating fault trees is not based on the traditional generation of cut sets: a pattern-recognition technique restructures the fault tree by using relations between the laws of Boolean algebra and regrouping certain patterns of events. For example, if replicated event C is input to two different OR gates A and B, and A and B are input to AND gate T, then event C can be taken out from A and B and put directly into gate T. Hence, the restructuring makes the replicated event C a regular primary event. Several other more complex patterns are recognized and changed to a new reduced form. The numerical evaluation starts with the direct calculation of independent branches, followed by a calculation of branches made independent through the reduction of common events. If the structure of the tree does not allow full reduction, the residual tree is processed by simulation. Unavailability is simulated by failure in a proportionate number of trial periods of 0.01 year. The simulation terminates after 10 occurrences of the top event.

Input is very simple and free in format. Events in the tree can be input by up to 10-character alphanumeric names. The output can be displayed on the terminal and includes the new structure of the tree and the numerical results.

SIFTA is a code that is simple to use and can handle trees with multiple top events. Merging of fault trees is possible during interactive inputting of the fault tree. There is thorough error checking. Noteworthy is

the ability to untangle loops caused by errors in logic or by the misspelling of event codes. SIFTA can plot the fault tree by first displaying it on the screen of the terminal. The displayed tree can be edited and plotted fully or partially by a multicolor plotter. The SIFTA method, however, is highly dependent on the logical structure of the tree. It would be very difficult for the code to handle large trees (1000 events and up) or trees with a significant number of dependences. SIFTA is written for HP-1000 computers but is being implemented on a CDC computer.

TREEL AND MICSUP

TREEL and MICSUP are used to obtain minimal cut or path sets from fault trees with AND and OR gates (Pande et al., 1975). They find minimal cut sets of up to a specified order (maximum length specified by the user).

The algorithm is similar to that used in MOCUS, except that, working from the top event down, MICSUP (MInimal Cut Set UPward) starts with primary inputs of the lowest-level gate and works upward to the top event. TREEL is a preprocessor that checks the tree for errors and determines in advance the maximum number and order of the cut and path sets.

Required input consists of control information and a description of the fault tree; optional input includes eight-character alphanumeric names for the primary events in the fault tree. The output can be printed. This output includes the list of minimal path and cut sets.

In most of their characteristics, TREEL and MICSUP are similar to MOCUS. However, because of the bottom-up algorithm used in MICSUP, the path and cut sets of intermediate gates can be more easily found. TREEL has an extensive error-checking procedure. TREEL and MICSUP are written in Fortran IV for the CDC 6400 computer.

WAMCUT

WAMCUT is used to obtain minimal cut sets and to quantify the gates and top events of fault trees (Leverenz and Kirch, 1978). It finds cut sets of any length for fault trees with AND, OR, NOT, NOR, NAND, ANOT, ONOT, and K-of-N gates.

WAMCUT consists of two parts: WAM and CUT. WAM is a preprocessor that reads the fault-tree description and checks for logic and syntax errors. CUT is the cut-set finder routine, which takes the restructured input fault tree from WAM and finds the cut sets of each gate, working from the bottom to the top of the tree.

Required input consists of control information and a description of the fault tree; optional input includes 10-character alphanumeric names for primary events and gates, primary-event probabilities, and the number of cut sets to be generated. Output includes a list of cut sets and the probability of each. The cut sets can be saved for use in SPASM for uncertainty analysis.

WAMCUT is very easy to use. It can process large fault trees with up to 1500 gates and 1500 primary events without large expenditures of computer

time. The number of cut sets per gate is limited to 2000. Probability truncation of cut sets makes the code practical for PRA. Written in Extended Fortran IV for the CDC 7600 computer, WAMCUT has been used in several PRA studies.

WAMCUT-II

WAMCUT-II, an advanced version of WAMCUT, is used to obtain minimal cut sets and to quantify the gates and top events of fault trees (Putney and Kirch, 1981). It finds cut sets of any length and handles the types of gates handled by WAMCUT.

WAMCUT-II evaluates the fault tree by restructuring the logic to obtain and replace independent portions (independent subtrees) with pseudocomponents and to optimize the tree, thus reducing the amount of processing. Events that are input to several gates (replicated events) are moved up as far as possible toward the top of the tree without violating Boolean-algebra rules. After the tree is restructured, cut sets are obtained with a top-down algorithm. The fault-tree minimal cut sets are in terms of the independent subtrees. This form of cut sets is superior to the minimal cut sets of primary events for quantitative calculations. However, if requested, WAMCUT-II can expand the minimal cut sets of independent subtrees to obtain the cut sets of primary events. In certain cases this process can be costly and may be unnecessary.

Input to WAMCUT-II is the fault-tree description and, optionally, the probability of failure for each primary event. The output consists of the fault-tree minimal cut sets and failure probabilities for intermediate gates and the top event.

WAMCUT-II is very similar to WAMCUT in its capabilities, but is usually much faster. The process of restructuring the tree and removing the independent subtrees can reduce the running time considerably. WAMCUT-II is written in Fortran IV for the CDC 7600 computer; it is currently being converted to run on the IBM 370 computer.

6.6.2 COMPUTER CODES FOR QUANTITATIVE ANALYSIS

A variety of codes have been developed for the quantification of accident sequences. Like the qualitative codes discussed in Section 6.6.1, each one has its own uses, and which one is used depends greatly on the size and the complexity of the fault tree. Most of the quantitative programs discussed here are used to make point estimates of the probability of fault-tree top events; several codes, however, have special applications or characteristics. A summary of the quantitative-analysis codes is presented in Table 6-4.

Codes developed for calculating point-estimate probabilities indicate the relative safety of the system by establishing a probability for the top event. A point-estimate code should be capable of describing the relative safety of the system with a numerical value, and it should provide a list of probabilities associated with the dominant minimal cut sets or primary

Table 6-4. Computer codes for quantitative analysis

Code	Input	Quantitative calculations	Importance calculation	Other features	Type of computer and availability[a]
FRANTIC, FRANTIC II	Reduced system equation or minimal cut sets, primary-event failure data	Time-dependent calculation; nonrepairable, monitored, and periodically tested primary events are handled; uncertainty analysis for failure rates in conjunction with time-dependent calculation	No	Can model human-error and dependent-failure contributions; FRANTIC II can handle time-dependent failure rates and incorporates effect of renewal on aging	IBM 360/370 Available from Argonne Software Center
GO	GO chart[b] and fault-tree failure data	Only time-independent calculations for gates and top event; nonrepairable or periodically tested primary events are handled	No	Cut sets for selected gates and probability truncation of cut sets up to order 4	CDC 7600 Available from EPRI Code Center
ICARUS	Reduced system equation, choice of testing scheme, failure data	Average unavailability, optimal test interval, relative contributions of testing, repair, and random failures	No	Three testing schemes available: random testing, uniformly staggered testing, and nearly simultaneous testing	IBM 360/370 Available from Argonne Software Center
IMPORTANCE	Minimal cut sets, primary-event failure data	Top-event point-estimate probability or unavailability	Can calculate the following: Birnbaum, criticality, upgrading function, Fussell-Vesely, Barlow-Proschan, steady-state Barlow-Proschan, sequential contributory	Can rank cut sets and primary events on basis of each importance measure	CDC 7600 Available from Argonne Software Center
KITT-1, KITT-2	Minimal cut sets supplied directly or by MOCUS or PREP; primary-event failure data	Time-dependent unavailability for primary events, minimal cut sets, and top event; failure rate, expected number of failures, and unreliability for top event and minimal cut sets	Fussell-Vesely importance calculations for primary events and minimal cut sets	KITT-2 allows each component to have unique time phases and thus failure and repair to vary from phase to phase	IBM 360/370 CDC 7600 Available from Argonne Software Center

Table 6-4. Computer codes for quantitative analysis (continued)

Code	Input	Quantitative calculations	Importance calculation	Other features	Type of computer and availability[a]
RALLY	Fault-tree description, control information, failure data	Average unavailabilities and failure frequencies for top event; time-dependent calculation possible through use of minimal cut sets; uncertainty analysis possible by using minimal cut sets; normal, lognormal, Johnson, extreme value-1, Weibull, gamma, and exponential distributions are handled	Code CRESSEX in RALLY can perform importance calculations	Can handle up to 1500 components and 2000 gates; can determine minimal cut sets using either a simulative or analytical way	IBM 360/370
RAS	Fault-tree description or minimal cut sets; failure and repair rates	Time-independent unavailability, expected number of failures, and frequency of top event	No	Phased-mission analysis possible; if fault tree is input, minimal cut sets will be calculated	CDC 7600 Available from Argonne Software Center
SUPERPOCUS	Minimal cut sets, component failure data, time at which calculations are performed	Time-dependent unavailability, reliability, and expected number of failures for minimal cut sets and top event	Yes	Ranks minimal cut sets on basis of importance; can read cut sets directly from MOCUS or PREP	IBM 360/370 CDC 7600 Available from Dept. of Nuclear Engineering, University of Tennessee
WAM-BAM	Fault-tree description, primary-event failure data	Point unavailability for top event and intermediate gates; no time-dependent analysis possible	No	Extensive error checking possible through WAM; probability truncation of fault tree; sensitivity analysis possible by using WAM-TAP preprocessor instead of WAM	CDC 7600 Available from EPRI Code Center

[a]All the codes listed here are written in Fortran IV.

[b]A GO chart (see Section 3.6.3) is a chart that resembles a schematic of system primary events and their relations via a set of 16 Boolean operators.

events that contribute to system failure. Other quantitative results that are calculated by these codes are importance measures for primary events, minimal cut sets, and modules of the tree; sensitivities; and time-dependent unavailability or reliability. The application of these calculations is discussed elsewhere in this guide.

Computer codes that perform quantitative analyses can be divided into two major groups: those requiring minimal cut sets as input, and those able to perform analyses without computing cut sets as a necessary intermediate step. The latter are called direct-evaluation codes.

FRANTIC

The FRANTIC (Formal Reliability Analysis including Testing Inspection and Checking) code computes the average and time-dependent unavailability of any general system model like a fault tree (Vesely and Goldberg, 1977). It can be used to assess the effects on system unavailability due to test downtimes, repair times, test efficiency, test bypass capabilities, test-caused failures, and different test staggerings. The primary events handled by FRANTIC are primary events involving periodically tested, nonrepairable, and monitored components; human-error and dependent-failure contributions can also be modeled.

FRANTIC uses a system equation that represents the general system model much as a fault tree does. The system equation must be formulated by the user before the FRANTIC run. The primary events of the system equation are assigned an exponential distribution to describe hardware failures. At different instants of time the unavailability associated with each primary event is calculated. A Monte Carlo version of FRANTIC can be used to input sampling distributions for primary-event failure rates.

The input to FRANTIC consists of the system equation, primary-event failure rates, and test and repair characteristics; other inputs include the time period for the calculations as well as print and plot options. The output consists of system unavailability at different instants of time and, if requested, Calcomp plots of the time-dependent system unavailability.

A second version of the code, FRANTIC II, has been developed to enhance the capability to model the time-dependent unavailability of primary events and systems over their total in-service lifetime (Vesely et al., 1981b). The effects of the initial burn-in period, the time region of normal operation, and finally the wearout period can be modeled (the bathtub model). For this FRANTIC uses the Weibull distribution, which has a time-dependent failure rate. In addition, FRANTIC II allows the investigation of discontinuous changes in the failure rate as a function of the number of tests performed. This is essentially a demand-related, rather than a time-related, burn-in and wearout model. FRANTIC II also incorporates the effects of renewal on aging by introducing "good as new" or "good as old" primary events.

The FRANTIC and FRANTIC II codes are very simple to use. There is essentially no limit on the number of primary events in the system equation, but the construction of a system equation for a large system containing a large number of primary events is a nontrivial task. FRANTIC and FRANTIC II are written in Fortran IV for the IBM 360/370.

GO

GO (see Section 3.6.3) calculates the probabilities of all operating and nonoperating states for a system (Gateley et al., 1968). It uses a set of standardized functional operators to model physical primary events with mathematical entities that are easily identified as primary events. The modeling technique produces the GO chart, which corresponds closely to the physical layout, diagram, or schematic.

In the modeling procedure, 16 GO operators are used. Some of them are similar to fault-tree gates, but in addition to logic functions, time delays and switches can be modeled as well as complementary events and mutually exclusive states. The development of the GO chart consists of selecting the functional operators and connecting them with arrows to represent the flow of information. The GO code performs the logical connections and generates the minimal cut sets.

Required input is the GO chart and probabilities associated with the possible operational modes of each primary event, which is analogous to applying probabilities for the primary events of a fault tree. The output consists of probabilities for several output events in several operating states. In addition, cut sets of up to order 4 are generated.

Like WAMCUT, the GO code reduces storage requirements by eliminating low-probability paths at an intermediate stage of the processing and at the same time keeps track of the total of the discarded path. Because of the diversity and detail of the GO operators and the need to include all system primary events, the modeling process is quite complex. Furthermore, a change in probabilities often requires a complete rerun. However, the GO chart can be useful for design and system engineering.

ICARUS

The code ICARUS (Vaurio and Sciandone, 1979) is capable of calculating the average unavailability, optimum test interval, and relative contributions of testing, repair, and random failures for any one of three testing schemes: random testing, uniformly staggered testing, and nearly simultaneous testing.

ICARUS was developed to handle only primary events involving periodically tested components that are constantly unavailable, nonrepairable, or monitored. It is capable of calculating the average unavailability only in the asymptotic state. Consequently, the user must choose one of the three available testing schemes rather than create a particular testing scheme through the input. ICARUS evaluates the average unavailability analytically, and in this regard it is capable of calculating the optimum test interval by direct differentiation.

The input consists of the choice of the testing scheme and various failure rates, testing downtime, and probabilities of failure detection for the primary events in the system under study. The output consists of the testing scheme and failure-mode probabilities specified by the user and the average unavailability, optimum test interval, and average unavailability

at the optimum test interval for the system. Also provided are average unavailability contributions due to testing, repair, and random failures.

The advantage of ICARUS over a similar code like FRANTIC includes the use of analytical treatment for calculating unavailability, which avoids any inherent numerical error. It also includes three failure modes not found in FRANTIC: failure to start on a true demand, failure to detect a failure, and failure to repair a failure. Disadvantages include the ability to handle only periodically tested primary events and the restriction to only three testing schemes. ICARUS is written in Fortran IV for the IBM 360/370.

IMPORTANCE

IMPORTANCE was developed to rank primary events and cut sets according to various available importance measures (Lambert and Gilman, 1977). It is capable of handling fault trees with time-dependent primary events under the assumption that primary events are statistically independent and that their failure and repair distributions are exponential in time.

The importance measures that are included in the IMPORTANCE code are as follows:

1. Birnbaum (Birnbaum, 1969).
2. Criticality (Lambert, 1975).
3. Upgrading function (Lambert, 1975).
4. Fussell-Vesely (Fussell, 1975).
5. Barlow-Proschan (Barlow and Proschan, 1975).
6. Steady-state Barlow-Proschan (Barlow and Proschan, 1975).
7. Sequential contributory (Lambert, 1975).

The input is a list of cut sets and primary-event failure data. (The cut sets generated by FTAP or SETS can be received directly.) Input events can have up to eight-character alphanumeric names. The output consists of the probability, importance, and ranking of top events, primary events, and cut sets on the basis of one or more of the above-mentioned measures.

IMPORTANCE is written in Fortran IV for the CDC 7600. It has been used in several PRA studies.

KITT-1 and KITT-2

KITT-1 and KITT-2 are used for the quantitative reliability analysis of systems. They calculate time-dependent reliability characteristics for primary events, minimal cut sets, and the top event. The calculated characteristics include unavailability, failure rate, expected number of failures, unreliability, and importance.

The KITT codes calculate conservative approximations of the top-event reliability characteristics or can be used to bracket these characteristics. If the bracketing is carried to completion, the exact values are calculated. KITT-1 assumes that the primary events have exponentially distributed times to failure and constant repair times (if the event is repairable). Inhibit conditions have a constant probability of occurrence.

KITT-2 assumes that the primary events have exponentially distributed times to failure and repair (if repairability is applicable). The parameters in these distributions (failure and repair rates) can be changed at times specified by the user. Primary events can also be assigned probabilities of being failed initially if the assumption is not made that they are working at time zero. Inhibit conditions have a constant probability.

Required input consists of control information, primary-event information (failure rates, repair times, and optional names), time points at which the characteristics are calculated, and the list of minimal cut sets. The control parameters control the bracketing options and allow multiple-parameter runs to be performed. Each parameter run uses the same minimal sets, but if one or more of the primary-event failure rates or repair times are changed, the reliability characteristics are recalculated. The output from KITT-1 and KITT-2 consists of unavailability, unreliability, and the expected number of occurrences for primary events, minimal cut sets, and the top event; failure rates for minimal cut sets and the top event; and importance for primary events and minimal cut sets.

KITT-1 and KITT-2 are written in Fortran IV for the IBM 360/370 and the CDC 7600 computers. Some input-error checking is available. The cut sets can be input directly from PREP or MOCUS. However, for large fault trees the use of the KITT codes would be very limited. No external routine is required for running these codes.

RALLY

The code package RALLY, developed by the Gesellschaft fuer Reaktorsicherheit (1978), is capable of evaluating fault trees with up to 1500 primary events and 2000 gates, including AND, OR, NOT, and K-of-N gates.

RALLY consists of the codes TREBIL, TIMBER, CRESSEX, FESIVAR, CRESSC, CRESSCN, SLAP-MP, KARI, and STREUSL. TREBIL was based on the PREP preprocessor for fault-tree synthesis, optimization, and data acquisition for the other programs of the RALLY package. TIMBER plots the fault tree. CRESSEX calculates the average unavailability and failure frequency by means of Monte Carlo simulation. FESIVAR is similar to CRESSEX except that it is capable of performing importance calculations. Minimal cut sets are calculated by either a simulation method with CRESSC (or CRESSCN if the fault tree contains NOT gates) or an analytical method with the programs SLAP-MP and KARI. STREUSL performs a time-dependent fault-tree quantification based on the minimal cut sets of the tree. The average unavailability and failure rate are calculated by the AVAGS code, which is used in conjunction with STREUSL. For uncertainty calculations the following distributions are handled by AVAGS: normal, lognormal, Johnson-SL, extreme value 1, Weibull, gamma, and exponential. RALLY was used in the German Risk Study and is written in Fortran IV for the IBM 360/370.

RAS

RAS (Reliability Analysis System) is an integrated package of computer codes for the quantification of fault trees (Rasmuson et al., 1977). It is based on MOCUS, POCUS, KITT-1, SRTPAN, and COMCAN. The package can be used for an entire system analysis with up to five system phases, or it can be

used to do only one section of the analysis, such as cut-set determination or fault-tree reduction.

The amount of input required for RAS depends on whether all capabilities will be used or if only one or two tasks will be required. Either the fault-tree logic equation or the cut sets can be input. Failure-rate data (exponential distribution assumed), repair rates, and mission times are required for calculating unreliability. If more than one mission is desired, this information must be input for each phase. It should be noted that the phases refer to system phases; cut sets or fault-tree equations can be different for each phase. In this RAS differs from KITT-2, which allows the phasing of primary-event information only and cannot readily handle system phases.

The output of RAS can be in printed, punched-card, or file-storage form. The input information is printed out, which allows for easy error detection. The specific information output depends on the options called within the program. The output from cut-set algorithms includes a fault-tree summary, the number of cut sets of each size. A listing of the cut sets is obtained by requesting it as an input option.

If KITT-1 or POCUS is called, reliability characteristics--including availability, expected number of failures, and failure rates--are listed for the top event. This information can be output for the cut sets and primary events by exercising an option. POCUS also has the option of ranking the primary events and cut sets by their unavailability importance.

When phased missions are required, the mission cut sets can be reduced by a cancellation subroutine and mission unreliability bounds can be optionally calculated. The output from the bounds calculation includes the time-dependent upper bound on mission unreliability for each phase.

RAS is convenient to use because one package can perform a wide range of options without additional input, and several codes that accomplish different functions are included. However, several of these codes have long running times, and their use may not be justified. It should also be noted that, as these codes are relatively new, there is little actual operating experience, which could affect the overall practicality of their use. The RAS package is written in Fortran IV for the CDC 7600 computer.

SUPERPOCUS

SUPERPOCUS is used for quantitative reliability analyses (Fussell et al., 1977). It calculates time-dependent unavailability, unreliability, unavailability importance, and unreliability importance for primary events and ranks the primary events by importance. For the minimal cut sets, this code calculates unavailability, reliability, expected number of failures, unavailability importance, and unreliability importance. The calculated top-event characteristics are unavailability, expected number of failures, and failure rates.

SUPERPOCUS is very efficient because it uses a tightly bounding approximation method. The approximations always overpredict failure characteristics. Primary events are assumed to have exponentially distributed

times to failure and repair (if the event is repairable). Inhibit conditions have a constant probability of occurrence.

Input consists of control information, primary-event information (failure rates, repair times, and optional names), time points at which the characteristics are calculated, and the list of minimal cut sets. The control parameters can be used to edit the output. Any or all of the output information can be suppressed except the top-event information.

The SUPERPOCUS algorithm is superior to that of the KITT codes. It accepts the initial unavailability for primary events. It can read the cut sets directly from MOCUS or PREP output. SUPERPOCUS is written in Fortran IV for the CDC 7600 and the IBM 360/370 computers. A routine for input-error checking is available, and no external routine is required.

WAM-BAM

WAM-BAM, which calculates point probabilities for the top events (Leverenz and Kirch, 1976), actually consists of four codes: WAM, WAMTAP, BAM, and CUT. WAM and WAMTAP are input preprocessors for the evaluation code BAM (Boolean Arithmetic Model). The WAM preprocessor is designed to ease the input description of the fault tree and the event probabilities. If requested, the input to BAM can be saved and subsequently modified by WAMTAP. WAMTAP allows the probability of single or grouped primary events to be changed for sensitivity studies. The use of WAM-CUT has been already discussed in this section.

The evaluation code BAM uses a combination of concepts from the GO method and fault-tree analysis: it uses the GO computational scheme but models the operations as gates on a fault tree. The probability of the top event is computed by forming a truth table, each line of which represents a product term (P-term) event disjoint from all the other P-terms. The product of the probabilities of the event in each P-term gives the probability of the P-term, and the union of the applicable P-term gives the probability of the top event. Like the GO code, WAM-BAM keeps track of the total probability of the discarded path during a probability truncation.

WAM-BAM is very easy to use. However, the new version of WAMCUT may be faster and more efficient for calculating top-event probabilities. WAM-BAM is written in Fortran IV for the CDC 7600 computer.

6.6.3 CODES FOR UNCERTAINTY ANALYSIS

Uncertainty analyses are important parts of PRA studies because of the statistical uncertainty in the failure and event-frequency data. To model statistical uncertainties first, failure and initiating-event frequency-data distributions are selected. Then, based on the logical relationship (e.g., cut sets) of these distributions, they are combined.

The computer codes developed to deal with uncertainty analysis can be divided into two categories: codes that perform the analysis through Monte

Carlo simulation (e.g., SAMPLE and STADIC-II) and codes that perform the analysis by mathematically combining the distributions (e.g., BOUNDS and SPASM). Most of the uncertainty programs can handle a variety of statistical distributions, normal, lognormal, uniform, and empirical distributions being most commonly used. However, some of these codes use more sophisticated distributions, such as the beta, gamma, Student-t, and Johnson distributions. Table 6-5 presents a summary of the codes for uncertainty analysis.

BOUNDS

BOUNDS is used to find probability intervals of system unavailability (Lee and Apostolakis, 1976). Multiple system functions with multiple data input descriptions can be processed in one run.

In the first step of the procedure BOUNDS computes, from primary-event failure rates, the first two moments of primary-event probabilities. Next it uses the information on minimal cut sets to obtain the moments for the occurrence probabilities of the minimal cut sets. From this last step, it calculates the moments of the top-event probability. It matches these moments to produce the Johnson-type distribution that possesses the same moments and then uses the fitted distribution to obtain the uncertainty bounds of the top event.

In input and output BOUNDS is similar to SAMPLE. The code can handle up to 1000 primary events. The number of input minimal cut sets is limited to 500, and it is assumed that there are no more than 5 primary inputs in any minimal cut set. BOUNDS is written in Fortran IV for the IBM 360/370 computer.

MOCARS

MOCARS is a Monte Carlo code for determining the means, the standard deviation, and distribution for fault-tree models (Matthews, 1977). It is essentially the same as SAMPLE, with added capabilities. The method for Monte Carlo simulation is the same as that in SAMPLE, but MOCARS can also handle the following distributions: exponential, Cauchy, Weibull, Pearson type IV, and empirical.

Input is a system-unavailability function specified either in Fortran statements or in terms of cut sets. The output is similar to that of SAMPLE, but MOCARS has the additional option of microfilm plotting with the integrated graphics system and the ability to perform a Kolmogorov-Smirnov goodness-of-fit test. This test shows whether the output distribution resembles a normal, lognormal, or exponential function. The probability distribution for the top event of the fault tree can be plotted as an optional output.

MOCARS is no more complex to use than SAMPLE. Its extra capabilities (e.g., plotted output) give it advantages. Because of the added capabilities, MOCARS is considered applicable to PRA programs and could be more useful than SAMPLE. It is written in Fortran IV for the CDC 7600 computer.

Table 6-5. Computer codes for uncertainty analysis

Code	Input	Method of uncertainty analysis	Type of statistical distribution	Other features	Type of computer and availability[a]
BOUNDS	Reduced system equation or minimal cut sets, primary-event failure data	Mathematical combination of uncertainties; output includes two moments of minimal cut sets and the top event	Johnson, empirical	Can handle multiple system functions with multiple data input descriptions; can fit Johnson-type distribution to the top event	IBM 360/370 Available from University of California at Los Angeles
MOCARS	Minimal cut sets or reduced system equation, primary-event failure data	Monte Carlo simulation	Exponential, Cauchy, Weibull, empirical, normal, lognormal, uniform	Microfilm plotting of output distribution; Kolmogorov-Smirnov goodness-of-fit test on output distribution is possible	CDC Cyber 76 Available from Argonne Software Center
PROSA-2	Reduced algebraic function for system representation, failure data	Monte Carlo simulation	Normal, lognormal, uniform, any distribution in the form of a histogram, truncated normal, beta	Can correlate input parameters; no sorting necessary to obtain the top-event histogram	IBM 370 Available from Argonne Software Center

Table 6-5. Computer codes for uncertainty analysis (continued)

Code	Input	Method of uncertainty analysis	Type of statistical distribution	Other features	Type of computer and availability[a]
SAMPLE	Minimal cut sets or reduced system equation, primary-event failure data	Monte Carlo simulation	Uniform, normal, lognormal	Used in the Reactor Safety Study; output is a probability distribution for the top event	IBM 360/370 Available from Argonne Software Center
SPASM	Fault tree or reduced system equation, component-failure data	Mathematical combination (similar to BOUNDS)	Lognormal	Works in conjunction with WAMCUT	CDC 7600 Available from EPRI Code Center
STADIC-II	Reduced system equation, primary-event failure data	Monte Carlo simulation (similar to SAMPLE)	Normal, lognormal, log-uniform, tabular input distribution	Has a better and efficient method of sorting the probabilities obtained in each trial	PRIME, UNIVAC-1180, CDC 7600 Available from General Atomic Company

[a]All the codes listed here are written in Fortran IV.

PROSA-2

PROSA-2 is an advanced version of PROSA-1, which provides a response-surface solution to probability distributions for the consequences of postulated nuclear accidents (Vaurio, 1981). However, the code can be used for uncertainty analysis by a direct simulation of general analytical functions.

The method for Monte Carlo simulation is similar to that of SAMPLE, but PROSA-2 has a different selection of input distributions; it can handle partially correlated input parameters and forms the top-event histograms without the comparative sorting method used in SAMPLE. The following types of distributions are available in PROSA-2: uniform, truncated normal, exponential, beta, and lognormal.

The input data include the simplified system equation, failure data, and the type of distribution used for the events in the equation. The output includes the probability-distribution histograms for top events and the statistical-error estimates for the histograms.

PROSA-2 can handle dependent (correlated) input parameters and can calculate conditional distributions. The maximum number of variable input parameters that can be analyzed simultaneously is 12. The correlations (if any) between the input parameters are limited to linear correlations. The program can plot the output histogram. PROSA-2 is written in Fortran IV for the IBM 370 computer.

SAMPLE

SAMPLE calculates the mean, the standard deviation, the distribution, and the probability bounds of a function. It was used in the Reactor Safety Study (USNRC, 1975). It uses the Monte Carlo simulation method and allows multiple system functions with multiple data input descriptions to be processed in one run.

The Monte Carlo simulation used in SAMPLE is performed by sampling primary-event values from their input distributions and finding the system-failure probability corresponding to this "trial." After many trials, the system-failure probabilities are sorted and probabilities corresponding to various confidence levels are obtained. SAMPLE can use primary-event data with either a normal, lognormal, or log-uniform distribution. Once selected, the same type of distribution is used for all primary events throughout the problem. After all these trials, results are sorted and the accuracy is tested. Finally, median and 90th percentile confidence bounds are calculated by using the sorted results.

Input includes a probability function derived from the logical configuration of the primary events, primary-event failure rates, and error factors. The output includes a listing of input data, the median value of the point estimates, as well as the system-failure probability in various increments and distribution confidence limits. The output distribution is presented in terms of estimated empirical probability percentiles from which the estimated median and upper and lower bounds can be easily read. The output also includes the estimated mean and standard deviation of the distribution and a tabular histogram of the system density function.

SAMPLE is inefficient with respect to its sorting procedure for the failure probabilities calculated in each trial. Also, the provision of the system unavailability function is a nontrivial task for large fault trees. However, it is very easy to use. SAMPLE is written in Fortran IV for the IBM 360/370 computer.

SPASM

SPASM (System Probabilistic Analysis by Sampling Methods) is an uncertainty-analysis code designed to complement the WAM series (Leverenz, 1981). Its main purpose is to provide an approximation to the top-event probability distribution from an input system model and primary-event probability distribution. While SPASM can be used in conjunction with WAMCUT, it can also be used independently. When using SPASM with WAMCUT, the system analyst chooses an option in WAMCUT that builds a system model for SPASM from the fault-tree Boolean equation. The analyst then inputs the model to SPASM together with the distributions for each event in the model.

The method used in SPASM is similar to that of BOUNDS. However, if SPASM is used in conjunction with WAMCUT, the process of preparing and inputting the system equation is eliminated. This makes the use of SPASM very easy and practical for large fault trees.

Input is very simple, especially if SPASM is used with WAMCUT (the input in WAMCUT stays essentially the same). Only the first and second moments of the fault-tree primary events must be input. If WAMCUT is not used, the analyst should construct the system model in Fortran IV to replace the cut sets generated by WAMCUT. The output consists of the first and second moments of the top event. SPASM is currently programmed for a CDC 6600/6700 computer, but conversion to IBM machines is in progress.

STADIC-II

STADIC uses a Monte Carlo simulation to generate pseudo-random-sample statistical distributions for user-defined output functions (Cairns and Fleming, 1977). STADIC-II (Orvis and Frank et al., 1981) is an improved version. STADIC was used in the uncertainty analysis of the PRA study for high-temperature gas-cooled reactors (General Atomic Company, 1978).

STADIC-II uses a "binning" procedure that eliminates the need to store and sort all of the sample (Monte Carlo trial) values generated for an output function. It also uses a very efficient algorithm for calculating normally distributed random variables. In the binning procedure the complete range of output-function variability, from the 0th to the 100th percentile, is partitioned into user-defined intervals called bins. The programmed default is 20 bins with intervals concentrated around the 50th and 95th percentiles. STADIC-II internally calculates bin boundaries in terms of the output-function values corresponding to the preselected percentiles. A counter is established for each bin. As each random-sample value of the output function is generated, it is compared with the bin boundaries, the bin within which it belongs is identified, and the corresponding counter is incremented by one. Up to 10 functions using up to 75 different variables can be analyzed simultaneously.

Input consists of the user-specified functions to be evaluated, the parameters of the selected statistical probability density function, and the number of trials desired. The user can specify any of the normal, log-normal, or log-uniform distributions--or select an arbitrary tabular distribution--for any of the variables. The input variables can have different distributions within the same functional expression. The output of STADIC-II consists of a complementary cumulative distribution function and a probability density function for each input function; the mean, variance, standard deviation, coefficient of skewness, and coefficient of kurtosis; and the Monte Carlo sampling error.

STADIC-II is considerably faster than SAMPLE because of the binning procedure. Another attractive feature is the flexibility of using different distribution functions for the variable of a given function. STADIC-II is written in Fortran IV and is currently available on UNIVAC-1180, CDC Cyber 7600, and PRIME computers.

6.6.4 CODES FOR DEPENDENT-FAILURE ANALYSIS

Dependent-failure analysis is becoming increasingly important in system reliability and safety studies, because it has been recognized that such failures can often dominate random hardware failures. Dependent-failure analysis attempts to identify the modes of system failure (i.e, minimal cut sets) that have the potential of being triggered by a single, more primary common cause; the minimal cut sets that need to be identified are those with two or more events, all of which are susceptible to a single common-cause failure mechanism.

Codes developed to deal with dependent failures are basically tracking and sorting codes. They are essentially a first effort at providing a formalized method of approaching the difficult problem of identifying and evaluating dependent failures. These codes are summarized in Table 6-6 and briefly described below.

BACFIRE

The BACFIRE code is used as an aid in common-cause failure analysis (Cate and Fussell, 1977). Its objective is to aid in identifying common-cause failures in a system and to point out why this failure potential exists. To this end, each minimal cut set is individually searched for a commonality among all the primary events in that cut set.

BACFIRE uses exactly the same method as COMCAN. However, BACFIRE allows the use of multiple locations for primary events involving components like pipes and cables. For example, if a cable passes through several different-susceptibility compartments, COMCAN can assign only one of these compartments to the cable, but BACFIRE can assign different compartments. The input and output characteristics of BACFIRE are similar to those of COMCAN.

Table 6-6. Computer codes for dependent-failure analysis

Code	Input	Method of common-cause analysis	Other features	Type of computer and availability[a]
BACFIRE	Cut sets, component susceptibilities and locations, and susceptibility domains	Examines cut sets for possible common generic causes or links between all components; prints out cut sets that are common-cause candidates	Has same features as COMCAN, but allows use of multiple locations for primary events (e.g., pipes and cables)	IBM 360/370 Available from Dept. of Nuclear Engineering, University of Tennessee
COMCAN	Cut sets, component susceptibilities and locations, and susceptibility domains	Examines cut sets for possible common generic causes or links between all components	Cut sets that are common-cause candidates can be ranked by significance of common-cause failure output	IBM 360/370 Available from Argonne Software Center
COMCAN-II	Fault tree, component susceptibilities and locations, and susceptibility domains	Same as COMCAN	FATRAM is used to generate cut sets before common-cause analysis; other features are similar to those of COMCAN	CDC 7600 Available from Argonne Software Center
MOCUS-BACFIRE	Fault tree, component susceptibilities and locations, and susceptibility domains	Same as BACFIRE	Similar to BACFIRE, but does not need cut-set input: cut sets are generated by MOCUS and automatically passed to BACFIRE	IBM 360/370 Available from Dept. of Nuclear Engineering, MIT

Table 6-6. Computer codes for dependent-failure analysis (continued)

Code	Input	Method of common-cause analysis	Other features	Type of computer and availability[a]
SETS	Fault tree	Adds generic causes and links to fault tree; cut sets that include one or more generic causes are obtained and identified as common-cause candidates	Can handle large fault trees and can identify partial dependency in cut sets; attractive features of SETS as cut-set generator justify use for dependent-failure analysis	CDC 7600 Available from Argonne Software Center
WAMCOM	Fault tree with susceptibilities added	Uses modularization and SETS to more effectively identify cut sets that contain critical events, critical random events, and significant common-cause events or to describe common-cause sets for each random failure	Can identify common total or partial links between fault-tree components; can handle very large fault trees	CDC 7600 Available from EPRI Code Center

[a]All the codes listed here are written in Fortran IV.

BACFIRE is written in Fortran IV for the IBM 360/370 and the CDC 7600 computers. There is an extensive error-checking routine, and no external routine is needed to run the code.

COMCAN

COMCAN is used to identify potential common-cause failures in a system or combination of systems (Burdick et al., 1976). It individually searches each cut set of system failures for commonality among all the primary events in that cut set.

A minimal cut set will be identified as a common-cause candidate by one of two criteria. The first criterion is met when all the primary events in a minimal cut set share a special condition that alone can result in the simultaneous failure of all the primary events in the cut set. An example of a common special condition is a common maintenance crew servicing all of the primary events implied by the primary events of a minimal cut set. The second criterion is met if all the primary events in a minimal cut set are susceptible to the same secondary-failure cause and are located in the same domain with respect to that failure cause. An example is a minimal cut set with primary events that will all occur when the associated components get wet and no water barrier exists between them.

The input consists of secondary-failure susceptibilities and applicable special conditions for primary events, domain maps for secondary-failure causes, and the list of minimal cut sets. The output provides the analyst a listing of minimal cut sets that have potential for dependent failures. The number of these common-cause candidates can be limited to those that are probably most important.

The method used in COMCAN does not provide partial common-cause dependences in systems under study. The inputting of cut sets (most often minimal cut sets are numerous) is very difficult. COMCAN is written in Fortran IV for the IBM 360/370 and has error-checking routines.

COMCAN-II, an improved version of COMCAN (Rasmuson et al., 1978, 1979), was developed to circumvent COMCAN's dependence on minimal cut sets that must be obtained by other codes. COMCAN-II uses the qualitative code FATRAM (discussed in Section 6.6.1) to obtain minimal cut sets before COMCAN analysis. This eliminates the cumbersome task of inputting all the cut sets. COMCAN-II is written in Fortran IV for the CDC Cyber 7600 computer.

MOCUS-BACFIRE

MOCUS-BACFIRE, obtained by coupling MOCUS and BACFIRE, is used to aid in identifying potential dependent failures in a system directly from the fault tree (Modarres et al., 1980a, b). It eliminates the need for generating cut sets before running BACFIRE and simplifies the input process.

The method used in MOCUS-BACFIRE is the same as described for MOCUS and BACFIRE individually. MOCUS-BACFIRE is written in Fortran IV for the IBM 360/370 computer. There is an extensive error-checking routine, and no external routine is required.

SETS

The SETS code, described earlier, can also be used for dependent-failure analysis (Worrell and Stack, 1978). The analysis is conducted in a manner similar to that of COMCAN by inputting generic cause susceptibilities for each primary event. A transformation of variables incorporates the dependent-failure susceptibilities into the Boolean equation for the top or any intermediate gate of the fault tree, and a few simple manipulations allow the user to display the cut sets that are dependent-failure candidates. The use of SETS for dependent-failure analysis has an advantage in that SETS can handle very large trees, which other dependent-failure codes are unable to do.

WAMCOM

The WAMCOM package (Putney, 1981) is designed for the dependent-failure analysis of large, complex fault trees. It can handle up to 2000-i primary events (i being the number of gates) affected by the common-cause events.

The WAMCOM dependent-failure analysis consists of model preparation and computer analysis. Model preparation is accomplished in four primary steps: the identification of potential common-cause events, fault-tree construction, the definition of individual common-cause events, and the creation of a primary-event effectivity table. The computer analysis is based on a package of three computer codes: LEVEL, WAMCOM, and SETS. LEVEL generates a solution structure for WAMCOM and SETS by partitioning the fault tree into subtrees, called "levels," that are solved individually. LEVEL develops a scheme for rebuilding the tree by reintroducing groups of subtrees into the main tree trunk. The WAMCOM program consists of a preprocessor and a SETS user's-program generator.

The SETS program generator writes a SETS user's program for each of four computer runs called "modes." This user's program is developed from the solution structure generated by LEVEL. Each mode uses as input the solution structure, fault-tree structure, and a component-susceptibility table along with information generated from previous modes.

The input consists of the fault-tree structure and a component-susceptibility table that identifies the susceptibility of each component to a generic cause. The output consists of a listing of the fault-tree input and level processing, the solution of the tree (top-down), and the output of various modes. The output of mode 1 consists of all critical common-cause events (a critical common-cause event is an event that can individually cause enough primary events to fail to place the entire system in a failed state) in the fault tree. Mode 2 output is a list of all critical random-failure events, all combinations of two significant common-cause events, and all combinations of significant common-cause events. (A significant common-cause event is an event that does not by itself lead to system failure but can cause enough primary events to fail such that the existence of a second event, either a dependent or a random failure, will place the system in a failed state.)

Mode 3 output consists of combinations of significant common-cause events with significant random-failure events affected by noncritical common-cause events. The output of mode 4 provides descriptive cause sets for each critical or significant common-cause event input.

The advantages of the WAMCOM package include a fast-running code for the dependent-failure analysis of large trees and the flexibility offered by the code to an experienced analyst. The WAMCOM package is written in Fortran IV for the CDC 7600 computer.

6.6.5 COMPUTER CODES FOR OTHER RELATED PROBABILISTIC ANALYSES

A variety of codes have been developed to aid in probabilistic analyses of accident sequences that are not used directly for qualitative or quantitative analyses of the sequences. Examples are codes that are used to perform a Bayesian updating analysis on failure data or codes that are used to perform cause-consequence analyses (to identify common characteristics among accident sequences). All of these codes are briefly discussed in this section. However, because of their variety and different applications, they were not included in Tables 6-3 through 6-6.

BROLS

BROLS (Orvis and Frank et al., 1981) is a small, fast-running code that facilitates calculations associated with using Bayes' theorem. The code calculates the posterior distribution given the prior and the likelihood distributions.

The user can choose from eight analytical statistical distributions programmed in the code, and different distributions can be selected for the prior and the likelihood. The available distributions are the normal, lognormal, Poisson, binomial, beta, hyperbinomial (or beta binomial), exponential, and uniform. An option allows the user to provide an arbitrary statistical distribution input as a discrete-probability histogram. The program makes extensive use of the International Mathematical and Statistical Library (IMSL) routines.

The input consists of the choice of the prior and likelihood functions, either one of the built-in functions or a user-specified histogram. The output consists of a table containing the probability histograms for the prior, the likelihood, and the posterior. The mean values of the prior and the posterior are also printed. BROLS is written in Fortran IV and is available for UNIVAC 1180, CDC Cyber 7600, and PRIME computers.

EXCON

EXCON is used to aid in performing risk assessments of engineered plants, facilities, or systems (Arendt et al., 1978). It is used in conjunction with cause-consequence analysis.

Required input consists of control information, consequence category descriptions, accident-sequence specifications, and search specifications.

Searches can be performed to analyze the entire set of accident sequences or any subset with the common characteristics specified in the search data.

The output information resulting from a search consists of the expected occurrence frequency of accidents resulting in consequences within each of the categories. A listing of major contributors to the expected occurrence frequency, by category, is also printed. This allows the analyst to determine which accident sequences are inconsistently large contributors to overall system risk. EXCON generates risk curves of the "Farmer" type.

EXCON is written in Fortran IV for the IBM 360/370 and the CDC 7600 computers. It has an extensive error-checking capability and requires no external routine.

6.7 DOCUMENTATION

The documentation of accident-sequence quantification goes beyond providing a frequency estimate for each plant-damage bin. It is very important to document the process by which these results were obtained. This documentation should include detailed descriptions of all simplifying assumptions and approximations used to obtain the results. If possible, the quantitative effects of the assumptions and approximations should be discussed. The documentation should also, by providing intermediate results, include enough information for the reader to reconstruct the accident-sequence frequencies from their dominant contributors. These results should identify the dominant contributions to bins, sequences, and systems. Pertinent sensitivities to primary-event point-estimate and uncertainty characteristics should be discussed.

If uncertainty studies have been performed, then the results of such studies and their ranges or distributions should be presented. The equations and distributions used for uncertainty analysis as well as the process of propagation (with pertinent assumptions) should be clearly described.

If the plant model was used to gain insights into plant reliability or the prevention of damage other than core melt, these insights should also be documented. In most cases this is the most valuable result obtained by the PRA study.

6.8 ASSURANCE OF TECHNICAL QUALITY

The assurance of technical quality can be promoted by a thorough documentation of the quantification process along with a comprehensive review of analysis results by the systems and sequence analysts. The details of the quantification should be recorded to permit a reviewer to reconstruct the quantification process and determine the validity of the steps taken and the

assumptions made. In addition to establishing the validity of the quantification process, the review of results by the systems and sequence analysts serves another objective: to ensure that the results are reasonable and that no important failure modes have been left out. Results from similar plants can be examined to establish whether they are consistent and, if they are not, the reasons for the difference.

REFERENCES

Apostolakis, G. E., and S. Kaplan, 1981. "Pitfalls in Risk Calculations," Reliability Engineering, Vol. 2, pp. 135-145.

Arendt, J. S., et al., 1978. EXCON--A Computer Program for System Risk Assessment, JBFA-129-78, JBF Associates, Inc., Knoxville, Tenn.

Barlow, R. E., and F. Proschan, 1975. Statistical Theory of Reliability and Life Testing, Holt, Rinehart, and Winston, Inc., New York.

Birnbaum, Z. W., 1969. On the Importance of Different Components in a Multi-System in Multivariate Analysis, Academic Press, New York.

Burdick, G. R., N. H. Marshall, and J. R. Wilson, 1976. COMCAN--A Computer Program for Common Cause Failure Analysis, ERDA Report ANCR-1314, Aerojet Nuclear Company.

Cairns, J. J., and K. N. Fleming, 1977. STADIC--A Computer Code for Combining Probability Distributions.

Cate, C. L., and J. B. Fussell, 1977. BACFIRE--A Computer Program for Common Cause Failure Analysis, NERS-77-02, Nuclear Engineering Department, University of Tennessee, Knoxville.

Corynan, G. C., 1982. STOP: A Fast Procedure for the Exact Computation of the Performance of Complex Probabilistic Systems, UCRL-53230, Lawrence Livermore National Laboratory, Livermore, Calif.

Engelbrecht-Wiggans, R., and D. R. Strip, 1981. On the Relation of Various Reliability Measures to Each Other and to Game Theoretic Values, SAND80-2624, Sandia National Laboratories, Albuquerque, N.M.

Erdmann, R. C., F. L. Leverenz, and H. Kirch, 1978. WAMCUT, A Computer Code for Fault Tree Evaluation, EPRI-NP-803, prepared by Science Applications, Inc., Palo Alto, Calif., for the Electric Power Research Institute.

Fussell, J. B., 1975. "How to Hand Calculate System Reliability Characteristics," IEEE Transactions of Reliability, Vol. R-24, No. 3.

Fussell, J. B., E. B. Henry, and N. H. Marshall, 1974. MOCUS--A Computer Program To Obtain Minimal Sets from Fault Trees, USAEC Report ANCR-1156, Aerojet Nuclear Company.

Fussell, J. B., D. M. Rasmuson, and D. Wagner, 1977. SUPERPOCUS--A Computer Program for Calculating System Probabilistic Reliability and Safety Characteristics, NERS-77-01, Nuclear Engineering Department, University of Tennessee, Knoxville.

Gateley, W. Y., et al., 1968. GO, A Computer Program for Reliability Analysis of Complex Systems, KN-67-704(R), Kaman Sciences Corporation.

Gesellschaft fuer Reaktorsicherheit, 1978. Deutsche Risikostudie Kernkraftwerke: Eine Untersuchung zu dem durch Stoerfaelle in Kernkraftwerken verursachten Risiko (German Risk Study), Verlag TUEV, Rheinland, Federal Republic of Germany.

Green, A. E., and A. J. Bourne, 1972. Reliability Technology, Wiley-Interscience, New York.

Hahn, F. E., 1966. Applied Boolean Algebra, 2nd edition, Macmillan, New York.

Henley, E. J., and H. Kumanoto, 1981. Reliability Engineering and Risk Assessment, Prentice-Hall, Inc., Englewood Cliffs, N.J.

Kaplan, S., 1981. "On the Method of Discrete Probability Distributions in Risk and Reliability Calculations," Risk Analysis, Vol. 1.

Lambert, H. E., 1975. Fault Trees for Decision-Making in Systems Analysis, Ph.D. thesis, UCRL-51829, Lawrence Livermore National Laboratory, Livermore, Calif.

Lambert, H. E., and F. M. Gilman, 1977. The IMPORTANCE Computer Code, ERDA Report UCRL-79269, Lawrence Livermore National Laboratory, Livermore, Calif.

Lee, Y. T., and G. E. Apostolakis, 1976. Probability Intervals for the Top Event Unavailability of Fault Trees, UCLA-ENG-7663, University of California, Los Angeles.

Leverenz, F. L., 1981. SPASM, A Computer Code for Monte Carlo System Evaluation, EPRI NP-1685, Electric Power Research Institute, Palo Alto, Calif.

Leverenz, F. L., and H. R. Kirch, 1976. User's Guide for the WAM-BAM Computer Code, EPRI-217-2-5 (PB-240-624), prepared by Science Applications, Inc., Palo Alto, Calif., for the Electric Power Research Institute.

Leverenz, F. L., and H. R. Kirch, 1978. WAMCUT--A Computer Code for Fault Tree Evaluation, NP-803, Electric Power Research Institute, Palo Alto, Calif.

Matthews, S. D., 1977. MOCARS: A Monte Carlo Simulation Code for Determining Distribution and Simulation Limits, ERDA Report TREE-1138, EG&G Idaho, Inc., Idaho Falls, Idaho.

McKnight, C. W., et al., 1966. Automatic Reliability Mathematical Model, NA-66-838, North American Aviation, Inc., Downey, Calif.

Modarres, M., N. C. Rasmussen, and L. Wolf, 1980a. Reliability Analysis of Complex Technical Systems Using the Modularization Technique, MITNE-228, Department of Nuclear Engineering, Massachusetts Institute of Technology, Cambridge, Mass.

Modarres, M., N. C. Rasmussen, and L. Wolf, 1980b. A User's Guide for MODCUT and PL-MODMC Computer Codes for Fault Tree Analysis, USNRC Report NUREG/CR-1461.

Olman, M. D., 1981. Quantitative Fault Tree Analysis Using the SET Evaluation Program (SEP), USNRC Report NUREG/CR-1935.

Olmos, J., and J. Wolf, 1977. A Modular Approach to Fault Tree and Reliability Analysis, MITNE-209, Department of Nuclear Engineering, Massachusetts Institute of Technology, Cambridge, Mass.

Orvis, D. D., M. V. Frank, et al., 1981. Guidebook for the Reliability and Maintainability Analysis of NWTS Repository Equipment, ONWI-334, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, Ohio.

Pande, P. K., M. E. Spector, and P. Chatterjee, 1975. Computerized Fault Tree Analysis: TREEL and MICSUP, ORC-75-3 (AD-A010 146), Operations Research Center, University of California, Berkeley.

Putney, B. F., 1981. WAMCOM, Common Cause Methodologies Using Large Fault Trees, NP-1851, Electric Power Research Institute, Palo Alto, Calif.

Putney, B. F., and H. R. Kirch, 1981. WAMCUT-II--A Fault Tree Evaluation Program, SAI-SR-234-81-PA, Science Applications, Inc., Palo Alto, Calif.

Rasmuson, D. M., and N. H. Marshall, 1978. "FATRAM--A Core Efficient Cut Set Algorithm," IEEE Transactions on Reliability, Vol. R-37, No. 4, pp. 250-253.

Rasmuson, D. M., N. H. Marshall, and G. R. Burdick, 1977. User's Guide for the Reliability Analysis System (RAS), ERDA Report TREE-1168, EG&G Idaho, Inc., Idaho Falls, Idaho.

Rasmuson, D. M., et al., 1978. COMCAN II: A Computer Program for Common Cause Failure Analysis, USDOE Report TREE-1289, EG&G Idaho, Inc., Idaho Falls, Idaho.

Rasmuson, D. M., N. H. Marshall, J. R. Wilson, and G. R. Burdick, 1979. COMCAN II--A Computer Program for Automated Common Cause Failure Analysis, USDOE Report TREE-1361, EG&G Idaho, Inc., Idaho Falls, Idaho.

Rooney, J. J., and J. B. Fussell, 1978. BACFIRE II--A Computer Program for Common Cause Failure Analysis of Complex Systems, Department of Nuclear Engineering, University of Tennessee, Knoxville.

Science Applications, Inc., 1978. Fire Related Accident Sequences at CRBRP, SAI-125-78-PA, Palo Alto, Calif.

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.

Van Slyke, W. J., and D. E. Griffing, 1975. ALLCUTS--A Fast Comprehensive Fault Tree Analysis Code, ERDA Report ARH-ST-112, Atlantic Richfield Hanford Company.

Vaurio, J. K., 1981. PROSA-2: A Probabilistic Response Surface Analysis and Simulation Code, ANL-81-33, Argonne National Laboratory, Argonne, Ill.

Vaurio, J. K., and D. Sciandone, 1979. Unavailability Modeling and Analysis of Redundant Safety Systems, ANL-79-87, Argonne National Laboratory, Argonne, Ill.

Vesely, W. E., 1977. "Reliability Quantification Technique Used in the Rasmussen Study," Reliability and Fault Tree Analysis, Society for Industrial and Applied Mathematics, Philadelphia, Pa., pp. 775-804.

Vesely, W. E., and F. F. Goldberg, 1977. FRANTIC--A Computer Code for Time-Dependent Unavailability Analysis, USNRC Report NUREG-0193.

Vesely, W. E., and R. E. Narum, 1970. PREP and KITT: Computer Codes for the Automatic Evaluation of a Fault Tree, USAEC Report IN-1349, Idaho Nuclear Corporation, Idaho Falls, Idaho.

Vesely, W. E., F. F. Goldberg, N. H. Roberts, and D. F. Haasl, 1981a. Fault Tree Handbook, USNRC Report NUREG-0492.

Vesely, W. E., et al., 1981b. FRANTIC II--A Computer Code for Time-Dependent Unavailability Analysis, USNRC Report NUREG/CR-1924.

Waddington, J. G., and A. Wild, 1981. The Fault Tree as a Tool in Safety Analysis in Nuclear Power Plants, INFO-0036, Atomic Energy Control Board, Ottawa, Canada.

Wagner, D. P., C. L. Cate, and J. B. Fussell, 1977. "Common Cause Failure Analysis for Complex Systems," in Nuclear Systems Reliability and Risk Assessment, J. B. Fussell and G. R. Burdick (editors), Society for Industrial and Applied Mathematics, Philadelphia, Pa.

Willia, R. R., 1978. Computer-Aided Fault Tree Analysis, ORC-78-14, Operations Research Center, University of California, Berkeley.

Worrell, R. B., and D. W. Stack, 1977. Common-Cause Analysis Using SETS, SAND77-1832, Sandia National Laboratories, Albuquerque, N.M.

Worrell, R. B., and D. W. Stack, 1978. A SETS User's Manual for the Fault Tree Analyst, SAND77-2051, Sandia National Laboratories, Albuquerque, N.M.

Worrell, R. B., and D. W. Stack, 1981. "A Boolean Approach to Common Cause Analysis," in 1980 Proceedings, Annual Reliability and Maintainability Symposium, San Francisco, Calif., pp. 363-366.

Chapter 7

Physical Processes of Core-Melt Accidents

7.1 INTRODUCTION

This chapter describes procedures for predicting the progression of core-melt accidents and the associated physical processes, developing containment event trees, and quantifying probabilities for branches of the containment event tree for each accident sequence. It discusses the various physical processes that must be analyzed in a risk study that includes the estimation of accident consequences, the degree to which these processes are understood, unresolved issues, and available methods of analysis. Because of the state of development of core-melt analysis, the procedures provide for considerable flexibility in the selection of models. The need for sensitivity studies is emphasized.

Although this chapter provides guidance for the analysis of physical processes, the procedures described here are not as prescriptive as those in some other chapters of the guide. It is the intent of this guide to reflect commonly accepted practice, not to develop new procedures. The state of the art in this particular area is advancing rapidly.

The analyst should recognize that experience in the analysis of core-melt accidents is very limited and that improvements in methods can be expected over the next few years. Very few risk analyses that have been performed to date have attempted a complete analysis of consequences. The principal example of consequence analysis in a risk study is the Reactor Safety Study (USNRC, 1975). At the time of the Study, however, the methods available for analyzing the physical processes of core-melt accidents were primitive. Considerable experimentation and model development have occurred since the Reactor Safety Study, but the methods of analysis are for the most part not validated. There has also been very little experience in the use of these computer codes in risk analyses. The most complete treatment of the physical processes of reactor accidents in a recent PRA has been the Zion study (Commonwealth Edison Company, 1981).

The procedures described in this guide for the analysis of physical processes are limited to core-melt accidents. Two questions are frequently asked about such accidents: "If these accidents are expected to be very unlikely, why is there so much emphasis on their analysis?" and "Should not primary emphasis be placed on understanding the consequences of accidents that are more likely to occur?" The results of the Reactor Safety Study have indicated that the consequences of core-melt accidents are potentially so much greater than those of more likely accidents that the contribution from these accidents dominates the predicted risk to the public (Hall et al., 1979). It cannot be assumed, however, that the relative risk from core-melt accidents will exceed the risk from less severe accidents for all plant designs. Although methods for analyzing degraded-cooling sequences

that are arrested before a complete meltdown of the core are not addressed in this chapter, the treatment of such sequences is not precluded by the guide.

This chapter discusses a number of physical processes that must be evaluated in a risk study of core-melt accidents. The computer codes that are available for performing these analyses are also described. The analyst should recognize that these models are continually being upgraded and that other models are being developed by ongoing research. Because of these rapid developments in modeling, this guide does not recommend the use of any specific set of computer codes.

7.2 OVERVIEW

Figure 7-1 shows the tasks involved in analyzing the physical processes of severe core-damage accidents. As described in Section 7.10, the activities performed in each of these tasks can differ with the intended application of the PRA.

The first two tasks are concerned with the collection of data and the modeling of the plant for analysis. They require a good understanding of the plant, which can be obtained through close cooperation with the utility and through plant visits. Also necessary is interaction with the system analysts who are defining the success and failure criteria for safety systems. Frequently, analyses will be required to determine which plant conditions or accident sequences result in core melt. Modeling of the plant cannot be completed until after the methods of analysis and the specific sequences for analysis have been selected. In the third task, potential failure mechanisms for the containment and levels of failure are investigated in preparation for the construction of the containment event tree. The potential failure mechanisms must also be recognized before the methods of analysis are selected.

The accident sequences provided to the analyst of physical processes for analysis are the output of the system event trees. To reduce the number of sequences that must be analyzed, these sequences can be grouped into plant-damage states or bins. Alternatively, the selection of accident sequences for analysis can be based on their likelihoods. In the binning process, sequences are grouped according to accident characteristics that affect the response of the containment and the release of radionuclides into the environment. The development of bins and the development of the containment event tree are therefore very closely related. The representative sequences are then analyzed with the core-melt codes, and the results (accident timing, temperatures, flows, pressures, and rate of leakage from containment) are supplied to the radionuclide-transport task. Conditions associated with the leakage from containment are also provided to the environmental transport and consequence analysts. Sensitivity studies are performed as required to quantify event-tree branching probabilities and to estimate the contribution of uncertainties in physical processes to the uncertainties in the total risk.

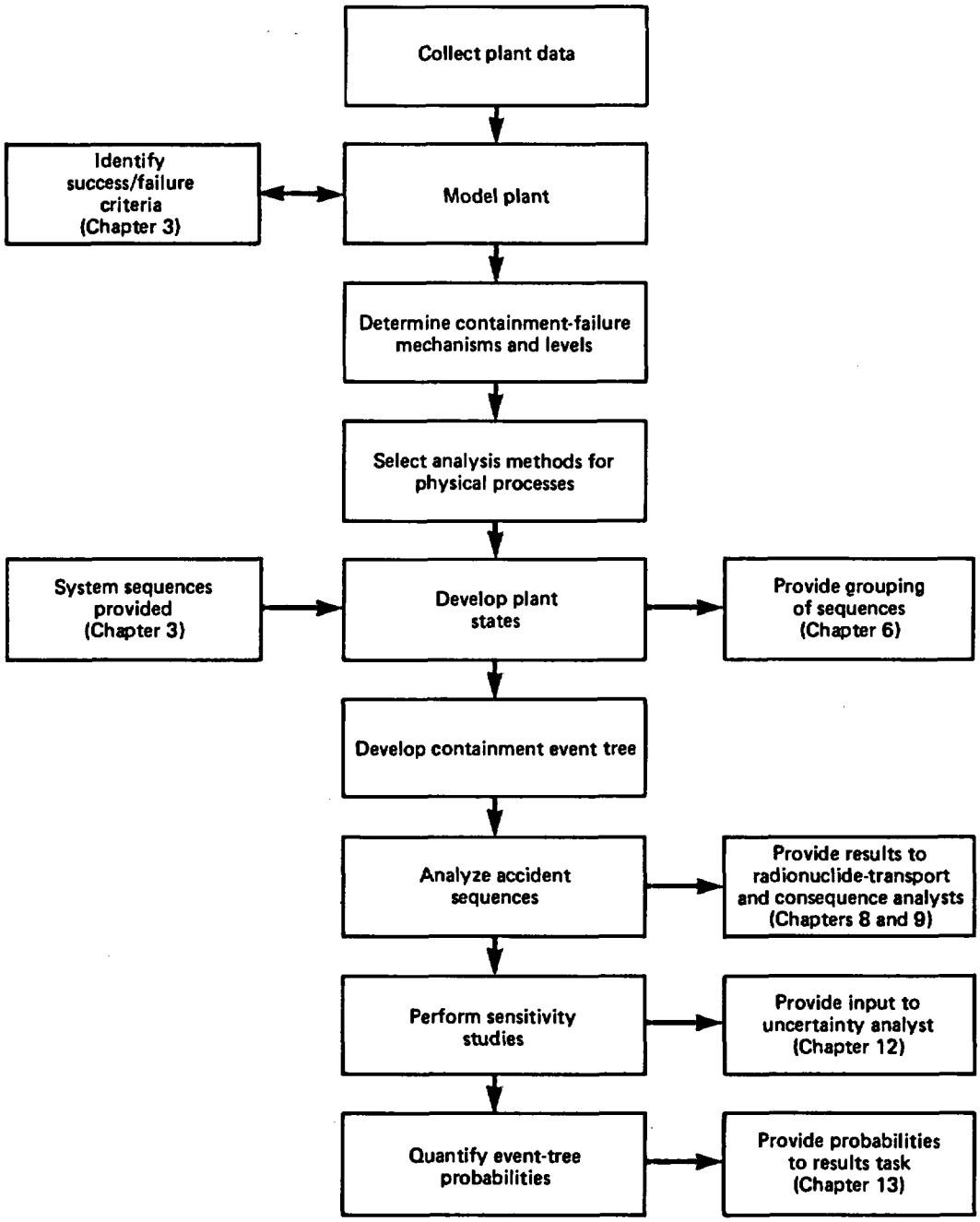


Figure 7-1. Activities diagram for the analysis of physical processes.

Figure 7-1 does not show the iterative nature of the effort. The analysis of accident sequences, for example, may lead to the need to modify assumed containment-failure mechanisms and the containment event tree. Sensitivity studies may also indicate the need for different methods of analysis. If sequence bins are not used, some iteration will be required with the system analysts to ensure that a sufficient number of accident sequences has been analyzed.

7.3 PHYSICAL PROCESSES OF CORE-MELT ACCIDENTS

Many of the physical processes of core-melt accidents are analyzed by the core-melt system codes, which integrate them with other processes. Because research into core-melt processes is limited and the computer codes have not been validated in some areas, the computer codes cannot be used routinely without understanding the limitations of the models and thoroughly understanding the physical processes involved in the progression of a core-melt sequence inside the containment. This section briefly reviews the physical processes and examines the status of current knowledge about the underlying phenomena.

7.3.1 IN-VESSEL BEHAVIOR

The core-melt sequences that are analyzed in PRAs typically involve an imbalance between the power level in the fuel and the availability of cooling water. In these sequences the inventory of water in the reactor-coolant system boils away and the fuel becomes uncovered, heats up, and melts. As already mentioned, degraded-cooling sequences (in which inoperative safety systems are restored in time to arrest the progress of core damage before the reactor vessel is penetrated) have not been analyzed in detail in PRAs in the past. The discussion that follows pertains primarily to accidents postulated to result in complete fuel melting. However, some modeling work that is in progress could be useful in determining the limits of degraded-core coolability (BMFT, 1980; Allison et al., 1981).

7.3.1.1 Pressurized-Water Reactors

The core of a PWR is characterized by an open array of Zircaloy-clad fuel rods. The fuel assemblies rest on a core-support plate that is suspended from the reactor vessel by the core barrel. Tubes containing neutron-absorber material fit into open spaces in each assembly of fuel pins (square arrays of 14 to 17 lattice positions) and are inserted as a cluster from above the core.

As the water level drops in the core region, the exposed fuel heats up. Phenomena affecting the rate of heating are the decay-heat level, fission power level in cases of failure to scram, zirconium oxidation at high temperatures, convective heat transfer to steam and hydrogen, radiative heat transfer to steam, and radiative heat transfer to structures. As the fuel heats, cladding swelling and rupture would occur. The extent of swelling and the failure temperature depend on the heating rate and reactor-coolant-system pressure. At sufficiently high temperatures (approximately 1900°C) the interaction between unoxidized zirconium and uranium dioxide at the cladding interface would result in the formation of a liquid phase (Hagen and Malauschek, 1979). A further rise in temperature could result in an expansion of the molten eutectic region and the melting of the zirconium, zirconium oxide, and uranium dioxide (Peehs et al., 1979). Molten material

would slump and resolidify in lower portions of the core, and some degree of flow blockage in the channels would occur.

Similarly, control rods would be heated by radiation transport and radiative heat absorption. For silver-indium-cadmium control rods in particular, melting might occur early in the core-melt accident and possibly influence the distribution of flow through the core.

A molten zone is expected to grow and progress downward, following the receding water level. The next major phase of the accident begins when the molten fuel leaves the original core region to enter the lower plenum. A variety of modes of fuel relocation can be postulated. Small portions of the fuel could conceivably fall out of the core region and into the lower plenum as they become molten. Or fuel relocation could take place on a larger scale but still progressively as portions of the core-support plate heat up and weaken sufficiently to release the fuel above it. Grid plates below the core-support plate might also impede the progress of the molten material entering the lower plenum. A massive relocation of molten fuel could take place when a portion of the support structure fails, with the molten fuel subsequently pouring into the lower plenum. Alternatively, the molten core material might progress radially outward, overheat the core barrel, and drain into the lower plenum.

When the molten fuel in the lower plenum comes into contact with the water, an interaction will occur, dispersing the fuel and generating steam. Under some conditions, a particularly energetic reaction, referred to as a "steam explosion," could occur with the potential to threaten containment integrity. The possibility of steam explosions is discussed in Section 7.3.4. After the molten fuel has drained into the lower plenum, the remaining water is expected to boil away in a comparatively short time. Additional reactions of steam with zirconium or steel could occur during this period, producing more hydrogen and heat. The core debris would then reheat and begin to attack the reactor vessel. Heat from the core debris would be transferred into the walls of the vessel by conduction, and after the fuel has remelted, heat transfer would be enhanced by internal convection. The fraction of the bottom head of the vessel exposed to fuel debris would be heated in this manner. Under the stress loads of the weight of the core and possibly a high internal pressure in the vessel, the bottom head could then yield and fail. In addition to the general attack on the bottom head, a localized attack on the in-core instrumentation tubes that penetrate the vessel would also occur. A failure of the instrumentation tubes would lead to a small available flow area and a more protracted release of core material, followed by steam and hydrogen, into the reactor cavity.

The release of radionuclides and inert aerosols from the fuel depends on the time-temperature history of fuel heatup. Unfortunately, the modeling of the fuel-melting regime in the existing core-melt system codes is not very mechanistic. More-detailed models are therefore being developed to improve predictions of the temperature of the fuel as a function of time after the start of melting (Allison et al., 1981; Tuerk et al., 1980). None of the currently available codes describe the temperatures and rates of flow throughout the reactor-coolant system during core degradation, but improvements are being made in the MARCH code to perform this analysis.

Since many of the processes described above are treated approximately by existing core-melt codes, a number of modeling efforts are under way to treat some of the processes more mechanistically. Examples are the SCDAP code (Allison et al., 1981) being developed by the Nuclear Regulatory Commission, the MAAP code (Fauske and Henry, 1982) being developed by the Industry Degraded Core Rulemaking (IDCOR) Program, and the CORMLT code (Denny, 1982) being developed by the Electric Power Research Institute. Some aspects and uncertainties regarding in-vessel behavior are described in more detail by Rivard et al. (1981) and in the Zion PRA (Commonwealth Edison Company, 1981).

7.3.1.2 Boiling-Water Reactors

In a BWR, the in-vessel behavior of a melting core is expected to differ in some respects from that described for the PWR. The fuel assemblies in a BWR have fewer pins (49 or 63) than those of a PWR and are enclosed in a shroud. The enrichment of uranium in the fuel pins varies with location within a bundle. A cruciform control blade is inserted upward from the lower plenum between a set of four neighboring bundles. Each bundle is supported by the associated control-rod-drive housing.

Because each bundle is enclosed at elevations above the grid plate, coolant flow cannot redistribute between bundles so long as the shrouds remain intact. In addition, water levels can vary among the bundles. Core melting would proceed in a manner quite similar to that described for the PWR except that radiative heat transport to the shroud and neighboring control blades would provide a major heat sink. Cooling of the shroud by core sprays or bypass flow can be effective in cooling fuel within the bundle. As the shrouds and control blades become molten during core melt, communication would be established among the bundles. Since the fuel bundles are individually supported, there is little potential for a "coherent" dumping of the molten fuel into the lower plenum. In addition, the lower plenum is closely packed with an array of housings for the control-rod drives. The progression of the molten core behind a receding water level would proceed in a series of slumping, solidification, and remelting steps; in the lower plenum of a BWR, its behavior might be similar to that expected in the core region.

The housings of the control-rod drives penetrate the bottom head of the reactor vessel. This appears to be the most likely pathway for the release of molten fuel to the cavity beneath the vessel. A gross attack on the vessel head could also lead to the failure of the head.

7.3.2 IN-CONTAINMENT BEHAVIOR

The accident processes that are described in this section begin with the entry of hot core debris into the region beneath the reactor vessel. Included in this discussion is the pressure and temperature response of the containment.

7.3.2.1 Pressurized-Water Reactor: Large Dry Containment

After the reactor vessel has failed, hot and possibly molten core material would drop or be injected under pressure into the reactor cavity. The subsequent behavior would depend on whether water is present in the reactor cavity and on the geometric configuration of the cavity region.

Interactions Between Molten Fuel and Water

The presence of water in the reactor cavity would depend on both the design and the accident sequence. In some sequences, the activation of the accumulators after vessel penetration would introduce water into the cavity. As molten core material interacts with the water, fuel fragmentation, rapid heat transfer, and steam production would ensue. Depending on the rate of steam production, the total quantity of steam produced, and the strength of the containment, the potential for containment failure could exist at this time. Moreover, this interaction of the core debris with water could lead to a dispersal of the debris throughout the containment. This may make the debris more coolable and avert its attack on concrete, but it may also lead to a greater release of radionuclides, rapid heating of the containment atmosphere, etc.

The generation of containment-threatening missiles from a steam explosion does not appear to be a possibility at this stage of the accident. The size of the fuel particles produced in the interaction could, however, have a major effect on the subsequent coolability of the resulting debris bed if the debris remains in the reactor cavity. (See the Zion PRA (Commonwealth Edison Company, 1981) for a more detailed discussion.) If pathways and mechanisms for the release of steam from the cavity and for the refluxing of condensed water back into the cavity exist, it is possible that a coolable debris bed could result (Commonwealth Edison Company, 1981; Lipinsky, 1980). Considerable experimentation and correlation development have been undertaken, particularly in the LMFBR program, in regard to the coolability of debris beds (Baker et al., 1977; Dhir and Catton, 1977; Hardee and Nilson, 1977; Lipinsky, 1980; Rivard, 1978; Squarer et al., 1981). The potential for arresting further core degradation and concrete attack at this stage is important not only because it would remove a possible containment-failure mode (basemat penetration) but also because it could limit the long-term generation of combustible gases once the core debris is cooled.

For some designs, if the reactor vessel is at an elevated pressure at the time of vessel penetration, some of the core debris could be swept out of the cavity to other regions of the containment (Commonwealth Edison Company, 1981). Since this material would be widely distributed, it would probably be cooled without attack on concrete or after limited attack.

Debris-Concrete Interaction

If water is not initially present in the reactor cavity or is boiled away--or if the core-debris bed is not coolable--the hot core material will attack the concrete basemat, not only eroding the concrete but also inducing the generation of hot (including combustible) gases. Experimental

observation and analysis have indicated that the attack on concrete by the hot fuel could go through a number of phases (Muir et al., 1981; Powers and Arellano, 1981; Powers et al., 1977). The initial attack is expected to involve rapid gas generation and vigorous mixing of the molten core material. At first, the oxide phase of the molten material is expected to be at the bottom of the pool if stratification occurs. After a short time, however, the products of concrete decomposition will mix with the oxidic phase, lowering its density. The layers will then invert, leaving the metallic phase on the bottom.

The principal source of radioactive-decay heat will remain in the oxidic phase, but several significant chemical reactions will take place between the products of concrete decomposition and the metallic phase. As steam and carbon dioxide are released from the concrete, they will pass through the metallic phase and be reduced to hydrogen and carbon monoxide. As the heat source decreases with time and is diluted with inert materials, the metallic phase and later the oxidic phase will solidify. The concrete will probably continue to erode, but at a reduced rate. To date, most of the experiments and model development have been concerned with the concrete-attack phase that precedes solidification. Eventually, the concrete basemat might be penetrated, introducing a pathway for the release of radionuclides.

Containment Pressurization

Throughout the accident, steam and noncondensable gases will be released to the containment atmosphere. All large dry PWR containments have spray systems that could act to condense this steam. Of course, for a given sequence, the spray system could be inoperative or ineffective in condensing steam. Many designs also have fan coolers. Regardless of whether these engineered safety features are operable in a given sequence, steam would condense on steel and concrete structures in the containment.

In the early stages of an accident, heat transfer to structures may be controlled by the flux of steam to the walls. Later in the accident, the heat transfer is limited by the conduction of heat within the structure itself. If an adequate means for removing heat from the containment is not available in an accident sequence, the containment will eventually overpressurize and fail.

Some of the conditions and phenomena that must be considered in performing the containment-response analysis for a core-melt accident include the following:

1. Gas composition (steam, oxygen, combustible gases, and inert gases).

2. Coefficients of condensing heat transfer to structures.

3. Temperature profiles in structures.

4. The effects of containment sprays, containment coolers, and suppression systems.

5. Hydrogen combustion.

6. Heat-source redistribution.

7. Flows between compartments.

The output of the containment analyses is required as input to the analysis of radionuclide transport as well as for predicting containment-failure modes. It may therefore be necessary to have a multicompartment capability for the analysis of containment conditions.

Burning of Combustible Gases

The combustible gases generated in a core-melt accident can threaten the integrity of the containment through combustion as a source of heat or flame impingement, through deflagration as a source of elevated temperature and pressure, and through detonation as a source of shock waves. The conditions leading to various modes of combustion for hydrogen are being investigated experimentally by both the industry and the government (USNRC, 1981a; Berman, 1981a,b). Reports on NRC- and EPRI-sponsored workshops provide a good review of the state of the art (Berman, 1981c). Current plants with large dry containments do not have engineered safety features designed to control the rapid rates of hydrogen generation associated with a core-melt accident. Important sources of hydrogen during an accident arise from the oxidation of metals by high-temperature steam, including the Zircaloy cladding, the steel internals, the steel in the lower head of the vessel, and rebar in the concrete basemat. Carbon monoxide and some methane can also be produced by attack on the concrete. Over a longer term, additional sources of hydrogen can arise from radiolysis and corrosion. The combustion of gases in the containment atmosphere may be possible at various stages of the accident. One critical time period follows the meltthrough of the vessel head and the rapid release of hydrogen from the vessel. Rapid steam generation could follow shortly after a deflagration event, superimposing the two sources of pressure.

High concentrations of steam have the effect of suppressing ignition. In a large dry PWR containment, steam inerting is frequently predicted for accident sequences in which containment safety features are inoperative. If the steam concentration is reduced in such an accident sequence--for example, by the delayed actuation of air coolers--the conditions in the containment atmosphere could rapidly move into a highly combustible range (USNRC, 1981b).

7.3.2.2 Pressurized-Water Reactor: Ice-Condenser Containment

Because of the arrangement of the reactor cavity and the sump, it is unlikely that the cavity would be filled with water at the time of vessel meltthrough. In some sequences, the accumulators would discharge after head failure, releasing water into the cavity. In general, however, conditions in the cavity and the subsequent attack on the concrete would be expected to be similar to the dry-cavity scenario discussed in the preceding section.

The features of the ice-condenser containment that particularly affect its response to core-melt conditions are the ice bed and the low design pressure. As long as it remains, the ice is expected to effectively condense steam even in those sequences where electric power is not available. Because of the small pressure-volume capacity, however, the production of noncondensable gases could eventually cause containment failure even with no significant partial pressure of steam. The low design pressure also makes the containment susceptible to failure in a hydrogen-burning event. For example, if a stoichiometric mixture of hydrogen and air burns rapidly in the containment, the resulting pressure rise will be approximately 150 psi, more than enough to fail the containment. In the future, all ice-condenser designs will have mitigation features that are intended to prevent the accumulation of hazardous levels of hydrogen.

7.3.2.3 Boiling-Water Reactor

There are some differences between the in-containment behavior postulated for core-melt accidents in PWRs and BWRs. Except for the physical layout of the drywell and the suppression pool, the design characteristics of the Mark I and Mark II BWR containments are quite similar. Both have small volumes and are therefore susceptible to failure through overpressure due to the generation of noncondensable gases. Since both are operated with inerted atmospheres, hydrogen burning is not possible except for unlikely circumstances involving a failure of the inerting function.

During the period of time preceding bottom-head failure, steam and hydrogen will be released to the drywell or directly to the suppression pool. If the blowdown from the reactor-coolant system flows into the drywell, the pressure in the drywell will increase to the point at which the vent lines to the suppression pool are cleared, and flow will be established from the drywell into the suppression pool.

After the vessel has failed, molten fuel will enter the cavity. The presence of water in the reactor cavity at the time of vessel meltthrough will depend on the accident sequence. In a LOCA, water is expected to be present in the cavity. In a transient sequence, however, RCS blowdown would be directed to the suppression pool and no water would be present in the cavity, except for cases where the suppression pool is overheated and steam may be condensed in the drywell. Because of the small containment volume, a steam explosion in the reactor cavity, rapid RCS blowdown, or the sweep-out of the fuel from the reactor cavity could threaten the integrity of the containment.

The progress of the molten-core attack on the basemat will be similar to that of the scenario described for the PWR. For the Mark II design, penetration of the concrete pad would be followed by entry into the suppression pool, fragmentation, rapid steam production, and possibly a cooled debris bed. The high temperatures produced in the drywell during the core attack on concrete could affect the integrity of the penetration seals.

Although the volume of the Mark III containment is larger than that of the earlier designs, the design pressure is lower. Thus, noncondensable-gas

generation and hydrogen combustion are potential causes of containment failure. In the future, all Mark III designs will have some type of safety feature for hydrogen control to prevent conditions that could result in deflagration (USNRC, 1981c).

Because of the small size of the drywell, very high temperatures could be produced during the period of concrete attack. Since the outer containment region surrounds the drywell in the Mark III design, leakage from the drywell would not be as critical as it would be for the other designs.

7.3.3 MECHANISMS LEADING TO CONTAINMENT FAILURE

The reactor containment building is a very effective safety feature. If it remains intact, the offsite consequences of the accident will be minor. Conversely, if the containment fails at about the time of core meltdown, current methods of analysis predict major consequences (USNRC, 1981d). For this reason, a risk estimate can be very sensitive to the treatment of containment-failure modes. Table 7-1 lists the mechanisms that might lead to containment failure in a core-melt accident. The mechanisms that are typically considered in risk studies are identified.

The manner and the location of containment failure can be very important. If the size of the breach is small, more time will be available for retention mechanisms to be effective before radioactive material leaks to the environment. Radionuclides may also be retained along the path of leakage. The location of failure can have a particularly large effect on the predicted consequences of accidents in pressure-suppression containments. A location that involves bypassing the pressure-suppression device could involve substantially larger releases to the environment. The elevation of release and the energy content of the gases leaving the containment also affect the offsite consequences.

Table 7-1. Potential containment-failure modes and mechanisms

Direct bypass[a]	Core-concrete interaction
Failure to isolate[a]	Basemat penetration[a]
Vapor explosions	Structural failure and tearout of penetrations
Blast	Blowdown forces
Missile generation[a]	Pipe whip
Quasi-static pressure rise[a]	Vessel thrust forces
Overpressurization	Pressure-vessel burst
Steam[a]	Missile generation
Noncondensable gases[a]	Meltthrough
Combustion processes (hydrogen, carbon monoxide, methane)	Direct contact of containment liner with fuel debris
Blast	
Missile generation	
Quasi-static pressure rise[a]	

[a]Mechanisms typically analyzed in risk studies.

A number of processes that could result in containment failure such as hydrogen combustion and core-concrete attack were described briefly in this section. Sections 7.3.4 and 7.4 discuss the potential for containment failure from a steam explosion and the containment structural response to overpressurization loads (resulting from steam generation, noncondensable-gas production, and the burning of combustible gases).

Two other failure modes that should be carefully considered are failure to isolate the containment and a direct bypass of the containment. The latter type of sequence was the single largest risk contributor identified for the reference PWR in the Reactor Safety Study (USNRC, 1975).

7.3.4 STEAM-EXPLOSION RESPONSE

One of the potential modes of containment failure considered in the Reactor Safety Study (USNRC, 1975) was a steam explosion in the lower plenum of the reactor vessel, which was postulated to cause a slug impact on the upper head of the vessel and the launching of the head as a missile. Steam explosions could be important risk contributors because of the potentially high consequences, even if their likelihood is low. Experimentation and analysis conducted since the Study have indicated that this scenario is very unlikely (Corradini, 1981). At present, it is not possible to give definitive advice to the analyst as to whether or not steam explosions should be considered in a risk study. There is good evidence that the triggering of steam explosions is suppressed at high system pressures (Corradini, 1981; Henry and Fauske, 1979). Sandia National Laboratories has attempted to refine the probability estimates for steam explosions resulting in containment failure that were presented in the Reactor Safety Study (Corradini and Swenson, 1981).

In considering steam explosions, a number of different effects should be evaluated: the rapid generation of steam, missile production, major vessel motion, and (possibly) shock-wave propagation. Although the primary concern in a steam explosion would be containment failure, the possibility of a steam explosion changing the course of an accident should also be considered. A steam explosion could result in an early failure of the reactor vessel, the dispersal of fuel, and a greater release of radionuclides from the fuel.

7.4 ANALYSIS OF CONTAINMENT CAPACITY

The integrity of the containment--or, in the event of containment failure, the mode and the time of failure--can have a major influence on the radiological consequences of a core-melt accident. The results of PRAs indicate that, in terms of public risk, the failure mode of greatest concern is containment overpressurization. Such a failure can result from the generation of steam and noncondensable gases or from the burning of combustible gases. This section discusses the response of different containment designs

to internal overpressurization transients.* The discussion is limited to events in which the rate of pressurization is small in comparison with the mechanical-response time of the structure.

7.4.1 CONTAINMENT DESIGNS

The function of the containment is to provide a leaktight barrier against the release of radionuclides to the environment. To perform this function, the containment must contain the pressure resulting from a blowdown of the reactor-coolant system in the event of an accident. In practice, pressure containment is achieved either by providing a sufficient design pressure capacity and containment volume to accommodate the steam released in an RCS blowdown or by using efficient heat sinks (a suppression pool or an ice bed) to remove steam from the containment atmosphere.

Two major types of structural designs are used in the United States: steel containments and concrete containments (Walser, 1980). Steel containments employ welded steel plate to provide both structural strength and a leaktight barrier. A reinforced-concrete building around the steel containment provides biological shielding and protection against external threats. In concrete containments, the structural strength comes from reinforcing bars or prestressing tendons. The concrete provides biological shielding against direct radiation. A thin steel liner is used to form a leaktight barrier against the release of radioactive material.

Steel containment buildings are designed in accordance with the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME, 1980), Section III, Division 1, Subsection NE for Class MC components. A number of concrete reactor containments had been constructed in the United States before an ASME Code committee was formed; these were designed and constructed in accordance with variations of the ACI codes. The current code is ASME Code Section III, Division 2 (ASME, 1980).

The codes embody certain safety factors in the relationships between the allowable working stresses and limiting-stress levels, such as yielding or ultimate failure. The specific safety factors vary with the nature of the loadings as well as with the applicable portions of the Code; for example, the safety factors for primary membrane stresses are different from those for secondary stresses. The design of containment structures must also take into account a variety of load combinations, internal as well as external, normal as well as those induced by accidents. Thus, for any containment structure it is not easy to determine the available safety margins between design loadings and those at which the structure can be expected to fail.

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*It may be necessary to also consider the response of the containment to external events (airplane impacts, tornadoes, or earthquakes--see Chapters 10 and 11), internal missiles, and hydrogen-detonation loads.

7.4.1.1 PWR Containment Designs

Most of the PWR designs in the United States are of the large dry type. These typically have large free volumes (~2 x 10^6 ft^3) and high design pressures (~30 to 45 psi for steel containments and ~45 to 60 psi for concrete).

The earliest steel containments were spherical. This design option was chosen for West German PWRs and is again becoming popular in the United States. A number of PWRs have cylindrical steel containments with elliptical bottoms and hemispherical heads. Many recent plants have used a hybrid design: a cylindrical steel containment supported by a steel-lined reinforced-concrete basemat.

The concrete PWR containment buildings are cylindrical, with a hemispherical or shallow dome and a flat-slab basemat. The first designs used conventionally reinforced concrete. More recently, prestressed-concrete containments have been constructed; these may be partially or fully prestressed.

The other type of PWR containment system uses packed beds of ice to condense the steam released from the reactor-coolant system. Nearly all of these ice-condenser designs have steel containments with typical volumes of 1.2 x 10^6 ft^3 and a design pressure of 12 to 15 psi.

7.4.1.2 BWR Containment Designs

Boiling-water reactors use suppression pools to condense the steam released from the reactor-coolant system in an accident. Three configurations--Mark I, Mark II, and Mark III--have evolved with time. In each configuration, the reactor-coolant system is located inside a drywell. In the event of an accident involving a break in the reactor-coolant system, steam would be released into the drywell and would flow into the suppression pool to be condensed. Noncondensable gases would flow into the vapor region of the wetwell. In other accidents, steam from the reactor-coolant system may be released directly to the suppression pool through safety relief valves.

The Mark I is called the "lightbulb-and-torus design" because the drywell is shaped like a lightbulb. The suppression pool is inside a torus that runs around the drywell at a lower elevation. The Mark I designs are steel structures with volumes of approximately 2.8 x 10^5 ft^3 and design pressures of 60 psi.

The Mark II is often referred to as the "over-and-under" design because the suppression pool is directly beneath the drywell. These reinforced-concrete containments have volumes of about 3 x 10^5 ft^3 and design pressures of 45 to 60 psi.

The most recent design variation, the Mark III, is a cylindrical containment that can either be concrete or a hybrid with a steel dome and body and a reinforced-concrete basemat. The suppression pool is in an annulus at the lower periphery of the containment. The vapor space of the wetwell is

much larger than that of the other two pressure-suppression designs and forms an outer containment volume that encloses the drywell. The volume of the containment is approximately 1.7×10^6 ft^3, and the design pressure is about 15 psi.

7.4.2 FAILURE PRESSURES, CRITERIA, AND MODES

7.4.2.1 Failure Criteria

In order to establish the pressure at which a structure will fail, it is necessary to define one or more failure criteria for the structure: a limiting stress, strain, or some other condition (Commonwealth Edison Company, 1981; Murfin, 1980). For idealized structures of well-defined material (e.g., a free-standing spherical steel shell) a failure criterion characterized by limiting stresses or strains would appear reasonable. As the geometry of the structure becomes more complicated (e.g., a cylindrical shell with a hemispherical or dished head and a flat rigid bottom), the definition of a failure criterion becomes more difficult, but may still be based on a limiting stress, strain, or the onset of instability. Real containments are, however, far removed from idealized structures, being characterized by gross as well as local geometrical discontinuities, local reinforcements, changes in wall thickness, and the like. Thus, the definition of failure criteria and the associated failure pressures is far from straightforward.

The problem becomes even more complex in reinforced-concrete containments, whose overall behavior depends on interactions among the reinforcement (or prestressing), the concrete matrix, and the leaktight liner. For small deformations (i.e., response in the linear range) the behavior of the composite structure is quite predictable. As increased loadings take part of such a structure into the plastic regime, a variety of failure modes can occur, and the overall behavior becomes more and more difficult to determine. While the reinforcing or prestressing is the principal strength member, it relies on the concrete matrix for support and the transmission of internal pressure loads; both of the former depend on the integrity of the liner for effective performance of their functions. A failure of any one of the three principal components will result in a functional failure of the structure. Although it is to be expected that a concrete containment will fail when the ultimate strength of the principal load-bearing members is exceeded, it must be recognized that a complex structure can also fail by other mechanisms, such as the tearing of the liner. The definition of a failure criterion for the establishment of a failure pressure should recognize the widely differing stress-strain characteristics of the several components of a concrete containment. Many actual containments consist of a combination of conventional reinforcement and prestressing, thus offering the possibility of different behavior in, and requiring different failure criteria for, the several parts of the total structure.

In order to establish failure criteria, the analyst may need to decide what constitutes a functional containment failure. On the one hand, it may not be sufficient to assume that the ultimate strength of the structure can be reached without any loss of leaktightness; on the other hand, minor

increases in leakage should not be defined as failure. In the Reactor Safety Study (USNRC, 1975) it was observed that containment leak rates of up to 100 vol.% per day would not significantly alter the response of the containment to some key accident sequences. This leak rate would of course depend on the containment design and the accident sequences considered.

7.4.2.2 Mode of Failure

Another difficult problem is the characterization of the mode of containment failure or the size of the hole associated with the failure. The mode of containment failure could be closely related to the failure criterion that is deemed to be appropriate. For example, if the failure criterion is related to the blowout or degradation of some of the penetrations, then the magnitude of the leakage can be defined reasonably well. If the failure criterion is associated with the ultimate strength of the structure, it may be necessary to assume a large break.

7.4.2.3 Distribution of Failure Pressures

For the purposes of PRA, a realistic failure pressure is of interest, not the nominal design pressure. Because of uncertainties in the conditions leading to failure, a specific failure pressure cannot be determined. For example, it should not be assumed that the gross ultimate strength of the structure can be reached without the prior loss of function. What is needed for a PRA is a density function describing the probability of failure as a function of loading (pressure) (USNRC, 1975). The shape of such a density function will vary with the containment design, level of analysis, and knowledge of the details of the actual containment. Among the variables that should be considered in assessing the uncertainty in the failure pressure are the validity of the selected failure criterion, the accuracy of the computational methods, the possibility of construction faults, and variations in material properties.

7.4.2.4 Analysis

Containments can generally be characterized as axisymmetric thin-shell structures. Thus, structural analyses at loadings close to design levels are relatively straightforward, and a wide variety of applicable analytical tools are available, ranging from hand calculations to detailed finite-element computer codes. At the high loadings where a structural failure of the containment can reasonably be expected (i.e., after the yielding of the principal load-bearing members) large deformations will typically be encountered, and simple analytical approaches or even many of the detailed design codes may no longer be applicable. While there are a number of sophisticated codes for elastic-plastic structural analysis that are capable of treating large deformations in complex structures (Commonwealth Edison Company, 1981; Murfin, 1980; Murray et al., 1979), these are generally not used by utility or architect-engineer firms in the design of plants.

There is little experimental data on which to base the predictions of the failure levels of containment structures. In the past, some work was done on the explosion-containment potential of steel structures; this is of little interest in the present context. A considerable number of scale-model experiments have been conducted for the prestressed-concrete pressure vessels of gas-cooled reactors; these, however, have been thick-walled structures with again limited applicability to the present problem. There have been only a few experiments in which scale-model containments have been tested to failure under the conditions of interest here (Aoyagi et al., 1979; Atchison et al., 1979; Donten et al., 1979; Rav, 1975). An experimental program (Von Riesemann et al., 1981) that has been initiated at Sandia National Laboratories should provide data that can be used in validating analytical models of containment response at loadings to failure.

The degree of effort and/or sophistication that should enter into the development of a failure pressure may vary with the scope of a particular PRA as well as the nature of the accident sequences that are found to be important. For example, for the class of accident sequences characterized by a loss of containment-heat removal, the pressure in the containment will, in the absence of recovery, increase monotonically to many times the design level, and failure may be a virtual certainty. If such sequences are found to be significant contributors to the risk profile for a particular design, it may be more meaningful to analyze the probability of recovering heat removal as a function of time than to try to pinpoint the failure level of the structure. As another example, the occurrence of large rapid hydrogen burns in certain types of containment can lead to pressures many times the design level. Here again, precise knowledge of the failure level would not be very important.

Analyses of core-melt accidents in various reactor and containment designs have indicated that pressures several times the design levels could be produced by a variety of mechanisms. At such pressure levels, the possibility of failure must clearly be considered. The degree of confidence to which a failure pressure or a failure criterion must be known must inevitably be related to the degree of reliance that is placed in the integrity of the structure at such extreme load conditions. At a minimum, analyses should be conducted to define the simple yield and ultimate-strength levels for the base structure. It should be possible to determine the former quite reliably with even a simplified analysis (Walser, 1980). While the ultimate strength of a thin-shell structure can also be determined quite simply, the validity of the results could be quite suspect since a simplified analysis would not account for nonlinear effects associated with large plastic strains and for interactions among the various components of a complex structure.

In addition to considering the gross behavior of the structure, special consideration should be given to localized conditions, such as the following:

1. Penetrations, including electrical penetrations and major openings (e.g., equipment and personnel hatches).

2. Major discontinuities, such as the transitions from the cylindrical shell to the top head and the basemat.

3. Layout and anchorage of the reinforcement.

4. Liner walls and anchoring.

5. Interactions with surrounding structures at large deformations.

While all of the above areas are considered in the design process, such considerations are limited to a well-defined envelope and may not be applicable far outside the normal region of consideration--for example, in situations involving large plastic strains.

In the analyses, actual property data should be used, where available, rather than general material specifications. In baselining the analysis, advantage may be taken of actual test data on the structure. For example, load-versus-deflection curves obtained during the strength testing could be compared against the analytical predictions, as could the concrete-cracking patterns that may be observed. The extent to which actual data can be used will obviously depend on the state of the plant for which the PRA is conducted. Clearly at the conceptual design stage general material specifications would have to be used, whereas for an existing plant actual measurements should be available.

While the internal pressure loading appears to be the principal determinant of potential containment failure, some consideration should be given to the possible effects of accident temperatures on the response of the containment. Temperature effects may be indirect, in that they may influence the strength characteristics of the structural materials, as well as direct: they may lead to the direct degradation of materials like penetration seals. The potential temperature effects on containment response would be expected to vary with the design of the structure; for example, large-volume containment structures may be less sensitive to temperature effects than are smaller structures, such as the BWR drywell, which has a smaller gas volume and heat sink for the superheated gases from a molten core.

7.5 GROUPING OF SEQUENCES

Chapter 3 describes the development of system event trees, whose end points represent plant conditions that can lead to accident sequences. In the preceding chapters these plant conditions were themselves called "accident sequences," but here the term "system sequence" is used; "accident sequence" is reserved for the end points of the containment event tree.* In a typical PRA, the number of system sequences that are identified is very large--much too large for the physical processes of each to be analyzed.

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*For a given plant condition, the containment event tree describes the various pathways that the accident might follow, particularly in terms of the physical processes that could lead to containment failure. A discrete pathway corresponds to a unique accident sequence.

Two approaches have been used to treat this problem: probability screening and the development of plant-damage bins. The former was used in the Reactor Safety Study Methodology Applications Program (Carlson et al., 1981). In this approach, a number of system sequences are selected by using point estimates to identify those with the highest frequencies. These are provided to the physical process analysts for evaluation. If the results of the analysis indicate that the spectrum of potential accident consequences is not well represented (e.g., there are no sequences that fall into large release categories), the level of discrimination is reduced and more sequences are analyzed. One problem with this approach is that it does require iteration and some judgment in deciding when the process is complete. It is consistent, however, with an approach to atmospheric dispersion and consequence analysis in which each dominant accident sequence is analyzed rather than grouped into release categories.

In the other approach, which has been used in a number of recent studies, the analyst develops groups of system sequences referred to as "plant-damage bins," "plant-damage states," or "plant event-sequence categories." The categories are identified by the characteristics of the system sequence that affect the release of radionuclides to the environment. All system sequences within a bin are assumed to have the same containment event tree, in that the branching probabilities are the same, and the end points are assigned to the same radionuclide release categories.

A potential problem with binning is that it presupposes a level of knowledge and skill that many analysts may not have. A combination of the two approaches might therefore be used: a variety of sequences are selected for analysis, and the binning is done after a significant number of sequences have been evaluated.

Some of the characteristics that are used to define bins are listed in Table 7-2 for a typical PWR. Other engineered safety features would, of course, be considered for a BWR or an ice-condenser plant. In practice, it is not necessary to consider a bin for each combination of these characteristics. Most bins would be vacant. In the Zion study, the system sequences were grouped into 21 plant-damage states.

Table 7-2. Bin characteristics[a]

Initiating event	Timing of core melt
Small LOCA	Early
Large LOCA	Late
Transients[b]	Performance of engineered containment safety features
Interfacing-systems LOCAs	No sprays or coolers
Vessel rupture	Coolers only
	Sprays only

[a]For a typical large dry PWR containment.
[b]A number of different types may be identified.

The development of bins requires interactions among the analysts involved in the activities described in Chapters 3, 6, 7, 8, and 9. The systems analysts (Chapter 3) provide a description of the initiating events and system faults of interest for the specific plant to the analysts of physical processes (Chapter 7). After some preliminary analysis, the analysts of physical processes identify the system-sequence characteristics that define the bins. This selection must be done cooperatively or in consultation with the analysis of radionuclide transport (Chapter 8) and environmental consequences (Chapter 9) because the ultimate criterion for grouping system sequences into one bin is the pattern of radionuclide release to the environment. The system sequences are then assigned to bins and returned to the quantification task (Chapter 6).

7.6 CONTAINMENT EVENT TREES AND THEIR QUANTIFICATION

By considering the success or failure states of active plant systems, the event trees described in Chapter 3 trace an accident sequence from the initiating event, through the onset of core damage, and to the point where a stable condition with intact fuel is achieved or where the fuel will overheat and proceed to melt. The containment event tree is developed to describe the progression of an accident sequence from the start of core melt to the release of radionuclides after containment failure, with particular emphasis on branch points that can result in containment failure or significantly affect the release of radionuclides.

The final branch points of the containment event tree are referred to as "accident sequences." The activities in performing a probabilistic risk analysis can be conceptually reduced to estimating the absolute frequency and consequences of all the sequences.

7.6.1 DEVELOPMENT OF CONTAINMENT EVENT TREES

Typically, containment event trees follow from the final branch points of system event trees. In the Limerick study (Philadelphia Electric Company, 1981) the concept of a bridge tree was used for special accident sequences in which there was an interaction between containment failure and subsequent core meltdown.

In the Reactor Safety Study (USNRC, 1975), the headings of the containment event tree were events postulated to lead to containment failure. However, it might be appropriate to include in the containment event tree events that significantly change accident consequences without failing the containment. For example, if an accident pathway could result in the formation of a coolable debris bed in the reactor cavity rather than attack on the concrete basemat, the consequences of the accident could be altered even if the modes of containment failure were unaffected.

7.6.1.1 Time and Location of Containment Failure

It might also be appropriate to recognize the potential for a particular type of containment failure or a particular event to occur at different times. Two important examples are steam overpressurization and hydrogen deflagration. In some sequences postulated for large dry PWR containments, the containment can be threatened by a rapid release of steam after the bottom head of the pressure vessel melts through. If the containment survives this steam spike, it may be challenged many hours later by the buildup of steam and noncondensable gases. The potential consequences of the later failure could be much smaller because of the time available for deposition processes to reduce the concentration of radionuclides in the containment atmosphere. Thus, a number of possible failure times should be included in the containment event tree.

Hydrogen combustion is subject to similar uncertainties. As the concentration of hydrogen in the containment increases, there could be a broad time period during which deflagration could occur, depending on the availability of an adequate ignition source. The potential for containment failure and the subsequent release of radionuclides would depend on the time of the ignition. The analyst might therefore decide to include a number of possible times for hydrogen-combustion events in the event tree. The containment event tree should not be expanded unnecessarily, however, because the number of subsequences that must be analyzed increases rapidly.

The location of containment failure can also be an important variable that can appear on the containment event tree. This is particularly true for pressure-suppression containments, in which the effectiveness of the suppression system (pools or ice beds) could be affected by the location of the failure.

7.6.1.2 Special Cases

The analysis may identify special cases that cannot be conveniently fit into the generalized containment event tree. An example is vessel rupture as an initiating event or as the result of a transient. Such an event could lead to missile generation and containment failure. This mechanism would not appear on the event tree for most accident sequences.

Another special case is containment isolation after an accident. This is an operation which would be expected to appear on the system event tree but which corresponds to a preexisting failure of the containment. In the Reactor Safety Study, containment-isolation failure appeared explicitly in the containment event tree. Indeed, for the BWR, several leak sizes as well as two isolation-failure locations were considered because of their possible influence on predicted behavior.

Another option, which keeps the functions of the system event tree and the containment event tree separate and more clearly defined, is to treat containment isolation as a separate case with its own containment event tree.

The interfacing-system LOCA sequence, in which the containment is bypassed before the meltthrough of the reactor vessel, is also a special case that can be assigned its own containment event tree.

It is convenient to set up the containment event tree in a time sequence because this allows logically nonsequential branches to be easily eliminated.

7.6.1.3 Examples of Containment Event Trees

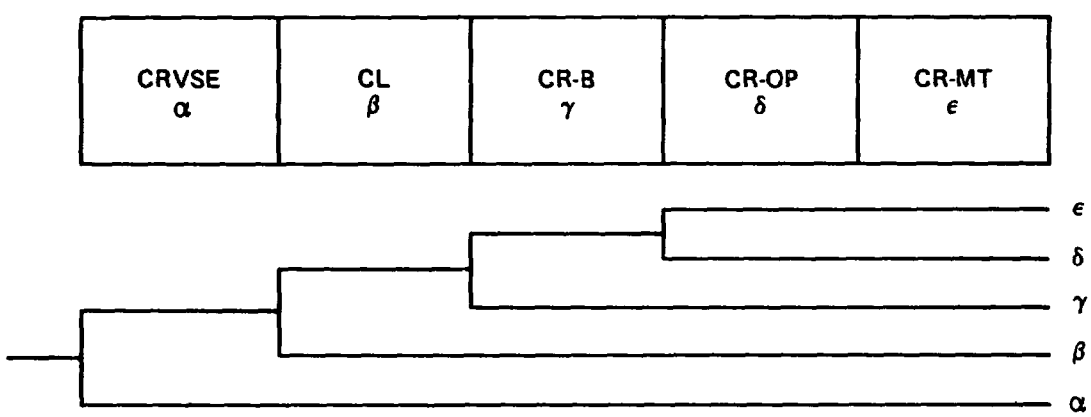
Containment Event Trees for PWRs

Figure 7-2 shows the simple containment event tree that was used in the Reactor Safety Study (USNRC, 1975). In the Zion PRA, a number of additional branch points were considered (Commonwealth Edison Company, 1981). The Zion event tree was first divided into key time periods of interest; for each time period the key binary branching decisions were identified (see Table 7-3). The specific binary decisions will depend on the design of the plant and the use of the PRA. Binary decisions that might be added to those in Table 7-3 are the following:

1. Does an in-vessel steam explosion result in containment failure?

2. Does an out-of-vessel steam explosion result in containment failure?

3. Does pocketing of hydrogen result in hydrogen detonation and subsequent containment failure?



CRVSE	Containment failure from in-vessel steam explosion
CL	Containment isolation failure
CR-B	Containment failure from hydrogen combustion
CR-OP	Containment failure from overpressurization
CR-MT	Containment failure through basemat penetration

Figure 7-2. Example of a containment event tree. From the Reactor Safety Study (USNRC, 1975).

Table 7-3. Typical binary branching decisions for the containment event tree of a large dry PWR containment[a]

EVENTS BEFORE CORE MELT

Is the containment pressure resulting from the initiating transient before any core degradation within the containment pressure limit?
Is sufficient hydrogen generated and released before core melt, and do conditions for the ignition of this hydrogen exist?
Is the containment pressure within the containment pressure limit?

EVENTS RELATED TO IN-VESSEL PHENOMENA

Does the postulated fuel melting progress noncoherently?
Is the pressure generated by the core debris-water interaction inside the reactor vessel within the pressure-boundary failure limits?
Is sufficient hydrogen generated and released before vessel failure, and do conditions for the ignition of this hydrogen exist?
Is the containment pressure within the containment pressure limit?
Do the conditions for in-vessel cooling of the core debris exist?
Is most of the core debris forcibly ejected after vessel failure?

EVENTS RELATED TO OUT-OF-VESSEL PHENOMENA AFTER VESSEL FAILURE

Is water present in the reactor cavity at the time of vessel failure?
Is the basemat perforated immediately after vessel failure?
Do the accumulators discharge, or does water return to the cavity after vessel failure?
Does the containment pressure from steaming alone exceed P_0 = 70 psia in the transient immediately after vessel failure?
Is the containment pressure from steaming alone within the containment pressure limit?
Is sufficient hydrogen available immediately after vessel failure, and do conditions for the ignition of this hydrogen exist?
Is the containment pressure within the containment pressure limit?

EVENTS RELATED TO ULTIMATE CORE DEBRIS DISPOSITION AND COOLABILITY

Does a coolable debris bed form initially?
Does the containment pressure remain within the containment pressure limit?
Is basemat failure prevented?

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[a]From the Zion Probabilistic Safety Study (Commonwealth Edison Company, 1981).

Containment Event Trees for BWRs

The form of the containment event tree depends on the design of the plant. The types of nodal questions that define the branch points in a BWR containment event tree will be different from those in a PWR event tree. Table 7-4 lists a number of nodal questions that would be appropriate for a Mark III BWR containment.

Table 7-4. Typical binary branching decisions for the containment event tree of a Mark III BWR containment

Does containment failure precede melting?
Does containment failure result in the disruption of the suppression pool?
Is the suppression pool bypassed?
Does a steam explosion occur during core meltdown?
Does a steam explosion cause containment failure?
Is the hydrogen-control system functional?
Does hydrogen combustion lead to containment failure?
Is there water in the reactor cavity before vessel meltthrough?
Does a steam explosion fail the containment?
Do drywell penetration seals fail because of high temperatures?
Does the suppression pool boil after containment failure?
Does basemat penetration occur?

7.6.2 QUANTIFICATION OF THE CONTAINMENT EVENT TREE

It is not possible to provide detailed guidance on the quantification of the branch points in the containment event tree. Some judgment will be required from the analyst. Since the state of knowledge about many of the key physical processes is changing rapidly, risk analysts will have to follow the results of research closely to remain abreast of developments.

The meaning of branching probabilities is frequently treated with some ambiguity. Because of the uncertainties in the prediction of physical processes, it is sometimes not possible to state with complete confidence which pathway an accident sequence will take. The branching probability in this sense represents a lack of knowledge about the physical processes that are involved.

In the real world one of the branches would be followed for all similar sequences. However, because of our inability to model the process with confidence, we cannot say which path that would be. Thus, we must judge the likelihood of each path being the correct one.

A branch point can also be attributed to variability in accident processes. For example, a specified composition of hydrogen and air may be within flammability limits, but, for burning to occur, an adequate ignition source must be present as well. The availability of ignition sources for practical purposes can be considered as a random process. Thus, in some accident sequences burning may occur, while in other, essentially identical sequences, burning would not occur. By grouping accident sequences in very narrow bins, some of the aspects of variability can be minimized.

Each approach that has been taken in assigning probabilities to the branch points of the containment event tree has required some degree of subjectivity. For some events it is possible to develop a prescription that can be applied to the results of analyses to determine the branch-point probabilities directly. In this approach the judgment enters in the development of the formulas for estimating the probabilities. For other events the analyst may not have sufficient information to do more than make a purely subjective judgment of branching probability. Some examples for various containment-failure modes are given in Sections 7.6.2.1, 7.6.2.2, and 7.6.2.3.

Another type of condition that can be encountered within the containment event tree is one in which the analyst has a high degree of confidence in the outcome, but recognizes a residual probability that he could be wrong. In this case, a small probability (e.g., 1×10^{-2} to 1×10^{-4}) may be assigned to the alternative branch point to ensure that this potential is recognized. This approach was used in the treatment of debris-bed coolability in the Zion study (Commonwealth Edison Company, 1981).

7.6.2.1 Overpressurization Failures

One of the most important types of containment failure involves overpressurization caused by the generation of steam (rapid or steady), the production of noncondensable gases, or the burning of hydrogen. The pressure level at which the containment would fail can be calculated (as discussed in Section 7.4), but only within some range of uncertainty; the same is true of the pressure history within the containment. In a prescriptive approach to determining failure probability for a specific accident sequence, the analyst develops a curve that shows the probability of failure as a function of containment pressure. Some judgment is required in developing this curve, but there are some points that can be determined easily. For example, at the design pressure, the probability of containment failure is near zero because the containment has been tested at this pressure; at the ultimate strength of the structure, the probability of failure must be unity.

In defining the rest of the curve, the analyst should consider the natural variability in material properties, uncertainties in analyzing the failure level for the structure, and the possibility of construction

defects. For a given sequence, the estimated probability of failure is the integral of the overlap of the peak pressure with its uncertainty distribution and the failure pressure with its uncertainty distribution. The approaches taken in the Reactor Safety Study (USNRC, 1975) and the Zion PRA (Commonwealth Edison Company, 1981) basically followed this type of procedure.

If the burning of hydrogen is involved in the particular containment overpressure failure, it might be appropriate to include a conditional probability of hydrogen combustion in the predicted failure probability. For the treatment of hydrogen combustion, it is possible to develop a prescriptive approach that is analogous to that described for overpressure failure. The conditions resulting in combustion (e.g., a flame-temperature criterion or a region on a ternary air-steam-hydrogen diagram) can be described with uncertainties. If the conditions predicted by the calculational method with their uncertainties overlap the conditions required for combustion, a probability of combustion can be estimated.

7.6.2.2 Steam-Explosion Failures

In the Reactor Safety Study, the probability of a steam explosion that would fail the containment was estimated by dividing the overall probability into three components: the probability of a coherent drop of a large mass of molten fuel into water, the probability of a steam explosion, and the likelihood that, given a steam explosion, the containment would fail. Each probability was determined subjectively but was based on a number of auxiliary calculations and a review of known steam-explosion mechanisms.

Recently, a more-detailed approach to estimating the probability of containment failure was undertaken for a PWR and a BWR (Corradini, 1981), using the results of experimental work performed since the Reactor Safety Study. In this work, the conditional probabilities of events that would result in containment failure were characterized as density functions and the overall probabilities were estimated by a Monte Carlo propagation.

7.6.2.3 Basemat Penetration

Although considerable research has been performed, major uncertainties remain in modeling the long-term behavior. In assigning a probability for this failure mode, the analyst must exercise considerable subjective judgment. In the Reactor Safety Study, it was assumed that basemat penetration was a certainty; the question that was considered was whether penetration would precede and prevent overpressure failure by releasing the gases in the containment into the ground. On the basis of parametric analyses, an uncertainty band (assumed Gaussian) was established around the time of basemat penetration. The time calculated for overpressure failure was then compared with this band to estimate the probability that penetration will precede containment failure through overpressure.

7.7 AVAILABLE METHODS OF ANALYSIS

The subsequent discussion of computer codes used in analyzing the physical processes of core-melt accidents covers only the codes that are publicly available. There is no intent to suggest that proprietary codes should not be used in PRAs. However, if proprietary codes are used, some documentation of the models will be required in the PRA, and it would be preferable to show how the results of the proprietary code compare with those of publicly available codes or with experiment.

The descriptions that follow are quite general. Furthermore, no attempt is made to list their limitations, because changes in the codes are being made so quickly that such a listing would be out of date by the time this guide is published. Some comparisons of code capabilities have been presented by other authors (see, for example, Rivard et al., 1981, Tables 5-II through 5-V).

The codes used in the analysis of physical processes are divided into three categories, according to their function: thermal-hydraulics, core melt, and core-concrete interactions. As shown in Table 7-5, they are available from the National Energy Software Center at the Argonne National Laboratory or the EPRI Software Center. Also discussed briefly are some of the codes that are used to analyze the structural response of the containment. These codes are not included in Table 7-5 because a great many codes with similar capabilities are available.

Table 7-5. Computer codes used in the analysis of physical processes

Code	Type	Source	Reference
RELAP5	System thermal-hydraulics transient	National Energy Software Center	Ransom et al. (1980)
RETRAN	System thermal-hydraulics transient	EPRI Software Center	Moore et al. (1978)
TRAC	System thermal-hydraulics transient	National Energy Software Center	Los Alamos National Laboratory (1981)
MARCH 1.1	Core-melt system code	National Energy Software Center	Wooton and Avci (1980)
RACAP	Core-melt system code	EPRI Software Center	Electric Power Research Institute (1981)
KESS	Core-melt system code	EPRI Software Center	Gulden et al. (1980)
CORCON-MOD1	Core-concrete interactions	National Energy Software Center	Muir et al. (1981)

7.7.1 CODES FOR ANALYZING THE THERMAL-HYDRAULICS OF TRANSIENTS AND LOCAS

The core-melt codes treat the initial phase of a core-melt accident simplistically, either by inputting tables of mass and enthalpy leak rates or by using a single-control-volume approximation. To provide assurance that the timing of core uncovering is not in significant error, it is advisable to analyze the early time period with a system thermal-hydraulics code like RETRAN (Moore et al., 1978), RELAP (Ransom et al., 1980), or TRAC (Los Alamos National Laboratory, 1981). The results of these analyses can be used as input to, or for the initialization of, the core-melt system codes, or they can be used to adjust parameters in the core-melt code.

Of the codes that are generally available for analyzing this phase of the accident, RETRAN has the most detailed treatment of the secondary and control systems. Some validation of the code has been made against mild plant transients. RELAP and TRAC are quite similar to RETRAN, however, in terms of the basic approach, strengths, and limitations. They all require a significant amount of computer time to analyze a typical transient or small-break LOCA. A sizable reduction in computation time has, however, been achieved with RELAP5, the newest version of the code.

7.7.2 CORE-MELT SYSTEM CODES

There are three core-melt system codes that are potentially available for use in performing risk analyses for LWR plants: MARCH, KESS (Gulden et al., 1980), and RACAP (EPRI, 1981). Of these, the MARCH code has been the most widely used and reviewed (Rivard et al., 1981). In addition, the MAAP code (Fauske and Henry, 1982), which is being developed by the Industry Degraded Core Rulemaking Program (IDCOR), should be available in the near future.

This guide does not recommend any one of these codes in preference to another. However, whichever code or method is selected, great care must be taken in its use. None of the codes have been validated against experimental data. Because the codes are in a developmental stage, many versions are in use. This can lead to ambiguity regarding the underlying assumptions of the model. The NRC has established a process for freezing reference versions of MARCH, updating the code, and informing users of identified problems.

This section describes the principal features of each of the core-melt system codes. Because of the developmental status of the codes, the comments made here about the limitations and the capabilities of the codes may become rapidly outdated. Before using one of these codes, the analyst should examine it in depth and should become thoroughly familiar with it. The codes should not be routinely applied without a continuing reevaluation of the applicability of models and assumptions for the conditions under consideration.

7.7.2.1 The MARCH Code

MARCH (Meltdown Accident Response Characteristics--Wooton and Avci, 1980) was written as a follow-on to the Reactor Safety Study. There were three principal objectives in its development:

1. To provide a consistent and integrated treatment of the stages of core melt.

2. To develop the capability to analyze transients and small-pipe-break accidents in addition to large-pipe-break accidents.

3. To develop generalized models capable of analyzing a variety of LWR containment designs.

MARCH was written to be compatible with the input needs of the CORRAL codes, which predict radionuclide transport and deposition inside the containment. Figure 7-3 shows the interfacing between MARCH and CORRAL II.

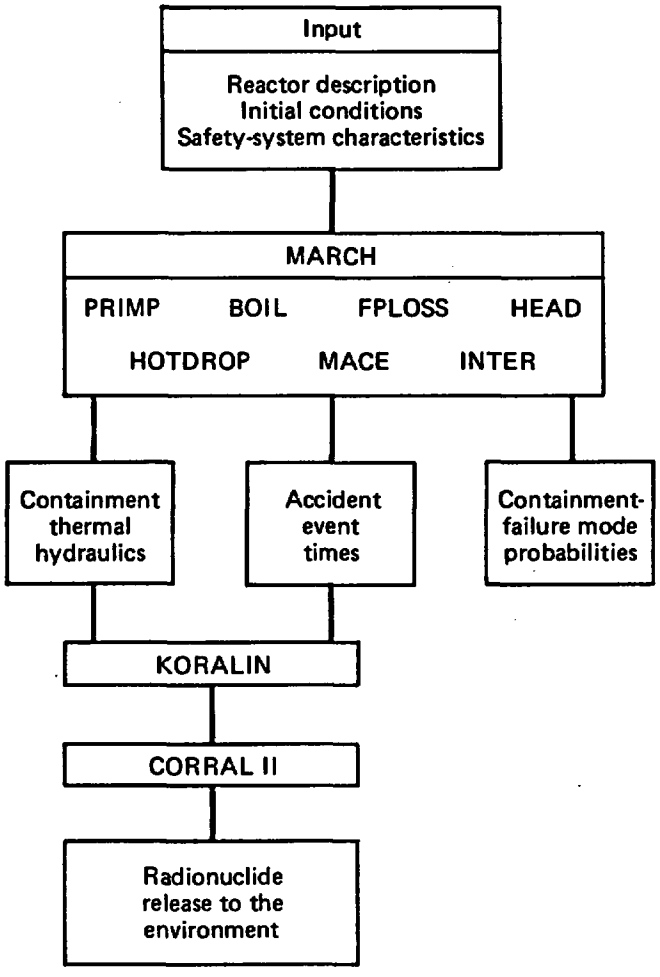

Figure 7-3. Flow diagram for MARCH/CORRAL analyses.

Subroutines in MARCH describe the major physical processes expected in a core-melt accident (see Figure 7-4). The analysis begins with the description of the hydraulic response of the system to the initiating event and proceeds to the uncovering of the core, the heatup of the fuel, the oxidation of the cladding, the liquefaction and slumping of the fuel, the boil-off of water in the lower plenum, attack on the reactor pressure vessel, interactions between the molten fuel and water in the reactor cavity, and attack on concrete. During the stages of core melt, the transient temperature and pressure history of the containment atmosphere is predicted. If the pressure exceeds an input criterion, containment failure is predicted to occur and the subsequent depressurization is analyzed. Hydrogen concentrations in the containment volumes are also followed, and the consequences of hydrogen combustion can be examined.

MARCH predicts the behavior of many complex physical processes. It contains, however, a number of well-recognized deficiencies. An examination of its limitations has been undertaken for the NRC (Rivard et al., 1981). Some limitations in the MARCH models arise from an inadequate supporting-data base. Others could have been corrected by improvements that are within the state of the art, but the necessary funding has not been available.

7.2.2.2 The RACAP Code

The RACAP code package (EPRI, 1981) includes modules for the analysis of the physical processes of core-melt accidents, radionuclide behavior, and offsite consequences. Interactions among the routines are shown in Figure 7-5. The INCOR part of RACAP corresponds to the MARCH code, and some of the INCOR modules are very similar to the subroutines in MARCH. The BOIL routines in MARCH and RACAP are both derivatives of the Reactor Safety Study's BOIL code. The INTER code is the basis for the modeling of core-concrete interactions in both code packages.

To predict the containment temperature and pressure transient, RACAP uses the CONTEMPT code. CONTEMPT performs a more rigorous treatment of intercompartment flow than does the MACE routine in the MARCH code. In particular, CONTEMPT can account for pressure differences between interconnected compartments during periods of rapid pressure change. The analysis of reactor-vessel meltthrough also differs from the treatment in MARCH.

7.7.2.3 The KESS Code

The KESS code package (Gulden et al., 1980) was developed in the Federal Republic of Germany. KESS is designed on a modular basis, using an executive-code management approach that allows for a number of modeling options and close coupling between models at some expense in computation time. The two levels of control and data transfer in the executive program are shown in Figure 7-6. The computer codes currently available in the KESS system are shown in Figure 7-7. In general, alternative modules can be

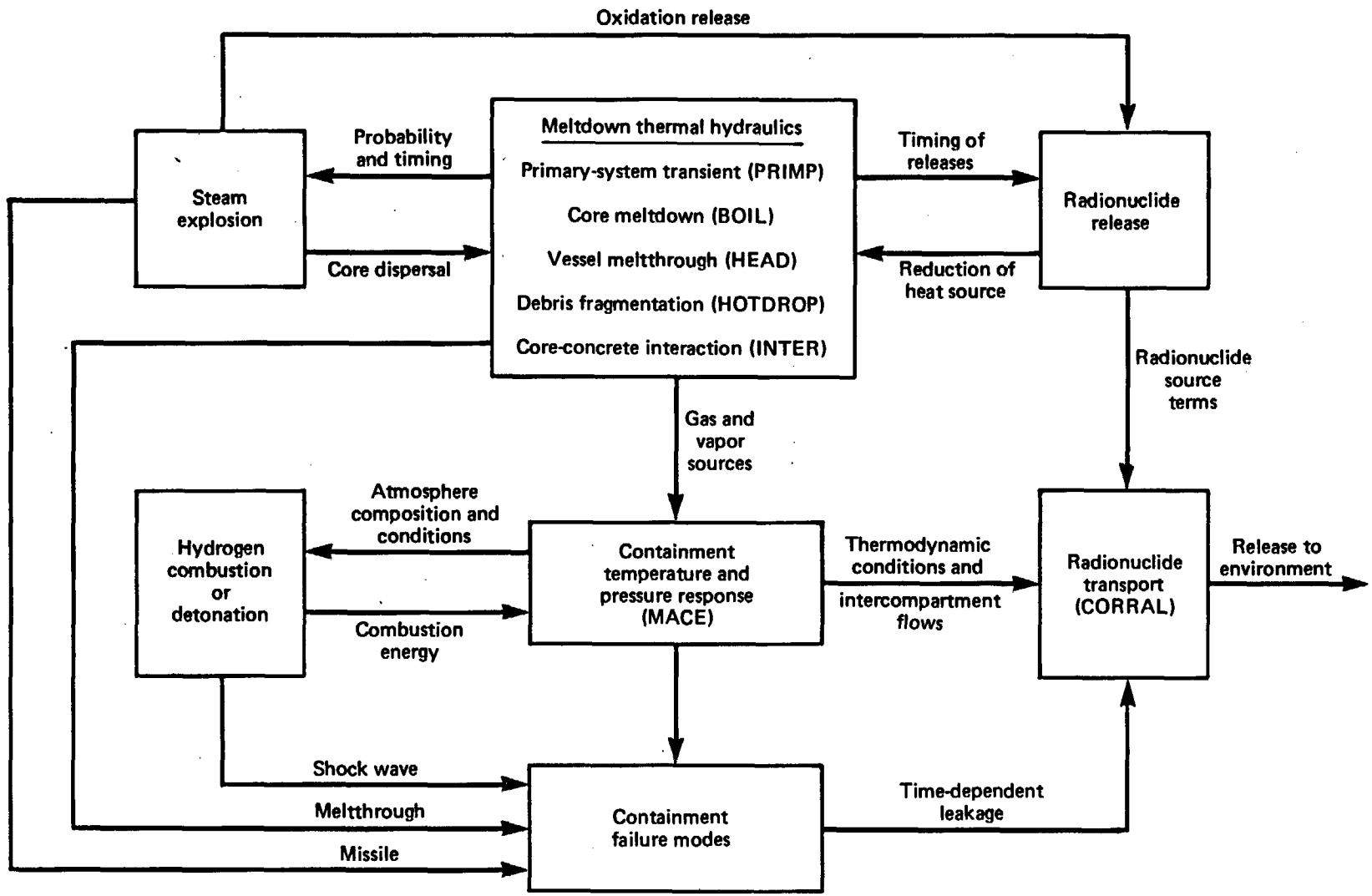

Figure 7-4. Core-melt processes.

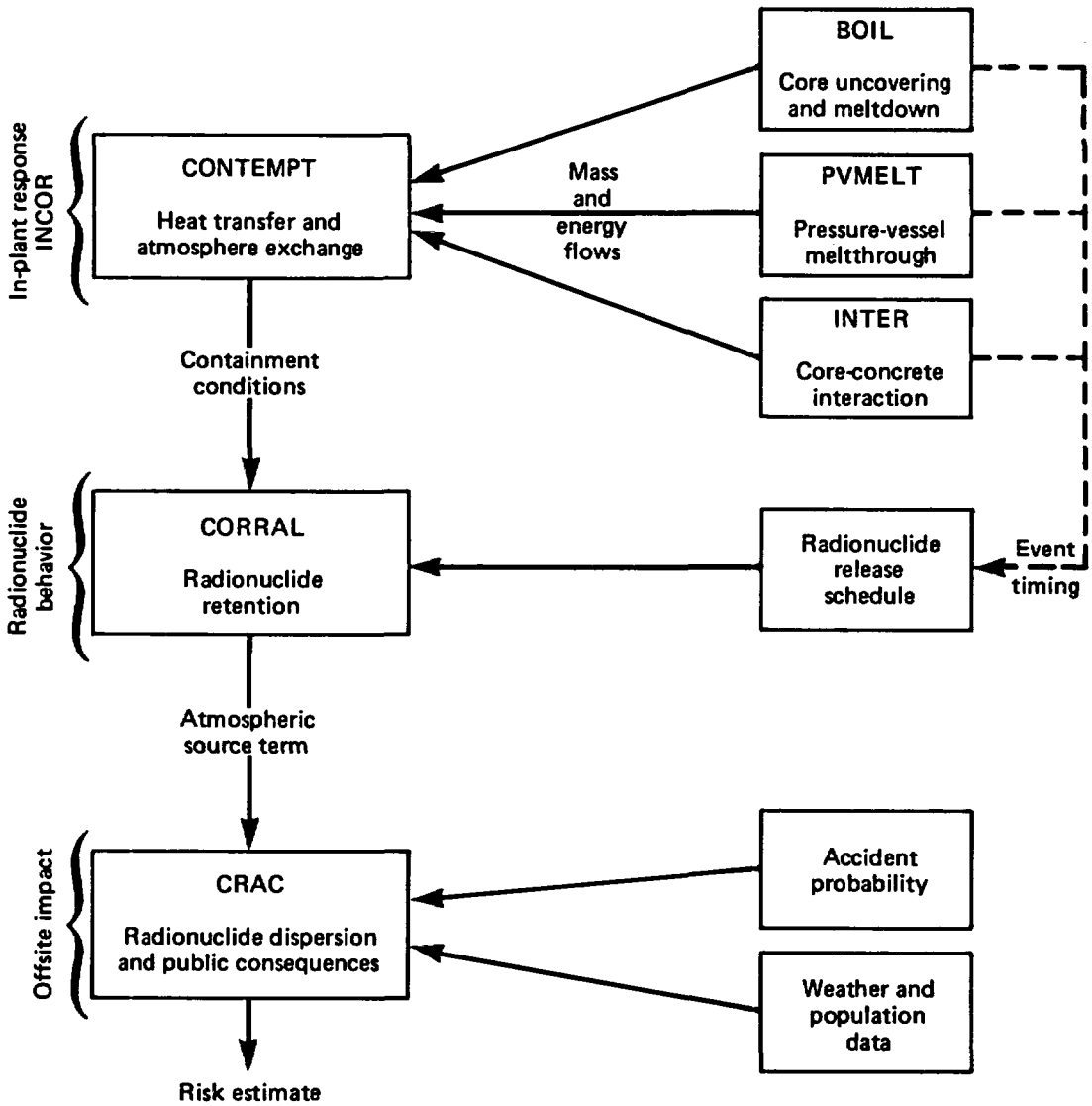

Figure 7-5. The RACAP code network for accident-consequence analysis.

selected for each phase of the accident--a simple model and a more complex model.

Some of the models in KESS are more detailed than the analogous models in MARCH. For example, the MELSIM model is more detailed in the description of fuel slumping than the slumping models in BOIL. Similarly, the RAUHZ module examines heat transfer from a pool in which natural convection is driven by internal heat generation. This mechanism is not considered in the HEAD routine of MARCH. Further experimentation and analysis are required, however, to determine which models are more appropriate under different conditions.

The KESS code has been made available in the United States through information-exchange arrangements with the NRC and EPRI. The currently

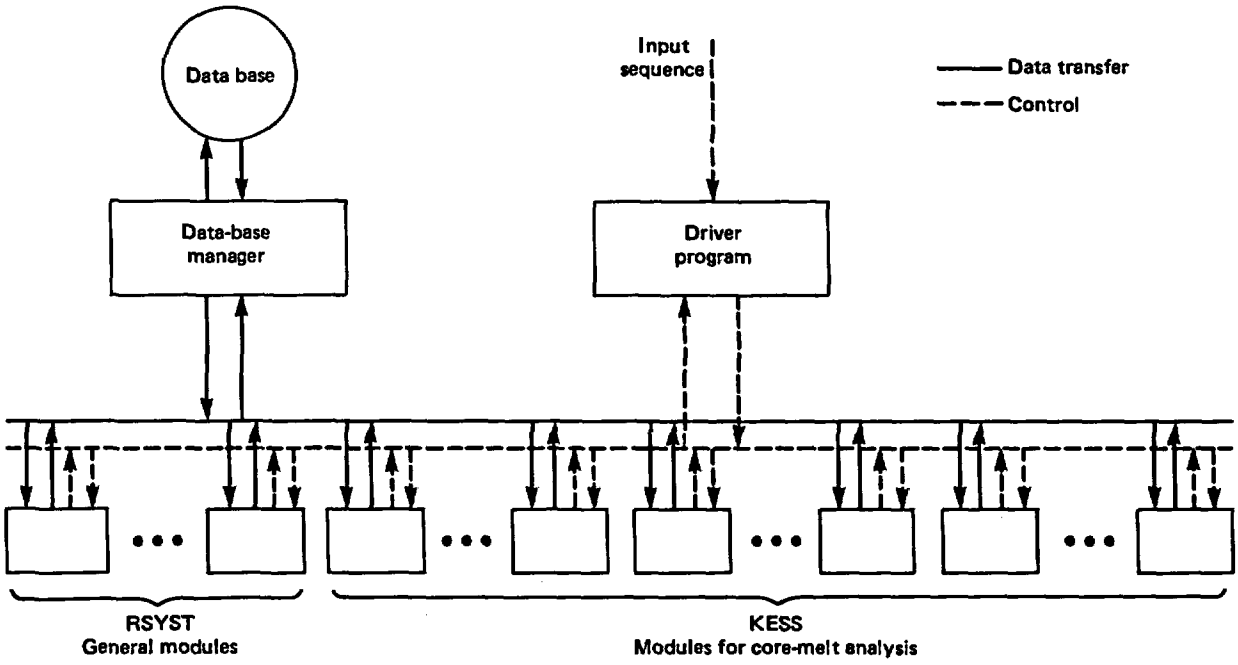

Figure 7-6. Control and data transfer in the KESS executive program.

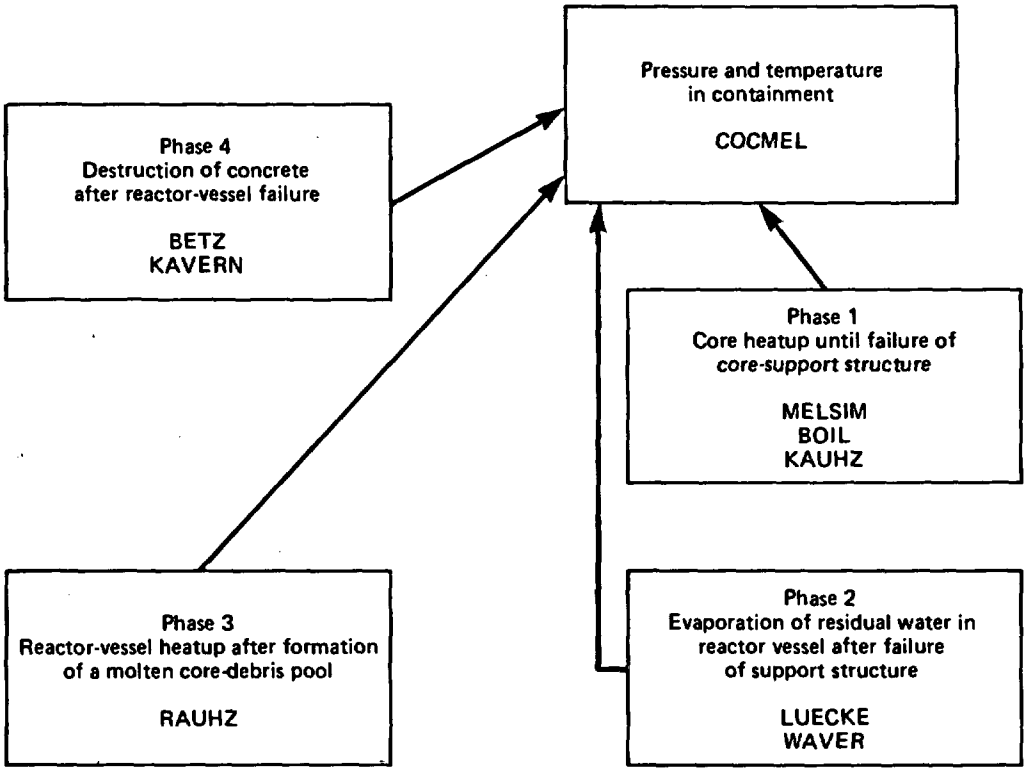

Figure 7-7. Computer codes incorporated into KESS.

available version has some significant limitations for use in risk analyses. This version does not have the capability to analyze transients and small-pipe-break accidents, nor does the containment code have spray, suppression-pool, or ice-condenser models. A number of additional modules are being added to KESS by the German researchers, and updated versions of the routines in KESS should be soon available in the United States.

7.7.2.4 Separate-Effects Codes

The conditions that would lead to a containment failure by overpressurization are well beyond those that are analyzed in the design of the containment. However, a number of computer codes are available for analyzing the behavior of the containment through the range of gross yielding. Among the codes that are generally available are HONDO (Key et al., 1978), ADINA (Bathe, 1978), NASTRAN (MacNeal, 1978), and MARC (Marcal, 1975). Codes of this type normally use finite-element methods and can perform two- or three-dimensional nonlinear analyses for complex materials like concrete. Multipurpose shock-hydrodynamics codes are also available to evaluate the impact on the containment wall of shock waves or missiles that could be generated in a steam explosion or a hydrogen detonation.

The most advanced American code for modeling core-concrete interactions is CORCON (Muir et al., 1981), which has been included in some experimental versions of MARCH. The principal components of the CORCON model are the concrete cavity, the molten pool of core debris, and the gas atmosphere and surroundings above the pool. CORCON considers mass and energy transport and conservation within this system. Analytical models are provided for the pertinent physical phenomena and chemical interactions, including heat transfer, the ablation of concrete and changes in the shape of the reactor cavity, heat transfer inside the molten pool and from the surface of the pool to the atmosphere and the surroundings, chemical reactions between the molten pool and gases, and decay-heat generation in the molten pool. The MOD1 version is applicable only to the high-temperature phase of core-concrete interactions, when the core debris is hot enough to be entirely liquid and to erode the concrete at a relatively rapid rate.

7.7.2.5 Codes Under Development

Three computer codes that are currently under development will examine specific aspects of core-melt processes. These codes are SCDAP, CONTAIN, and HECTR. The SCDAP code (Allison et al., 1981) was developed at the Idaho National Engineering Laboratory to model fuel behavior during core heatup and melting inside the reactor pressure vessel. It is being validated by comparison with the results of the NRC's experimental program on severe fuel damage.

CONTAIN (Senglaub et al., 1981) will analyze containment processes in severe accidents in either light-water reactors or liquid-metal fast breeders. The analysis couples radionuclide transport with the thermal-hydraulic behavior. The objective of this code, which is being developed at Sandia National Laboratories, is to provide detailed mechanistic models for containment processes--models that can be used as a benchmark for other codes.

The HECTR code (Berman, 1981a) is being developed at Sandia National Laboratories in close conjunction with the NRC research program on hydrogen behavior. It will treat the combustion of hydrogen in a more mechanistic manner than do the models in existing core-melt system codes.

7.8 UNCERTAINTY ANALYSIS

7.8.1 SOURCES OF UNCERTAINTY

Uncertainties in the analysis of the physical processes of core-melt sequences enter into the results of a probabilistic risk assessment in two ways. First, the uncertainties affect the estimates of the frequencies of accident sequences. These uncertainties are therefore reflected as the probabilities of branches in the containment event tree or as distributions on these probabilities. Second, the uncertainties appear as variations in the output variables from the analysis. These variables (e.g., temperatures, compositions, flows) are used as input to the models of radionuclide release and transport in the plant and in the environment. These uncertainties can therefore be propagated through the radionuclide-transport models and be represented as distributions on the radiological consequences of the accident sequences.

Some of the principal modeling uncertainties that affect the predictions of radionuclide release and transport are related to the following:

1. The thermal history of the fuel.

2. Temperature distributions and flows in the reactor-coolant system and the containment.

3. The relative timing of core melt and containment failure.

4. The mode of containment failure.

5. The coolability of core-debris configurations.

6. The generation and combustion of hydrogen.

7. Fuel-coolant interactions.

7.8.2 METHODS OF ANALYSIS

The use of some subjective judgment in assigning uncertainties to accident consequences appears to be unavoidable. For some input parameters, experimental data are available, and from these data distributions can be inferred. Frequently, however, the experimental bases for input variables and models of physical processes are quite limited. Judgment is therefore required in the assignment of uncertainties.

There is some concern that the subjective assignment of uncertainties implies more knowledge about the results of a PRA than actually exists. On the other hand, the estimated uncertainties in the risk provide very important insights even if they only represent the best judgment of the analyst. The effect of assumptions underlying the subjective representation of uncertainties can be determined by performing sensitivity studies. It is therefore recommended that, when a potentially controversial judgment is made regarding the progress of an accident, the effect of the judgment be evaluated by also performing the analysis with a different (possibly more conservative) assumption.

Chapter 8 presents the results of formal uncertainty analyses for the MARCH and CORRAL codes. This type of analysis is useful in determining how uncertainties in the input variables for physical process analysis affect radiological consequences. It can also be used to some extent to evaluate the implications of different models for radiological consequences. Since the analyses are performed within the context of existing models in computer codes, they cannot fully account for all sources of uncertainty. Informed judgment must therefore be used to extend the ranges of uncertainty obtained by formal methods of uncertainty analysis.

7.8.3 AVAILABLE INFORMATION ON UNCERTAINTY AND VARIABILITY

An uncertainty analysis has been performed with the MARCH and CORRAL codes for some specific accident sequences. Some of the results are tabulated in Chapter 8.

7.9 INFORMATION REQUIREMENTS

A large amount of plant design information is needed to analyze the physical processes of core-melt accidents. Since some of the output (compartment temperatures and intercompartment flows) is used as input for the analysis of radionuclide release and transport, the two groups of analysts must agree on the appropriate breakdown of the containment into control volumes.

The initial source of plant data is the final safety analysis report (FSAR). The FSAR will not contain all of the necessary information,

however. Liaison with the utility or the equipment vendor and architect-engineer must be established to obtain detailed plant drawings and specifications. At least one plant tour should also be made at a point midway in the data-acquisition stage to confirm assumptions and answer questions.

Table 7-6 identifies plant data that must be input to a core-melt code. Some analysis is required to convert raw plant data into this form. For the reactor-coolant system, the following information is required: fuel design, core power distribution, masses and quantities of different materials, the design of the upper and lower internals, and the design of the reactor vessel. For the containment it is necessary to know the overall dimensions, air volume, the dimensions and material compositions of heat sinks (noting whether heat sinks are one-sided or two-sided), and interconnections between subregions.

Table 7-6. Plant-data input to core-melt codes

System or component	Parameters included
General containment data	Total volume; number of compartments; volume and dimensions of compartments; initial pressure, temperature, and humidity
Heat sink	Number and compartment location of heat-sink slabs; materials in slab, including density, heat capacity, and thermal conductivity; heat-transfer area, thickness, and heat-transfer coefficient for the liner-concrete interface
Ice condenser (if applicable)	Mass of ice; temperature of ice; temperatures of water drained from ice bed; temperature of gas leaving ice bed
Suppression pool (if applicable)	Mass of water; temperature of water; water volume; air volume
Containment floor (for core-concrete interactions)	Thickness, density, thermal conductivity, temperature, and composition of concrete containment floor
ECC tanks	Pressure, temperature, and water mass of accumulators and/or upper head injection tanks
ECC pumps	Start time, nominal flow rate, nominal and shut-off pressure of all pumps, including high-pressure injection, safety injection, low-head pumps, and any additional pumps; minimum temperature to avoid pump cavitation
ECC heat exchangers	Heat-exchanger capacity; primary and secondary flow rates and temperatures for ECC and containment-spray heat exchangers
Containment coolers (if applicable)	Number and location of coolers; air-flow rate and inlet temperature; secondary flow rate and inlet temperature
Containment sprays	Flow rate, temperature, and spray-drop diameter of containment-spray system

Table 7-6. Plant-data input to core-melt codes (continued)

System or component	Parameters included
Auxiliary feedwater (if applicable)	Flow, temperature, and start time of auxiliary feedwater pumps
Water-supply parameters	Mass of water in condensate-storage tank; mass of water in the refueling-water storage tank (RWST); fractional value of RWST to start recirculation of ECC and containment sprays; minimum sump mass to avoid cavitation
Core	Initial thermal power; total number of lattice positions in core; total number of fuel rods in core; active fuel height; liquid level; mass of UO_2, Zircaloy, and miscellaneous metal; fuel-rod diameter; fuel-pellet diameter; hydraulic diameter; cladding thickness; density, conductivity; and heat capacity of core material; peaking factors
Vessel	Code diameter; flow area; cross-sectional area; mass, heat capacity, temperature, and heat-transfer area of internal structures; mass, diameter, and thickness of bottom head
Reactor-coolant system	Volume; initial primary steam volume; pressure; safety-relief-valve pressure setpoint and rated capacity
Steam generator	Initial mass of water in steam generator; volume of steam generator; setpoint of secondary steam-generator relief valve

Information is also needed about the engineered safety features (ESFs): the number, capacity, requirements for net positive suction head, failure mechanisms, and the temperature of the source water. The analyst must know the logic of ESF operation: What triggers their operation? Are there alternative operating modes? Does more than one system compete for the same source of water? Emergency operating procedures must be reviewed to determine how the operator will interact with the system for a particular accident situation. The analyst must remember that the intent of the analysis is realism. The flow rates and water-source temperatures provided in SARs are frequently conservative. The analyst must also decide what constitutes an operable state for a system. If the emergency core-cooling (ECC) system is operational and two of three pumps must function for success, should the analyst assume that two or three pumps are operating? In the Reactor Safety Study, a minimum safeguards assumption was made. This assumption does not necessarily represent the most likely mode of operation, nor is it necessarily conservative. Emergency operating procedures may provide quidance, but consideration should also be given to sensitivity studies.

7.10 PROCEDURES

The depth at which physical processes should be analyzed in a risk assessment depends on the use of the study. Procedures for two types of analysis are described in this section. The first set of procedures outlines the steps that would be undertaken in a detailed PRA in which an in-depth treatment of accident consequences is performed. For a reliability-oriented risk study, like those conducted in the Interim Reliability Evaluation Program (Mays et al., 1981), a more limited treatment of physical processes is sufficient. The steps in this type of analysis are presented as the second set of procedures. The depths of analysis that are described for the two sets of procedures are actually end points on a spectrum of possibilities. The analyst must decide the appropriate depth of analysis for the specific application. The major tasks are illustrated in Figure 7-1.

7.10.1 DETAILED ANALYSIS OF PHYSICAL PROCESSES

Task 1: Collect Plant Data

1. Review the FSAR. Collect data on system design, ESF operating levels, etc., as required to provide the data listed in Table 7-6.

2. Establish liaison with utility staff, the vendor of the nuclear steam supply system, and the architect-engineer. Obtain plant drawings and operating procedures. Provide a list of missing data (as early as possible).

3. Make one or two plant visits to answer questions and verify assumptions about the plant layout (e.g., connections to the sump, flow paths between compartments).

Task 2: Model Plant

1. Develop plant model to be used in core-melt analyses for each accident sequence (e.g., MARCH analyses). This must be done in cooperation with the radionuclide release and transport task (Chapter 8).

2. Develop models for separate-effects analyses (e.g., containment structure, shock-hydrodynamic analysis of hydrogen detonation, debris-bed coolability) as required. The level of detail in these models will depend on the specific application and the requirements of the analysis techniques.

3. Reduce plant data to the engineering units required as code input.

4. Assist in the development of success and failure criteria for engineered safety features. This step is usually the responsibility of the systems analysts (Chapter 3). However, separate-effects analyses are frequently required to determine which conditions or sequences result in core melt. In the Reactor Safety Study, it was

assumed that if the criterion for the peak cladding temperature (2200°F) is exceeded, core melt will result. From experimental data and the behavior observed in the accident at Three Mile Island, it may be possible to defend a criterion that is less conservative.

Task 3: Determine Containment-Failure Mechanisms and Levels

1. Identify a comprehensive list of potential containment-failure mechanisms (see Table 7-1 for example).

2. Perform structural analyses of the containment to determine the steady internal pressure resulting in containment failure. Identify possible modes and locations of failure. The analysis should recognize that a range of possible failure pressures could exist from some level above the design pressure up to the ultimate strength of the containment, including the potential for stress concentrations and manufacturing defects. A density function for failure pressure should be developed.

3. Perform separate-effects analyses for the other potential mechanisms of containment failure to determine (a) whether the mechanism is credible, (b) the conditions under which containment failure would result, and (c) the likely locations and modes of containment failure.

Task 4: Select Analysis Methods for Physical Processes

1. Identify analysis requirements. Consider the special features of the reactor design that could require separate-effects analyses or changes in existing codes.

2. Select a code for the core-melt analysis (e.g., MARCH) and separate-effects codes as necessary.

3. Develop models or modify codes as required.

Task 5: Develop Bins for Accident Sequences

1. Receive system sequences from the task of accident-sequence definition and system modeling (Chapter 3).

2. Identify the initiating events, ESF states, and core-melt characteristics that can be used to group system sequences (see Table 7-2). This should be done in consultation with the analysts of radionuclide release and transport as well as the analysts of environmental transport and consequences to ensure that the release categories assigned to different sequences within a bin are common and that the branching probabilities on the containment event tree are the same.

3. Assign system sequences to plant-damage bins and provide to the analysts who will perform the accident-sequence quantification (Chapter 6).

Alternative Procedure

Receive from the task of accident-sequence quantification a small set of dominant system sequences that have been identified by probability discrimination. After release categories for these sequences are determined, consider the need for analyzing more sequences.

Task 6: Develop Containment Event Tree

1. Divide the accident into the major time periods of interest as in Table 7-3.

2. Select event-tree headings. The nodal questions in Tables 7-3 and 7-4 can be used as a guide. Add or delete headings, depending on the special features of the plant.

3. Order event-tree headings and describe the structure of the tree. The size of the tree is affected by the order of events. In general, events should be ordered on the tree in the temporal sequence in which they would actually occur. Unnecessary or meaningless branches may be removed from the tree.

Task 7: Analyze Accident Sequences

1. Provide preliminary assistance to the analysts involved in accident-sequence definition and system modeling in identifying plant conditions leading to core melt as required.

2. Select a representative sequence for each bin.

3. Determine the status of operating systems for the accident sequence. This includes not only whether a system is operating but also the level of operation (e.g., two of three pumps at 150 gpm each). Describe initial and boundary conditions.

4. Perform analyses to describe the transient power, thermal, and hydraulic behavior before core damage. Benchmark or tune the core-melt code.

5. Identify the containment-failure modes to be evaluated for each sequence. An accident sequence associated with each containment-failure mode will be analyzed.

6. Analyze the physical processes for each accident sequence using the core-melt code. Separate-effects analyses may be necessary to determine the time and the conditions of containment failure. Provide the results to the analysts of radionuclide release and transport as well as environmental transport and consequences.

Task 8: Perform Sensitivity Studies

1. Identify potentially sensitive parameters. In particular, consider parameters that could affect the likelihood or the time of containment failure.

2. Perform sensitivity studies by varying assumptions and values of input parameters over the range of uncertainty.

3. Provide results to the uncertainty-analysis task.

Task 9: Quantify Containment Event Tree

1. Develop a systematic approach to the characterization of branch-point probabilities, with or without uncertainties, consistent with the overall philosophy of the study.

2. Compare the predicted pressure profile for each sequence with the distribution function for failure pressure to determine the probability of containment failure (including sequences with hydrogen burning).

3. On the basis of the results obtained in tasks 8 and 9, use subjective judgment to predict branch-point probabilities (and uncertainty bands).

4. Provide results to the uncertainty-analysis task and to the integration task.

7.10.2 LIMITED ANALYSIS OF PHYSICAL PROCESSES*

Task 1: Collect Plant Data

Collect and review FSAR data on the design of the containment and the nuclear steam supply system. Compare with analogous features from previous risk studies.

Task 2: Model Plant

1. Develop plant model to be used in core-melt analyses of selected accident sequences.

2. Reduce plant data to the engineering units required as code input.

Task 3: Determine Containment-Failure Mechanisms and Levels

1. Identify containment-failure mechanisms by analogy with similar plants.

2. Estimate the containment-failure pressure on the basis of building-code requirements for the specific structure (e.g., a factor of 2 to 3, depending on the type of design).

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*Since sensitivity studies would usually not be performed in this level of analysis, task 8 of Section 7.10.1 is omitted.

Task 4: Select Analysis Methods for Physical Processes

Select a code for core-melt analysis.

Task 5: Develop Bins for Accident Sequences

1. Receive system sequences from the task of accident-sequence definition and system modeling.

2. Select bin characteristics.

3. Assign system sequences to bins and provide to the task of accident-sequence quantification.

Alternative Procedure

Do not use binning approach.

Task 6: Develop Containment Event Tree

Develop a containment event tree by analogy with similar plant designs.

Task 7: Analyze Accident Sequences

1. Identify sequences for analysis. Sequences that are expected to have small consequences or are comparable to sequences analyzed previously in a similar plant would not be analyzed. The criteria for selecting sequences for analysis are as follows:

   a. It is unclear whether the sequence leads to core melt, or

   b. It is unclear whether the sequence leads to containment failure, or

   c. The sequence is substantially different from those analyzed previously.

2. Perform core-melt or separate-effects analyses as required.

Task 8: Quantify Containment Event Tree

Estimate branch-point probabilities by analogy with other studies or from the results for the few sequences analyzed specifically for the plant.

7.11 METHODS OF DOCUMENTATION

The amount of documentation that is required for the analysis of physical processes depends on the purpose of the study. In general, it should not be necessary to provide so much information that a reviewer can independently operate the computer codes to duplicate the calculations. It

should be assumed, however, that the risk analysis will be subjected to an extensive review by peers. Some of the information that should be documented in the report is listed below.

1. Sources of data.

2. Tables of plant-design data.

3. Computer codes (names and brief descriptions).

4. Major model options.

5. Tables of sequence probabilities and uncertainties.

6. Tables of accident event times and containment conditions.

7. Figures illustrating containment conditions for selected sequences.

8. A list of all assumptions.

9. A list of all limitations of the study.

10. Data documenting and justifying the containment event tree.

11. A justification and description of the basis for branching probabilities.

7.12 DISPLAY OF FINAL RESULTS

The outputs of this task are the thermal-hydraulics conditions for each accident sequence as required for the analysis of radionuclide transport and the conditional probabilities of sequences. If the core-melt code and the radionuclide-transport code are compatible (e.g., MARCH and CORRAL), the interface between the codes can be automatically determined--for example, by storing the output files of the physical process analysis on tape for later use in the radionuclide-transport code. If there is no formal link between the analysis methods, close liaison between the radionuclide-transport task and the physical process task will be necessary to ensure that the data are provided in a convenient format. Since a limited amount of thermal-hydraulics data produced by the core-melt codes is used in the transport codes, care must be exercised in the interpolation or averaging of the thermal-hydraulics data to be certain that the reduction process gives truly representative and reproducible results.

The probabilities of sequences can be characterized by point estimates or distributions, depending on the method selected for the propagation of uncertainties.

7.13 ASSURANCE OF TECHNICAL QUALITY

None of the currently available core-melt codes have been adequately validated against experiments. As the testing and validation of these codes progress, it would be advisable to use controlled versions of the codes that can be referenced. Since there are a number of available options in the core-melt codes, the selected options should be documented. Before any analyses are made it would be advisable to identify (list) all of the options available in the code. A conscious selection of options should then be made and frozen. If at a later time changes in options are advisable, they should be made with the approval of the project management and not left to the judgment of the analysts. This does not mean that a variety of code options should not be used to test the importance of modeling assumptions.

A formal procedure should be established for checking code input and results. The cost of review can be very high, and project management must decide the extent of review that is warranted. It should be recognized, however, that experience indicates a very high incidence of errors in preparing code input. Reference cases should be performed with the computer codes to demonstrate that the codes are operating correctly.

REFERENCES

Allison, C. M., B. O. Hagrman, S. Hsieh, E. T. Laats, and J. W. Spore, 1981. Severe Core Damage Analysis Package (SCDAP), Code Conceptual Design Report, EGG-CDAP-5937, EG&G Idaho, Inc., Idaho Falls, Idaho.

American Society of Mechanical Engineers, 1980. 1980 ASME Boiler and Pressure Vessel Code, An American National Standard, Section III Rules for Construction of Nuclear Power Plant Components, Philadelphia, Pa.

Aoyagi, Y., et al., 1979. "Behaviours of Reinforced Concrete Containment Models Under Thermal Gradient and Internal Pressure," in Transactions of the 5th SMIRT Conference, Paper No. J4/5, Berlin.

Atchison, R. J., G. J. K. Asmis, and F. R. Campbell, 1979. "Behavior of Concrete Containment Under Over-Pressure Conditions," in Transactions of the 5th SMIRT Conference, Paper No. J3/2, Berlin.

Baker, L., Jr., et al., 1977. Post Accident Heat Removal Technology, ANL/RAS 77-2, Argonne National Laboratory, Argonne, Ill.

Bathe, K. J., 1978. ADINA: A Finite Element Program for Automatic Dynamic Incremental Nonlinear Analysis, Report 82448-1, prepared by the Massachusetts Institute of Technology for the National Aeronautics and Space Administration. (Published in September 1975; revised in December 1978.)

Berman, M., 1981a. Monthly Progress Report on Sandia Hydrogen Programs, October and November of 1981, Sandia National Laboratories, Albuquerque, N.M.

Berman, M., 1981b. Light Water Reactor Safety Research Program Quarterly Report, January-March, 1981, USNRC Report NUREG/CR-2163 (SAND81-1216, Sandia National Laboratories, Albuquerque, N.M.).

Berman, M., 1981c. Proceedings of the Workshop on the Impact of Hydrogen on Water Reactor Safety, USNRC Report NUREG/CR-2017 (SAND81-0661, Sandia National Laboratories, Albuquerque, N.M.).

BMFT (Bundesministerium fuer Forschung und Technologie), 1980. Investigation of Partial Core Meltdowns, BMFT-RS-380, quarterly reports for 1979, Bonn, Federal Republic of Germany.

Carlson, D. D., et al., 1981. Reactor Safety Study Methodology Applications Program: Sequoyah #1 PWR Power Plant, USNRC Report NUREG/CR-1659, Vol. 1.

Commonwealth Edison Company, 1981. Zion Probabilistic Safety Study, Chicago, Ill.

Corradini, M. L., 1981. Analysis and Modelling of Steam Explosion Experiments, USNRC Report NUREG/CR-2072 (Sandia National Laboratories, Albuquerque, N.M.).

Corradini, M. L., and D. V. Swenson, 1981. Probability of Containment Failure Due to Steam Explosions During a Postulated Core Meltdown Accident in a LWR, SAND80-2132, Sandia National Laboratories, Albuquerque, N.M.

Denny, V. E., 1982. "Analytical Models for Core Heatup, Liquefaction, and Slumping," International Meeting on Thermal Nuclear Reactor Safety, August 1982.

Dhir, V. K., and I. Catton, 1977. Study of Dryout Heat Fluxes in Bed of Inductively Heated Particles, USNRC Report NUREG-0262.

Donten, K., et al., 1979. "Results of Strength Tests on a 1:10 Model of Reactor Containment," in Transactions of the 5th SMIRT Conference, Paper No. J4/8, Berlin.

EPRI (Electric Power Research Institute), 1981. RACAP-1: Reactor Accident Consequences Analysis Program, First Version, EPRI NP-1871, Palo Alto, Calif.

Fauske, H. K., and R. E. Henry, 1982. "IDCOR Modular Accident Analysis Program, An Enhanced Analytic Capability To Model Core Degradation Phenomena," Transactions of the American Nuclear Society, Vol. 41.

Gulden, W., et al., 1980. KESS, A Program System for the Analysis of Hypothetical Core Meltdown Accidents, USNRC Translation 837A.

Hagen, S., and H. Malauschek, 1979. "Bundle Experiments on the Meltdown Behavior of PWR Fuel Rods," Transactions of the American Nuclear Society, Vol. 33.

Hall, R. E., et al., 1979. A Risk Assessment of a Pressurized-Water Reactor for Class 3-8 Accidents, USNRC Report NUREG/CR-0603 (Brookhaven National Laboratory, Upton, N.Y.).

Hardee, H. C., and R. H. Nilson, 1977. "Natural Convection in Porous Media with Heat Generation," Nuclear Science and Engineering, Vol. 63, p. 119.

Henry, R. E., and H. K. Fauske, 1979. "Nucleation Processes in Large Scale Vapor Explosions," Journal of Heat Transfer, Vol. 101, pp. 280-287.

Key, S. W., Z. E. Bersinger, and R. D. Krieg, 1978. HONDO II, A Finite Element Computer Program for the Large Deformation Dynamic Response of Axisymmetric Solids, SAND78-0422, Sandia National Laboratories, Albuquerque, N.M.

Lipinski, R. J., 1980. Assessment of Core Penetration of a PWR Reactor Vessel and Particulate Debris Coolability in TMLB', S2D and ABG Accidents, USNRC Report NUREG/CR-1518 (SAND80-0701, Sandia National Laboratories, Albuquerque, N.M.).

Los Alamos National Laboratory, 1981. TRAC-PD2--An Advanced Best-Estimate Computer Program for Pressurized Water Reactor Loss-of-Coolant Accident Analysis, USNRC Report NUREG/CR-2054.

MacNeal, R. H., 1978. NASTRAN: Theoretical Manual Level 17.6, National Aeronautics and Space Administration, Washington, D.C.

Marcal, P. V., 1975. MARC User Information Manual, Vol. 1, MARC Analysis Corporation.

Mays, S. E., J. P. Poloski, W. H. Sullivan, J. E. Trainer, R. C. Bertucio, and T. J. Leahy, 1981. Draft Report--Risk Assessment for Browns Ferry Nuclear Plant, Unit 1, prepared for the U.S. Nuclear Regulatory Commission.

Moore, K. V., et al., 1978. RETRAN--A Program for One-Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, CCM-5, prepared by Energy Incorporated for the Electric Power Research Institute, Palo Alto, Calif.

Muir, J., et al., 1981. CORCON-MOD1: An Improved Model for Molten-Core/Concrete Interactions, USNRC Report NUREG/CR-2142.

Murfin, W. B., 1980. Report of the Zion/Indian Point Study, USNRC Report NUREG/CR-1410, Vol. 1, Chapter 3.

Murray, D. W., L. Chitnuyanondh, and C. Wong, 1979. "Modelling and Predicting Behavior of Prestressed Concrete Secondary Containment Structures Using BOSOR5," in Transactions of the 5th SMIRT Conference, Paper No. J3/5, Berlin.

Peehs, M., K. Hassmann, and S. Hagen, 1979. "Analysis of a Hypothetical Core Meltdown Accident of a PWR," Siemens Forschungs und Entwicklungsberichte, Siemens, Erlangen, Federal Republic of Germany.

Philadelphia Electric Company, 1981. Probabilistic Risk Assessment, Limerick Generating Station, Docket Nos. 50-352, 50-353, U.S. Nuclear Regulatory Commission, Washington, D.C.

Powers, D. A., and F. E. Arellano, 1981. Large Scale, Transient Tests of the Interactions of Molten Steel with Concrete, USNRC Report NUREG/CR-2282 (SAND81-1753, Sandia National Laboratories, Albuquerque, N.M.).

Powers, D. A., D. A. Dahlgren, J. F. Muir, and W. D. Murfin, 1977. Exploratory Study of Molten Core Material/Concrete Interactions, SAND77-2042, Sandia National Laboratories, Albuquerque, N.M.

Ransom, V. H., et al., 1980. RELAP5/MOD1 Code Manual, USNRC Report NUREG/CR-1826.

Rav, TVSR Appa, 1975. "Behavior of Concrete Nuclear Containment Structures up to Ultimate Failure with Special Reference to MAPP-1 Containment," Inelastic Behavior, Report 4-SM-THEME/75, Structural Engineering Research Centre, Madras, India.

Rivard, J. B., 1978. Post Accident Heat Removal: Debris Bed Experiments D-2 and D-3, USNRC Report NUREG/CR-0421 (SAND78-1238, Sandia National Laboratories, Albuquerque, N.M.).

Rivard, J. B., et al., 1981. Interim Technical Assessment of the MARCH Code, USNRC Report NUREG/CR-2285 (SAND81-1672, Sandia National Laboratories, Albuquerque, N.M.).

Senglaub, M. E., J. P. Odom, M. J. Clauser, J. E. Kelly, and P. S. Pickard, 1981. CONTAIN, a Computer Code for the Analysis of Containment Response to Reactor Accidents--Version 1A, draft USNRC Report NUREG/CR-2224 (SAND81-1495, Sandia National Laboratories, Albuquerque, N.M.).

Squarer, D., A. T. Pieczynski, and L. E. Hochreiter, 1981. "Dryout in Large Particle, Deep Debris Beds," Transactions of the American Nuclear Society, Vol. 36.

Tuerk, W., F. A. R. Schmidt, and H. Unger, 1980. "An Experimentally Verified Fuel Rod Meltdown Model," Transactions of the American Nuclear Society, Vol. 35.

USNRC (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.

USNRC (U.S. Nuclear Regulatory Commission), 1981a. Long Range Research Plan, NUREG-0740.

USNRC (U.S. Nuclear Regulatory Commission), 1981b. Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects, NUREG-0850, Vol. 1.

USNRC (U.S. Nuclear Regulatory Commission), 1981c. Federal Register, Vol. 46, No. 246, pp. 62281-62285.

USNRC (U.S. Nuclear Regulatory Commission), 1981d. Technical Bases for Estimating Fission Product Behavior During LWR Accidents, NUREG-0772, Washington, D.C.

Von Riesemann, W. A., et al., 1981. "Structural Safety Margins of Containments," Ninth Water Reactor Research Information Meeting, October 1981, U.S. Nuclear Regulatory Commission, Washington, D.C.

Walser, A., 1980. "An Overview of Reactor Containment Structures," Nuclear Engineering and Design, Vol. 61, pp. 113-122.

Wooton, R. O., and H. I. Avci, 1980. MARCH (Meltdown Accident Response Characteristics) Code Description and User's Manual, USNRC Report NUREG/CR-1711 (Battelle's Columbus Laboratory, Ohio).

Chapter 8
Radionuclide Release and Transport

8.1 INTRODUCTION

This chapter describes methods for the evaluation of radionuclide releases to the environment during degraded-core accidents at light-water reactors. Overheating or melting of the reactor fuel in such accidents can result in the release of radionuclides from the fuel and their eventual release to the environment. Structural materials from the core and the reactor-coolant system can also be released with the radionuclides and are likely to affect not only the behavior of radionuclides in the reactor-coolant system and the containment but also eventual releases to the environment.

The methods discussed in this chapter include those used for establishing the initial inventories of radionuclides and structural materials in the fuel and the reactor; the analysis of radionuclide and structural material releases from the core; and the analysis of radionuclide transport, deposition, and release in the reactor-coolant system and the containment. These steps in a PRA are usually preceded by an analysis of the physical processes that can occur during degraded-core accidents since the presently available radionuclide-behavior models require input information on the timing of various events and the thermal-hydraulic conditions in the reactor. The analyses needed to obtain such information were discussed in Chapter 7.

The principal output of the radionuclide release and transport calculations is a set of release fractions to the environment. These express the quantities of radionuclides released to the environment as a fraction of the reactor-core inventory at the beginning of the accident. This information is required for analyzing, as described in Chapter 9, the transport of radionuclides through the environment and the consequences of the accident to public health and safety.

Radionuclides can be released from the reactor either into the air or into the ground. The pathways of releases into the ground are likely to result in significant attenuation of most radionuclides during their passage through soil, and thus accident sequences involving releases into the air are of much greater radiological importance. If radionuclides come in contact with groundwater, however, they may dissolve and be transported much more readily. Only cursory analyses of releases into the ground have been performed to date. For completeness, a summary of the treatment of ground releases in the Reactor Safety Study (USNRC, 1975) is given in Appendix G. This chapter deals only with the evaluation of releases into the air.

8.2 OVERVIEW

The evaluation of radionuclide releases that result from severely degraded core accidents involves the four elements shown in Figure 8-1:

1. Inventories of radionuclides and structural materials.
2. Radionuclide and structural material source term from the core.
3. Transport, deposition, and release in the reactor-coolant system.
4. Transport, deposition, and release in the containment.

Also shown in Figure 8-1 are the input needed from the analysis of physical processes (Chapter 7) and the output provided for the analysis of environmental transport and consequences (Chapter 9). The analysis proceeds sequentially, starting with the inventories of radionuclides and structural materials. This involves the determination of the quantities of radionuclides and structural materials that are present at the beginning of the accident. The next step is the evaluation of the radionuclide and structural material source term from the core. This entails the determination of the quantities of radionuclides and structural materials that are released

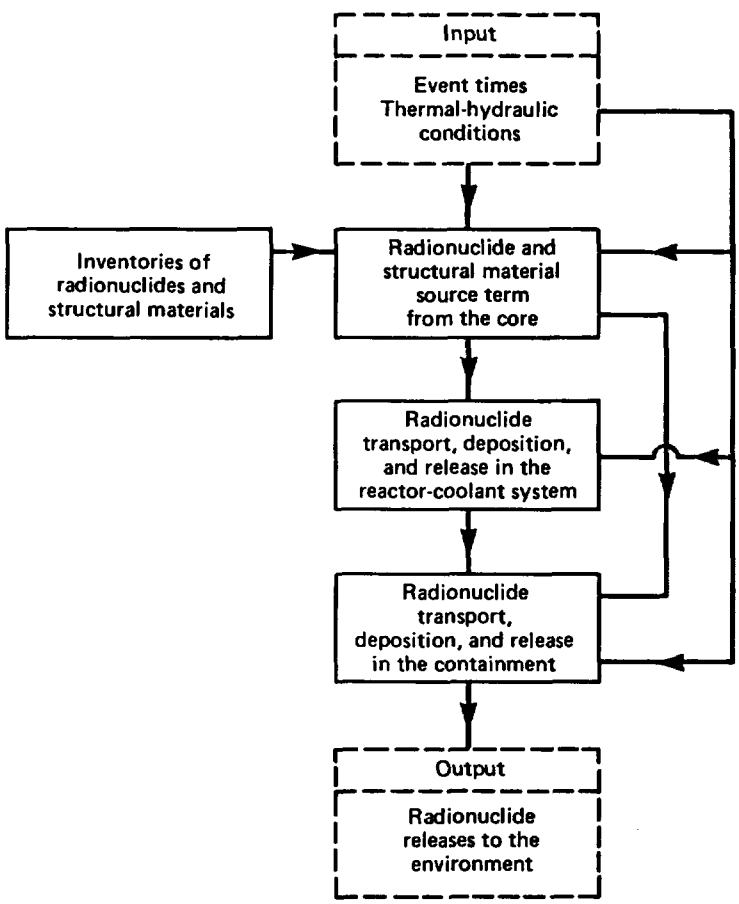

Figure 8-1. Elements in the analysis of radionuclide behavior in the reactor.

from the core to the reactor-coolant system or to the containment. (Direct releases of radionuclides and structural materials from the corium--the melted core and structural materials--to the containment can occur in core-melt accidents after the pressure vessel has melted through and the corium is interacting with the concrete basemat.) This source term is then used in the analysis of radionuclide transport, deposition, and release in the reactor-coolant system. The analysis considers the various deposition processes that can occur in the coolant system. The result is the source term for release from the coolant system to the containment; it is used in the analysis of transport, deposition, and release in the containment. This analysis takes account of the various deposition processes that can occur in the containment, and it estimates the quantities of radionuclides that are released from the containment to the environment.

It should be noted that, although the primary objective of the radionuclide transport and deposition calculations for the reactor-coolant system and the containment is the evaluation of source terms for releases to the containment and to the environment, respectively, the analyses can also provide information on the distribution of radionuclides deposited in the reactor-coolant system and the containment. This information may be of value for any cleanup and decontamination operations that may be necessary after an accident.

The four steps in the analysis of radionuclide release and transport are described in greater detail below.

8.2.1 INVENTORIES OF RADIONUCLIDES AND STRUCTURAL MATERIALS

The starting point in the analysis of radionuclide behavior during degraded-core accidents is the determination of the inventories of materials that can be released to the reactor-coolant system and the containment. This includes radionuclides, fuel, stable isotopes produced by the decay of radionuclides during reactor operation, and structural materials like cladding, control rods, core supports, and instrument tubes. Released structural materials can have a significant impact on the behavior of radionuclides in the coolant system and the containment, primarily by their effects on such aerosol behavior as agglomeration. It should be noted that the transport and deposition behavior of the stable isotopes of a particular nuclide is indistinguishable from that of the radioisotopes of the same nuclide. It is important to account for the effect of stable isotopes on mass-balance calculations since their inventories can be greater than those of the corresponding radioisotopes.

Radionuclide and stable-nuclide inventories can be determined with an isotope generation and depletion code that accounts for fission, transmutation, and decay. Such codes need nuclear constants (cross sections, decay rates, fission yields) and information on the initial nuclide inventory, the percentage uranium enrichment, the specific power of reactor operation, and burnup. Information on the quantities of structural materials present in the core can be found in documents on the reactor design, such as the safety analysis report and design drawings, or it can be obtained from the utility or the vendor.

8.2.2 RADIONUCLIDE AND STRUCTURAL MATERIAL SOURCE TERM FROM THE CORE

Releases of radionuclides from fuel can be expected to depend on the chemistry of the radionuclides within the fuel (kinetics and thermodynamics), the physical form of the fuel (e.g., cladding intact or failed, fuel solid or molten, fuel surface-to-volume ratio), and the environment to which the fuel is exposed (e.g., temperature, fluid composition, and steam/water/air/hydrogen ratio). The specification of the source term from the fuel should include not only the magnitudes of the releases but also the release rates and the chemical and physical forms (especially particle size) of the released materials. Releases can occur by a variety of processes. These release processes are classified here phenomenologically rather than mechanistically since this is more consistent with the state of knowledge of the subject. The release processes presently believed to be possible in degraded-core accidents are described below. It should be noted that each of these processes may actually represent several mechanisms of release.

Cladding-Rupture Release. When the fuel-rod cladding ruptures, which is usually considered to result from overheating, part of the radionuclide inventory that accumulates in the fuel-to-cladding gap during normal reactor operation will be released from the pressurized fuel rod. The radionuclides thus released will consist of the noble gases and the radionuclides that are in volatile form; there is also the possibility that any loose debris present in the gap, such as fuel powder, may be entrained in the gases flowing out of the rod. The cladding-rupture release can be expected to be a small component of the overall source term in a core-melt accident, but it could be an important contributor in other degraded-core accidents. Although it is usually considered to occur instantaneously at the time of cladding rupture, some diffusion may continue over a longer period of time. The release will not occur in all fuel rods at the same time, as it depends on the heatup rates of individual rods. This release process was called the "gap release" in the Reactor Safety Study (USNRC, 1975).

Diffusion Release. When the fuel is held at an elevated temperature and the cladding has failed, radionuclides will diffuse from inside the fuel matrix to the surface, where they will be released. Such a release may occur, for example, after the cladding fails and as the fuel is heating up but before the fuel melts. This release process can be important in degraded-core accidents where the core does not melt but stays at an elevated temperature for a significant period of time.

Leach Release. When water comes into contact with the fuel, radionuclides in the fuel will be leached into the water. This process can only occur, of course, after the cladding has failed. A release of this type may have occurred during the accident at Three Mile Island.

Melt Release. Occurring when fuel melts, this process involves the diffusion of radionuclides from inside the melt and their escape from its surface. It is believed that significant amounts of structural materials can be released with the radionuclides by this process. The rates of such a release can be expected to depend on the way the core melts down and will not be uniform across the core.

Melt/Concrete Release. This can occur after pressure-vessel melt-through, when the molten core and structural materials (corium) are in contact with the concrete basemat in the reactor cavity. Gases from the decomposition of the concrete will sparge the corium and can remove radionuclides contained within it. Large quantities of concrete components can be expected to be released with the radionuclides. This release was termed the "vaporization release" in the Reactor Safety Study. The process of vaporization can, and does of course, occur at other times--for example, during the melt release.

Fragmentation Release. Steam explosions that may result from the contact of a mass of fuel with water either in the pressure vessel or in the containment could result in the fragmentation of the fuel. (The likelihood of steam explosions is discussed in Chapter 7.) Other energetic events could cause a similar fragmentation; for example, the forces involved in pressure-vessel meltthrough may be sufficient for this purpose. If such an event occurs in an oxidizing atmosphere, a release of radionuclides from the dispersed fuel may result from (1) fuel oxidation with the attendant increase in surface area and a greater opportunity for the escape of radionuclides by diffusion or (2) the oxidation of radionuclides within the fuel. In particular, it has been suggested that there may be an enhanced release of radionuclides that have volatile oxides (USNRC, 1975), an example being ruthenium. (Molybdenum can also be oxidized by steam.) This release process was called the "oxidation release" in the Reactor Safety Study.

It should be noted that the volatile radionuclides that would be most likely to escape as a result of fuel oxidation may not be present in the fuel at the time of fuel oxidation, owing to their earlier release by other processes. The oxidizing atmosphere needed for this release derives either from air in the containment or air in the environment. It is also possible that fuel dispersal into an inert or reducing atmosphere may occur. In this case, the relocation of the fuel and the radionuclides it contains can be an important factor. This is also true of fuel dispersal in an oxidizing environment since not all radionuclides will escape from the fuel, and the location and ultimate fate of the remaining nuclides must be considered. If fragmented fuel becomes immersed in water, radionuclides can be released by leaching from the fuel with its increased surface area. For example, such a release could occur in the event of a steam explosion in the containment when there is water in the reactor cavity.

It should be noted that release by oxidation does not require fuel dispersal in a finely divided form. It may occur if the fuel is held at a high temperature in the presence of an oxidizing agent--such as oxygen, steam, or carbon dioxide--for a reasonable amount of time. Releases by this mechanism may contribute to some of the other release processes described above, for example, the melt/concrete release.

The release processes discussed above represent those believed to be possible during severely degraded core accidents. Not all processes will necessarily occur in every accident since they are dependent on the particular conditions of each accident. Although this discussion represents current understanding of the release processes, it is also possible that processes not discussed above may occur.

In order to analyze radionuclide and structural material releases from the core, information is needed on radionuclide and structural material inventories, the physical processes that occur, and physical and chemical data needed to model each of the release processes. Information on the physical processes determines which radionuclide-release processes occur, provides data on the atmosphere in the reactor-coolant system, describes the manner and timing of core degradation, and specifies the time at which various events occur (e.g., cladding failure, core-melt initiation and termination, pressure-vessel failure).

8.2.3 TRANSPORT, DEPOSITION, AND RELEASE IN THE REACTOR-COOLANT SYSTEM

The analysis of radionuclide transport and deposition in the reactor-coolant system (RCS) must consider both chemical and physical processes that may influence the behavior of the radionuclides. Radionuclides and structural materials can be released from the fuel to the reactor-coolant system as vapors or particulates. Vapors can condense on coolant-system surfaces, within the RCS fluid to form particulates, or on suspended particulates. Particulates or condensed materials can also be vaporized if appropriate temperatures are encountered. Particulates can also agglomerate to form larger particles. In addition, materials released from the core can react chemically with one another or with the components of the carrier fluid (steam, hydrogen, and possibly air). Vapor materials can be removed from the RCS atmosphere by interaction with water (e.g., injected emergency core coolant) or by natural deposition processes like sorption on surfaces. Particulate material can also be removed by interaction with water and by such natural deposition processes as diffusion, diffusiophoresis, thermophoresis, impaction, and gravitational settling. It should be recognized that deposited material can be resuspended. For example, particulates can be reentrained in the fluid flow, and deposited vapors can be revaporized.

In order to perform such analyses, information is needed on the radionuclide and structural material source terms released to the reactor-coolant system (quantities of materials, release rates, time dependence, chemical forms, particle-size distribution, and particle composition), physical conditions in the reactor-coolant system (e.g., pressure, fluid temperature, surface temperature, fluid flow rate, fluid composition, flow path), the geometric configuration of the reactor-coolant system and the materials of RCS surfaces, and the physical and chemical properties of the released materials (e.g., vapor pressures, chemical reaction rates).

8.2.4 TRANSPORT, DEPOSITION, AND RELEASE IN THE CONTAINMENT

The analysis of radionuclide transport and deposition in reactor containments is similar in many ways to the analysis of radionuclide behavior in reactor-coolant systems. In principle, the processes that can occur in the coolant system can also occur in the containment. However, the conditions in containments during degraded-core accidents are very different from the conditions in reactor-coolant systems. This means that the actual

behavior of radionuclides in the containment is likely to be quite different from that in the coolant system. Furthermore, material suspended in the containment can be removed by various engineered safeguards, such as sprays, filters, ice condensers, and suppression pools, depending on the design of the reactor.

Input information needed for the analysis is of the same general type as for the RCS analyses but also includes information on the characteristics and functionability of the engineered safeguards and information on the mode and timing of containment failure.

8.3 METHODS

Severely degraded core accidents are rare events for which an experimental data base on radionuclide releases consequently does not exist. Therefore, recourse must be made to analytical methods in order to evaluate radionuclide releases to the environment. Available methods are discussed below for the four parts of radionuclide release and transport analysis that were described in Section 8.2.

8.3.1 INVENTORIES OF RADIONUCLIDES AND STRUCTURAL MATERIALS

The ORIGEN computer code is often used to estimate radionuclide inventories in fuel (Bell, 1973; Croff, 1980); it analyzes fission, transmutation, and decay. Cross sections, which are averaged over ranges of neutron energy, are used in predicting the reaction rates for fission and transmutation. Standard descriptions of radioactive-decay chains and accepted values of nuclear constants, such as half-life and fission yield, have been incorporated into the code. ORIGEN cannot predict the spatial distribution of nuclides within the reactor; it can, however, be used to estimate either the average inventory in the reactor or the inventory in a particular region of the reactor if the power generation in that region is specified as a function of time. Predictions made with ORIGEN have been compared with measurements of the inventories of actual fuel rods (Croff, 1980). The agreement has been typically within approximately 30 percent of the measured value.

Input data needed by ORIGEN include the initial nuclide inventory, the percentage uranium enrichment, the specific power of reactor operation, and burnup.

Computer codes other than ORIGEN are available for calculating radionuclide inventories. Many of them are proprietary, but one that is in the public domain is CINDER. It differs from ORIGEN in the technique used to solve the decay equations and in the data base employed. A comparison of CINDER and ORIGEN calculations is under way (T. England, Los Alamos National Laboratory, personal communication, 1981).

8.3.2 RADIONUCLIDE AND STRUCTURAL MATERIAL SOURCE TERM FROM THE CORE

In order to characterize concisely the radionuclide source term from the fuel and to facilitate analyses of radionuclide behavior, it is desirable to classify the large number of fission and activation products that occur in reactor fuel into a small set of categories, each of which is similar in physical and chemical behavior and can consequently be represented by a single nuclide. Such a classification was employed in the Reactor Safety Study (RSS--USNRC, 1975), and it has achieved a measure of popularity. It is shown in Table 8-1.

Table 8-1. Radionuclide-classification scheme used in the Reactor Safety Study[a]

Noble gases	Xe, Kr
Halogens	I, Br
Alkali metals	Cs, Rb
Tellurium group	Te, Se, Sb
Alkaline earths	Sr, Ba
Transition metals	Ru, Mo, Pd, Rh, Tc
Lanthanides and actinides	La, Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, Nb

[a]In subsequent parts of this chapter, the various radionuclide groups will be denoted by the symbol for the first element listed in each group.

At present, there is no generally accepted comprehensive method for estimating radionuclide releases from the fuel during degraded-core accidents. However, several models are available in the literature. One of the earliest models is that used in the FRCRL2 computer code (Ritzman and Morrison, 1971). This code considers cladding-rupture, diffusion, and melt releases of radionuclides for a nodalized core (core divided into regions from each of which radionuclides are released independently). It is based on the earlier codes FRACREL and REGAP (USNRC, 1975). These models and available experimental data were used in the development of the RSS release fractions (see Table 8-2). The RSS source term has achieved some popularity, but the validity of the release-fraction values has been questioned. Furthermore, the RSS source term considers only the cladding-rupture, melt, vaporization, and oxidation release processes and does not consider possible structural material releases. In addition, the RSS source-term model assumes that the release processes apply to the whole core, with no core nodalization.

Available models for each of the six release processes discussed in Section 8.2.2 are described below.

8.3.2.1 Cladding-Rupture Release

An improved model for the cladding-rupture releases of cesium and iodine has been developed at the Oak Ridge National Laboratory (Lorenz et

Table 8-2. Release fractions used in the Reactor Safety Study for radionuclide releases from the fuel

Nuclide	Gap[a]	Melt[a]	Vaporization[b]	Oxidation[b]
Xe, Kr	0.03	0.87	0.1	0.9
I, Br	0.017	0.885	0.1	0.9
Cs, Rb	0.05	0.76	0.19	0
Te[c]	1.0×10^{-4}	0.15	0.85	0.6
Sr, Ba	1.0×10^{-6}	0.1	0.01	0
Ru[d]	0	0.03	0.05	0.9
La[e]	0	0.003	0.01	0

[a]Fraction of initial core inventory released.
[b]Release fraction applies to the core inventory remaining after previous releases.
[c]Release fractions also apply to Se and Sb.
[d]Release fractions also apply to Mo, Pd, Rh, and Tc.
[e]Release fractions also apply to Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, and Nb.

al., 1979, 1980). Experiments were performed to measure radionuclide releases from several types of LWR fuel rods into which defects had been introduced. The experiments were conducted in steam over the temperature range 500 to 1200°C, and a model was fit to the collected data. It was found that in the temperature range 700 to 900°C, this release can be expressed by

$$M_B = \alpha V_B \left(\frac{M_0}{A}\right)^a \exp\left(-\frac{C}{T}\right)$$

where

M_B = mass of radionuclide released in the burst (g).
V_B = volume of plenum gas vented at 0°C and system pressure (cm^3).
M_0 = radionuclide inventory in the fuel-to-cladding gap (g).
A = internal area of the cladding associated with M_0 (cm^2).
T = temperature at rupture location (K).

Values of the adjustable constants α, a, and C were obtained by fitting the model to the experimental data (see Table 8-3).

In addition to the burst release, a longer term diffusion release was measured in the experiments and fit, over the temperature range 500 to 1200°C, with the model

$$M_D = M_0 \left[1 - \exp\left(-\frac{R_0 t}{M_0}\right)\right]$$

Table 8-3. Values of parameters in burst and diffusion release models for cesium and iodine[a]

Parameter	Cesium	Iodine
α $[(g/cm^3)\cdot(g/cm^2)]$	3.49	0.163
a	0.8	0.8
C (K^{-1})	7.42×10^3	3.77×10^3
σ $[(g\cdot MPa/\mu m\cdot hr)\cdot(g/cm^2)^{-a}]$	1.90×10^3	1.22×10^2
γ (K^{-1})	1.98×10^4	1.48×10^4

[a]From Lorenz et al. (1980).

where M_D is the mass of radionuclide released by diffusion (g), t is the time at diffusion temperature (hr), and R_0 is the initial rate of release by diffusion (g/hr), given by

$$R_0 = \sigma\left(\frac{W}{P}\right)\left(\frac{M_0}{A}\right)^a \exp\left(-\frac{\gamma}{t}\right)$$

where W is the width of the radial gap (μm) and P is the system pressure (MPa). Again, values of the adjustable constants σ and γ were obtained by fitting the model to the experimental data (see Table 8-3). In general, the models were found to represent the data on which they were based within a factor of 3.

The models require knowledge of the initial radionuclide inventory in the gap. This inventory can be estimated on the basis of experimental observation or by using analytical methods. Several techniques were discussed in the Reactor Safety Study, including the use of the REGAP computer code.

The models were applied to the analysis of a loss-of-coolant accident (LOCA) for a typical pressurized-water reactor. Release fractions for iodine and cesium were found to be 5.3×10^{-4} and 2.5×10^{-4}, respectively--one to two orders of magnitude lower than the RSS release fractions. However, the cladding-rupture release is a very small contributor to the total source term for a meltdown accident.

In using these models, it is necessary to recognize their limitations. Their validity can only be ensured when applied to situations within the range of the test parameters used in the experiments on which they are based. These experiments used short sections of fuel rods with low gap inventories, but the authors of the model believe the model is applicable to full-length rods.

8.3.2.2 Diffusion Release

Classical models can be employed to model diffusion (Booth, 1957), but a supporting data base of diffusion coefficients is needed. A computer code called GRASS can be used for a mechanistic analysis of the diffusion of radionuclides from fuel to the fuel-to-cladding gap (Rest, 1978). It treats such processes as gas-bubble nucleation, diffusion, fuel microcracking, and grain boundary diffusion. The code does not treat radionuclides other than the noble gases, but extensions are planned (R. Sherry, U.S. Nuclear Regulatory Commission, personal communication, 1981). Although it was developed for steady-state conditions, GRASS has also been applied to transients (Rest, 1982).

Some experimental data of a scoping nature have been developed for irradiated LWR fuel heated to 1300-1600°C in steam. The tests simulated fuel rods with ruptured cladding. Heating times were short (0.4 to 10 minutes), and the fuel was of high burnup (30,000 MWd/MT) and low initial gap inventory (0.3 percent for cesium and iodine). The results showed a large increase in the release of noble gases, cesium, and iodine when the fuel is heated uniformly to a minimum of 1350 to 1400°C. Within 2 minutes at 1400°C, approximately 4 to 9 percent of the noble gases, cesium, and iodine in the fuel rod was released. Releases at 1- to 10-minute heating times were estimated to differ by factors of 0.8 to 1.2, respectively, from those at 2 minutes. At 1600°C, the releases in 2 minutes were about 17 to 25 percent of the total inventory. LWR fuel with different irradiation histories can be expected to give different release results in the temperature range 1300 to 1600°C.

Experimental work in progress should provide more data at higher temperatures (T. Kress, Oak Ridge National Laboratory, personal communication, 1981).

Diffusion releases of radionuclides in degraded-core accidents have not received a great deal of attention, because they have been viewed as unimportant in meltdown accidents. Their actual importance depends on the accident sequence that is modeled, and it is quite possible that they may be significant contributors to the total radionuclide source term for some accidents.

8.3.2.3 Leach Release

Until recently, leach releases have received little attention in analyses of degraded-core accidents owing to the perception that they are not important. Their actual importance depends on the details of the accident. The accident at Three Mile Island has helped focus attention on their potential contribution to the radionuclide source term.

Data on the leaching of radionuclides by water from fuel are sparse. Some work on spent fuel has been done at the Pacific Northwest Laboratory (Katayama et al., 1980). The leaching of cesium and strontium from corium-concrete mixtures has been studied by Johnstone and Braithwaite (USNRC, 1978). Powers and Westrich (USNRC, 1981) have also investigated the leaching of a variety of species from corium-concrete mixtures.

8.3.2.4 Melt Release

Experiments investigating the melt release have been conducted for the past several years in the Federal Republic of Germany, in the SASCHA facility of the Kernforschungszentrum Karlsruhe (KfK) (Albrecht et al., 1978, 1979). The experimental apparatus consists of a high-frequency induction furnace in which corium is heated to melting in a thoria crucible under air, argon, and steam atmospheres. The materials released from the corium are trapped in a collection train for analysis. Experiments have been reported for small samples (30 to 150 grams) of corium and corium traced with fission products. The experiments are designed to determine the melt release of both radionuclides and structural materials from the corium. Releases were generally found to increase on changing the atmosphere from steam to argon to air. The release fractions in air at 2700°C were found to be 0.004 to 0.007 for Fe, Cr, and Co and 0.04 to 0.11 for Sn, Sb, and Mn; for air at 2150°C, release fractions in the range 0.2 to 0.4 were found for Se, Cd, Te, and Cs. Generally, the melt temperature had the greatest effect on the releases, but chemical reactions among the melt constituents and with the atmosphere also played a significant role. The most probable sizes of the aerosol particles formed in air at temperatures between 1800 and 2700°C were less than 0.5 micrometer. Species of low volatility were concentrated in the larger particles, while those of high volatility were concentrated in the smaller particles.

Recent experiments at KfK were performed with sample sizes of 150 to 250 grams (Albrecht and Wild, 1981). Release information was obtained for a variety of species. More than 90 percent of iodine and cesium was released when a temperature of 1700°C was maintained for 10 minutes. Since total radionuclide releases for actual accident sequences will be dependent on specific time/temperature histories, release rates expressed as percent release per unit time at a particular temperature were calculated where possible for use in source-term evaluations.

The results of these experiments were compared with the melt-release values of the Reactor Safety Study. They indicate that the RSS melt-release values are underestimated for Te and Sb by a factor of 3 to 5 and overestimated for Ba, Mo, Zr, Ru, La, Ce, Pr, Nd, and Np by a factor of about 10.

The KfK release rates were also used to estimate the amounts of structural materials that would be released in a core-melt accident at the Biblis-B PWR. It was predicted that, of the total fuel and structural material inventory of 181 metric tons, a total aerosol mass of 3.5 metric tons would be formed before pressure-vessel meltthrough. Of this, 1.8 metric tons was estimated to come from the silver in the control rods and about 0.45 metric ton each from UO_2 and Fe/FeO.

In experiments planned at the Oak Ridge National Laboratory (ORNL), sections of irradiated fuel rods will be heated to melting and material releases measured (M. Silberberg, U.S. Nuclear Regulatory Commission, personal communication, 1981). These experiments will be extensions of the work that produced the ORNL cladding-rupture release model described in Section 8.3.2.1.

The results of these various melt-release experiments will eventually be used to develop a new model for this release process. However, no such model is available at present. In the interim, it is possible that the presently available results can be used to provide some indication of appropriate melt-release fractions. If such data are used, the user must consider their extrapolation to the actual conditions of full-scale core meltdowns.

A model that improves on the RSS model and accounts for diffusion and melt releases was recently proposed (USNRC, 1981). The model has the form

$$\frac{dM_x}{dt} = -k_x(t)M_x$$

where M_x is the mass of material x in the corium, k_x is a temperature-dependent release-rate coefficient, and t is time. This model allows the radionuclide releases from the fuel to be related to the core-heatup time. Release-rate coefficients were determined for several radionuclides by fitting to a wide range of experimental data (see Figure 8-2). Coefficients for fuel, cladding, and other structural material were developed from data collected at the SASCHA facility (see Table 8-4). The fractional radionuclide-release rates of Figure 8-2 and Table 8-4 were approximated by the equation

$$k(t) = Ae^{Bt} \tag{8-1}$$

where A and B are constants determined by curve-fitting procedures (see Table 8-5).

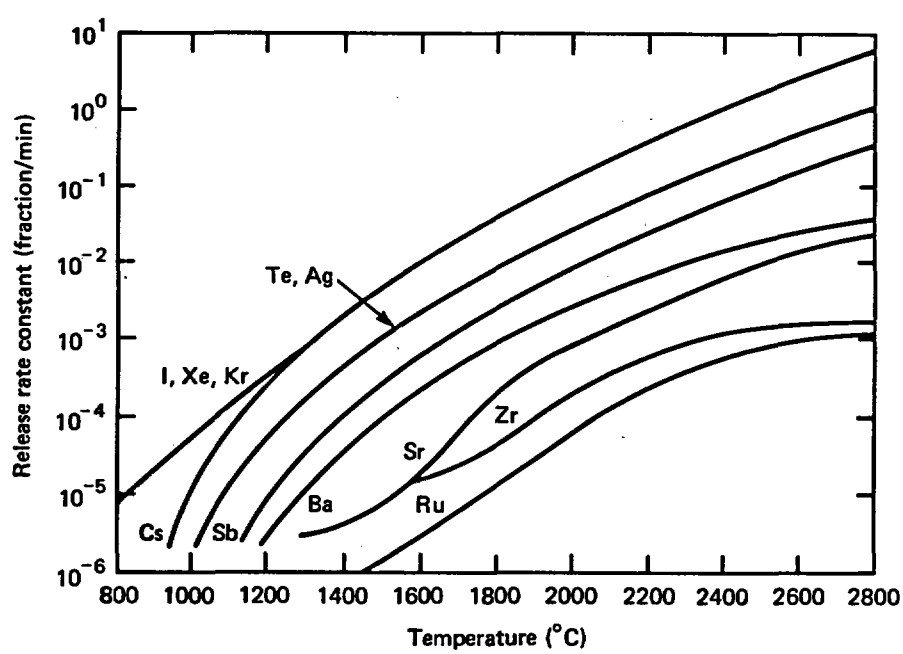

Figure 8-2. Release-rate coefficients for various radionuclides. From NUREG-0772 (USNRC, 1981).

Table 8-4. Release-rate coefficients for inert material[a]

Material	Temperature (°C)	Coefficient (1/m)
Fuel	2400	1×10^{-6}
	2700	1×10^{-5}
Cladding	2200	1×10^{-6}
	2500	1×10^{-5}
Structure	1800	1×10^{-6}
	2200	1×10^{-5}

[a]From NUREG-0772 (USNRC, 1981).

A comparison of results obtained with this new model and the results reported in the Reactor Safety Study for the large-pipe-break meltdown-accident sequence AB showed general agreement for all radionuclides except Te and Sb, for which the new model predicted considerably higher releases (USNRC, 1981). The usefulness of the model depends on the accuracy of the release-rate coefficients. The available values are thought to be quite uncertain (USNRC, 1981).

Table 8-5. Values of the constants A and B for release-rate coefficients[a,b]

Element	1000°C < T < 2200°C		T > 2200°C	
	A	B	A	B
Fuel (UO_2)	1.0×10^{-14}	0.00768	Same	Same
Cladding (Zr, Sn)	4.6×10^{-14}	0.00768	Same	Same
Structure (Fe)	3.2×10^{-11}	0.00576	Same	Same
Ru	1.36×10^{-11}	0.00768	8.49×10^{-7}	0.00262
Zr	8.3×10^{-10}	0.00622	1.44×10^{-5}	0.00173
Ba	7.28×10^{-11}	0.00677	6.40×10^{-7}	0.00377
Sb	1.0×10^{-8}	0.00667	1.55×10^{-6}	0.00303
Te, Ag	2.96×10^{-8}	0.00677	1.17×10^{-5}	0.00404
Cs, I	1.65×10^{-7}	0.00667	1.89×10^{-5}	0.00451

[a]From NUREG-0772 (USNRC, 1981).
[b]See Equation 8-1.

8.3.2.5 Melt/Concrete Release

For the past several years, an experimental program has been under way at Sandia National Laboratories to investigate the interaction of molten corium with concrete. Its principal objective has been to study the rate of concrete decomposition and the behavior of the melt. Information has also been obtained on aerosol generation (USNRC, 1980). It was found that the aerosol was composed mostly of nonfuel material and that the multimodal particle-size distribution of the aerosol was sharply peaked at a mean aerodynamic diameter of 2 micrometers. A preliminary model has been developed for the rate of aerosol release from the surface of molten corium interacting with concrete. This model has the form

$$\frac{dM}{dt} = C_a A_s V_s$$

where

M = released aerosol mass (g).
C_a = aerosol concentration in the plume rising above the melt (g/m^3).
A_s = melt surface area (plume cross section) (m^2).
V_s = superficial gas velocity (m/sec).
t = time (sec).

The aerosol concentration, C_a, was related empirically to the melt temperature and the superficial gas velocity by

$$C_a = A_0 \exp\left(-\frac{E}{RT}\right)\left(\beta V_s + \alpha\right)$$

where R is the universal gas constant = 1.987 cal/mole, T is the melt temperature (K), and the empirical constants E, β, α, and A_0 have the following values:

$$E/R = 19{,}000$$
$$\beta = 24$$
$$\alpha = 3.3$$
$$A_0 = 10^4$$

In order to estimate the aerosol release, a knowledge of the geometric configuration of the melt and a thermal analysis of the melt-concrete interaction are needed. The WECHSL (Reimann and Murfin, 1978) and CORCON (Murfin, 1977) computer codes are presently being developed to perform such analyses. Examples of the application of this model are the Zion and Indian Point probabilistic risk assessments (USNRC, 1980).

It should be noted that the model does not provide information on the release of radionuclide aerosols from the melt. It applies to materials like the oxides of silicon, calcium, and aluminum, which derive from the concrete. The model is also based on only a limited data base, and it depends on the results of a thermal analysis that are somewhat uncertain.

It is likely that the model provides order-of-magnitude accuracy, but it becomes worse at low (<1700°C) and high (>2600°C) melt temperatures (D. A. Powers, Sandia National Laboratories, personal communication, 1981).

Experiments are planned in the Federal Republic of Germany to examine material releases that result from corium-concrete interactions (Albrecht and Wild, 1981).

8.3.2.6 Fragmentation Release

Little information is available on the oxidation release that results from fuel fragmentation beyond that provided in the Reactor Safety Study. This release process was called the "oxidation release" in the Reactor Safety Study. The RSS release fractions for fragmentation were based on measurements of radionuclide releases during fuel oxidation by air at elevated temperatures.

8.3.2.7 Fuel Oxidation Release

A preliminary model has been published recently for the release of radionuclides from damaged fuel rods in a steam environment (Cubicciotti, 1981). The model describes a release that occurs as a result of fuel oxidation by steam and the ensuing grain growth. It is based on the experimental observation that the rate of sintering of UO_2 is significantly greater in a steam atmosphere than in an inert or reducing atmosphere and that the release of noble gases from heated UO_2 fuel is enhanced in the presence of steam. For the noble gases the model has the form

$$F = 1 - \left[1 - 4\left(\frac{\tau_H}{\pi}\right)^{1/2}\right]\left[1 - 4\left(\frac{\tau_\rho}{\pi}\right)^{1/2} + \tau_\rho\right] \qquad (8\text{-}2)$$

where

F = fractional release of radionuclide.

$\tau_L = D_c t/L^2$.

D_c = chemical diffusion constant representing the penetration of oxidant into the UO_2 (m^2/sec): $D_c = 9.9 \times 10^{-3} \exp(-28{,}600/T)$, T being the temperature (K).

L = height (H) or radius (ρ) of a fuel pellet (m).

t = time (sec).

An extension of the model to handle volatile radionuclides has been proposed (Cubicciotti, 1981). It was suggested that the factor $[1 - \exp(-P_i/P_T)]$, where P_T is the total pressure in the system and P_i is the vapor pressure of the volatile radionuclide, be used for that

purpose. This factor is unity for the noble gases and highly volatile radionuclides. For less volatile materials, the factor depends on the vapor pressure of the chemical form of the material. Preliminary calculations indicate that releases in steam are one to two orders of magnitude greater than releases in inert atmospheres (Cubicciotti, 1981).

8.3.2.8 Important Issues and Work in Progress

An experimental program has recently been started at the Oak Ridge National Laboratory to study the release of material from fuel during heatup to melting (T. Kress, Oak Ridge National Laboratory, personal communication, 1981). No results are presently available, however. In a separate analytical program at Battelle's Columbus Laboratories, a computer code called START is being developed to predict the releases of radionuclides and structural materials during degraded-core accidents, including meltdown accidents. Using semimechanistic models, the code presently accounts for releases by the cladding-rupture, diffusion, leach, melt, vaporization, and oxidation processes. It provides a detailed time dependence for material releases. A paper describing a preliminary version of the START code has recently been published (Baybutt et al., 1981).

Little information is available on the chemical forms of the materials that may be released from the core or the sizes of particulates. The Reactor Safety Study (USNRC, 1975) assumed that iodine will be released as elemental iodine (I_2) in vapor form and other radionuclides as particulates, but no information was provided on particle sizes. It has been suggested, on the basis of recent experiments, that iodine will be released from fuel not as elemental iodine but as cesium iodide (Campbell et al., 1981). It is possible that this could significantly change any resulting release of iodine to the environment. This issue is discussed further in Section 8.4. Lack of knowledge of chemical forms represents a significant uncertainty in the evaluation of radionuclide and structural material source terms.

Another major uncertainty is the timing of radionuclide and structural material releases. This depends partly on the timing of the physical processes that occur, especially the rate at which the fuel heats up, and partly on the chemical and physical properties of the radionuclides and structural materials. It should also be noted that the radionuclide-release rates will affect the core-heatup rate. For a given release process, it is quite possible that different materials will have different release rates owing to their different properties. Little work has been done to date to study such differences.

Since wide variations in physical conditions can be expected across degraded cores, it is important, in performing analyses of releases from the core, to partition, or nodalize, the core into regions within which the physical conditions are approximately uniform. The analyses are then performed for each individual region. It is quite possible that different regions of the core will experience different release processes at the same time.

8.3.3 TRANSPORT, DEPOSITION, AND RELEASE IN THE REACTOR-COOLANT SYSTEM

The TRAP computer code is the only model that is presently available to analyze radionuclide transport and deposition in reactor-coolant systems during degraded-core accidents (Baybutt and Jordan, 1977; Jordan et al., 1979). Analyses of the transport of radionuclides through reactor-coolant systems were performed in the Reactor Safety Study (USNRC, 1975) for both particulates and elemental iodine, and it was concluded that, except in two special BWR cases, the retention of radionuclides in the coolant system would be minimal. However, the analyses were based on the thermal-hydraulics information available at the time, which was not detailed and did not consider the effects of structural material releases on radionuclide behavior in the reactor-coolant system. The coagulation of structural material aerosols with radionuclides could result in significant radionuclide removal.

The TRAP code models mechanistically the behavior of both radionuclide vapors and particulates (Jordan et al., 1979). It includes models for vapor sorption on surfaces, vapor condensation and evaporation onto and from particles and surfaces in the reactor-coolant system, particle deposition by diffusion from laminar and turbulent flow, inertial particle deposition from turbulent flow, particle deposition by thermophoresis, and particle agglomeration by Brownian and turbulent processes. The reactor-coolant system is represented in the code as a set of interconnected compartments (control volumes) within which the thermal-hydraulic conditions are uniform at any instant in time and the radionuclides are well mixed. Radionuclide transport is superimposed on the fluid flow between compartments without being coupled to it. The control volumes can be connected arbitrarily by fluid flow, and a source term of radionuclides can be placed in any volume. The modeling of radionuclide transport is based on the concept of a radionuclide state in which a particular physical form is associated with a radionuclide location (e.g., particulates suspended in steam). The transport of radionuclides can occur among the states of an individual control volume or between certain states of different control volumes if these are connected by fluid flow. Radionuclide-transport rates are modeled by using correlations for mass-transfer coefficients in a system of differential equations.

Data required as input to TRAP include the physical properties of the radionuclides, the geometric configuration of the reactor-coolant system and the material of surfaces, the source term from the core, the flow path through the coolant system, and thermal-hydraulic conditions. TRAP provides as output the radionuclide masses present in each state within each control volume as a function of time. This includes the amounts of radionuclides released to the containment from the breach in the reactor-coolant system.

TRAP does not model the gravitational agglomeration of particles, which can be important if, as is likely, large amounts of structural materials are released with the radionuclides. Neither does it account for chemical reactions that may occur during the transport of radionuclides through the reactor-coolant system, the sorption of vapors on particulates, radioactive decay, particle resuspension, or the interaction of radionuclides with water in the coolant system. The code was designed for accidents in which there is no water in the flow path to the containment. The TRAP code is being

developed further in a continuing program at Battelle's Columbus Laboratories (J. A. Gieseke, Battelle's Columbus Laboratories, personal communication, 1981). In particular, some of the processes identified above are being incorporated into the code.

To support the development of TRAP, experiments are being performed at Sandia National Laboratories to determine the vapor pressures of radionuclide compounds and to identify the chemical compounds that can result from reactions among materials released from the fuel to the reactor-coolant system and from reactions between these materials and the atmosphere of the reactor-coolant system (R. M. Elrick, Sandia National Laboratories, personal communication, 1981). For materials typical of the surfaces in the reactor-coolant system, radionuclide-vapor deposition velocities under conditions characteristic of accidents are being measured in a project at Battelle's Columbus Laboratories (S. L. Nicolosi, Battelle's Columbus Laboratories, personal communication, 1981). The results of these experiments will be incorporated into TRAP as they become available.

8.3.4 TRANSPORT, DEPOSITION, AND RELEASE IN THE CONTAINMENT

There are several computer codes that describe radionuclide transport and deposition in reactor containments. One of the earliest of these was MIRA, which accounts for the removal of iodine by natural deposition, filtration, PWR sprays, and scrubbing in a BWR wetwell (Ritzman, 1971). This code was superseded by CORRAL, which was developed as part of the Reactor Safety Study. CORRAL treats all radionuclides that can be released from the fuel and employs the classification of Table 8-1. It assumes that iodine is present in the containment as elemental iodine or organic iodide (e.g., methyl iodide). Other radionuclides, except the noble gases, are assumed to be present as particulates. Noble gases and organic iodides are assumed to pass through the containment without attenuation. Models for the removal of methyl iodide by charcoal filters and sprays were described in the Reactor Safety Study but were not programmed in the CORRAL code.

Models for the removal of elemental iodine and particulates by both natural processes and the operation of engineered safeguards are included in CORRAL. These models are semiempirical and are based largely on results obtained in the Containment Systems Experiments (Postma and Johnson, 1971). Natural deposition of elemental iodine is modeled by natural convection to the containment walls as a result of a temperature gradient between the containment atmosphere and walls. For particulates, natural deposition is modeled by gravitational settling. The code accounts for the removal of both elemental iodine and particulates by sprays, filters, and suppression pools. The containment is represented as a set of interconnected compartments (control volumes) within which the thermal-hydraulic conditions are assumed to be uniform at any instant in time, and the radionuclides are well mixed. Radionuclide transport is superimposed on the fluid flow between compartments without being coupled to it.

CORRAL places some restrictions on the time dependence of the source term that is used. The cladding-rupture and fragmentation/oxidation

releases are assumed to occur instantaneously, the melt release is assumed to occur in 10 equal parts spaced equidistantly over a time period specified by the user of the code, and the melt/concrete release is assumed to occur in two parts, each composed of 10 exponentially decreasing amounts characterized by an empirical half-life. The size of the particulates is assumed to decrease linearly from 15 to 5 micrometers over a period specified by the user. This behavior approximates results obtained in the Containment Systems Experiments and is believed to represent the evaporation of water from particulates on which water had condensed (Postma and Johnson, 1971).

Data required as input to CORRAL include the geometric configuration of the containment, the source term to the containment, the flow path through the containment, thermal-hydraulic conditions, the times of various events that occur during the accident, and information on the operation of containment safeguards. CORRAL provides as its principal output a set of cumulative radionuclide-release fractions for environmental transport analyses. It also provides information on the quantities of radionuclides deposited in each compartment.

CORRAL does not model the behavior of any structural materials released to the containment, nor does it account for any chemical reactions that may occur among released materials or reactions between released materials and the atmosphere or the materials in the containment. It does not model explicitly the agglomeration of particles or the condensation of steam on particles, although, since the code is empirically based, their effects can be considered to be included to some degree. Furthermore, CORRAL does not consider radioactive decay, phase changes in radionuclides, or the resuspension of deposited radionuclides. The removal of radionuclides in ice condensers and during their passage through leak pathways in the containment to the environment is not specifically modeled, but can be accounted for by using intercompartmental decontamination factors that are supplied by the user. The removal of particulates by diffusion, thermophoresis, and diffusiophoresis is not modeled, nor is the sorption of vapors on particulates.

The accuracy with which CORRAL predicts the actual radionuclide behavior that would occur in a degraded-core accident naturally depends on how far the conditions in the containment during the accident depart from those used in the Containment Systems Experiments, on which CORRAL is based. These experiments were performed in an isothermal environment, where radionuclide deposition by convective flow and diffusiophoresis is likely to be less than would be experienced under actual accident conditions. A limitation of CORRAL is that it is not designed to treat situations where airborne mass concentrations of particulates are high and agglomeration becomes a controlling factor. The aerosol concentrations employed in the Containment Systems Experiments were well below those that can be expected for some degraded-core accidents. CORRAL is best suited for those cases where steam condensation on particles occurs in the containment.

The version of CORRAL that was used in the Reactor Safety Study was tailored specifically to the Surry and Peach Bottom reactors. The version available from the National Energy Software Center is designated CORRAL-2 and has been generalized to accommodate other reactor designs. The radionuclide-behavior models do not differ from the RSS version.

The NAUA computer code has been developed at Kernforschungszentrum Karlsruhe in the Federal Republic of Germany to describe the behavior of aerosols in containments during core-melt accidents (Bunz and Schoeck, 1980; Bunz et al., 1981). The code is based on first principles. NAUA treats the Brownian and gravitational coagulation (agglomeration) of particles, the condensation or evaporation of water vapor onto or from particles, and the deposition of particles by sedimentation (gravitational settling), diffusion, and thermophoresis. The code assumes homogeneous mixing of the atmosphere in the containment, which it treats as a single volume. It can handle all possible particle-size distributions, such as lognormal, Gaussian, and monodisperse, with any time dependence for the aerosol source term to the containment. NAUA takes into account a size-dependent composition of the particles. The particle contents of water and solid material are averaged over the size of each size fraction but not over the whole size distribution. The radioactive nonvolatile nuclides are assumed to be homogeneously distributed over the solid fraction of the particles. The possible reaction of volatile radionuclides with the particles and droplets is not modeled, nor are the transport and deposition of radionuclide vapors or the resuspension of particulates. Radionuclide decay is not modeled, nor is the removal of particles by engineered safeguards like sprays. Application of the code depends on a detailed knowledge of the thermal conditions in the containment that control steam condensation.

Data required as input to NAUA include the geometric configuration of the containment, thermal-hydraulic conditions, and aerosol and steam source terms to the containment. Output includes the mass and number concentration of the aerosol as a function of time and particle size, and the quantities released to the environment.

An experimental program in the Federal Republic of Germany is aimed at providing data for the further development of NAUA (Bunz and Schoeck, 1980). Particular attention is being focused on water-vapor condensation on particulates and walls. Measurements of condensation on particulates have been made. Work is continuing on the examination of wall condensation and the dynamics of latent heat transfer. Aerosol experiments are also being performed in the NSPP facility at ORNL to investigate steam-condensation effects (T. Kress, Oak Ridge National Laboratory, personal communication, 1981).

The COSMO code has been written at the Japanese Atomic Energy Research Institute to analyze the removal of inorganic, organic, and particulate iodine (Nishio et al., 1981). It represents an extension of the MIRA code (Ritzman, 1971). A code called FISSCON has also been written recently in Canada (Fluke, 1981). It analyzes the behavior of radionuclides in the containments of CANDU reactors. FISSCON is very similar to CORRAL.

Aerosol-behavior codes, such as HAARM-3 (Gieseke et al., 1978) and QUICK (Gieseke and Lee, 1980), which were developed for the containment analysis of liquid-metal fast breeder reactors (LMFBRs), have been applied to LWR containments when steam condensation is not excessive (USNRC, 1981). HAARM-3 includes models for Brownian, gravitational, and turbulent agglomeration; gravitational, diffusional, and thermophoretic deposition on surfaces; particle removal by filtration; and leakage to the environment. The code assumes that aerosol concentration is spatially homogeneous throughout

the containment and that the aerosol-size distribution is lognormal. It cannot handle multiple compartments, the behavior of radionuclide vapors, or the condensation of steam on particles.

QUICK is similar to HAARM-3 in the processes it treats, but no simplifying assumptions are made regarding the aerosol-size distribution. An extended version of QUICK, called ZONE (Jordan et al., 1980), has the provision to treat the containment as three compartments interconnected by fluid flow. In addition, a code called MSPEC, which is similar to QUICK but also treats many chemical species, is under development (H. Jordan, Battelle's Columbus Laboratories, personal communication, 1981). Other LMFBR aerosol codes that are available include PARDISEKO (Jordan et al., 1974), HAA-3 (Hubner et al., 1973), AEROSIM (Walker et al., 1978), and MAEROS (Gelbard, 1981). However, not all of these codes have yet seen extensive use. Several of them have been shown to predict reliably experimental results with aerosol concentrations of less than 30 g/m^3 in vessels up to 850 m^3 in volume (Reed et al., 1980). However, it is possible that local aerosol concentrations higher than 30 g/m^3 may occur in some accident sequences, and the present codes have not been tested against experiments in this higher concentration regime. Consequently, care should be exercised in the use of these codes when the aerosol concentrations depart significantly from those for which the codes have been validated.

The ability of engineered containment safeguards to remove radionuclides is an important element in modeling the behavior of radionuclides in the containment. Radionuclide removal by filters and PWR sprays is relatively well understood. However, significant uncertainties exist for ice condensers and BWR suppression pools.

Experimental data for the removal of elemental iodine in ice beds have been obtained by the Westinghouse Electric Corporation (Malinowski, 1970). The fraction of air mixed with the steam was found to have a major effect on the decontamination factor for iodine. The effect of different additives to the ice on the amount of iodine retention was also investigated. There are no directly relevant data for particulate removal, however.

Presently, there are no detailed models available for radionuclide removal by scrubbing in BWR suppression pools, although a code called SUPRA is under development (I. B. Wall, Electric Power Research Institute, personal communication, 1981). It is also possible that existing models for bubble rise in steam generators (Baybutt et al., 1980) could be adapted. Pool scrubbing is usually treated with empirically obtained decontamination factors. Their values are very sensitive to the conditions of the experiments. In particular, the extent of radionuclide removal can be expected to depend very sensitively on whether the suppression-pool water is subcooled or boiling. More experimental and model-development work is needed. Recently, a program to perform measurements and develop models for decontamination factors was started (J. C. Cunnane, Battelle's Columbus Laboratories, personal communication, 1981).

The risk-dominant accidents of the Reactor Safety Study involved a gross failure of the containment, which provides a leak pathway with a large cross-sectional area for the escape of radionuclides to the environment.

For accidents involving a containment leak path with a smaller cross-sectional area, it is quite possible that the leak could be plugged by condensing steam or escaping material, or that some fraction of the escaping material would deposit along the leak path. It should be noted that our present understanding of the structural response of the containment building during accidents is not well developed and consequently represents a significant uncertainty in the modeling of radionuclide behavior.

In the Containment Systems Experiments, it was observed that leaks were plugged by condensed steam (Witherspoon and Postma, 1971). Measurements were made on the decontamination of radionuclides during passsage through leakage pathways, and significant attenuation was found (Hilliard and Postma, 1981).

A simple model has been developed for the plugging by aerosol deposits of ducts that are circular in cross section (Vaughan, 1978). This model has been recently compared with experimental data on the behavior of aerosols passing through leaks. The comparison indicated that the model was valid for a variety of aerosols over a range of duct diameters from 100 micrometers to 30 centimeters (Morewitz, 1981, 1982). In the process of plugging leak paths, the aerosols attach to the walls or to previously deposited aerosols. Some of the agglomerates can break off and be resuspended in the air stream so that the sizes of aerosols exiting from leaks can be increased (Morewitz, 1981).

It should be noted that the model described above was developed for idealized cracks. Most vessel cracks or leak pathways in a containment can be expected to be irregular, and therefore the actual aerosol removal may be higher than that predicted by the model.

A new code, MATADOR, that improves on CORRAL-2 is being developed at Battelle's Columbus Laboratories (Baybutt and Raghuram, 1981). The CONTAIN computer code, which was developed for the analysis of aerosol behavior during LMFBR accidents, is being extended at Sandia National Laboratories to handle radionuclide behavior in LWR containments during degraded-core accidents (Clauser et al., 1981). The TRAP code, originally developed for the analysis of radionuclide behavior in reactor-coolant systems and described in the preceding section, is being extended to treat the containment.

8.4 CURRENT ISSUES IN RADIONUCLIDE BEHAVIOR

The phenomena that occur during degraded-core accidents are complex. As a result, we do not have a complete understanding of the processes that occur, and there are a number of questions about radionuclide behavior that remain unanswered. As more research is performed, we can expect these issues to be resolved, but it is likely that new questions will arise. It is important that any analysis of radionuclide behavior take account of unresolved issues--for example, by providing estimates of the uncertainties they cause in the results of the analyses. This can, however, be a difficult task.

Some issues that remain unresolved at present and their probable impacts on public risk are listed in Table 8-6 and briefly discussed below.

Table 8-6. Unresolved issues in radionuclide behavior and their probable impacts on public risk

Issue	Probable impact
Aerosol generation from structural materials	High
Agglomeration of aerosols	High
Radionuclide removal by water pools and ice condensers	High to medium
Resuspension of deposited radionuclides	Medium
Chemical form of the radionuclide	High
Presence of organic iodides	Medium to low
Hydrogen combustion	Medium to low
Chemical reactions of radionuclides with materials in the containment	Medium to low
Radioactive decay	Medium
Radiation effects	Low
Coupling of thermal-hydraulics and radionuclide-behavior models	Medium
Verification and validation of computer codes	High to medium

8.4.1 AEROSOL GENERATION FROM STRUCTURAL MATERIALS

A variety of structural materials are present in the reactor. They include fuel cladding, control rods, core supports, and instrument tubes. Some of these materials can be released to the reactor-coolant system and the containment with radionuclides. In addition, the components of concrete can be suspended in the containment atmosphere as a result of the interaction between the molten core and the concrete basemat. These released structural materials can exert a significant impact on the behavior of radionuclides in the reactor-coolant system and the containment, primarily by their effects on such aerosol behavior as agglomeration.

Some data on structural material releases are available from fuel-melt experiments (Albrecht et al., 1978, 1979, 1981). These data, however, were obtained in small-scale experiments, and the extrapolation of the data to prototypic core-meltdown conditions is questionable. Larger-scale experiments, currently in progress, may help to solve this problem.

Limited experimental data on aerosol generation as a result of core-concrete interactions are available, and a preliminary model has been developed (USNRC, 1980). However, the model is accurate at best to within

an order of magnitude (D. A. Powers, Sandia National Laboratories, personal communication, 1981). Its application also requires a knowledge of the geometric configuration of the melt and a thermal analysis of the melt-concrete interaction, both of which are subject to significant uncertainties.

8.4.2 AGGLOMERATION OF AEROSOLS

It is likely that dense aerosols (with high concentrations per unit volume) will be generated in the reactor-coolant system and possibly parts of the containment as a result of the release of structural materials with radionuclides. Such aerosols will agglomerate and form particles of much larger sizes, which will then be subject to more rapid settling by gravitational deposition. In turn, this would significantly reduce the radionuclide release to the environment.

Little experimental work has been done under conditions appropriate for degraded-core accidents. However, some experiments are now under way (T. Kress, Oak Ridge National Laboratory, personal communication, 1981).

The importance of agglomeration in the reactor-coolant system has recently been assessed with the QUICK code (USNRC, 1981). For accident sequences with low aerosol concentrations and short residence times in the coolant system, such as the sequence AD in the Reactor Safety Study (USNRC, 1975), it was found that agglomeration and settling would result in less than 1 percent of the released materials being retained in the reactor-coolant system. For other accident sequences with longer residence times and higher aerosol concentrations, such as TMLB', QUICK predicted that 99 percent or more of the released materials will be retained in the reactor-coolant system. The importance of agglomeration as a contributor to particle retention thus depends on the characteristics of the particular accident sequence.

8.4.3 RADIONUCLIDE REMOVAL BY WATER POOLS AND ICE CONDENSERS

Little information exists on the removal of radionuclides by BWR suppression pools, PWR ice condensers, or water pools that may exist in PWRs (e.g., in the pressurizer). If the radionuclides released from the fuel do pass through water pools, significant radionuclide attenuation may occur. Whether such attenuation occurs depends on the nature of the accident. Of particular importance are the radionuclide-flow pathway to the environment and whether the pool water encountered is subcooled or boiling.

Some experimental data for the removal of elemental iodine in ice beds have been obtained by the Westinghouse Electric Corporation (Malinowski, 1970), but there are no directly relevant data for particulate removal. Existing information on the removal of radionuclides by water pools is poor. A program of research to measure aerosol attenuation by pool scrubbing has just been initiated at Battelle's Columbus Laboratories (J. C. Cunnane, Battelle's Columbus Laboratories, personal communication, 1981) with funding by the Electric Power Research Institute. No data are available yet, however.

A full understanding of radionuclide interactions with water pools requires a knowledge of the aqueous chemistry of radionuclides. The aqueous chemistry of iodine has been summarized in a recent report (USNRC, 1981).

8.4.4 RESUSPENSION OF DEPOSITED RADIONUCLIDES

Radionuclides that are deposited on surfaces in the reactor-coolant system or the containment can be resuspended; for example, particulates can be reentrained in fluid flow, and deposited vapors can be revaporized. To date, few analyses of radionuclide behavior have considered resuspension. In principle, the vaporization of condensed vapors is easily handled, and indeed the TRAP code treats this process. However, little information is available on the resuspension of chemisorbed vapors. Some data are available on the resuspension of particulates, but this process has not received attention in analyses of radionuclide behavior.

Radionuclides that dissolve in water can also become resuspended in the reactor atmosphere. The processes responsible for this resuspension are partitioning effects and the flashing of the water to steam to leave the dissolved materials. Other than the equilibration of elemental iodine in the water of the containment sprays, treated in CORRAL, these processes have not been modeled in analyses of radionuclide behavior.

8.4.5 RADIONUCLIDE CHEMICAL FORMS

The chemical form in which a radionuclide is released from the fuel can be expected to influence its subsequent behavior in the reactor-coolant system and the containment as well as the quantity that is eventually released to the environment. (It is also likely that the chemical form will influence the behavior of radionuclides in the environment.) The properties that exert an effect on radionuclide behavior include volatility or vapor pressure, solubility, and chemical reactivity, which vary among the possible chemical forms of a given radionuclide. At present, there is little information on the chemical forms of the radionuclides released from the core, and thus considerable uncertainty in analyses of radionuclide behavior can be introduced from this source.

Over the past year or so, questions have been raised on the chemical form of iodine released from fuel in degraded-core accidents.* For many years it has been assumed that iodine is released in elemental form.

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*The interested reader can consult the following correspondence, available in the NRC Public Document Room: letter from W. R. Stratton, A. P. Malinauskas, and D. O. Campbell to NRC Chairman J. Ahearne, dated August 14, 1980; letter from Chauncey Starr to NRC Commissioner J. Hendrie, dated September 2, 1980; and letter from the Nuclear Safety Oversight Committee to President Jimmy Carter, dated December 21, 1980.

However, experiments at the Oak Ridge National Laboratory suggest that the actual form is cesium iodide (Campbell et al., 1981). Thermodynamics calculations performed in the Reactor Safety Study and more recently (USNRC, 1981) support this conclusion. Some evidence has also been found for the formation of cesium iodide deposits in the fuel-to-cladding gap (Cubicciotti and Sanecki, 1978). However, the experimental evidence is not definitive, and thermodynamics conclusions alone cannot necessarily be expected to determine chemical forms (kinetics is also important).

The Nuclear Regulatory Commission attempted to resolve this issue by funding a study to examine the state of the technology of iodine behavior. The results of the study have been recently reported (USNRC, 1981). The report summarized and evaluated available information on radionuclide releases from fuel, the chemistry of cesium and iodine, and radionuclide transport in the reactor-coolant system and the containment. The primary objectives were to determine, if possible, the most likely chemical form of iodine and to determine the effect of chemical form on the quantity of iodine released to the environment. The report concluded:

> The current data base suggests that cesium iodide will be the expected predominant iodine chemical form under most postulated light water reactor accident conditions. The current evidence regarding the chemical form of iodine released from fuel at high temperatures (>1400°C) is inconclusive. However, thermodynamic calculations predict that formation of CsI should occur in the gaseous reducing atmosphere in the reactor coolant system following release from fuel even if iodine is not released from the fuel as CsI. The formation of some more volatile iodine species (e.g., elemental iodine and organic iodines), however, cannot be precluded under certain accident conditions.
>
> The assumed form of iodine (either cesium iodide or elemental iodine) was not predicted to have a major influence on the estimated magnitude of iodine attenuation in the containment for severe accident sequences with early containment failure in which there is little time for natural fission product retention mechanisms to be effective. However, the assumed chemical form of iodine can influence the predicted attenuation within the reactor coolant system, where, in general, the attenuation factor will be greater for cesium iodide than for elemental iodine (i.e., less iodine will escape into the containment).

It should be recognized that, though these conclusions are based on the current state of the technology, there are significant uncertainties in the analyses and the supporting data base, and thus the conclusions cannot be regarded as definitive.

Questions can also be raised about the chemical forms of other radionuclides and their effects on radionuclide releases to the environment. This is an area in which experimental research is badly needed. Some work is in progress at Sandia National Laboratories to investigate the radionuclide chemical forms that may be present in reactor-coolant systems during degraded-core accidents (R. M. Elrick, Sandia National Laboratories, personal communication, 1981). Some insights into likely chemical forms can

be obtained by consulting thermodynamics tables or performing thermodynamics calculations with such computer codes as SOLGASMIX (Bessmann, 1977). However, these tables and codes do not contain data for all likely radionuclide chemical forms. Moreover, thermodynamics considerations alone cannot be used to predict chemical forms since chemical kinetics also plays an important role. A summary of available information on the chemistry of volatile radionuclides has recently been compiled (USNRC, 1981).

8.4.6 PRESENCE OF ORGANIC IODIDES

It is possible that iodine will react with organic materials (e.g., lubricating oils present in the containment), after it has been released from the fuel, and form organic iodides, such as methyl iodide. In the Reactor Safety Study, for example, it was assumed that a small percentage of the iodine in the containment is converted to organic iodides (USNRC, 1975). Both radiolytic and nonradiolytic formation mechanisms were considered. The importance of this phenomenon is that organic iodides differ from other chemical forms in their transport and deposition behavior. It is also possible that volatile forms of iodine, such as hydrogen iodide (HI) and hypoiodous acid (HOI), may be formed. Some recent work has provided evidence for the existence of HOI (C. C. Lin, General Electric Company, personal communication, 1981). The presence of such forms can have important implications for the transport and deposition of iodine and for the partitioning of iodine between water and the reactor atmosphere.

The information base on this topic is quite sparse, and the estimation of the quantities in which such materials may be formed is subject to large uncertainties. More experimental work is needed to resolve these issues.

8.4.7 HYDROGEN COMBUSTION

During degraded-core accidents, hydrogen can be formed by metal-water reactions or possibly by radiolysis under boiling conditions (see Chapter 7). In the latter case, free oxygen gas can be liberated. Hydrogen can burn or detonate in the containment if air is present, certain hydrogen concentrations are reached, and an ignition source is present. If such an event were to occur, it might affect the deposition rates and the chemical forms of the radionuclides in the containment. No information is presently available on such effects.

8.4.8 CHEMICAL REACTIONS OF RADIONUCLIDES WITH MATERIALS IN THE CONTAINMENT

A wide variety of materials are present in the containments of light-water reactors, from lubricating oils to paints on containment surfaces. It is possible that these materials will react chemically with the radionuclides released into the containment and thus possibly alter their behavior. The formation of organic and other forms of iodine provides a specific example. Little is known about such reactions.

8.4.9 RADIOACTIVE DECAY

Most analyses of radionuclide behavior do not account for radioactive decay because it has been assumed that decay would exert little effect on the results, owing to the relatively short time periods involved. However, for certain accidents, the time periods can be on the order of days, and there are some transformations, such as that of tellurium to iodine, that could be important. The treatment of radioactive decay is especially important if short-lived isotopes are to be considered.

8.4.10 RADIATION EFFECTS

It is possible that the radiation fields in reactor accidents may influence radionuclide behavior either by physical (e.g., charge) or chemical effects. Studies of charging effects on aerosol behavior have concluded that there is minimal impact (Reed et al., 1977). The formation of organic iodides is controlled, in part, by radiolytic mechanisms (Postma and Zavadoski, 1972). It is also possible, as discussed earlier, that the formation of hydrogen and oxygen by the radiolysis of water may affect radionuclide behavior. Little other work pertinent to reactor-accident conditions has been done.

8.4.11 COUPLING OF THERMAL-HYDRAULICS AND RADIONUCLIDE-BEHAVIOR MODELS

The state of the art in the modeling of radionuclide behavior artificially decouples the evaluation of thermal-hydraulic conditions. The computer codes that describe radionuclide behavior use information on thermal-hydraulic conditions as input. Radionuclide transport and deposition are superimposed on fluid flow. Although at present this approximation is believed to be reasonable, considering the level of sophistication of current models, it is possible that, as the state of the art in radionuclide behavior is advanced, an integration with the thermal-hydraulics analyses may be warranted. The assumption of well-mixed, homogeneous control volumes, which is used in virtually all radionuclide-behavior codes, may also need to be reassessed.

8.4.12 VERIFICATION AND VALIDATION OF COMPUTER CODES

It is important that computer codes for the analysis of radionuclide behavior and the models they contain be verified (independently assessed to determine that they function as specified and there are no coding errors) and validated (assessed to determine the accuracy of the analyses by comparison with experimental results). Few of the available codes have either been verified or validated, in the latter case mainly owing to the lack of experimental data.

8.5 INFORMATION REQUIREMENTS

This section discusses the information needed for analyses of radionuclide behavior and describes where such information can be obtained. Each of the steps in the analysis is addressed in turn.

8.5.1 INVENTORIES OF RADIONUCLIDES AND STRUCTURAL MATERIALS

Computer codes that calculate radionuclide inventories require information on the operating history of the reactor (Bell, 1973; Croff, 1980). Such information is available from the utility operating the reactor. They also require a set of nuclear constants; these are often incorporated in the codes as data libraries. Information on the amounts of structural materials in the system can be found in documents containing design data, such as safety analysis reports, or in design drawings, or it can be obtained from the utility or the vendor.

8.5.2 RADIONUCLIDE AND STRUCTURAL MATERIAL SOURCE TERM FROM THE CORE

The analysis of the release of radionuclides and structural materials from the core requires information on the following:

1. Inventories of radionuclides and structural materials (taken from the previous step).

2. The physical processes of core-melt accidents (provided by the analyses described in Chapter 7).

3. Physical and chemical data needed by each of the release models (sources were discussed in Section 8.3.1).

Information on the physical processes of core-melt accidents specifies which radionuclide-release processes will occur, provides data on the atmosphere in the reactor-coolant system (e.g., amount of hydrogen present), describes the manner of core degradation, and specifies the times at which various events occur (e.g., cladding failure, core-melt initiation and termination, and pressure-vessel failure).

8.5.3 TRANSPORT, DEPOSITION, AND RELEASE IN THE REACTOR-COOLANT SYSTEM

This element of the analysis requires information on the following:

1. Radionuclide and structural material source terms from the core. Data on quantities, release rates, time dependence, chemical forms, and particle-size distribution and composition are needed. (Provided by the previous step.)

2. The geometric configuration of the reactor-coolant system and the materials of its surfaces. (Provided by sources of information on the reactor design, such as the safety analysis report.)

3. Physical conditions. Information is needed on fluid flow rates and flow paths, fluid composition, fluid temperatures, surface temperatures, and system pressure, all as a function of time. (Provided by the analyses described in Chapter 7.)

4. Physical and chemical properties of radionuclides for the models of radionuclide behavior. (Sources were discussed in Section 8.3.2.)

8.5.4 TRANSPORT, DEPOSITION, AND RELEASE IN THE CONTAINMENT

The information needed for this task can be summarized as follows:

1. Radionuclide and structural material source terms. Data on quantities, release rates, time dependence, chemical forms, and particle-size distribution and composition are needed. (Provided by the previous step.)

2. Steam source term. Data on the quantity and release rate are needed. (Provided by the analyses described in Chapter 7.)

3. The geometric configuration of the containment and the materials of its surfaces. (Provided by sources of information on the reactor design, such as the safety analysis report.)

4. Physical conditions. Information is needed on fluid flow rates and flow paths, fluid composition, fluid temperatures, surface temperatures, system pressure, and steam condensation, all as a function of time. (Provided by the analyses described in Chapter 7.)

5. Engineered safeguards. Information is needed on their functionability (provided by the analyses described in Chapter 7) and operational characteristics, such as spray flow rates and water-droplet size, and filter efficiencies. (Provided by sources of information on the reactor design, such as the safety analysis report.)

6. Containment failure. Information is needed on the time and the mode of containment failure and such leak-path characteristics as length, cross-sectional areas, and tortuosity. (Provided by the analyses described in Chapter 7.)

7. Physical and chemical properties of radionuclides for the models of radionuclide behavior. (Sources were discussed in Section 8.3.3.)

8.6 UNCERTAINTIES IN THE ANALYSIS OF RADIONUCLIDE BEHAVIOR

8.6.1 SOURCES OF UNCERTAINTY

Uncertainties are present in both the data and the models used in analyzing the behavior of radionuclides. Data may be imprecise or unavailable, and models may only approximate the processes they are intended to describe. The omission of important processes because certain phenomena are not completely understood or because they cannot be modeled represents another source of uncertainty. Such sources of uncertainty were identified in Section 8.3. The most significant sources are summarized in Table 8-7.

All these sources of uncertainty propagate through to the results of the analyses. It is important for the assessment and utilization of the results of a PRA study that uncertainties in the results be evaluated and presented with the risk estimates. It is also important to represent the sources of uncertainty and to present their contributions to the total uncertainty because this information is of value in establishing priorities for further work and providing insights into the results of the probabilistic risk assessment.

8.6.2 RECOMMENDED PROCEDURES FOR UNCERTAINTY ANALYSIS

Ideally, in a probabilistic risk assessment, uncertainties should be assessed for the actual analyses that were performed. On some occasions a purely qualitative assessment will suffice, while on others, numerical estimates will be needed. The level of quantification can range from the simple application of engineering judgment for evaluating bounds on the predicted radionuclide releases into the environment to the development of input uncertainty estimates and their propagation through the computer codes by one of several available methods.

Various techniques of uncertainty analysis are described in Chapter 12. One method that has been developed for the analysis of uncertainties in radionuclide behavior uses statistical design and response-surface methods (Baybutt et al., 1981). The response-surface method is described in Chapter 12. Since relatively little work has been done in this area, it is not possible to recommend any specific techniques of uncertainly analysis as most appropriate. Consequently, the analyst of uncertainties in radionuclide behavior should select a technique from those discussed in Chapter 12, basing his choice on the needs of the probabilistic risk assessment being performed and the methods used for the analysis of radionuclide behavior.

One simple way to estimate uncertainties in releases to the environment is to use values that have been developed in other studies. Obviously, care must be taken to ensure that the selected values are appropriate for the case at hand. The section that follows provides a summary of such information that is available.

Table 8-7. Significant sources of uncertainty in the analysis of radionuclide behavior

Element of analysis	Sources of uncertainty
Inventories of radionuclides and structural materials	No significant uncertainties
Radionuclide and structural material source term from the core	Mode of core degradation and core-melt behavior Quantities of structural material released Chemical forms of released radionuclides Timing of radionuclide and structural material releases Adequacy of experimental data base on releases Validity of extrapolation of correlations based on small-scale experiments to prototypic reactor-accident conditions
Transport, deposition, and release in the reactor-coolant system	Source term from the core (magnitude, physical and chemical form, timing) Particle agglomeration Chemical reactions Water scrubbing of radionuclides Thermal-hydraulic conditions Vapor pressures of radionuclides Validity of computer codes
Transport, deposition, and release in the containment	Source term from the reactor-coolant system (magnitude, physical and chemical form, timing) Removal of radionuclides by ice condensers and BWR suppression pools Thermal-hydraulic conditions, particularly steam condensation on particles and hydrogen combustion Particle agglomeration Radionuclide attenuation during passage through containment cracks Chemical reactions Validity of computer codes

8.6.3 AVAILABLE INFORMATION ON UNCERTAINTIES

Only beginning efforts at the quantification of uncertainties in the analysis of radionuclide behavior have been made. One such effort was a project at Battelle's Columbus Laboratories to develop and apply methods for evaluating uncertainties in the predictions of the MARCH and CORRAL codes of radionuclide releases to the environment (Baybutt and Kurth, 1980). A series of reports on this work are being prepared, and some of the results have been published (Kurth et al., 1980).

Tables 8-8, 8-9, and 8-10 show the results of uncertainty analyses for three meltdown-accident sequences from the Reactor Safety Study: TMLB'-δ, ACDF-α, and TC-γ. Typically, standard deviations were 40 to 60 percent of the mean release fractions. Both data and model uncertainties were considered, but only the variables and models that were believed to be the dominant contributors were included in the analyses. Thus the actual uncertainties are likely to be larger. The results of the calculations also depend on the variable and model input uncertainties that were used and the method of uncertainty analysis that was employed. Furthermore, the analyses do not include uncertainties associated with the validity or the completeness of the MARCH and CORRAL codes. These can be expected to be substantial; consequently, the results of Tables 8-8, 8-9, and 8-10 should not be used as estimates for uncertainties in radionuclide releases to the environment without modification to account for the additional sources of uncertainty.

Similar methods have been applied to the evaluation of uncertainties in TRAP calculations of radionuclide deposition in the reactor-coolant system (Baybutt et al., 1980). Calculations were made for the BWR accident sequence TC-γ with a source term containing elemental iodine, cesium hydroxide, and plutonium dioxide. The results are shown in Table 8-11, where the standard deviation expressed as a percentage of the mean deposition fraction is seen to range from 48 to 75 percent. The qualifications given for the MARCH/CORRAL uncertainty estimates also apply to those for TRAP.

8.7 RELEASE CATEGORIES

Ideally, in a comprehensive risk assessment, analyses of radionuclide behavior should be made for all accident sequences of interest. However, such an exercise can become prohibitively expensive. In order to circumvent this problem, it is possible to categorize sequences by their characteristics in such a way that members of the same category have similar radionuclide-release fractions. A set of release categories is then defined such that all accidents assigned to the same category are assumed to have the same set of release fractions. It is then necessary to perform analyses of radionuclide behavior for only one accident sequence in each category in order to determine the set of release fractions for that category. This is the approach that was used in the Reactor Safety Study (USNRC, 1975) for radionuclide releases to the environment, and the release categories defined in the Study are shown in Table 8-12 as an example.

Table 8-8. Uncertainty estimates for the environmental radionuclide-release fractions of the TMLB'-δ PWR meltdown-accident sequence[a]

Radionuclide group[b]	Best-estimate release fraction[c]	Mean release fraction[d]	Standard deviation
I	0.59	0.18	0.08
Cs	0.55	0.38	0.17
Te	0.18	0.35	0.16
Sr	0.07	0.05	0.03
Ru	0.02	0.06	0.03
La	0.003	0.01	0.006

[a]From analyses by P. Baybutt, D. C. Cox, and R. E. Kurth, Battelle's Columbus Laboratories (work in progress). See qualifications in the text of Section 8.6.3 on what these estimates do and do not include.

[b]See Table 8-1 for definitions.

[c]Obtained from the CORRAL code with best-estimate input data.

[d]Obtained from a statistical analysis.

Table 8-9. Uncertainty estimates for the environmental radionuclide-release fractions of the ACDF-α PWR meltdown-accident sequence[a]

Radionuclide group[b]	Best-estimate release fraction[c]	Mean release fraction[d]	Standard deviation
I	0.49	0.38	0.06
Cs	0.36	0.38	0.16
Te	0.19	0.34	0.15
Sr	0.04	0.06	0.03
Ru	0.19	0.29	0.12
La	0.002	0.01	0.006

[a]From analyses by P. Baybutt, D. C. Cox, and R. E. Kurth, Battelle's Columbus Laboratories (work in progress). See qualifications in the text of Section 8.6.3 on what these estimates do and do not include.

[b]See Table 8-1 for definitions.

[c]Obtained from the CORRAL code with best-estimate input data.

[d]Obtained from a statistical analysis.

Table 8-10. Uncertainty estimates for the environmental radionuclide release fractions of the TC-γ BWR meltdown-accident sequence[a]

Radionuclide group[b]	Best-estimate release fraction[c]	Mean release fraction[d]	Standard deviation
I	0.04	0.08	0.05
Cs	0.15	0.25	0.11
Te	0.11	0.27	0.12
Sr	0.02	0.03	0.02
Ru	0.01	0.05	0.03
La	0.001	0.01	0.006

[a]From analyses by P. Baybutt, D. C. Cox, and R. E. Kurth, Battelle's Columbus Laboratories (work in progress). See qualifications in the text of Section 8.6.3 on what these estimates do and do not include.

[b]See Table 8-1 for definitions.

[c]Obtained from the CORRAL code with best-estimate input data.

[d]Obtained from a statistical analysis.

Table 8-11. Uncertainty estimates for the reactor-coolant-system deposition fractions[a] of the TC-γ BWR meltdown-accident sequence[b]

Radio-nuclide	Mean deposition fraction	Standard deviation
Iodine	0.04	0.03
Cesium	0.29	0.14
Plutonium	0.1	0.06

[a]Defined as the ratio of material deposited to that released.

[b]From Baybutt et al. (1980). See qualifications in the text of Section 8.6.3 on what these estimates do and do not include.

Table 8-12. Radionuclide-release categories used in the Reactor Safety Study

Release category	Fraction of core inventory released[a]							
	Noble gases	Organic iodine	I	Cs	Te	Ba	Ru	La
PWR-1	0.9	6×10^{-3}	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR-2	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR-3	0.8	6×10^{-3}	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR-4	0.6	2×10^{-3}	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR-5	0.3	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR-6	0.3	2×10^{-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}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR-8	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR-9	3×10^{-3}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR-1	1.0	7×10^{-3}	0.40	0.40	0.70	0.05	0.5	5×10^{-3}
BWR-2	1.0	7×10^{-3}	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR-3	1.0	7×10^{-3}	0.10	0.10	0.30	0.01	0.02	3×10^{-3}
BWR-4	0.6	7×10^{-4}	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1×10^{-4}
BWR-5	5×10^{-4}	2×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-11}	0	0

[a]See Table 8-1 for definitions of the radionuclide groups I, Cs, etc.

Such categorizations are likely to be dependent on both the design of the reactor and the methods used for radionuclide-behavior analyses. Furthermore, their establishment and use involves subjective judgment. It is possible that generic release categories applicable to several different reactor designs could be developed.

Little work has been done on constructing release categories or establishing procedures for the development of release categories since the time of the Reactor Safety Study. However, some work on this topic is being performed as part of the Oconee PRA (W. J. Parkinson, Science Applications, Inc., personal communication, 1982).

8.8 PROCEDURES

A step-by-step set of procedures for analyzing radionuclide behavior in degraded-core accidents is provided below.

Task 1: Establish level of analysis to be performed. The level of analysis can range from the simple use of radionuclide-release categories developed in previous studies to the use of the most sophisticated methods available. The appropriate level of analysis depends on the objectives of the PRA, available resources, and time constraints.

Task 2: Select techniques to be used for analysis. Presently available methods are described in Section 8.3. The selection of techniques is governed by the level of analysis to be performed, the availability of needed data, the objectives of the PRA, available resources, and time constraints. The techniques that are selected may need to be adapted to the specific problem at hand. The analyst should also determine that the methods account for all phenomena likely to be of importance. Any improvements made in the analytical techniques subsequent to the publication of this procedures guide should be reviewed for possible inclusion in the work.

Task 3: Collect needed input data. The information identified in Section 8.5 as needed for analyses of radionuclide behavior must be collected.

Task 4: Determine inventories of radionuclides and structural materials. Available methods are described in Section 8.3.1. Often the ORIGEN computer code (Bell, 1973; Croff, 1980) is employed to estimate the radionuclide inventory in the core at the outset of the accident. Alternative codes like CINDER should be considered for possible use. The choice depends on how accurately the core inventory must be predicted, and therefore it is necessary to establish the degree to which the codes considered agree with experimental results. The quantities of structural materials can be determined from sources of information on the reactor design, an example being the safety analysis report.

Task 5: Determine radionuclide and structural material releases to the reactor-coolant system. Available methods are described in Section 8.3.2.

There are no generally accepted techniques. New methods are under development, but until they become available, special assessments of radionuclide and structural material releases will have to be made case by case, using whatever information is available when the assessments are made.

Task 6: Determine radionuclide releases to the containment. Available methods are described in Section 8.3.3. The only computer code presently available for performing such analyses is TRAP (Baybutt and Jordan, 1977; Jordan et al., 1979). However, this code does not include all processes that are likely to be important for all accidents (see Section 8.3.3). If necessary, for the cases to be analyzed, code modifications or special assessments must be made to account for important processes not modeled in TRAP. Aerosol processes like gravitational agglomeration can be modeled with the QUICK or HAA codes.

Task 7: Determine radionuclide releases to the environment. Available methods are described in Section 8.3.4. Several computer codes are available. CORRAL (USNRC, 1975) has been used widely but has a number of deficiencies. New codes are under development. In the meantime, codes like CORRAL and NAUA (Bunz et al., 1981) can be used, but an assessment must be made of the validity of their results in light of their deficiencies or processes not treated. The use of both CORRAL and NAUA may provide more reliable results than either code does individually.

Task 8: Determine uncertainties in radionuclide releases to the environment. A procedure for uncertainty analysis must be selected, and input data and model uncertainties must be quantified. The procedure is then used to propagate the input data and model uncertainties through the analyses of radionuclide behavior to determine the uncertainties in the results of the analyses, namely, the radionuclide releases to the environment.

Task 9: Provide data to analyses of environmental transport and consequences. The output of the analyses of radionuclide behavior is provided as input to the analyses of environmental transport and consequences. This consists of the magnitude of radionuclide releases to the environment as a function of time, particle sizes and compositions, and chemical forms.

8.9 METHODS OF DOCUMENTATION

This section provides an outline of the information that should be in a final report describing the results of the analysis. Sufficient detail should be provided for purposes of peer review.

1. Introduction. The objectives of the radionuclide behavior analysis should be described in light of the overall objectives of the probabilistic risk assessment being performed. The level of analysis should be specified, and any special requirements of the analysis for the reactor design being analyzed should be described.

2. Overview. The elements involved in the analysis should be summarized, and a description should be provided of the procedure that was followed. The methods used in the assurance of technical quality should be discussed.

3. Analytical techniques. A description of the methods used should be provided, with a justification for their selection. Details should be given of any modifications made to existing methods.

4. Input data. References should be provided to the sources of input data employed. A summary should be given of key data.

5. Assumptions. Any assumptions made in the analyses should be specified and discussed.

6. Presentation of results. This section should discuss the particular accidents analyzed and present the results of each step in the procedure. The final results presented should be those needed as input to the analyses of environmental transport and consequences (Chapter 9).

7. Summary. Any pertinent observations on the analyses or their results should be given here.

8.10 DISPLAY OF FINAL RESULTS

This section describes the format of the results of the radionuclide-behavior analyses that are provided as input to the analyses of environmental transport and consequences. The information that is needed consists of (1) the magnitude of radionuclide releases to the environment, (2) the physical form of the released radionuclides, and (3) the chemical form of the released radionuclides. The required format is briefly described below.

In preparing the results on release magnitudes, the radionuclide classification of Table 8-1 should be employed or one that is compatible with the analytical methods used for the evaluation of environmental transport and consequences. Radionuclide releases should be expressed as a fraction of the original core inventory at the beginning of an accident. The time dependence of these releases should be specified.

In presenting results on the physical form of the radionuclides, the distribution of particle sizes and composition as a function of particle size should be given. The chemical forms in which radionuclides are released to the environment should be specified for each radionuclide.

8.11 ASSURANCE OF TECHNICAL QUALITY

Little formal work has been done to develop methods of ensuring that PRAs are performed correctly. Consequently, this section presents only some general suggestions on the procedures that can be used for the assurance of technical quality (AOTQ) in analyses of radionuclide behavior.

The function of ensuring technical quality should be provided at several levels: the levels of the analysts (the persons actually performing calculations), the task leader (the individual directing the analysts), the PRA program manager (the individual directing the overall probabilistic risk assessment), the plant operating personnel, and peer reviewers. In general, these persons should perform AOTQ activities appropriate to their level in the organization and check on the execution of such activities by their immediate subordinates. For analysts these activities would include checking calculations and input data to computer codes and maintaining a written record of all calculations. The task leader would perform spot checks, and the PRA program manager would review the results for any inconsistencies or apparent errors. Review of results for their reasonableness is a valuable approach at all levels. Plant operating personnel should review the analyses to ensure that the design and operation of the plant have been represented properly. Peer reviewers should be used to provide a truly independent appraisal of the analyses performed.

Part of the AOTQ program must address the verification and validation of any computer codes or other tools used in the analyses. In addition, whenever computer codes are implemented on a new computer, they should be thoroughly checked using test cases.

The formal procedures described in Chapter 2 for the assurance of technical quality should be implemented for analyses of radionuclide behavior.

REFERENCES

Albrecht, H., and H. Wild, 1981. "Investigation of Fission Product Release by Annealing and Melting of LWR Fuel Pins in Air and Steam," in Proceedings, ANS Topical Meeting on Reactor Safety Aspects of Fuel Behavior, Sun Valley, Idaho, August 2-6, 1981, American Nuclear Society, La Grange Park, Ill.

Albrecht, H., V. Matschoss, and H. Wild, 1978. "Investigation of Activity Release During Light Water Reactor Core Meltdown," Nuclear Technology, Vol. 40, p. 278.

Albrecht, H., V. Matschoss, and H. Wild, 1979. "Experimental Investigation of Fission and Activation Product Release from LWR Fuel Rods at Temperatures Ranging from 1500 to 2800°C," paper presented at IAEA Specialists Meeting on the Behavior of Defected Zirconium Alloy Clad Ceramic Fuel in Water Cooled Reactors, Chalk River, Canada, September 1979.

Baybutt, P., and H. Jordan, 1977. "TRAP: A Computer Code for the Analysis of Radionuclide Transport in LWR Primary Systems During Hypothetical Terminated LOCA's," in Proceedings of Topical Meeting on Thermal Reactor Safety, Sun Valley, Idaho, July/August 1977, CONF-770708, Vol. 3, p. 249.

Baybutt, P., and R. E. Kurth, 1978. Uncertainty Analysis of Light Water Reactor Meltdown Accident Consequences: Methodology Development, Topical Report for the U.S. Nuclear Regulatory Commission, Battelle's Columbus Laboratories, Columbus, Ohio.

Baybutt, P., and S. Raghuram, 1981. "MATADOR--A Replacement for the CORRAL Computer Code for the Analysis of Radionuclide Behavior in LWR Containments," paper presented at the Ninth Water Reactor Safety Research Information Meeting, Gaithersburg, Md., October 26-30, 1981.

Baybutt, P., J. A. Gieseke, H. Jordan, and S. Raghuram, 1980. "An Assessment of LWR Primary System Radionuclide Retention in Meltdown Accidents Using the TRAP Computer Code," in Proceedings, ANS/ENS Topical Meeting on Thermal Reactor Safety, Knoxville, Tennessee, April 1980, CONF-800403, Vol. 2, p. 1322, American Nuclear Society, La Grange Park, Ill.

Baybutt, P., R. E. Cudnik, and S. Raghuram, 1980. "Iodine Scrubbing in Steam Generator Tube Rupture Accidents," Proceedings, ANS/ENS Topical Meeting on Thermal Reactor Safety, Knoxville, Tennessee, April 1980, CONF-800403, Vol. 2, p. 1306, American Nuclear Society, La Grange Park, Ill.

Baybutt, P., S. L. Nicolosi, and S. Raghuram, 1981. "Radionuclide Source Terms for Degraded Core Accidents in Light Water Reactors," in Proceedings, ANS Topical Meeting on Reactor Safety Aspects of Fuel Behavior, Sun Valley, Idaho, August 2-6 1981, American Nuclear Society, La Grange Park, Ill.

Bell, M. J., 1973. ORIGEN--The ORNL Isotope Generation and Depletion Code, ORNL-4628, Oak Ridge National Laboratory, Oak Ridge, Tenn.

Besmann, T. M., 1977. SOLGASMIX-PV, A Computer Program To Calculate Equilibrium Relationships in Complex Chemical Systems, ORNL/TM-5775, Oak Ridge National Laboratory, Oak Ridge, Tenn.

Booth, A. H., 1957. Report AECL DCI-27, Atomic Energy of Canada Ltd., Ottawa, Canada.

Bunz, H., and W. Schoeck, 1980. "Measurements of the Condensation of Steam on Different Aerosols Under LWR Core Meltdown Accidents," paper presented at the CSNI Specialists Meeting on Nuclear Aerosols in Reactor Safety, Gatlinburg, Tenn., April 1980.

Bunz, H., and W. Schoeck, 1980. "The Natural Removal of Particulate Radioactivity in an LWR Containment During Core Meltdown Accidents," in Proceedings, ANS/ENS Topical Meeting on Thermal Reactor Safety, Knoxville, Tennessee, April 1980, CONF-800403, Vol. 2, p. 1328, American Nuclear Society, La Grange Park, Ill.

Bunz, H., W. Schikarski, and W. Schoeck, 1981. "The Role of Aerosol Behavior in Light Water Reactor Core Melt Accidents," Nuclear Technology, Vol. 53, p. 141.

Campbell, D. O., A. P. Malinauskas, and W. R. Stratton, 1981. "The Chemical Behavior of Fission Product Iodine in Light Water Reactor Accidents," Nuclear Technology, Vol. 53, p. 111.

Clauser, M. J., M. E. Sanglaub, J. E. Kelly, J. P. Odom, M. F. Young, P. J. Cooper, K. K. Murato, and P. E. Rexroth, 1981. "Development of CONTAIN for LWR Containment Analysis," paper presented at the Ninth Water Reactor Safety Research Information Meeting, Gaithersburg, Md., October 26-30, 1981.

Croff, A. G., 1980. ORIGEN2--A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code, ORNL-5621, Oak Ridge National Laboratory, Oak Ridge, Tenn.

Cubicciotti, D., and J. E. Sanecki, 1978. "Characterization of Deposits on Inside Surfaces of LWR Cladding," Journal of Nuclear Materials, Vol. 78, p. 96.

Cubicciotti, D., 1981. "A Model for Release of Fission Gases and Volatile Fission Products from Irradiated UO_2 in a Steam Environment," Nuclear Technology, Vol. 53, p. 5.

Fluke, R. J., 1981. FISSCON: A Fission Product Behavior Code, Report 79317, Ontario Hydro, Toronto, Canada.

Gelbard, F. MAEROS: Input/Output Manual, draft USNRC Report NUREG/CR-1391.

Gieseke, J. A., H. Jordan, and K. W. Lee, 1980. Aerosol Measurements and Modeling for Fast Reactor Safety, USNRC Report NUREG/CR-1165.

Gieseke, J. A., K. W. Lee, and L. D. Reed, 1978. HARM-3 User's Manual, USNRC Report BMI-NUREG-1991 (Battelle Memorial Institute, Columbus, Ohio).

Hilliard, R. K., and A. K. Postma, 1981. "Large-Scale Fission Product Containment Tests," Nuclear Technology, Vol. 53, p. 163.

Hubner, R. S., E. U. Vaughan, and L. Baurmash, 1973. HAA-3 User's Report, AI-AEC-13038, Atomics International.

Jordan, H., J. A. Gieseke, and P. Baybutt, 1979. TRAP-MELT User's Manual, USNRC Report NUREG/CR-0632.

Jordan, H., W. Schikarski, and H. Wild, 1974. Nukleare Aerosole in Geschlossenem System, KfK-1989, Kernforschungszentrum Karlsruhe, Federal Republic of Germany.

Jordan, H., P. M. Schumacher, J. A. Gieseke, and K. W. Lee, 1980. Multiple Zone Aerosol Behavior Model, USNRC Report NUREG/CR-1294.

Katayama, Y. B., D. J. Bradley, and C. O. Harvey, 1980. Status Report on LWR Fuel IAEA Leach Tests, PNL-3173, Pacific Northwest Laboratory, Richland, Wash.

Kurth, R. E., P. Baybutt, and D. C. Cox, 1980. "Determination of Environmental Radionuclide Release Uncertainties for LWR Meltdown Accidents," Transactions of the American Nuclear Society, Vol. 34, p. 444.

Lorenz, R. A., J. L. Collins, and A. P. Malinauskas, 1979. "Fission Product Source Terms for the Light Water Reactor Loss-of-Coolant Accident," Nuclear Technology, Vol. 46, p. 404.

Lorenz, R. A., J. L. Collins, and A. P. Malinauskas, 1980. Fission Product Source Terms for the LWR Loss-of-Coolant Accident, USNRC Report NUREG/CR-1288.

Lorenz, R. A., J. L. Collins, A. P. Malinauskas, M. F. Osborne, and R. L. Towns, 1980. Fission Product Release from Highly Irradiated LWR Fuel Heated to 1300-1600 C in Steam, USNRC Report NUREG/CR-1386.

Malinowski, D. D., 1970. Iodine Removal in the Ice Condenser System, WCAP-7426, Westinghouse Electric Corporation, Pittsburgh, Pa.

Morewitz, H. A., 1981. "Fission Product and Aerosol Behavior Following Degraded Core Accidents," Nuclear Technology, Vol. 53, p. 120.

Morewitz, H. A., 1982. "Leakage of Aerosols from Containment Buildings," Health Physics, Vol. 42, p. 195.

Murfin, W. B., 1977. A Preliminary Model for Core/Concrete Interactions, SAND77-0370, Sandia National Laboratories, Albuquerque, N.M.

Nishio, G., M. Tanaka, K. Hashimoto, Y. Motoki, M. Naritomi, and S. Kitani, 1981. "Containment Spray Model for Predicting Radioiodine Removal in Light Water Reactors," Nuclear Technology, Vol. 54, p. 68.

Postma, A. K., and B. M. Johnson, 1971. Containment Systems Experiment Final Program Summary, BNWL-1592, Battelle Northwest Laboratories, Richland, Wash.

Postma, A. K., and R. W. Zavadoski, 1972. Review of Organic Iodine Formation Under Accident Conditions in Water-Cooled Reactors, WASH-1233, U.S. Atomic Energy Commission, Washington, D.C.

Powers, D. A., and H. Westrich, 1981. Advanced Reactor Safety Research Quarterly Report, January-March 1980, USNRC Report NUREG/CR-1594.

Reed, L. D., H. Jordan, and J. A. Gieseke, 1977. "Charging of Radioactive Aerosols," Journal of Aerosol Science, Vol. 8, p. 457.

Reed, L. D., K. W. Lee, and J. A. Gieseke, 1980. "The Behavior of Contained Radioactive Suspensions," Nuclear Science and Engineering, Vol. 75, p. 167.

Reimann, M., and W. B. Murfin, 1978. "Calculations for the Decomposition of Concrete Structures by a Molten Pool," paper presented at the PAHR Information Exchange Meeting, Ispra, Italy, October 1978.

Rest, J., 1978. GRASS-SST: Fission Gas Behavior in UO_2, USNRC Report, U.S. Nuclear Regulatory Commission, Washington, D.C.

Rest, J., 1982. "The Prediction of Transient Fission-Gas Release and Fuel Microcracking Under Severe Core-Accident Conditions," Nuclear Technology, Vol. 56, p. 553.

Ritzman, R. L., and D. L. Morrison, 1971. FRCRL2--A Computer Code for Calculating Fission Product Release in Reactor Accident Analyses, BMI-1913, Battelle Memorial Institute, Columbus, Ohio.

Ritzman, R. L., 1971. MIRA--Methods for Iodine Removal Analysis in Reactor Containment Systems, BMI-1915, Battelle Memorial Institute, Columbus, Ohio.

USNRC (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., Appendix VII.

USNRC (U.S. Nuclear Regulatory Commission), 1978. Light Water Reactor Safety Research Program Quarterly Report, January-March 1978, NUREG/CR-0324, Washington, D.C.

USNRC (U.S. Nuclear Regulatory Commission), 1980. Report of the Zion/Indian Point Study, Vol. I, NUREG/CR-1410, Washington, D.C.

USNRC (U.S. Nuclear Regulatory Commission), 1981. Technical Bases for Estimating Fission Product Behavior During LWR Accidents, NUREG-0772, Washington, D.C.

Vaughan, E. U., 1978. "Simple Model for Plugging of Ducts by Aerosol Deposits," Transactions of the American Nuclear Society, Vol. 28, p. 507.

Walker, B. C., C. R. Kirby, and R. J. Williams, 1978. Discretization and Integration of the Equation Governing Aerosol Behavior, SRD-R-98, Safety and Reliability Directorate, United Kingdom Atomic Energy Authority, London, England.

Witherspoon, M. W., and A. K. Postma, 1971. Leakage of Fission Products from Artificial Leaks in the Containment Systems Experiment, BNWL-1582, Battelle Northwest Laboratories, Richland, Wash.

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16. ABSTRACT (200 words or less)

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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