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EVALUATION OF SEVERE ACCIDENT RISKS: METHODOLOGY FOR THE CONTAINMENT, SOURCE TERM, CONSEQUENCE, AND RISK INTEGRATION ANALYSES

Volume 1

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Please forward the attached document to the PDR.

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ABSTRACT

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1. INTRODUCTION

The U. S. Nuclear Regulatory Commission (NRC) has recently completed a major study to provide a current characterization of severe accident risks from light water reactors (LWRs). The characterization was derived from the analysis of five nuclear plants. The report of that work, Severe Accident Risks: An Assessment of Five U. S. Nuclear Power Plants (hereafter referred to as NUREG-1150[1]) is based on extensive investigations by Sandia National Laboratories (SNL) and other NRC co ractors. Several series of reports document these analyses and their results in detail.

The risk assessments can generally be characterized as consisting of four analysis parts, an analysis integration step, and an uncertainty analysis:

- Systems analysis: the determination of the likel and nature of accidents that result in the onset of core damage;
- Accident progression and containment analysis: an investigation of the core damage process both in and outside the reactor vessel and the resultant impact on the containment;
- Source term analysis: an estimation of the radionuclide releases associated with the progression of the accident;
- Consequence analysis: the calculation of the offsite consequences in terms of health effects and financial loss;
- Risk integration: the combination of the output of the previous tasks into an overall expression of risk; and
- Uncertainty analysis: estimate of uncertainty in the risk calculation due to uncertainty in knowledge of important physical and chemical phenomena.

This report is the first of seven volumes of the NUREG/CR-4551 series that explain the supporting analysis for the last five items listed above, covering the progression of the accident once damage is initiated, through to an integrated estimate of overall risk and uncertainty in risk. This particular volume describes the methods used in these analyses while the remaining volumes focus on inputs and results for the particular plants and on inputs to the uncertainty analysis.

This volume contains all of the information needed to understand why particular methods were selected or developed, how they were employed and any special characteristics of the results. There is very little repetition of this discussion of methods in the other volumes which report results of the analyses; this report should be the reference for any questions concerning how the study was performed.

1.1 Background and Objectives

The overall objectives of the NUREG-1150 program are given below.

1. Prepare a current assessment of the risks to the public from severe accidents at five nuclear power plants, which will:

   - Provide a "snapshot" of the risks reflecting plant design and operational characteristics, related failure data, and severe accident phenomenological information extant in 1988;

   - Update the estimates of the NRC's 1975 risk assessment, the Reactor Safety Study;[2]

   - Include quantitative estimates of risk uncertainty, in response to the principal criticism of the Reactor Safety Study; and

   - Identify plant-specific risk vulnerabilities, in the context of the NRC's individual plant examination process;

2. Summarize the perspectives gained in performing these risk analyses, with respect to:

   - Issues significant to severe accident frequencies, consequences, and risks;

   - Uncertainties for which the risk is significant and which may merit further research; and

   - Comparisons with NRC's safety goals;

   - The potential benefits of a severe accident management program in reducing risk; and

   - The potential benefit of other plant modifications in reducing risk.

3. Provide a set of methods for the prioritization of potential safety issues and related research.

These ambitious goals required special consideration in selecting and developing analysis methods. This report describes those special considerations and the solutions implemented in the analyses supporting NUREG-1150.

1.2 Changes Since the Draft Reports

NUREG-1150 and its supporting documents were issued as drafts in early 1987. The draft series, NUREG/CR-4551, was issued in five volumes in the same time frame.[3-7] In addition to the solicitation of public comments on these draft reports, the NRC and its contractors initiated other review processes, both internal and external. As a result of all of these reviews, a number of changes to the methods were incorporated, updated information was included, and all analyses were almost completely redone.

The specifics of the responses to the comments are reported under separate cover.[8] For perspective, however, it is useful to have an overview of the major changes. This discussion focuses on analyses that are the subject of this volume.

The most controversial aspects of the draft analyses were the use of expert opinion to determine the uncertainty in the various input parts of the analysis and the representation of the resulting uncertainty in the risk. However, there was general support of the need to more comprehensively include uncertainties, particularly for the phenomenological aspects of the probabilistic risk analysis (PRA), which include many highly uncertain models and inputs. In the final analyses, considerable effort was concentrated on developing an uncertainty approach that would satisfy many of the previous objections, be robust, and statistically support and represent the full range of uncertainty in the various components.

The method selected maintains a reliance on expert judgment for the largest uncertainties. There is ample evidence and precedent to support the use of expert opinion for highly uncertain processes where detailed investigations are impractical. The need to provide a "snapshot" in time of the current status of uncertainty mandates that expert opinion be used since resolution of many of the uncertainties is years away. There is, however, a substantial difference in the way that expert opinion was elicited and manipulated in this final analysis as compared to the draft. The elicitation process was formalized, bringing in many of the characteristics of successful and accepted applications of this technology in the past, as well as conforming to the current theory and practice of this technology.

In addition, the input was obtained with clear objectives for its use in uncertainty characterization, eliminating another problem in the draft in which many of the experts eschewed a statistical interpretation of the input. In addition to major improvements in the elicitation, the other major problem cited by reviewers was lack of complete representation by the lear safety community. This problem was also addressed directly by ming new groups with a broader representation and by attempting to maintain a balance of any opposing viewpoints on each review group. While this part of the process was subjective and limited due to personnel unavailability within the timeframes needed, every effort was made to have relatively complete representation for important issues. Another concern of a number of reviewers was the possibility that the results did not represent the most current information. Owing to the significant time involved in completing a program of this size, this was true for some areas of high uncertainty where additional steps have recently been taken to better understand the phenomena. This study has been updated to reflect the most recent information. The expert elicitations were conducted in the first part of 1988 and any new information up to mid-1988 was considered.

The accident progression and containment analysis was performed with the same basic methods in the draft, but all of the accident progression event trees were revised to: (a) reflect changes in information; (b) improve the tracking of dependencies, particularly steam and hydrogen balances; (c) ensure consistent levels of detail and consistent treatment of identical phenomena in all trees; (d) and correct any errors in the drafts.

The source term analyses for each plant were upgraded for reasons similar to those of the containment analysis. Changes to ensure consistency and to accommodate the new input for the uncertainty issues were the most important.

Finally, many of the comments on the draft reports dealt with quality assurance and review of the analyses and the computer codes. Additional review activities were carried out in the interim and a formal review and quality assurance process was implemented as all the analyses were done.

Since the draft reports there were othe , more subtle, changes which are discussed, where appropriate, througnout this report. The analyses presented here supercede the previous ones completely. This report contains a complete discussion of the methods used and no reference to the draft reports is needed other than for purposes of comparison.

1.3 Scope of the Analysis

Five plants were analyzed: Surry Unit 1, Peach Bottom Unit 2, Sequoyah Unit 1, Grand Gulf Unit 1, and Zion Unit 1. The first four plants were analyzed by the staff at SNL while the Zion analyses were completed by Brookhaven National Laboratory (BNL) and Idaho National Engineering Laboratory (INEL), both using the same methodology. The scope of the accident sequence evaluation is described in NUREG/CR-4550. Two of the plants, Sequoyah and Peach Bottom, included external events as accident initiators (earthquakes, fire, flood, etc.) while the other three studies were limited to internal events as initiators.

The methods reported here carry the analysis forward once core damage is initiated. The progression of the accident and the effect on the containment is studied up to a point where the threat of additional radionuclide releases is negligible. While all of the basic inputs and outputs are described in this series of reports, it should be recognized that there were many other documents and calculations specifically in support of this program. These other sources are referenced where appropriate, or summaries are provided as appendices.

The uncertainty analyses were important components of these studies. Detailed uncertainty analyses, representing uncertainties in phenomenology, were included in all parts of the analysis except for the offsite consequence evaluation. Resource limitations precluded the inclusion of consequence phenomenological uncertainties. However, stochastic uncertainties in weather data have been included in the consequence analyses.

1.4 <u>Organization of this Report</u>

This report is intended as the sole source for describing the tasks listed previously. An overview of the entire process is provided in the following section as a prelude to more detailed discussions of individual analysis parts. This overview establishes some of the basic nomenclature that is used throughout the study and illustrates the very important interfaces between successive tasks.

Section 3 briefly describes the interface of this analysis with the accident sequence frequency analysis. This is the transition from the methods report in the NUREG/CR-4550 series to the rest of the analysis which is reported in this series (NUREG/CR-4551). Section 4 describes the accident progression and containment analysis, Section 5 describes the source term estimation, and Section 6 describes the consequence analysis.

As described above, the uncertainty analyses were an important part of the project activities. A general discussion of uncertainty followed by a presentation of the specific methods developed for this program is described in Section 7. The assembly of all of the task results to calculate risk is described in Section 8, as are the specific procedures for calculating the uncertainty in risk. Risk reduction is discussed in Section 9. Finally, the steps taken to ensure that the methods input and results were accurate and correct are described in Section 10.

1.5 References

1. U.S. Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants, NUREG-1150, Washington, DC, December 1988.

2. U.S. Nuclear Regulatory Commission, Reactor Safety Study--An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants, WASH-1400 (NUREG-75/014), Washington, DC, 1975.

3. A. S. Benjamin et al., Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Surry Power Station, Unit 1, Sandia National Laboratories, Albuquerque, NM, NUREG/CR-4551, Vol. 1 (Draft), February, 1987.

4. A. S. Benjamin et al., Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Sequoyah Power Station, Unit 1, Sandia National Laboratories, Albuquerque, NM, NUREG/CR-4551, Vol. 2 (Draft) February, 1987.

5. C. N. Amos et al., Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Peach Bottom Power Station, Unit 2, Sandia National Laboratories, Albuquerque, NM, NUREG/CR-4551, Vol. 3 (Draft), February 1987.

6. C. N. Amos et al., Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Grand Gulf Power Station, Unit 2, Sandia National Laboratories, Albuquerque, NM, NUREG/CR-4551, Vol. 4 (Draft), April, 1987.

7. Author. Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Zion Unit 1, Brookhaven National Laboratory, NUREG/CR-4551, Vol. 4 (Draft), April 1987.

8. Responses to comments on draft 4551 NEED REFERENCE

2. OVERVIEW OF THE CALCULATION OF RISK

The methods used to estimate risk are described in detail in this report. This section is an overview of the entire process, providing a context for the different parts of the calculation and introducing the interrelationships between the parts. It should be recognized that this summary does not include the subtleties and complexities of the methods used to perform the various analysis parts and any conclusions concerning the adequacy or correctness of the methods should be based on a reading of the more detailed sections devoted to each part.

The analysis methods were selected or developed to satisfy some special objectives of this program, in addition to the objectives of a typical PRA. In particular, the following were important considerations in the selection of methods:

- The need to incorporate uncertainty in the calculations;
- Development of the accident progression analysis in more detail than in the past;
- Calculation of intermediate results and detailed breakdown of the risk results, with clear traceability throughout the computation;
- Computational tractability;
- Potential for calculating the impact of plant modifications.

The selection of the methods also benefited from experience obtained while conducting the analyses presented in the draft versions of the NUREG/CR-4551 reports.[1-4] Changes since the draft include improvements in the basic technology and, where possible, incorporation of the comments of the reviewers of those draft documents.

2.1 Objective of the Risk Calculations

The risk calculations performed in this study provide an estimate of the risk to the public from all "severe" accidents at a given nuclear power plant. The accidents considered are those initiated by a well-defined set of events (loss of electric power to the plant, a leak in the primary system piping, etc.). All possible accidents resulting from the initiating events are included and their probability of occurrence calculated. The consequence of the accident (number of early deaths, number of latent cancers, etc.) is also calculated for each accident. Risk is measured by the average consequence of all possible accidents, weighted by the probability of occurrence of the accident.

2.2 Overview of the Risk Analysis Parts

The analyses comprising the risk calculation are illustrated in Figure 2-1. Four principal analyses that are needed to support the risk calculation can

Figure 2-1. Risk Analysis Parts and Interfaces be defined: (1) systems analysis; (2) accident progression analysis; (3) source term analysis, and (4) consequence analysis. Each analysis traces all the accidents through a certain stage or aspect of the accident. For example, the systems analysis traces the accidents from the initiating events to the onset of core damage. The accident progression analysis traces the accidents from the onset of core damage to the release to the environment of radioactive nuclides. The source term analysis follows the radioactive nuclides for each accident from the onset of core damage to their release to the environment. The consequence analysis follows the radionuclides in the environment to calculate their impact on it.

The transfer of information between analysis parts is critical, thus three interfaces are also illustrated in the figure. Each distinct continuous line that can be followed from the left of the illustration to the box marked "Risk Calculation" corresponds to a distinct group of accidents with a particular set of characteristics in each analysis part. Each of the analysis parts produce results that are useful for understanding the plant's response to that stage or aspect of the accident, and each part also provides an ingredient necessary to the calculation of the overall risk.

Each of the analysis parts is supported by a variety of information sources and supporting analyses. An ideal study might use comprehensive mechanistic models to calculate the entire sequence of events leading to core damage, release of radionuclides and exposure to the public, for each possible accident. However, a large variety of accidents will be possible because there are a variety of initiating events and because "random" events occurring during the accident can change the progress of the accident. It is presently neither practical (too many possible accidents to follow) nor possible (mechanistic models do not exist for many parts of the process) to conduct such a study. The current PRAs therefore have relied on the development of a variety of simple models and calculational tools to fill in where integrated mechanistic calculations are not available. Some of the tools assemble results from several existing mechanistic calculations to yield a more comprehensive result. Some of the models provide simplified mechanistic models with as much of the detailed analysis as possible, but which are able to efficiently calculate results for the wide range of conditions needed to examine all possible accidents.

The systems analysis relies on the use of probabilistic evaluations of fault- and event-tree models to estimate the frequency of accidents that result in core damage. As there are many different accidents possible, in terms of the actual failures and the timing of events, there is a need to coalesce the end states of the systems analysis in order to pass the information to the accident progression analysis. The interface is accomplished through the definition of plant damage states (PDS) on Figure 2-1. A PDS is a collection of accident cut sets (a specific set of failures leading to core damage) that have in common characteristics that are important to a determination of the subsequent accident progression and containment response.

The accident progression analysis investigates the physical processes affecting the core after core damage is initiated. In addition, this part of the analysis tracks the impact of the core-damage progression on the containment. The principal tool used for delineating and characterizing the possible scenarios in this study is called the accident progression event tree. The event tree is a computational tool used to assemble a large variety of analysis results and data to yield a comprehensive result for each of many accidents. The event tree is particularly suited for the study of processes that are not understood completely, permitting the study of many alternative phenomenologies. The output of the accident progression event tree (APET) is a listing of numerous different outcomes of the continued accident progression. As illustrated in Figure 2-1, these outcomes are grouped into accident progression bins (APBs) that, analogous to PDS, allow the collection of outcomes into groups that are similar in terms of the characteristics that are important to the next stage of the analysis, in this case source term estimation Once the accident progression event tree is constructed, the probabilities of the paths through the APET are evaluated by a computational tool, EVNTRE. EVNTRE also groups them together into bins. The accidents that are grouped into a single bin are similar enough in terms of timing, energy, and other characteristics that a single source term estimate suffices for estimating the radiological impact of any of the individual paths within that bin.

The next step is the source term analysis. Once again a relatively simple model was developed to allow consideration of alternative inputs and the assembly of information from many sources. In this study, a different model was written for each of the individual plants. This model is termed XSOR in the figure to illustrate the basic premise. A plant-specific code has been developed for each of the plants, with the suffix SOR built into the code name. For example, SURSOR is the source term model for the Surry plant. The results of the source term analysis are release fractions for groups of chemically similar radionuclides for each accident progression bin. As with the previous analyses, a large number of results are calculated, too many for direct transfer to the next part. The interface in this case is accomplished through the calculation of source term groups. The large number of XSOR results are plotted in terms of their important parameters (e.g., immediate health threat potential, delayed or latent health threat potential, and timing and energy of the release) and groups are defined which represent a whole collection of the individual XSOR results.

The consequence analysis in this study is performed with the MACCS code (MELCOR Accident Consequence Code System).[5] This code has been developed as part of other research programs, and is a replacement for the CRAC2 code, which has previously been used to estimate consequences for nuclear plant risk assessments.[6] The MACCS calculations are performed for each of the source term outcomes defined by the source term groups. MACCS provides probabilities of consequence levels for a number of consequence measures, early fatalities being one example.

The final stage of the analysis is the assembly of the outputs into an expression of risk. The calculation of risk can be written in terms of the outputs of the individual analyses:

$$Risk_l = \Sigma_h \Sigma_i \Sigma_j \Sigma_k \; fIE_h \; P(IE_h \rightarrow PDS_i) \; P(PDS_i \rightarrow APB_j) \; P(APB_j \rightarrow STG_k) \; cSTG_{lk}$$

where

$Risk_l$ = expected value for consequence measure l (consequences/year)

fIE_h = frequency (yr^{-1}) for initiating event h,

$P(IE_h \rightarrow PDS_i)$ = probability that initiating event h will lead to PDSi

$P(PDS_i \rightarrow APB_j)$ = probability that PDSi will lead to accident progression bin j

$P(APB_j \rightarrow STG_k)$ = probability that accident progression bin j will lead to source term group k

$cSTG_{lk}$ = value of consequence measure l conditional on the occurrence of source term group k

Because of the large information handling requirements of all of these analysis parts, computer codes are used to manipulate the data. Figure 2-2 illustrates the computer codes used in the risk assembly process in this study. The purpose of each of these codes is illustrated and will be discussed in more detail in the sections that follow.

Each of these four analysis areas is described in more detail below. In addition, the features of the uncertainty analysis that are critical to this study are also described, followed by a presentation of the means used to calculate the risk profiles.

2.3 Systems Analysis

A detailed description of the systems analysis methods used in this study is given in a separate document.[7] That description will not be repeated here. Only those aspects of the systems analysis which are directly related to the subsequent analyses will be discussed in this document.

2.3.1 Input to the Systems Analysis

The first task of the systems analysis is to gather (a) information about the configurations of the systems required to mitigate normal and abnormal occurrences at the relevant plant, (b) information about the normal and emergency operation of these systems, (c) information about the dependencies among these systems (power, actuation, cooling, etc.), (d) and informatiion about previous failures of the components within these systems. The information typically comes from final safety

Figure 2-2. Schematic of Computer Code Usage in the Risk Analysis.

analysis reports (FSARs), engineering diagrams, piping and instrumentation diagrams (P&IDs), licensee event reports (LERs), plant data files, and one or more plant visits. Good communication between the utility and the analyst is essential for a complete understanding of the current operation of the systems.

Some of the above information is also used as input to the accident progression analysis. An example is the probability of power recovery. The probability of power being recovered is important not only to the systems analysis, but also to the containment and source term analysis due to the impact of the availability or unavailability of containment equipment. To ensure consistency of the entire analysis it is important that a consistent set of power recovery curves is used in all parts of the analyses.

2.3.2 Systems Analysis Models

Logic models are constructed using the system information collected and traditional PRA techniques. Event trees are constructed define the system response to specific accident initiating events [e.g., loss of offsite power (LOSP), loss of condenser vacuum, loss-of-coolant accident (LOCA), etc.]. Each event tree leads to several outcomes (sequences). Each sequence results in core damage, a safe shutdown, or a core vulnerable sequence (core cooling is successful, but something happens to the containment which jeopardizes the coolant injection.)

The top events in the event trees (usually systems or operator actions) are modeled using fault trees or simplified Boolean expressions. These models describe the combination of basic events (pump failure, valve failure, operator failure to actuate a required system, etc.) that are necessary to fail the system. Point estimates of the probability for each of the basic events are used to quantify the top event models using the SETS computer program. The representations of the systems relevant to the core damage and core vulnerable sequences are combined and quantified using the SETS program. The results are in point estimate sequence frequencies. Additional uncertainty analyses are performed using the TEMAC computer code.

2.3.4 Output from the Systems Analysis

The initial output from the SETs code is a list of cut sets (combination of basic events that result in core damage) for each sequence defined in the event tree. Each cut set has an associated point estimate of its frequency. The sum of the frequencies of all of the cut sets that define a sequence is the frequency of the sequence. The sum of the sequence frequencies is the total core damage frequency.

The TEMAC code calculates a distribution around the core damage frequency (either by sequence or for the plant) using the uncertainty information provided for the basic events. Various basic event importance measures are also calculated by TEMAC--importance in reducing risk, importance in increasing risk, and the importance to uncertainty.
It is often necessary to regroup the cut sets for the accident progression analysis. The definitions of the sequences that lead to core damage do not

always include the information necessary to proceed with analysis of the subsequent severe phase of the accident. For example containment sprays, which may not impact the accident up to the point of core damage, where the accident sequences are defined will affect the accident after core damage has begun. The cut sets are therefore grouped by the parameters important to the subsequent analysis, the accident progression analysis. The cut sets groups are called PDSs.

As illustrated in Figure 2-2, mechanistically the output of the TEMAC code which processes and quantifies the system cut sets and groups the accidents into PDSs, is the frequencies of the PDSs. As illustrated, the result of this process is the product of the first and second inputs to the risk calculation Σ_h fIE_h $P(\,IE_h \rightarrow PDS_i)$, which yields the probability of each PDS_i.

2.4 Accident Progression Analysis

The purpose of the accident progression analysis is to track the progression of the potential accident from the onset of core damage until it is assured that no additional release of radionuclides from the containment will occur. Thus the core damage process is studied in the reactor vessel, as the vessel is breached, and outside the vessel. At the same time the analysis tracks the impact of the entire progression on the containment, with particular focus on the threat to containment integrity posed by pressure loadings or other physical processes.

2.4.1 Interface of the Systems Analysis with the Accident Progression Analysis

Potentially many thousands, even millions, of distinct accidents can be identified as a result of the systems analysis. Many of these accidents are of low probability and are thus unimportant in the subsequent risk analyses. Nevertheless, a large number of accidents remain for additional consideration. The remaining accidents may or may not be found to be risk-significant in the subsequent analyses. The systems analysis often requires details for individual accidents that are not needed in the accident progression analysis. Thus, before continuing the risk analysis the important accidents are grouped according to properties required in subsequent analyses. This grouping is conducted as part of the systems analysis, although the definition of the grouping characteristics is determined as part of the accident progression analysis. The accident groups are called PDSs and serve as the interface between the systems analysis and the accident progression analysis. The PDSs are different from the accident sequences, discussed in the previous section, since the characteristics determining the grouping are based on the needs of the subsequent analyses, rather than a traditional grouping of accident characteristics.

2.4.2 Input to the Accident Progression Analysis

The requirements of an ideal accident progression analysis would be knowledge, probably in the form of the results of mechanistic calculations from validated codes, of the characteristics of all possible accidents resulting from each of the PDS. More than one accident would result from each PDS since, as indicated in Section 2.1, random events (hydrogen detonations, for example) occurring during the accidents can alter the course of the accidents. Given the probability of the PDS and the frequencies of the random events one could determine the outcomes and frequencies of all possible accidents.

Knowledge of the characteristics of all possible accidents resulting from each PDS is clearly not available with current technology. A large number of mechanistic codes is available that can predict some aspects of the accident progression. For example, MELPROG and CONTAIN can be used to track in-vessel and containment details, respectively, of very explicit scenarios. Less detailed, but more comprehensive codes, such as as MAAP, the STCP and, more recently, MELCOR have been developed to predict generalized characteristics of more aspects of the accident in an integrated fashion. While these codes are very useful for developing detailed understanding of accident phenomena and how the different phenomena interact, they do not meet the constraints imposed by a PRA: the ability to analyze a very wide range of scenarios with diverse boundary conditions in a timely and cost efficient manner. In addition, the number of code calculations necessary to investigate uncertainty and sensitivity to inputs, models, and assumptions would be prohibitively expensive. Further, these codes have not been fully validated against experiments. Thus codes developed by different groups (for example, national laboratories and industry contractors) frequently include contradictory models and give different results for given sets of accident boundary conditions. Finally, these codes also do not contain models of all phenomena that may determine the progression of the accident. For example, none of these codes mechanistically models the response of the containment structure to dynamic pressure loading.

The information and/or models that were available with which to conduct the accident progression analysis for this study consists of the diverse body of research results from about ten years of severe accident research within the reactor safety community. This includes a large variety of severe accident code calculations, other mechanistic analyses and experimental results. Much of the information represents basic understanding of some important phenomena. Because of the expense of developing and running large integrated codes, less information is in the form of integrated accident progression analyses. That which is available is usually confined to analyses of a few types of accident sequences. All existing codes were recognized to have some limitations in their ability to mechanistically model severe accidents.

Many new calculations were conducted specifically for this study. In particular, many new CONTAIN calculations were conducted to assess pressure loading on the containment and sensitivity of the pressure loading calculations to various phenomenological assumptions. Most of the new calculations are described in other sections of this series of reports. Volume 8 (Supporting Calculations) contains a comprehensive listing and description of the new supporting calculations. For the most part, the new calculations were intended to fill the largest gaps in our knowledge of accident progression for the most important accidents.

2.4.3 Accident Progression Analysis Models

The accident progression analyses were conducted using plant-specific event trees, called APETs. These APETs are themselves models. They describe the accident in a very general and flexible way. They consist of a series of questions about physical phenomena affecting the progression of the accident. A typical question would be "What is the pressure rise in the containment at reactor vessel breach?" A complete listing of the questions that make up the event tree for each power plant can be found in the relevant NUREG/CR-4551 Volume (Evaluation of Severe Accident Risks, Section 2.3) for that plant. Typically, the event trees for each plant consist of about 100 questions. The questions can have multiple outcomes or branches.

The APETs are general enough to efficiently calculate the impact of changes in phenomenological models on the accident progression, in order to study uncertainties. This generality adds complexity to the analysis since, with the ability to consider different models, some paths through the tree, which would be forbidden for a specific model, must be included when a variety of models is considered. The multiplicity of possible accident progression results caused by the consideration of multiple models for some of the accident phenomena is amplified at each additional stage of the accident progression, since in addition to creating more possible outcomes, a wider range in boundary conditions at the subsequent events is made possible.

Because of the flexibility and generality of the APETs, basic principles, such as hydrogen mass conservation, steam mass conservation, etc., were incorporated into the event trees in order to automatically eliminate pathways for which the principles are violated. This is accomplished since parameters, such as hydrogen concentrations in various compartments, are passed along in the tree as each accident pathway is evaluated. At some questions in the tree, the parameters can be manipulated using computer subroutines. The branch taken in each question can depend on the values of passed parameters. The consistency of phenomenological treatment throughout each accident is also enhanced by allowing questions to depend on the branches or parameters taken in previous questions.

Generally, phenomenological models are not directly substituted into the event trees (in the form of subroutines) at each question. Rather, the results of the model calculations are entered into the trees through the assigned branching probabilities, the dependencies of the questions on previous questions (the "case structure") and/or tables of values that are used to determine parameters passed or manipulated by the event tree. Some questions in the trees, such as those concerning the operability of equipment and availability of power, were assigned probability distributions derived from data, analogous to the process in the systems analysis. Timing of key events is identified through a review of available code calculations and other relevant studies in the literature. The process of assigning values to the branching probabilities, creating the case structure, writing the user functions and supplying parameter values or tables is referred to as "quantification" of the tree.

The APETs used in this study represent a significant advance over previous methods used for PRA in this portion of the analysis. Typically, PRAs have employed a logic model called a containment event tree to track, at a high level, the key phenomena in containment. The number of questions in the event tree has been small (6 to 10, typically), the questions usually had two branches with yes/no outcomes and did not depend on previous questions taken. Parameters were not passed or manipulated.

Figure 2-3 illustrates schematically the APETs used in this study. The first section of the tree (about 20% of the total number of questions) is used to automatically define the input conditions associated with the individual PDS. Thus, if one of the characteristics of a PDS is the pressure in the vessel at the onset of core damage, a question will be included to set the initial condition according to that variable. The next part of the tree is then devoted to determining whether or not the accident is terminated before failure of the reactor vessel. Questions pertinent to the recovery of cooling and coolability of the core are asked in this part of the tree. The next section of the tree continues the examination of the accident progression in the reactor vessel. As illustrated in Figure 2-3, there are two chief areas of investigation for this part of the analysis: in-vessel phenomena which determine the radionuclide release characteristics and events that impact the potential for containment loads. The example illustrated shows the phenomena associated with the release of hydrogen during the in-vessel process, and the resultant escape of that hydrogen into the containment.

The next stage illustrated continues the examination of the accident during, and immediately after, vessel breach. This includes the continued core meltdown in the vessel and the simultaneous loading and response of the containment. A good example would be an examination of the coolability of the debris once out of the vessel, followed by questions concerning the loading of the containment as a result of core-concrete interactions.

The final stage of the tree illustrated is related to the final status of containment. Long-term overpressurization, threats from combustion events, and similar questions are asked concerning this stage of the accident. For convenience, some questions are included which summarize the status of the containment at specific times during the accident.

This explanation has delineated the general flow of the APET. What is not immediately apparent in this summary is the degree to which dependencies can be accounted for. An example of the dependency treatment are the questions that relate to hydrogen combustion. The outcomes of the event tree questions that ask whether hydrogen deflagration occurs sometime after

Figure 2-3.

vessel breach, and the resulting pressure load from the burn are highly dependent on previous questions. The individual values for the probability of ignition and the pressure rise are dependent on:

- Previous hydrogen burn questions (the amount consumed in each previous burn is tracked, and the concentration at the later time is calculated consistent with all previous hydrogen events);

- Questions concerning the steam loading to determine whether the atmosphere is steam inert; and

- Questions concerning the availability of power, which influences the probability of ignition.

In turn, these questions all have further dependencies on each other and on other questions. For example, the steam loading questions are dependent on the power and equipment availability, since heat removal would impact the steam concentration. Section 4 of this report includes more explicit examples of this process.

2.4.4 Output from the Accident Progression Analysis

Once an APET, with its list of questions, their branches and their case structure, its subroutines and its parameter tables, has been constructed by an analyst, it is evaluated using the computer code EVNTRE. EVNTRE can automatically track the different kinds of dependencies associated with the accident progression.[8] This code was also built with specific capabilities for analyzing and investigating the tree as it is being built, allowing close scrutiny of the development of a complex model. For each PDS, EVNTRE evaluates the outcomes of all subsequent accidents predicted by the APET and their probabilities.

EVNTRE groups paths through the tree into bins. PSEVNT is a "rebinner" that further groups the initial set of bins produced by EVNTRE. Groupings can be chosen which clearly illustrate the importance of some aspect or other of accident phenomenology, system performance or operator performance as long as that aspect is a distinct part of the APET. To meet the needs of the subsequent source term analysis the results are grouped into "accident progression bins," which are described in the next section. As illustrated in Figure 2-2, the result of this process is the third input to the risk calculation, $P(PDS_i \rightarrow ACP_j)$ the conditional probability of accident progression bin j given PDS_i.

2.5 Source Term Analysis (Gary Boyd)

The goal of the third part of the risk analysis is the estimation of the radionuclide release and the conditions of the release (timing and energy) so that consequences may be estimated in the final step of the analysis. As described above, the interface between this part and the previous part (the accident progression bin) is defined to efficiently transfer the important information, maintaining a manageable set of calculations. The conditions defined by a specific accident progression bin are similar enough that a single source term estimation is possible for all accident progression paths that are in that bin.

2.5.1 Interface of the Accident Progression Analysis with the Source Term Analysis

It is fairly easy to see that the number of possible paths through the APET is astronomical, since there are over 100 questions or events, most of which have more than two outcomes. It should also be recognized that although there are many paths through the tree, the probabilistic evaluation reduces the problem. Not all paths (or accidents) have any appreciable probability of occurrence. Nevertheless, the number of important paths is very large, typically over a thousand. Thus, in a manner similar to the grouping of important accidents from the systems analysis into PDS, the important accidents that are identified in the accident progression analysis are grouped into "accident progression bins." The accident progression bins are defined in terms of characteristics which are important to the subsequent analysis, the source term analysis. The number of accident progression bins that could be defined by all combinations of the characteristics needed for the source term analysis can be quite large (hundreds). However, typically only 20 or so are of importance because some combinations are physically precluded and other combinations are of vanishing probability.

The bins were defined through interactions between the accident progression analysts and the source term analysts. Characteristics of the bins include timing of release events, size and location of containment failure, availability of equipment and processes which scrub radionuclide and other similar features of the physical progression of the accident, and the impact on containment. Therefore, the bins are blind to many of the individual questions in the tree as they focus on the ultimate outcomes, and through the use of these bins, the paths through the tree were greatly reduced in terms of the number of unique outcomes.

Once again, in a risk assessment project it is not practical to analyze every scenario, in this case every accident progression bin, with a detailed code calculation. The method selected for this part of the analysis had to be efficient enough to calculate source terms for many (thousands) accident progression bins, and had to be flexible enough to allow direct incorporation of phenomenological uncertainties. In this case a simple parametric algorithm was developed that allows the calculation of source terms over a wide range of conditions.

A different model has been developed for each plant, although the basic algorithm is largely the same, with the code being customized to reflect specific plant conditions and any special feature that could impact the source term. (As noted in Figure 2-1, the codes that manipulate these algorithms are called XSOR, where the X refers to a plant-specific abbreviation, for example the code for Peach Bottom is PBSOR.) Initially, these models were developed through detailed examination of the results of Source Term Code Package (STCP) analyses of selected accidents done specifically for this program.[9-13] However, accomplishment of the second objective, treatment of the full range of source term uncertainty, led to changes to these models because the uncertainties reflected ranges of values outside the STCP and the uncertainties include phenomena not yet included in the STCP.

Figure 2-4 illustrates the basic processes considered in the XSOR algorithm. The release is broken up into constituent parts in order to allow changes to each part to reflect different boundary conditions, or to allow the input of a range of uncertainty within individual processes. Two basic release paths are defined and modeled: releases from the core in-vessel and releases ex-vessel, most notably during core-concrete interactions.

The in-vessel process starts during the core damage and continues until the bottom of the vessel is breached. The algorithm starts with a nominal list of radionuclide inventory at the time of the accident. Then the fraction of each radionuclide that is released from the core to the in-vessel environment is estimated. To allow a manageable calculation, the radionuclides are treated in terms of radionuclide groups that have similar properties, the same nine groups that are defined in the STCP. The model allows both for uncertainty in this release fraction, as well as for the effects of important boundary conditions, such as timing or temperature history. The next step in XSOR models is calculating the fraction of this in-vessel release that is subsequently released from the vessel. Once again this is affected by uncertainties in the basic processes as well as by the boundary conditions. The model includes a single parameter to account for deposition in the vessel. The algorithm also distinguishes the impact of the boundary conditions, for example, the possibility of a high-pressure in the vessel with no leakage as opposed to a case of low pressure with a large LOCA leakage rate. Other complexities are also involved. For example, as shown in the figure, there is the possibility of an additional release at vessel breach due to high pressure ejection of the core material.

Once released from the vessel, the concern is how much of the inventory gets released from the containment. The XSOR code therefore accounts for in-containment processes that effect the release, such as deposition or decontamination due to the operation of containment sprays. Finally, the fraction of the original release that escapes from containment is calculated. This is dependent on all of the previous processes as well as the timing and mode of any containment failure or leakage.

As described above, the boundary conditions (containment status, pressures, etc.) are accounted for in each of the sub-models where appropriate. The information concerning these boundary conditions is passed from the previous part through the accident progression bins.

The other basic release path in the algorithm is associated with the core-concrete interaction releases. As with the other processes, the model allows for the input of uncertainty information concerning the basic releases for each of the radionuclide groups. The impact of other containment conditions such as the availability of overlying water or the

Figure 2-4.

operability of sprays is then incorporated. Finally, the timing and mode of containment failure or leakage is considered in order to calculate a release from containment to the environment.

The algorithm also includes other complexities, two of which are illustrated in Figure 2-4: late revolatilization from the vessel, and late release of iodine from water pools. Simple equations are used to track these secondary sources of radionuclides that were removed in earlier processes. As with the other parts of the algorithm, the uncertainty in the amount of releases associated with these processes can be directly input to the XSOR models.

The actual SOR algorithms are somewhat more detailed than illustrated in Figure 2-4, because other complexities and plant-specific aspects are built into each model. However, it should be fairly obvious from this description that these codes are not too detailed or at all mechanistic. Simple mathematical expressions are used to track the various portions of the release and the processes adding to or subtracting from the release.

Once the basic algorithm was defined, it was necessary to supply basic parameters and release fractions, analogous to the quantification of the APET in the previous part. Originally (in the draft version of this study) these inputs were derived from STCP results that were used to obtain the information on the parameter level in the XSOR models. This is still the case for some of the parameters in the model, but the increased emphasis on the uncertainty analysis aspect of this part has lessened the role of STCP information in favor of inputs from the experts, as discussed in Section 2.5. It was the goal of this process to define uncertainty issues for all parameters that could significantly affect the consequences. This set of issues was created through review of the previous work on these plants, judgment of the analysts and judgment of the expert panels. For parameters that were not considered either particularly important or that are not highly uncertain, the XSOR model uses a parameter derived from STCP runs, adjusted as needed for the boundary conditions associated with the accident progression bins.

The source term calculations correspond one-to-one with the accident progression bins. This number of calculations is too great for the next step in the part, the consequence analysis. In this case the interface was defined to reduce the hundreds or thousands of source terms calculated to a more manageable number, on the order of 20. The clusters were determined by another post-processor code (see Figure 2-2) called PARTITION. The source term estimates include a number of variables, including timing, energy, and release fractions for nine groups of radionuclides. In order to select representative source terms, the number of variables had to be reduced. This was done by expressing the release in terms of dose equivalents of ^{131}I for early exposure effects and ^{137}Cs for chronic effects. The other important aspect for defining the impact of a release is the timing. Thus, the releases can be characterized in terms of two health measures, one for early effects and the other for chronic effects, over distinct time periods. Reduced in this fashion, the results can be plotted and partitioned such that representative source terms (clusters) are defined, with each cluster being sufficient for representing many of the individual source terms produced by the calculation.

The result of this process is the third input to the risk calculation, $P(STC)_k/APB_j$, the conditional probability of each source term cluster k, given each accident progression bin j. The actual release fractions for each of the radionuclide groups are also provided for each of the clusters.

2.6 Consequence Analysis

The final step of the analysis process, the consequence estimation, was performed using the MACCS code. This code is an improvement over the code previously used for this part, CRAC2. Although a relatively recent development, the MACCS code has been verified and benchmarked against the previous codes used for this part of the analysis. This part of the process was a straightforward application of the code for each of the source term clusters. Some development work was done to more effectively use the code within this program. A method was developed to break down the source terms for each cluster to treat evacuation timing and participation in evacuation. A means was also developed for calculating and displaying the effect of weather variability across the consequences for each of the source term clusters.

The MACCS code and its use are described in a separate report.[5] The code requires as input:

- A source term--release fractions for radionuclide groups, along with timing of release as well as sensible heat associated with the release;
- The inventory at reactor scram of all isotopes important to offsite consequences;
- The population distribution around the reactor site;
- Weather, land-use, and economic data for the region around the reactor site; and
- Emergency response parameters and assumptions (evacuation speed, non-participation, etc.)

Given those inputs, MACCS predicts the following:

- The downwind transport, dispersion, and deposition of the radioactive materials released from the failed containment;
- The radiation doses received by the exposed populations via direct (cloudshine, inhalation, groundshine) and indirect (ingestion, inhalation) pathways;
- The mitigation of those doses by emergency response actions (evacuation, sheltering, and relocation of people), interdiction of milk and crops, and decontamination or interdiction of land and buildings;

- The early fatalities and early injuries expected to occur within one year of the accident, and the latent cancer fatalities expected to occur over the lifetime of the exposed individuals;

- The total population dose received by the people living within some distance (50 miles) of the plant and the early fatality risk to persons living near the plant (within one mile); and

- The offsite costs of emergency response actions, and of the interdiction and decontamination of land, buildings, milk and crops.

By performing calculations for combinations of representative sets of source terms, weather sequences, and exposed populations, statistical distributions of consequence measures are developed that depict the range and probability of consequences. For this study, the uncertainties from the previous three parts were propagated through the consequence analysis, but only the stochastic uncertainty due to weather was considered in the consequence analysis. Therefore, although there are uncertainty parameters associated with the consequence measures, those uncertainties do not include the substantial uncertainty in the calculation of the consequences. For example, the dose conversions to health effects are set to nominal values that have been arrived at through a process involving experts in the field, but these factors are not varied over the range indicated by the uncertainty in these parameters. (If desired, the uncertainties associated with offsite consequence estimates can be developed by a variation of input parameter values using structured Monte Carlo sampling techniques, but such sensitivity studies were not performed as part of this study due to resource limitations.)

Through the use of the MACCS code, the final part of the risk calculation was developed: C_i/STC_k, the mean consequence (for measure 1) given the source term cluster k.

2.7 Characterization of Uncertainties

Although there have been significant advances in all areas of risk-assessment technology, there remains significant uncertainty in each of the analysis parts. A significant fraction of the effort in this program was devoted to the investigation of these uncertainties and the calculation of the uncertainty in the result of each part area as well as the uncertainty in overall risk. The appropriate means by which to characterize this uncertainty remains a topic of substantial debate within the technical community, and the methods chosen for this study were developed through review of the current technology, and direct response to comments made on the uncertainty treatment in the draft versions of these reports. The most important results of the analyses reported in this document are engineering and scientific insights that become evident after the completion and integration of each of the steps in the program and thorough review of the results. However, the significance of many of these insights can often be better understood within a quantitative framework. It is therefore essential that a clear presentation be made of the elements considered to be uncertain, and of the potential effects of these uncertainties on the results. The formulation of the uncertainty presentation for the results therefore had the following objectives:

- To provide decision-makers with engineering- and/or scientifically-based information that allows them to understand the analysts' treatment of important issues and the impact on the analysis of the range of viewpoints that experts in the field hold for these issues;

- To develop a quantitative estimate of uncertainty that reflects a credible and realistic range in which the analysts have a reasonable confidence that the correct answer lies;

- To identify as completely as possible the key sensitivities and sources of uncertainty for each portion of the analysis, including those that have the most impact on the calculated risk measures; and

- To evaluate the quantitative impact on the risk measures of the uncertainty in each part of the analysis and the combinations of the uncertainties and sensitivities for the different portions of the analysis.

There are many ways to include uncertainty considerations in analyses. In the past and in particular situations conservative analyses were performed to limit the impact of any uncertainties. There are difficulties in applying this method since it is difficult to always prejudge what is conservative when uncertainties are interdependent and multiplied through an analysis. The alternative is to obtain a best-estimate analysis that includes uncertainty parameters. Detailed codes that follow the entire accident process can also be used to investigate uncertainty. However, this approach suffers from the limitation that the codes have many inputs set to best estimate values, not all of which are easily changed. The array of studies needed to support an assessment of the impact of all combinations of uncertainties is also impractical in most cases. In some cases there are also no codes for particular phenomena. Broader views of uncertainty can be examined by comparing the results of different codes which model the same phenomena. Once again this is a time-consuming practice and still sheds little light on which alternative model is correct. For the scope of analysis in a PRA, the only practical solution is the use of expert judgment. This allows a more complete spectrum of uncertainty to be included in a considerably smaller expenditure of resources. This also allows a current measurement of the uncertainty. Most of the experts would prefer to go off and investigate an uncertain problem to arrive at a model and solution, but in order to set the priorities of which problems are most deserving of this analysis (in terms of risk relevance) a "snapshot" in time of current understanding is required. Expert judgment, derived from a formal elicitation process, therefore played a large role in this study.

The first step of the uncertainty assessment was to define the scope of the analysis. In this study the uncertainties in the systems analysis include the uncertainties arising from incomplete or inconsistent data pertaining primarily to equipment-failure rates. In addition, there are numerous assumptions made in the systems analysis, some of which pertain to the lack of data or verified models for important phenomena, a good example being the behavior of the reactor coolant pump seals under loss of cooling conditions. In the containment analysis the largest uncertainties are associated with phenomena that are poorly understood. It was the intent of this program to directly account for these modeling uncertainties. The containment analysis also has other uncertainties including those due to variability in materials properties or failure rates of equipment. The scope of this project was intended to include all of the uncertainties in the containment analysis that have a significant impact on the risk results. The source term uncertainties are similar, in that there are major uncertainties in the modeling of the processes as well as statistical variability in boundary conditions that impact the estimated release. Once again the scope was selected to include all of the uncertainties that could change the risk results, with a focus on the modeling uncertainties since past studies have shown those to be the most significant. Finally, as mentioned above, the consequence uncertainty portion of the analysis was limited to the variability in the weather, since resource limitations precluded detailed investigation of all other uncertainties in this stage of the analysis.

Once the scope was defined, the next stage was the definition of the specific uncertainties. This process was based on a review of the current body of knowledge to identify the uncertainties found most significant in other studies and in the opinions of experts in the field. This process was initially done in the analysis reported in the draft versions of these reports. The results of the draft analyses, and the internal and external reviews helped to further identify the truly uncertain and important issues. In addition, more literature reviews and the advice of the outside expert panels served to further develop the uncertainty list. However, there are numerous uncertainties associated with all aspects of the analysis, particularly when considered on a low level, i.e., at the basic physics level. With the level of detail of the basic analysis tool in a PRA it is not possible to directly consider these detailed uncertainties. For practical representation of uncertainties in risk it was necessary to both reduce the number of uncertainties and to increase the level of the uncertainty characterization to a level consistent with the models. Through the use of sensitivity studies, expert judgment (including the external panel of experts) and a process of coalescing uncertainties, the large list of basic uncertainties was culled to a smaller list of higher-order uncertainties. For example, all of the uncertainties in the basic physics of the vessel breach process were collapsed into an "issue" concerning the pressure rise at vessel breach. Issues were defined in each of the analysis areas included in the uncertainty scope: systems analysis, accident progression and containment analysis and source term analysis. Table 2-1 lists the uncertainty issues for the accident progression and containment analysis, and for the source term analysis.

Once the uncertainties were identified, it was necessary to develop uncertainty ranges. The experience in the draft version of this study clearly identified some uncertainties as being so great that the experts in the field hold nearly polarized opinions on the possible outcomes of the phenomena. In order to capture the range of uncertainty for these uncertainties, it was determined that expert input was needed from a spectrum of experts representing the range of opinion on each important issue. This process was implemented in the draft analyses under some severe constraints. Comments on the draft analysis supported the use of expert elicitation as a valid method, but were highly critical of the methods used in the draft, as well as the incomplete representation of the full reactor safety community. In response to that criticism, a considerably more formalized method was developed and implemented for the results presented in the final reports, and the expert representation was changed to attempt to include a broader spectrum of viewpoints within the community. The details of this greatly improved process are provided in Section 7 of this report.

Practical limitations precluded outside expert elicitation of all uncertainties. Sensitivity analysis was used to determine which other inputs were most important and expert elicitation was again performed, but within the project team. Although the direct representation of differing viewpoints may not be as great for these less important uncertainties, the participants did use available resources on the subjects as input to their own elicitations, and the full range of opinion should generally be represented. Finally, the reliability data assigned to the events in the fault trees and event trees include uncertainty distributions derived from the original data source, or from judgment in the case of events not particularly well supported by data.

Many methods of uncertainty calculation were considered in the development of the methods for this program, and a Monte Carlo approach was selected as being suitable with the form of the input and compatible with the overall method of calculation of risk. The Monte Carlo method produces results that can be analyzed with a variety of techniques, e.g., regression analysis, and it allows consideration of essentially any variable that is part of the input or output of the calculation. This sampling-based technique also allows for consideration of uncertainties with wide ranges, as well as correlation between uncertain variables.

With a problem of this magnitude, Monte Carlo sampling always poses a resource limitation threat. This was limited in this case through basic properties of the models--the use of relatively fast running models for each part of the analysis with well-defined interfaces. In addition, the Monte Carlo sampling was performed with a very efficient sampling technique: latin hypercube sampling (LHS). LHS has proven an effective technique when compared to other, more costly, methods. The key to the LHS method is creation of a sampling scheme that is constrained or stratified

Table 2-1

Uncertainty Issues For External Expert Elicitation: Accident Progression and Containment Analysis and Source Term Analysis

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Structural Issues

Static Containment Failure Pressure and Mode: All Plants
Reactor Building Bypass Probability: Peach Bottom
Ice Condenser Failure Due to Combustible Gas Detonation: Sequoyah
Drywell and Wetwell Failure Due to Combustible Gas Detonation: Grand Gulf
Reactor Pedestal Failure Due to Erosion by Core Concrete Interactions: Grand Gulf

In-Vessel Accident Progression Issues

Temperature-Induced Failure of the Hot Leg: PWRs
Temperature-Induced Steam Generator Tube Rupture: PWRs
In-Vessel Hydrogen Production: BWRs
In-Vessel Hydrogen Production: PWRs
Temperature-Induced Bottom Head Failure: BWRs
Temperature-Induced Bottom Head Failure: PWRs

Containment Loading Issues

Loads Due to Combustion Before Vessel Breach: Grand Gulf and Sequoyah
Loads Due to Combustion in the Reactor Building: BWRs
Loads at Vessel Breach: All Plants

Molten Core-Containment Issues

Drywell Meltthrough: Peach Bottom
Pedestal Failure Timing: LaSalle
Mark III Containment Failure Via Pedestal Failure: Grand Gulf

Source Term Issues

In-Vessel Fission Product Release and Retention: All Plants
Ice Condenser Decontamination Factor: Sequoyah
Revolatilization From the Vessel and Reactor Coolant System, Early and Late: All Plants
Core-Concrete Interaction (CCI) Releases: All Plants
Release of CCI Species From Containment, Aerosol Agglomeration: All Plants
Late Sources of Iodine: Grand Gulf
Reactor Building Decontamination Factor: Peach Bottom
Releases as a Result of High Pressure Ejection/Direct Heating: All Plants

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by the input information. The LHS method is described in more detail in Section 7 of this report as well as in separate reports.[14,15]

Figure 2-5 illustrates the incorporation of uncertainty into the risk calculation. The large list of basic uncertainties was reduced to the smaller list of higher order uncertainties--issues. These issues were then presented to the experts to obtain their input. Although the list of issues was reduced to on the order of 40 per plant, the experts were encouraged to create as many subcases of the issues as needed to accurately characterize the uncertainties under the different boundary conditions that might be expected. As illustrated in the figure, the next step was aggregation of the experts' inputs, each given equal weight. The resulting aggregate distribution formed the basis of the sampling for each issue. The actual sample creation was done by LHS. It should be noted that the figure does not represent all of the information manipulation. Many of the issues were found not to be independent, and correlations were introduced that then became part of the sampling scheme. The output of this process was an LHS sample consisting of approximately 200 sample members. Each sample member is defined by a specific set of distinct outcomes for each uncertainty issue, as sampled from the aggregate distributions. As such, each sample member defined a set of input variables that can be used to calculate a complete risk result.

2.8 Calculation of Risk

The inputs to the risk calculation have been defined in the previous sections. As illustrated in Figure 2-2, a number of codes were used to generate the necessary output, and these outputs are then processed by an additional code, PARAMIS, to calculate a risk result. PARAMIS is actually a matrix manipulation code. As illustrated in Figure 2-6, the elements of the risk equation can be represented in a vector/matrix format. The frequency of a PDS [F(PDS)] is a vector of frequencies for each individual PDS_i. There are n of these vectors, one for each sample member. As illustrated in Figure 2-5, each sample member represents a unique set of values for each uncertainty issue. For this study, there are approximately 20 PDSs with appreciable frequency (i = 20) and approximately 200 sample members (n = 200).

The plant damage vector is multiplied times the accident progression tree output matrix [P(APB)/PDS]. This i by j matrix represents the conditional probability of each of j accident progression bins, for each of i PDS. For this study, there are approximately 150 accident progression bins that contribute to the risk result (j = 150). There are also n of these matrices, one for each sample member. The result of this calculation is multiplied by a third matrix that represents the outcome of the source term analysis [P(STC)/APB]. This k by j matrix represents the probability of a source term cluster k, given each accident progression bin. There are approxi- mately 20 source term clusters (k = 20).

The final input in the equation is a vector representing the consequences for each of the source term clusters [P(C)/STC]. There are 1 of these vectors, one for each consequence measure. For this study, 9 consequence

Figure 2-5.

measures were calculated (1 = 9). It should be recognized that, because consequence uncertainty was not included in the list of issues and the LHS sampling, only one set of 1 consequence vectors is required; the last term illustrated in Figure 2-6 is the same for each and every sample member.

This matrix manipulation was done with the PARAMIS code. The risk calculation is a fairly straightforward process, but it is obvious that the number of numerical manipulations is rather great. After calculation of all of the risk measures for each sample member, the set of n (in this case approximately 200) results from a distribution in risk space that represents the uncertainty associated with the issues. As noted above, the Monte Carlo-based techniques are amenable to statistical examinations to provide insights concerning the result. Through examination of the results (with the SAS code package??) with statistical techniques such as regression analysis, the relative importance of the issues to overall uncertainty can be determined. Other measures of risk importance can also be calculated. The individual sample members can also be examined. For example, if the final distribution contains some results that are quite different from all the others (say 5 sample members an order of magnitude higher in consequences than any other sample members) the individual five sample members can be examined as separate complete risk analyses to determine the important effects causing the overall result.

One of the key developments in this program is the automation of the risk assembly process. The most significant advantage of this methods package is the ability to recalculate an entire risk result very efficiently, even given major changes in the constituent analyses. The manipulation of these models in sensitivity studies allows efficient, focused examination of particular issues, and significant ability for examining changes in the plants or in the analysis.

The objectives of the program included calculation and conclusions concerning the risk results; intermediate results are also quite important. Each of the analysis steps included intermediate outputs. These outputs are also manipulated to maximize the efficiency of consideration of intermediate results. The nomenclature and representation of the results described in this section are used consistently throughout the documentation of both the methods and the results for a specific plant. The same intermediate results will be illustrated for each facility and the terminology used to describe those results is consistent with that developed here.

Note: This section will need a description of the high-level results that form the actual presentation of results in NUREG 1150. Currently this section treats the actual analysis steps and the interfaces as they were used in the calculation. The translation of these results to the format used in 1150 (super damage states, reduced accident progression bins, etc.) will need to be explained so that the reader can follow the connection between the 1150 document and the support documents.

Figure 2-6

2.9 Review and Quality Assurance

There were significant comments concerning the quality assurance measures of the study published in the draft report, and additional review and quality assurance steps were taken for this analysis. Each piece of analysis has been reviewed internally by cross review of project analysts, and separate quality assurance review groups were formed for each principal part area. These reviews covered the input, the manipulation of the information, and the result. The codes have been formally reviewed also. Section 9 of this report discusses the quality assurance measures that were implemented for this program. In addition, Reference 16 describes the specific activities and findings of the review group.

The methods, inputs, and results of this program were also developed with specific consideration of the public review comments on the draft. Although not all comments were directly incorporated (indeed many of the comments themselves conflicted in their advice) each comment was seriously considered and a response was generated, as reported in Reference 16.

2.10 References

(These references were in two different places. They have been combined and alphabetized; now they need to be identified with the correct numbered citations in the text.)

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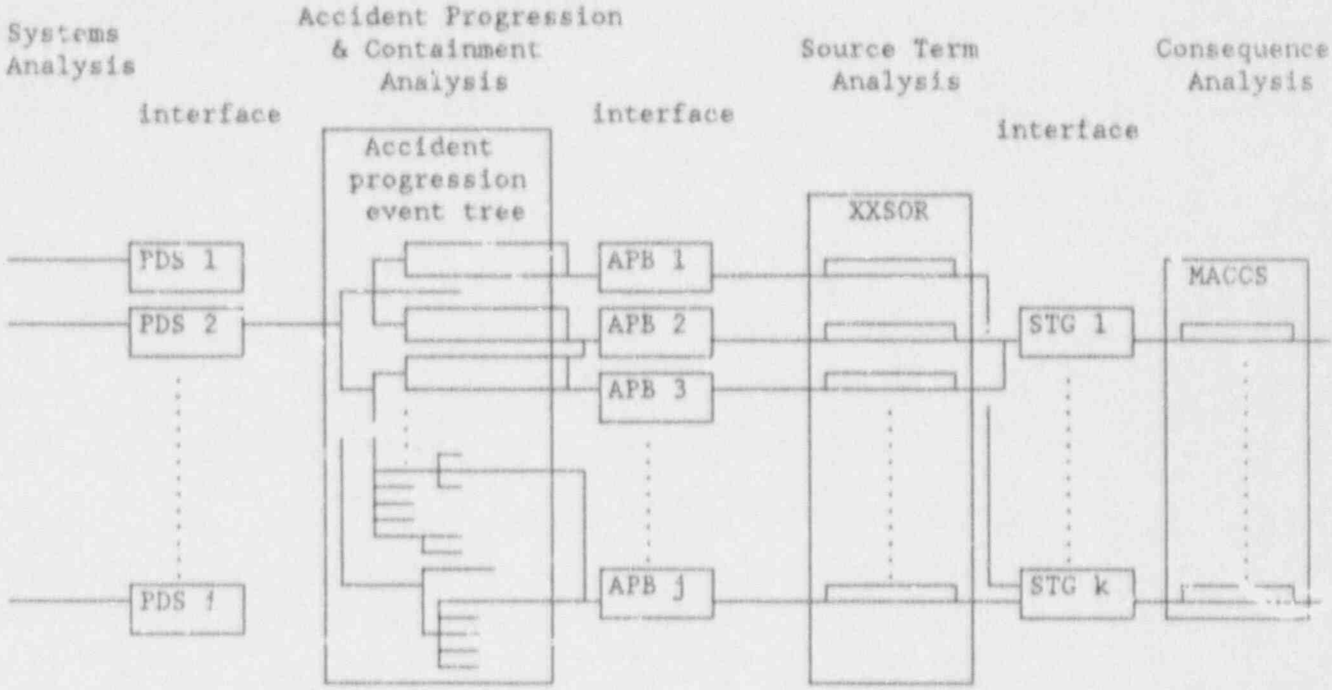
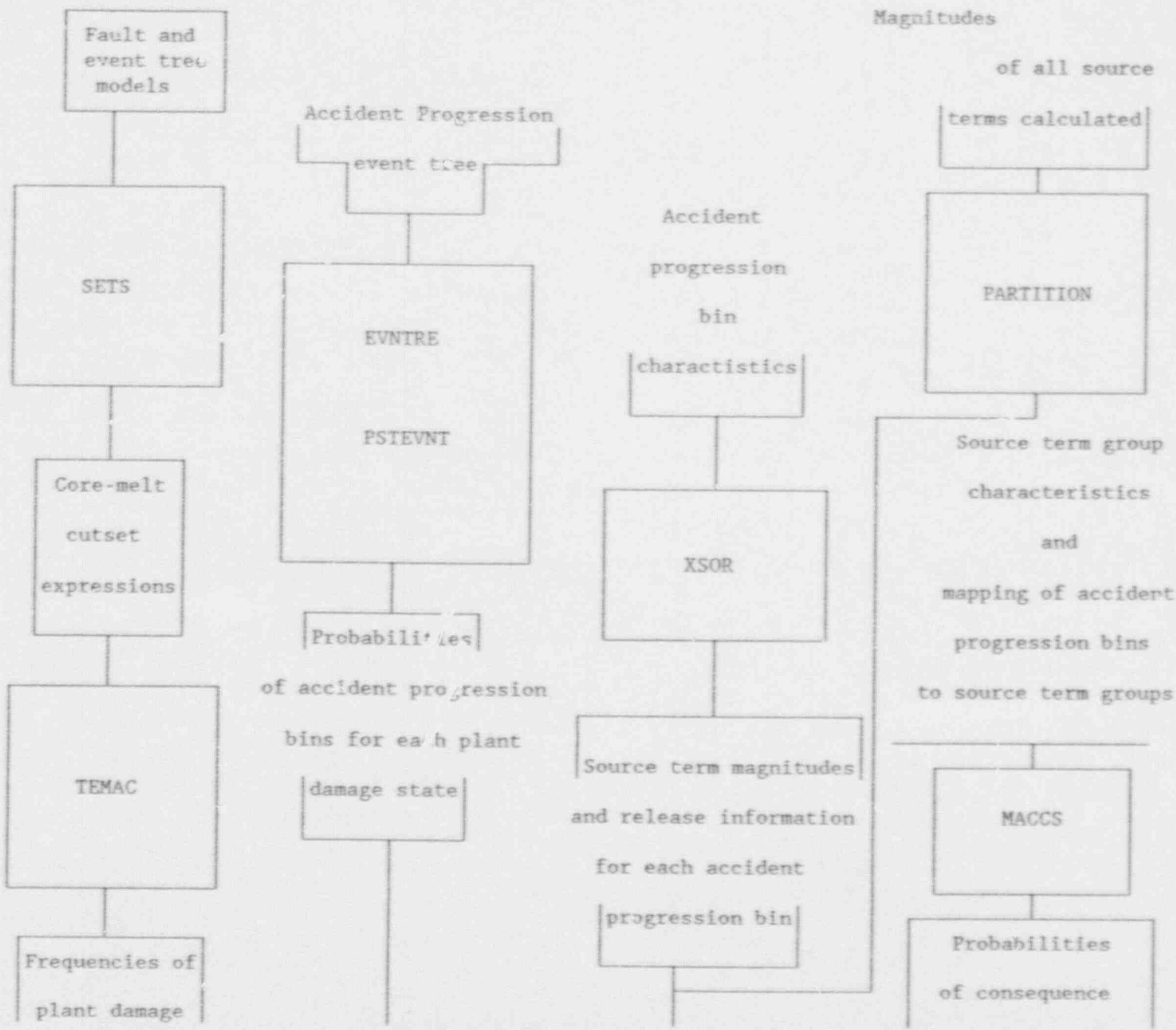


Figure 2-1 Risk Analysis Parts and Interfaces

SYSTEMS ANALYSIS	ACCIDENT PROGRESSION ANALYSIS	SOURCE TERM ANALYSIS	CONSEQUENCE ANALYSIS


Fault and event tree models
SETS
Core-melt cutset expressions
TEMAC
Frequencies of plant damage
Accident Progression event tree
EVNTRE
PSTEVNT
Probabilities of accident progression bins for each plant damage state
Accident progression bin charactistics
XSOR
Source term magnitudes and release information for each accident progression bin
Magnitudes of all source terms calculated
PARTITION
Source term group characteristics and mapping of accident progression bins to source term groups
MACCS
Probabilities of consequence

$$(\Sigma_h \Sigma_i \Sigma_j \Sigma_k \, fIE_h \; P(IE_h \rightarrow PDS_i) \; P(PDS_i \rightarrow APB_j) \; P(APB_j \rightarrow STG_k) \; cSTG_{lk} = Risk_l)$$

PRAMIS

3. INTERFACE WITH THE ACCIDENT SEQUENCE FREQUENCY ANALYSIS

As described in Section 2, there was a need for close interaction between the systems analysts and the accident progression analysts. This interaction was needed for the efficient transfer of information from one area to the other, as well as for a consistent treatment of assumptions across the interface. This section briefly decribes the primary interface activities. The details of the systems analysis methods are described under separate cover.[1]

In addition to the interactions described here, one additional interface was during the automated calculation of risk with propagation of uncertainties. The actual means used to quantify the results while maintaining consistency in assumptions and inputs are described in Section 8 of this report.

3.1 Development of PDS

This report includes a discussion of the methods for analyzing the accident after core damage has been initiated. As illustrated in Section 2, the interface between the accident sequence frequency analysis and the accident progression analysis was accomplished by defining the PDSs. The output of the accident sequence frequency analysis is a listing of cut sets (each cut set is a unique set of events, including system failures, human actions, and recovery failures) that describe the initiation of core damage. A PDS is a group of cut sets which presents a unique set of initial and boundary conditions to the accident progression and containment analysis. These damage states were defined by identifying the characteristics of the accident sequences that most affect the continued progression of the accident or the containment response. The cut sets within a PDS can be sufficiently represented by the same evaluation of subsequent accident progression.

The development of the characteristics that define the accident progression bins (APBs) was based on an understanding of the important attributes of the progression and containment analysis. In large part, this identification is based on knowledge from other studies, and in this case the analysis completed for the draft report was very useful in understanding what initial and boundary conditions were most important. The accident progression analysts and the systems analysts worked together to develop a set of characteristics that would allow grouping of all cut sets. Each characteristic has a distinct number of outcomes that are defined in enough detail to specify the conditions for the subsequent analyses. For some of the characteristics, the outcomes are binary (for example, failed or not failed), while in other cases a continuum of outcomes is divided into discrete categories. An example of the PDS definition is listed in Table 3-1, in this case for the Peach Bottom plant.

Table 3-1
Example of PDS Definitions

Characteristics	Outcomes
Initiating Event	Large LOCA Medium LOCA Small & very small LOCA Transient Transient with scram failure IORV
Offsite Power Availability	Seismic LOSP Other LOSP No LOSP
Station Blackout	Yes (no diesels operating) No
DC Power Availability (Early)	No (all dc power failed) Yes (at least one train available)
Safety/Relief Valve Status	At least one stuck open None stuck open
Status of High Pressure Injection System and Reactor Core Isolation Cooling Systems	Both failed Either or both working
Control Rod Drive System Status	Failed Available (but not operating) Working
Initial Vessel Pressure	High and automatic depressurization have failed High, automatic depressurization available Low
Status of Low Pressure Injection Systems	Both failed Not operating but recoverable Operating but not injecting Either or both systems are operating and injecting

Table 3-1 (Continued)

Characteristics	Outcomes
Residual Heat Removal Systems Status	All modes are failed Recoverable (with power recovery). At least one mode is operating
Status of Condensate System	Failed Recoverable Available, but not injecting Injecting
Status of High Pressure Service Water	Failed Recoverable (with power recovery) Available for manual lineup Operating
Containment Spray System Status	Failed Recoverable Available for manual actuation Operating
Containment Vent Status	Containment vent Drywell vent Wetwell vent Drywell vented but pressure high Wetwell vented but pressure high
Level of Containment Leakage	None in excess of tech specs Leak after accident Rupture after accident Leak before accident or isolation failure Rupture before accident or isolation failure
Location of Leakage	Containment intact Drywell Drywell Head Wetwell

The set of characteristics theoretically allows a very large number of PDSs. In reality, considerably fewer PDSs are defined because the characteristics are not independent and many combinations of the outcomes of the characteristics are mutually exclusive. In addition, the accident sequence analysis is probability-based, and many combinations have too low a frequency to be considered.

The results of the accident sequence analyses are reported in terms of PDS. A unique evaluation of the accident progression event tree (APET) can be reported for each PDS. In order to maintain a single APET rather than one for each PDS, the PDS characteristics are built into the front of the tree. The branch point probabilities for subsequent questions in the tree can then be conditioned on the PDS characteristics, allowing a single evaluation of the tree to cover all damage states.

In order to ensure correct implementation of the PDS, there was regular interaction between the systems analysts and the containment analysts. The actual sorting of cut sets was done with the TEMAC code. A description of the processing of the information may be found in Reference 1.

3.2 Resolution of Accident Outcomes

In addition to the transfer of information embodied in the PDS, there were some special cases which required additional interaction between the systems analysis and the accident progression analysis. These are scenarios in which the outcome of the containment response to an accident affects the probability that core damage will occur. Specifically, these accidents involve situations where the core injection systems are operating but containment heat removal is unavailable. If containment heat removal is not recovered there is a possibility that the containment would fail. The containment failure could then have a negative impact on the core cooling systems through some direct physical impact on the operating equipment.

In this situation, the systems analysis alone cannot resolve the outcome of the accident into whether or not core damage eventually occurs. Typically, the outcome of the core damage frequency analysis for this type of accident is termed a "core vulnerable" sequence. The probability of containment failure and the probability of equipment failure given containment failure must be ascertained in the accident progression analysis. In this program, the APET is used to investigate the effects of the loss of containment cooling. The results of the evaluation are then passed back to the systems analysts in the form of a probability of failure of the operating systems given the loss of containment cooling. There may be several cases involved, depending on the specific initial and boundary conditions associated with the accident.

This feedback link was established through direct interactions between the systems and accident progression analysts. The uncertainties and other dependencies between the analyses were treated directly in the logic of the APET.

3.3 References

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4. ANALYSIS OF ACCIDENT PROGRESSION AND CONTAINMENT RESPONSE

After the frequency of core damage is calculated, the next task is the modeling of the accident progression both in the reactor vessel and after the core debris leaves the vessel. At the same time, the effect on the containment of the core damage progression must be studied. The methods used in this task are described in this section.

4.1 Objectives of the Analysis

The accident progression and containment analysis is designed to allow efficient transfer of information to the source term and consequence tasks. This information is needed both to calculate risk and to provide intermediate findings, insights, and conclusions concerning the ability of the containment systems to mitigate core damage accidents and the effect of the physical progression of the accident on the release of radionuclides. The possible impact on containment of the continued meltdown process is critical to a determination of offsite risk since the timing, location, and type of containment failure (if the containment does fail) are very important considerations in the offsite release of radioactivity. The analysis also examines the operability or effectiveness of plant features such as containment sprays or the suppression pool, which act to reduce the radionuclide releases. Uncertainty analysis was also a primary objective for all of the significant uncertainties including the basic phenomenologies of the meltdown in the vessel and in the containment.

It was a specific goal of this program to improve upon the technology for the evaluation methods involved in this task. In particular, past studies have not modeled many of the complexities and dependencies among the different phenomena, and the models were not well adapted to change; much of the model had to be rethought if one of the inputs changed. It was intended that the models of this study be more amenable to further analysis. In addition, one of the programmatic objectives involved the calculation of the effects of modifications to the facility, and it was a goal here to enable such analyses directly without major restructuring of the basic model. The models also had to be suitable for the uncertainty analysis including the capability to maintain consistency throughout the uncertainty analysis and to handle the dependencies between uncertainty issues.

4.2 Selection of the Modeling Approach

As discussed in Section 2, it would be desirable to have a detailed model for this part of the process that mechanistically evaluated the actual progression of the accident. The highest level of detail would be afforded by mechanistic computer codes such as MELPROG and CONTAIN. Integrated simulation codes such as MELCOR are not as detailed as the mechanistic codes, but would also provide results that could be tied to more mechanistic models. Although these models do exist, they are not practical for use in a PRA where a full array of scenarios must be evaluated efficiently, because mechanistic code calculations involve considerable resource expenditures and calendar time. In addition, these codes do not include models for all phenomena, nor is there a universally accepted code that experts believe is correct.

The PRA requires both more flexibility and more efficiency, and these requirements have generally been satisfied through the use of an event tree in past PRAs. The event tree allows the creation of a logic structure that can discriminate endstates in terms of severity, a necessity for the source term assessment. In addition, the event tree structure is well suited for analysis of uncertainties since the uncertainty in any given phenomena can be treated as alternative outcomes of an event, and two (or more) whole new progression pathways can be represented. The event tree can be thought of as a probabilistic framework used to synthesize the results of the mechanistic models. However, the event trees used in past PRAs did not satisfy all of the other objectives of this program, particularly the need to more completely track dependencies and the ability to recalculate the entire logic under different assumptions, a necessity for sensitivity analysis and analysis of risk reduction.

In the methods development for this program, it was determined that there were many positive attributes to the event tree approach, provided that additional complexity could be included and the process could be automated to allow efficient reevaluation. To satisfy these conditions, a new code was developed for use in this program. This code, EVNTRE, is not specific to accident progression analysis, but rather is a powerful and flexible manipulator of event tree logic. This code is capable of handling very complex relationships between event tree questions and has numerous ways of examining the tree structure and results.

The use of an event tree does not eliminate the use of the mechanistic and simulation codes. Indeed, these codes are used to establish the basic structure of the tree, to determine what events should be included and to help establish specific parameters for specific sets of input and boundary conditions.

Previous analyses have used the term "containment event tree" for the model for this part of the analysis, because the focus of the study is to determine the containment effects. In this study, the model is termed the "accident progression event tree" to recognize the importance of the entire meltdown process on the subsequent radionuclide release and the containment response. Another increased focus on accident progression in these trees is the consideration of the possibility of recovery of the core damage before the vessel is breached. In any case, the terminology is not critical and the reader is free to consider these models as being equivalent in purpose to previous containment event trees.

The reader familiar with containment event trees from previous PRAs may have some initial difficulty in reviewing these accident progression event trees, because although each question is easy to comprehend, the size of the tree and the interrelationships can be difficult to grasp in terms of the "big picture." The presentations of the individual trees in the other volumes of this report series include reduced tree representations and considerable discussion of the inputs and results. A working familiarity with the trees is the best way to become comfortable with this advancement in the containment analysis.

4.3 Event Tree Mechanics and Evaluation Capability

Before the discussion of the event tree structure, there is a need to define the capabilities of the EVNTRE program used to manipulate and quantify the trees. There is a separate user's guide for the code itself, and the details will not be repeated here; but a summary of the capabilities and their effect on tree construction will be provided.

While event tree quantification schemes have been used in the past, EVNTRE represents a significant advance in capabilities for manipulating any event tree logic structure. The specific features include:

- Multiple branches at each question or node, rather than a limitation of only two outcomes;
- Branch probabilities dependent on the path through the tree;
- Representation of continuous processes with automatic tracking of parameters, such as pressures and temperatures; and
- Flexible classification of the results (binning) to sort the output to a manageable set.

In addition to these capabilities, the EVNTRE code can handle very large event trees, more that 100 multiple-outcome questions. This allows more effective modeling of accident progression and allows separation of the problem into time regimes of interest, each of which can be represented with a unique set of questions. The value of this size increase can be illustrated through the treatment of gas combustion processes which can be examined at several time regimes in the model, with concentrations and pressures tracked automatically, and uncertainty issues treated consistently across all questions.

The program allows for consideration of eight types of questions, differentiated by the dependencies on other questions, the source of quantification information (supplied by the analyst or calculated from previous parameters) and later use of the output of the question (no use versus use in a calculation in a later question). These types are listed below:

> Type 1. This question is the typical event tree question where the branch point probabilities are supplied by the analyst and are independent of other events in the tree. The most typical use of this type is in the setup of boundary conditions through the PDS, e.g., "What is the status of the containment sprays at the start of the core damage?" In this particular case three outcomes could be defined: sprays operating, sprays failed, or sprays not failed but unavailable (e.g., due to power loss) The quantification of this

event would be determined by the input from the systems analysis and would not be dependent on other event tree questions.

Type 2. For this type, the quantification of the branch points is dependent on previous branches, in a manner specified by the analyst. For example, if the question relates to the probability of ignition of combustible gas in a given timeframe, the branch point probabilities can be made dependent on a previous event such as whether or not electric power is available.

Type 3. This type is independent of previous questions, but in this case the question not only carries information concerning the probabilities of various outcomes (supplied by the user), it also tracks other parameters for use in later questions. An example would be a question concerning the impact of the release of an accumulator inventory at some point in the accident. The outcomes could be defined to be either a large pressure increase or small pressure increase. In this case the parameter that would be tracked would be steam pressure, with a unique value associated with the two outcomes. If more resolution was required, three or more pressure outcomes could be defined.

Type 4. This is similar to Type 3, but the question is dependent on the outcomes of previous questions. In the previous example, if the accumulator dump pressure rise were dependent on an earlier question such as spray operability, it would be a Type 4 instead of a Type 3.

Type 5. A Type 5 question is independent of all previous questions, but the branch point probabilities are calculated rather than supplied as direct input. The algorithms for calculating the probabilities are called user functions. For example, a pressure rise due to a combination of events can be added and compared to a threshold that may represent the capability of the structure to withstand the increase. Additional discussion of user functions is included in Section 4.6.

Type 6. The Type 6 question is identical to the Type 5, except that the question is dependent on previous branch points. For example, a pressure rise question could be dependent on the operability of the sprays, with the resultant pressure rise compared to a threshold.

Type 7. The Type 7 question is similar to a Type 5, in that it is independent of previous questions and the branch ratios are calculated. However, in addition to the calculation, the parameter values are retained for reference in a future question. Therefore, if a question concerning a pressure rise was compared to a threshold for containment failure and found to be below the threshold, that base pressure may be used in a future question which concerns an event that adds another pressure increment. These types of questions are very useful for tracking hydrogen combustion, as the amount of gas remaining after each burn question can be retained to ensure consistent treatment in all future hydrogen questions.

Type 8. The Type 8 question is the same as Type 7, but it is dependent on previous question. A good example would be a question regarding a pressure increment associated with a hydrogen burn in the third time regime of interest. The pressure rise calculation could automatically account for the availability of hydrogen after previous burns plus any additional hydrogen generation due to physical events in the new time regime, and the probability of ignition could be dependent on previous ignition questions as well as questions concerning hydrogen flamability.

Two of the capabilities require special emphasis. The first is that the dependencies on previous questions may be very simple or quite complex. The simplest dependency divides a question into two cases. As illustrated below in Figure 4-2, the outcomes for question 2 are dependent on the outcomes for question 1.

example here (to be provided)

However, the code also allows for considerably more complex dependencies that can be entered as Boolean expressions. For example, a two-case system could be defined by the expression below for one case, with all other paths through the tree belonging to case 2:

```
Case 1 Question:  4        1        1        6        1        6
       Outcome:   1    *  [(3   +   4)   *   2    +   2    *   1]
                      and      or       and      or       and
```

Case 2 All other paths through the tree

The first case only applies if question 4 takes branch 1 and either question 6 takes branch 1 and question 1 takes branch 2, or question 6 takes branch 2 and question one takes either branch 3 or 4. This example illustrates the power of the tree to track dependencies, although it also illustrates the ability to develop a tree that is quite complex.

The second capability that makes these trees unique is the ability to include parameters within the tree structure, automating the tracking of pressures, steam concentrations, combustible gas concentrations and other parameters required for a consistent treatment of phenomena. For example, the mass balance of hydrogen may be preserved without having to rethink all of the hydrogen-related questions each time a change is made.

Due to the complexity allowed by the flexibility of the EVNTRE code, special emphasis was also placed on provisions to assist in the development of the trees and the diagnosis of errors. The code includes checks on inputs and writes error messages if the logic structure or inputs contain errors. Another important feature is the annotated echo of the input which repeats the input in a straightforward fashion, with a listing of the meaning of each piece of input information for every question. The tree can be examined question by question with a frequency report that delineates the complete split between outcomes for each and every case, as well as the realized split over all cases. With this capability the analyst can work through every branchpoint, ensuring that the intended modeling of dependencies is being correctly implemented. Typically, the analyst also inputs a cutoff frequency such that paths through the tree that are calculated to fall below this frequency are deleted from further consideration. The other capability important when building the tree is the binning of pathways. As described in Section 2, the APBs define endstates that are of sufficient resolution that a single source term calculation will suffice for all paths collected into that bin. During the development of the tree, any binner can be defined to aid the analysts in checking the tree. For example, a binner could be devised that concentrates solely on hydrogen phenomena, and the modeling of the hydrogen can be checked through this capability. The ability to bin the millions of paths through the tree into any combination of bins allows focus on particular aspects, once again to help ensure the analyst that the model is correct.

4.4 General Process of Tree Development

A great deal of information is incorporated in the accident progression and containment analysis. The event tree is used to distill the available information into a systematic format that allows a probabilistic delineation of the possible paths that the accident might take once core damage is initiated. The event tree does not mechanistically model the processes such as thermohydraulic flows or concrete attack by molten core material. It represents these processes in terms of events or questions that list the possible outcomes of the phenomena, as related to the outcomes of previous events. These trees also have a major difference from the typical event tree in that each question can have more than one outcome. For example, one tree question is concerned with the pressure in the vessel before vessel breach, and this question has four outcomes. Whereas the actual pressure could be anywhere in a continuum of values from very low up to the safety valve set points, the outcomes have been grouped into four distinct categories: safety valve setpoint pressure, high pressure, intermediate pressure, and low pressure. In the analysis, each category is associated with a range of pressure values. The selection of discrete outcomes is subjective and depends on the requirements for further use of the values in the rest of the analysis.

This analysis was initiated by collecting information concerning the design of the specific plant being considered. This included details concerning the primary system or reactor vessel, the containment structures, and any containment systems that operate to mitigate the effects of an accident. The next step was gathering information relevant to the study of the accident progression and contain⁻ nt response. This included:

- Results of detailed code calculations for partial and entire accident sequences;

- Results of the simulation codes such as the source term code package (STCP);

- Studies of particular phenomena such as hydrogen combustion with detailed, specialized codes,

- Previous risk assessments; and

- Experimental studies of relevant phenomena.

This background information is not necessarily specific to the individual plant, since even analyses on very different plants could provide insights useful for developing the tree logic. For the analyses reported in this series of reports, the collection of information was initially done several years ago, culminating in the development of the draft event trees reported in the NUREG/CR-4700 series of draft reports. For the final trees, the collection of information focused on updates since the original data collection plus the use of any more recent sources and those noted in the comments on the draft report.

4.4.1 Interfaces with Other PRA Tasks

Section 3 of this report described the interface of this analysis with the accident sequence frequency analysis. As noted there, the definition of the PDS was an iterative process, because the characteristics of the damage states are defined to be those that could significantly affect the accident progression and containment analysis. This process would be very difficult for an entirely new problem, but is somewhat easier with the availability of other PRAs. The previous PRAs and the output of mechanistic codes for similar plants or relevant accident scenarios were used to determine what characteristics of the initial and boundary conditions were most important. This process was initially done for the draft versions of the event trees. The damage states for the final analysis were changed to reflect new information was well as additional insight into important characteristics that was obtained through the performance of the analysis reported in the drafts.

As described in Section 2, the interface with the consequence analysis was through the definition of APBs which are used to group paths through the tree into bins that can be treated with one source term calculation. While these APBs are not needed until the final stage of the analysis, the interaction between the source term analysts and the accident progression and containment analysts was needed early on. The APET must include enough discrimination such that the source term and consequence analyses have sufficient information to appropriately calculate source terms. Four basic types of information are needed by the downstream analyses:

- Time regime information important to the timing of evacuation and other offsite events, as well as to the time for deposition, settling out, and decay which affect the radionuclide species;

- The physical progression of the accident in-vessel and ex-vessel, important to the determination of the release fractions during physical events;

- Presence of mitigating features, such as the containment sprays, which act to reduce radionuclides; and

- Integrity of containment during all of the above processes.

As with the rest of the steps of the PRA process, the most important elements that are needed to accurately calculate the source term and consequences can only be discovered through iterative processes. Once again, the availability of other studies allowed a first cut at the definition of the core melt progression characteristics that were most critical to source terms and consequences. These parameters were refined as the analysis of the downstream tasks was completed. The draft studies were very useful in this study in setting the stage for the identification of the aspects of the core melt progression most critical to the rest of the study.

Interactions with source term analysts were therefore an important part of the methodology. This process was initiated very early in the development of the event tree and was continued through to the calculation of the final result.

4.4.2 Definition of Time Regimes

As mentioned previously, the EVNTRE code allows the tree to be developed in enough detail to consider time regimes. One of the first steps in the tree development process is the selection of these time regimes. The time regimes selected are subjective, and are based on conveniently defined intervals and with consideration of the timing that is important to the source term and consequence analyses. Some time regimes may be quite long, while others, such as the time of vessel breach may be short but are developed in detail because of their importance. The selection of the time regimes can be made plant specific if particular features have effects at certain times, although there was an effort to maintain consistency across plants as much as possible to assist in comparing inputs and results.
As illustrated in Section 2 (Figure 2-3), the tree can be viewed in several time regimes. At least four time periods are considered: the boundary conditions at the start of core damage; in-vessel core melt progression; ex-vessel core melt progression; and the final outcome of the accident. In

practice, these time periods are generally expanded to provide additional focus on particularly important phenomena. For example, the in-vessel melt progression may be separated into sections dealing with: (a) the possibility of recovery of the core before the melt progresses to the point where it will fail the vessel, (b) the in-vessel meltdown process, (c) and the phenomena at vessel breach. The transition from in-vessel to ex-vessel, the vessel breach modeling, is usually detailed in these event trees. The ex-vessel portion of the analysis may also be broken up into periods: during critical core-concrete interactions, from vessel breach up to about 2 to 3 h; late, in the period following the principal core concrete interaction, and very late, up to about a day following the accident. The specific times associated with these intervals vary with the plant. Some of the time regimes are most important from the point of view of the accident progression while others are more critical with respect to the source term and the operability of equipment (e.g. sprays) that would reduce the source term. These time periods may be supplemented by others if needed to further resolve the outcome of the events.

4.4.3 Layout of the Tree Structure

There are no specific rules for the initial development of the APET. Once the initial PDSs were defined through questions, the time regimes of interest were generally considered in detail in chronological order. The process for the development of each part of the tree is the same. The detailed code analyses, experimental results, and all of the other sources with information pertinent to the time regime are examined. These sources identify the major physical events to be considered, and a review of a variety of sources indicates where there are uncertainties in the process.

Many Source Term Code Package (STCP) analyses were done in support of this program. Early in the program, the full project staff agreed on a set of STCP runs for each plant, anticipating the outcome of the core damage frequency assessment and attempting to cover the most important types of scenarios. These scenarios roughly corresponded to some of the PDSs, therefore relatively detailed analyses of certain PDSs were obtained from this process. The resultant report contained a great deal of information concerning timing of the physical events and identification of the major phenomena. These detailed analyses allowed the development of a general tree layout for the core melt progression in each time regime. This was only a first step; however, since these initial trees did not cover all PDSs and they did not reflect alternative hypotheses concerning many of the phenomenologies.

The next step was to determine the effect of the other PDSs characteristics on the basic progression of the accident. This involved the creation of new questions to cover types of events not considered in any of the STCP analysis, for example, the addition of questions to treat steam generator tube ruptures if they were not one of the specific scenarios analyzed with the STCP. In other cases, the impact of different boundary conditions had to be estimated in order to set up case structures for each question that would discriminate important differences in effects. For example, the reference STCP calculation may refer to an accident with a large LOCA initiating event, but another damage state might include intermediate LOCAs. The outcomes of the phenomenological questions in the tree would be examined to determine if the change in LOCA size would change the accident progression. In some cases sensitivity studies with the STCP generated insights useful in establishing the dependencies on previous questions, while in other cases no relevant STCP runs were available. Without sensitivity studies judgment, interpolation, and use of analyses from other studies had to be relied on to identify specific effects.

Through this process a basic tree structure was developed including the development of a case structure that allows dependencies to be tracked. Although the initial tool for layout of the tree structure was the STCP, the next step of the process expanded the trees to account for the other information sources collected as part of this task. Other mechanistic code output was reviewed for relevant information concerning the tree development. In some cases the resources existed to perform some code calculations, for example with CONTAIN, to assist in the further delineation of the trees. In other cases information on specific subjects was examined to see if the phenomenologies represented in the tree could account for alternative views. For example, if a specific calculation with a code other than the STCP, for example, a HECTR analysis of a combustion event, either suggested a different magnitude of phenomena or introduced a phenomena not in the STCP the tree was adjusted to allow consideration of the alternative paths created by these new phenomena.

This process of examining other sources was quite extensive. In the initial tree development, inputs from a number of other studies besides the STCP analyses played a role in the tree development:

- Containment Loads Working Group;
- Containment Performance Working Group;
- Severe Accident Uncertainty Analysis Program (SAUNA);
- Severe Accident Sequence Analysis Program (SASA);
- Various reports from the Industry Degraded Core Program (IDCOR);
- Steam Explosion Review Group;
- High pressure ejection test series;
- Analyses supporting the unresolved safety issues; and
- Other analyses of specific phenomena.

Although the initial tree layout was based on the STCP, the final tree represents a much broader view of the possible alternatives of accident progression and containment response. Experimental results are taken into account wherever possible. Most experiments are done on a small scale and involve only a portion of the accident progression scenario because of the cost and complexity of experiments of this type. The experimental results generally are considered in the tree development indirectly, by changing the outcomes of particular branchpoints or by influencing the interpretation of the output of one of the mechanistic codes. The experimental evidence was also part of the consideration of the experts when developing uncertainty distributions for the important uncertainties, as discussed in Section 4.7.

The final trees presented in the other volumes of this report were completely redone since the initial draft tree development. This reanalysis took up where the other trees left off, but included updates in information and steps to make the analyses of the different plants more parallel. The EVNTRE code capabilities were expanded, and the ability described earlier to track parameters such as steam and hydrogen concentrations automatically was built into the codes. The dependency cases were expanded to account for additional interactions. Finally, in the interim since the draft, a number of new information sources became available, allowing a more complete representation of the possible event tree paths. The most useful sources of information included:

- Some additional STCP runs to evaluate scenarios not well covered previously;
- Comments on the draft report by a wide range of organizations and individuals;
- Some new experimental evidence;
- New inputs from the groups involved in expert elicitations for this program;
- New analyses of issues found important in the draft study, including mechanistic calculations.

For example, the NRC redirected some of its contractor analysis toward resolving some of the important issues discovered or highlighted in the draft. A good example would be the numerous CONTAIN calculations done by SNL to investigate high pressure ejection and direct heating phenomena. An array of studies was carried out to attempt to better bound the characteristics of direct heating and the role of initial plant conditions and certain plant features. For example, before the draft analysis there was no information available concerning direct heating in the ice condenser, but in the interim CONTAIN has been used to study the phenomenon.

One of the other sources of information was the new elicitation of expert opinion for highly uncertain issues. These elicitations could bring about the need to modify the tree in a mechanistic way (the trees had to be adapted to fit the form of the expert input), or the experts sometimes identified dependencies not considered originally. In any case, the final trees are consistent with the inputs of the expert groups.

With the draft APETs as a starting point, the new information available particularly on the most important issues, and the new capabilities of EVNTRE it was possible to create new trees that reflect the current understanding and the uncertainty of accident progression and containment response.

4.5 Description of the Tree

The description of the individual event trees for each plant are included in the plant-specific reports. As an introduction to those trees, this section describes the principal parts of the tree and includes examples of the typical questions asked in the event tree. As discussed previously, the event trees may be considered in parts that generally correspond to time regimes or key events in the meltdown process. The exact breakdown of the tree is dependent on analyst preference as well as plant-specific attributes. In this section, the following breakdown is discussed: PDS definition, resolution of core vulnerable accidents, in-vessel recovery, in-vessel processes and containment effects, vessel breach, core concrete interactions, late containment effects, and summary questions.

4.5.1 PDS

Once defined, the PDS had to be incorporated into the analysis. For smaller event trees used in the past, it was possible to develop a separate event tree for each unique set of initial and boundary conditions, i. e., for each PDS. Due to the size of this tree, it would be unwieldy to create a separate tree for each PDS. Also, the basic tree structure remains the same for nearly every PDS. In this study, a single tree was used to represent all PDS, a possibility afforded by the capability of the EVNTRE code to accommodate a case structure for dependencies. Using a case structure for each event allows the question and outcomes to be the same, but the quantification of each branch to be dependent the specific scenario, for example, as defined by the PDS. If the outcomes would be expected to be different for each PDS, the number of cases would equal the number of PDS. In practice, there are considerably less cases, since an event is generally only dependent on one or two of the characteristics that make up the PDS. Using the case structure approach, a great deal of customization of the tree to meet individual boundary conditions is possible in a very efficient manner.

To establish the formal mechanism for establishing the case structure in the tree, the first stage of the event tree is used to delineate the PDS. Figure 4-2 illustrates an example of this process. The characteristics of the PDS are the event tree questions, and the branches for each question represent the possible outcomes for the characteristics. In the example illustrated, the first PDS characteristic is the size and location of any break in the RCS at the time of core uncovery. There are six outcomes defined for this location, each of which is important to the remaining accident progression or source term calculation. The next question concerns the availability of the injection systems, with three branches defined: operating, failed, not failed, but unavailable (power loss). This part of the tree continues until all of the characteristics are covered.

Figure 4-2.

The example illustrates other features of this part of the analysis. While the tree could be used to theoretically examine every combination of characteristic outcomes for the PDS, in reality there are considerably fewer because some combinations are precluded. For example, the definition of the V sequence, the interfacing systems LOCA, already precludes the availability of the ECCS and only one path is possible. There are many other combinations that are physically precluded. The other reduction in the actual number of damage states as opposed to the indicated number is that some combinations are of very low probability. The output of this portion of the event tree will be a sorting of the PDS that have significant frequency. The example also illustrates the difficulty of graphically representing the trees used in this program, because the large number of paths associated with an even a small number of multiple-path questions.

With the inclusion of these events that define the PDS in the tree, one APET can be used to cover all PDSs. Later questions in the tree can refer to the questions in the first part of the tree to establish a case structure dependent on the damage states. For example, if the question referred to the pressure in the vessel at some later time period, the outcomes of the question would be dependent on the outcomes of the first question listed which described the physical integrity of the vessel or RCS at the beginning of the core damage process. For example, the large LOCA leakage rate outcome would preclude further vessel pressurization.

The trees for each of the plants have the first 5 to 20 questions devoted to the PDS definition. The questions are different because the PDS characteristics are different for each plant. The order of the questions is not of significance; the analyst has arranged then in a convenient fashion.

4.5.2 Resolution of the Core Vulnerable Scenarios

The core vulnerable accidents are those where the systems analysis ends with a successful cooling of the core, but continued cooling is dependent on the effect of certain events on the containment. As described previously, feedback between the accident progression analysis and the systems analysis is needed to resolve these sequences into those that cause core damage and those that do not. The exact nature of these accidents is dependent on the plant, but the most typical sequence involves the failure of all containment heat removal mechanisms, with successful core cooling. The issue to be resolved in the tree is whether or not loss of all containment cooling will have a deleterious effect on the other core cooling systems in the long term. These questions are generally included right after the PDS questions described in the previous section. Typically the modeling to consider these core vulnerable sequences will consider the impact of the loss of containment cooling in terms of pressures, temperatures, and threat to containment integrity and possible recovery actions such as venting the containment. The event tree also considers the possibility of a direct physical threat to the core cooling systems of these containment events, including the possibility of containment failure by rupture or leakage, or any adverse effects of venting. For example, if the containment failed due to loss of cooling, there is a possibility that depressurization could fail the operating pumps either through NPSH problems or through the possibility of direct damage to the piping in the case of a catastrophic containment failure. The outcome of this portion of the tree is used to provide the systems analyst with the conditional probability that a core vulnerable accident will become a core damage accident

4.5.3 In-Vessel Recovery

One of the unique features of these event trees is that they address the recovery of the core before a serious threat to the integrity of the vessel. This part of the tree is not developed in great detail, but is included to recognize the possibility that even though core damage has been initiated, there is the opportunity to arrest the damage before vessel breach. There is a great deal of uncertainty in the timing and needs for successful cooling during this phase of the accident, and these uncertainties are reflected in the event tree model The trees only consider this possibility for accidents involving power loss, where equipment is unavailable rather than failed and there is the possibility of recovery when power is restored. (The loss of power accidents were important contributors to the core damage frequency of each of the plants studied in this program.)

4.5.4 In-vessel Processes and Containment Response

The next section of the tree deals with the continued degradation of the core, up to the vessel breach. There are many considerations during this phase:

- What is the loading of containment from steam for the different types of accidents?
- What are the conditions important to the release of radionuclides from the core and from the vessel?
- What are the physical conditions in the vessel before vessel breach?
- How is hydrogen being produced by the core degradation?
- Where is this hydrogen going, and does the hydrogen burn affect the containment?
- How does the operability or non-operability of containment cooling systems, containment sprays, or special features such as an ice condenser affect each of the questions above?

Just by this listing of considerations, it is easy to illustrate how the event tree quickly becomes very large. A few typical questions for this portion of the tree are discussed below.

For the PWR, there is the strong possibility that the core degradation process will increase temperatures enough to affect the structural integrity of the coolant system before direct breach of the bottom vessel head. Since this would affect containment loading at vessel breach, the release of hydrogen from the vessel and the release of radionuclides during this phase of the accident, the possibility of this type of phenomenon is critical to the estimation of risk. The exact nature of the questions dealing with this event are subjective. The first question that was asked is what are the critical outcomes that need to be discriminated for the rest of the analysis. After review of the literature and consideration of downstream events, it might be decided that only certain failure possibilities are at all likely and that it would be sufficient to know the following outcomes: hot leg failure, RCP seal failure, steam generator tube failure, or no failure. The possibility of these events could then be asked in either a series of questions, one with yes or no results for each outcome, or in a single question that has four outcomes. Once this structure is decided, the next step is identifying the cases needed to identify the different probabilities of these events. For example, one case would eliminate the possibility of any induced rupture for PDS where the vessel is already breached, since the heating process of natural circulation would not be present and the critical outcome, the integrity of the RCS, has already been determined. Another case might be defined to determine if RCP seal cooling is available, since this would affect the probability of an induced seal LOCA.

Many of the questions in this time regime are concerned with hydrogen gas production. Ordinarily, there is a question that asks how much hydrogen is produced in the vessel. This would be a Type 4 question, since it depends on previous questions, the distribution for the outcomes is an input (in this case, a probability distribution from the expert elicitation), and a parameter (amount of hydrogen) is associated with the outcomes for use in a later question. This would be followed by a question that delineates how much of the hydrogen produced is released from the vessel during this time period (before vessel breach). This would be a Type 6 question, because it would be dependent on previous questions, but the outcomes could be calculated internally based on how much gas was produced in the vessel. The next question might ask if the resultant concentration of the hydrogen in the containment is flammable, which is another Type 6 question because it depends on previous questions (integrity of containment, steam concentration in containment, and amount of hydrogen released from the vessel), and the outcomes are calculated from a formula that compares steam and hydrogen concentrations to flamability limits. The formula is supplied as a user function, a topic discussed in more detail in Section 4.6. The next question may ask if ignition occurs during this time frame and a followup questions would calculate the amount of hydrogen consumed and the pressure rise from the burn. These latter two questions might use the concentration information from the previous questions to calculate a burn size and pressure, based on analyst input concerning burn completion.

This discussion has identified the basic premise of this portion of the tree. The exact questions are dependent on the specific of the plant design and on the analyst's choice concerning how the questions are asked. The objective of this portion of the tree is to identify the conditions of the core and containment just before vessel breach. Steam and hydrogen concentrations are tracked within the event tree to ensure consistent treatment in later questions. For example, if an early hydrogen burn occurs, that amount of gas is deleted from the balance available to burn in later events. As is obvious from the simplified description, this is not a mechanistic treatment, and the physical event represented in single questions may be considerably more complex in reality. For example, all pre-vessel breach hydrogen burns may be modeled through a single question in this time regime. Multiple burns would only be modeled if it was necessary to obtain and accurate picture of containment effects or if the source term analysis needed this information.

4.5.5 Vessel Breach

All time regimes are not equally represented in terms of the number of associated questions. A good example is the time during and immediately after vessel breach. This time period is important because both the radiological release and the threat to containment integrity in this time period are quite important in terms of overall risk. This part of the tree generally includes questions that list the possible failure modes of the vessel when the core debris is released to containment. For example, there is a possibility of failure of the bottom head en masse, or a few holes may open, ablating to larger holes, and depending on conditions in the vessel the material may come out only by gravity or be forced out by pressure in the system. After vessel breach the event tree is concerned with loads on the containment from steam hydrogen and any other sources such as direct heating. All of these questions involve cases that describe the effect of the outcomes of earlier question on the probabilities of the outcomes of these questions. The operability of containment systems that could mitigate pressure loadings or radionuclide releases is also considered.

The questions in this time regime are similar to those described previously in that dependencies in parameters such as hydrogen concentrations are tracked automatically. A question concerning the capability of the containment to withstand the pressure loading is also asked. This question allows a direct comparison, within the EVNTRE code, of the total pressure due to the integrated effects of the accident progression to a containment pressure capability curve that is an input. A Type 5 question could be used for this purpose. Additional questions may be included to specify the possible types of failure mode of the containment, if failure should result from the loading at vessel breach. This could include both size and location of the failure, both of which may be important to the source term and consequence analysis.

4.5.6 Core-Concrete Interactions

The next stage of the tree typically deals with the physical process involving the core after the vessel breach, with a focus on core-concrete interactions. This section of the tree includes questions to account for:

- Additional combustible gas generation and its flammability;
- Possibility of debris coolability;
- Boundary conditions that would affect the release of radionuclide during this phase, especially the presence of overlying water or sprays;
- Loading of containment due to steam generation and gas combustion; and
- Effect on containment of both pressure loading or any direct damage due to the core-concrete interactions.

The questions are similar to those in the previous time regimes, with continued maintenance of the user functions to track combustible gas and steam concentrations. As with the rest of the tree, the dependencies on the previous paths of the accident are built into the case structure.

4.5.7 Late Containment Effects

Typically, a tree will also include some events to account for slowly-evolving accidents, such as long-term pressurizations that would take tens of hours to threaten containment. Another possibility is deinerting of the containment, a reduction in the steam concentration due to the late operation of cooling systems that allows the hydrogen concentration to pass to the flamability limit. Another phenomena in this time regime involves the possible breach of the containment by meltthrough of the basemat.

The late effects questions are generally only asked for pathways that have not involved other serious releases or containment failures. This philosophy is generally true throughout the tree development. For example, an early containment failure will preclude much of the downstream analysis. There are some events that are considered even for these early containment failures, depending on what aspects are most important to a full treatment of the scenario in the source term analysis.

4.5.8 Summary Questions and Final Tree Outcome

In reviewing the tree structures, one should be aware that it is possible that summary questions are placed at various locations in the tree structure and possibly at the end. These summary questions are an analyst's convenience for keeping track of the tree development. These summaries also provide useful cross-references for future dependencies. For example, if a question in the CCI time regime is dependent on the status of containment immediately after vessel breach, it is easier to set up dependency cases based on a single question that summarizes the status of containment up to that time period, as opposed to cross-referencing the cases of several combinations of outcomes of previous questions. Very often at the end of the tree it helps in the analysis of the results to have a summary question or two that identify the chief outcomes, basically the containment failure mode and the time of failure for those paths through the tree that do not have an intact containment.

4.6 User Functions

Questions Types 5 to 8 include the calculation of the branch point probabilities based on a user-supplied function. Therefore these questions can be considerably more complex than the other event tree questions in which probabilities and possibly parameter values are part of the input. The event tree program includes the capability for direct selection of some of the most typical user functions, or the EVNTRE program can call a subroutine that contains as user function of more complexity that can be provided by the analyst. The directly supplied user function capabilities are as follows:

- AND or ADD, the parameter values are added (e.g. adding of two pressure increments);

- MAX or MIN , find the maximum or minimum of the referenced parameters; and

MULT, the inputs are multiplied together.

If these are not used, the analyst can supply an alternative function. In this study the most frequent use of the user function involves the tracking of combustible gas. While straightforward, the user functions can be made quite complex in their ability to calculate parameters. The hydrogen combustion user functions track hydrogen stream and oxygen concentrations, over all time regimes of the tree. For example, a late deflagration question will first consider the depletion of the concentrations in any previous burns. The user function can also distinguish types of burns by the concentration limit, for example, diffusion flame events can be distinguished from deflagrations and detonations. The same user function is used to calculate the pressure rise associated with the burn. User functions are written in FORTRAN and are automatically called by EVNTRE. The specific user functions for each plant are described in the appropriate plant-specific reports.

The other aspect of the calculated branchpoint questions is the comparison function. The calculated parameter values (for example, two pressures combined together) can be compared with a third parameter (such as a containment pressure capability). Once again this part of the process may be done by EVNTRE-supplied comparison methods, or by a user function. The comparison types that are built in EVNTRE include:

- EQUAL, which uses the output of the user function to directly specify the branching ratios;

- NORMAL, which allows the comparison of a parameter with a normal distribution. A number is randomly selected from the distribution and compared to the input parameter. For example, the pressures added together in the first part of the process can be compared to a containment failure distribution for a fail/no fail assignment of split fractions for containment failure.

- THRESH, which compares the combined parameter from the built in function or a user function to a single threshold value. A comparison of concentration to a flamability limit would be an example.

- GTHRESH, allows a discritization of the reference parameter to compare to a series of thresholds, thus allowing consideration of more than two outcomes for a question.

The user functions offer great flexibility in tree development as well as freeing the analysis of the burden of tracking parameters needed to establish split fractions for paths well into the event tree.

4.7 Quantification of the Accident Progression Event Tree

Each question in the tree requires input for the quantification process. As illustrated by the discussion above there are several different types of questions in the tree, each of which may have a different type of quantitative input. Each of these types of questions are discussed below. The uncertainty representation was a critical objective of this study. As described in Section 2, a Monte Carlo approach was taken for the calculation of the uncertainty across the risk analysis tasks. For this part of the analysis, uncertain inputs were assigned distributions which were then sampled in the Monte Carlo calculations. The sources of input are described in this section, but Sections 7 and 8 of this report are the source for detailed explanations of the uncertainty process and the overall risk calculations.

It is important to remember that the quantification of the tree is done through case structures that allow separate quantifications for specific conditions established by the path through the tree up to each question. This is an important element of the analysis, and the quantification for a single question in the event tree may involve different inputs for different cases. The quantification can be made dependent on any or all of the preceding questions, allowing all of the complex interrelationships to be tracked automatically and considered in the quantification.

4.7 1PDS Questions

The first questions in the tree that sort PDS are quantified based on the results of the systems analysis. For a given PDS, each of the initial questions takes a single path. For example, if the first question sorts the size and location of the RCS failure, a large LOCA damage state will have a probability of 1.0 for that outcome, and a probability of 0.0 for all other outcomes.

4.7.2 System Reliability/Recovery Questions

Some events in the tree are concerned with the reliability of equipment or the possibility of recovery. These events are quantified with reliability data or human reliability analysis, just as the system analysis models are quantified. For example, the case structure defines whether or not the spray system has failed previously, and the probability of spray failure is applied to those cases where the system is available and called to operate. Similarly, the trees all contain questions referring to the probability of offsite power recovery. The systems analysis task developed a curve of recovery versus time. Each event tree question regarding power recovery is assigned a value from the power recovery curve for the probability of recovery in the interval since the last time the recovery event was asked. Similarly, operator actions in the tree are quantified with the methodology used in the system analysis. Some of these events can also be uncertainty issues, meaning that their quantification in the event tree is based on sampling from a distribution.

4.7.3 Phenomenological Questions

These questions form the basis for much of the tree, because the uncertainty in phenomenology is the reason for the alternative branchpoints at many of the questions. There is considerable discussion of these events in Section 8 of this report, because the description of the uncertainty in these events is somewhat subjective. For example, a phenomenology question might have two alternative outcomes. In some cases, given a set of input conditions, the phenomenology might always be expected to have one of the outcomes, but there is uncertainty as to which one. (In an series of experiments involving core damage, all cases with this set of input conditions would take one of the paths and not the other.) In other cases, given a set of input conditions, the phenomenology might be expected to take either path, although the probability of one path may be much greater than the other. (In the same series of experiments involving identical initial conditions, some fraction of the events would have one outcome for this phenomenology, while others would have the other outcome.) In practice, the situation is more complicated for two reasons. The first is that the initial conditions cannot always be specified exactly (the tree would be too large if all cases were considered). This creates a situation where the uncertainty in a specific question include both types mentioned above, some due to the inability to specify exact conditions and some due to the phenomenological uncertainty. The other complication is that individual analysts may have differing viewpoints on what type of uncertainty is involved.

For this study, most of the phenomenological questions were quantified with distributions, based on the evidence available from all current information, including the mechanistic analyses used to construct the tree. As described in Section 8, the most important uncertainties in the viewpoint of experts in the field and as identified by the draft studies, were quantified through a process of expert elicitation. Special expert panels were formed for key areas of the analysis. Each expert provided a view of a specific phenomenological question, including the dependencies on previous issues, in a formal elictation process. Then the inputs of the experts were convoluted to obtain a distribution representing the uncertainty in the issue, with a different distribution for each distinct set of previous dependencies. The quantification of the tree was based or a sampling of those distributions.

In cases of events with somewhat less uncertainty, the expert elicitation process was carried out within the project team. The process was he same, and an attempt was made to represent the full range of uncertainty by reviewing the available literature and experimental results of the topics. These questions were also sampled in the final quantification.

There are other questions which were quantified with distributions generated by the individual plant analyst. This method was applied only to questions which were verified through sensitivity analyses to have very limited importance.

As described in Section 8, the representation of uncertainty involved several hundred complete risk estimates (sample members) including separate event tree quantifications. It should be noted that the results of the tree quantification may be examined for any sample member. For purposes of tree development and review, mean inputs can be used for the issues.

4.7.4 Containment Failure Questions

One special case constitutes the questions regarding the capability of the containment. The experts provided their viewpoint of the location and failure pressure as a function of loadings. The combined distribution for the probability of failure (and location) as a function of pressure was then input as part of a Type 5 question. This type of question automatically adds the appropriate pressure from events contributing to the pressure rise and compares it to the curve to determine the split fraction for whether or not the containment fails. In this question or as a separate question, the expert inputs regarding location of failure are also used to generate the probability of the possible outcomes.

4.7.5 Combustible Gas Modeling

Another special case of the phenomenological question are those regarding the combustion of gas, principally hydrogen. As noted earlier, these are evaluated with user functions that track the concentrations of steam and hydrogen. As for as quantification of these events the user function automatically provides the output, but the function itself must be input. Some aspects of the user function may be subject to limited uncertainty, for example the calculation of concentration as a function of amount of gas. These aspects of the user function are input as values with no uncertainty. There may be other aspects of a user function that do involve uncertainty, for example a combustion limit for some specific set of conditions. The uncertainty in any of the inputs to the user function could be treated as an uncertainty issue and sampled from a distribution.

4.7.6 Summary Questions

As described previously, there are questions in the tree that are used for analysts' convenience in summarizing previous outcomes. These questions do not require additional quantitative input as the case structure determines the outcome.

4.8 Binning

An important part of the analysis process is the binning to reduce the number of paths through the event tree to a manageable set for the downstream analyses of source terms and consequences. As noted earlier, there are binning steps that the analysts use in tree development and review and then there are the formally defined accident progression bins which are used to transfer information to the source term analysis. In essence, a binned result can be thought of a reduced tree with only the events of interest as questions, and with all other questions internalized in terms of their effect on the bin characteristics.

The trees included in the plant-specific report include the bin definition information (the binner). The binner lays out the Boolean expressions that define the bins. The paths through the tree then are sorted into these bins. The EVNTRE code lists an error if a path through a tree cannot be matched to a defined bin.

Consider a very simple bin scheme that has two characteristics: size and time of containment failure. Each of these characteristics has a set of outcomes. For example, there might be three defined sizes of containment failure and two time periods of interest. In this example there are six possible bins. The binner defines the conditions needed to arrive at the different outcomes in terms of the outcomes of questions in the tree. The paths through the tree are automatically sorted into these bins by a logic structure supplied by the analyst. In the two-characteristic example, there are six bins that are possible from a strict combination of the possible outcomes. In reality, one or more combinations may be precluded (perhaps there is no possibility of a catastrophic containment failure at a late time period) and other combinations may not be likely.

This same logic applies to the larger binning schemes used in these studies. The APBs may have 10 or more characteristics, each with several outcomes. The number of possible combinations of bins is very large, but the actual number of bins after removal of those that are precluded and dropping those of very low probability becomes a manageable set. The cutoff frequency for what can be dropped is dependent on what is important to risk at the specific facility.

The results can also be rebinned to meet other specific needs. For example, the 10 characteristic bins are somewhat difficult to comprehend at a high level, and are quite difficult to compare to the results of past containment analyses. Therefore the results of these studies are also presented in terms of these reduced bins that allow a simpler representation of the results. The two-characteristic bins defined above concerning containment failure size and timing could be one example of a high-level binning that would allow insight into some of the basic outcomes of the analysis. The results of each study have been considered in terms of these high-level bins, and these results are the ones actually reported in NUREG 1150.

4.9 Computer Processing

Several references have been made to the use of the EVNTRE code in this analysis task. The EVNTRE User's Guide describes the input formats used in this study as well as the use of the code. Each of the plant-specific reports includes a detailed description of the tree input and a presentation of the actual code input. Once familiarity with the format is established, the annotated tree input file is sufficient for understanding the logic of the tree, although the detailed description of the quantification provided in text is needed for understanding the justification of the quantification of each branchpoint. Table 4-1 illustrates some typical EVNTRE input. The EVNTRE User's Guide should be referenced for a more thorough discussion.

Once input, the trees are processed to obtain the frequencies for each of the pathways, binned as appropriate. As described above, the creation of the tree involved iteration and review of intermediate branchpoints, so in reality the trees are solved many times. In these intermediate solutions, the analyst may create bins and bin sorting logic that focus on particular aspects. The quantification of the tree during the development stage may be based on nominal or mean values at each branchpoint, to avoid the need to run a sample that samples distributions for each question.

When the final tree was ready, the LHS code was used to generate a sample for the inclusion of uncertainty. (See Section 7 for a complete discussion of the uncertainty methodology in this program.) The LHS sample, consisting of several hundred sample members, is used to represent the uncertainty in this part of the analysis. Each sample member includes the selection of single values for each branch point from the uncertainty distribution supplied as discussed in Section 4.5. Each sample member therefore has a unique quantification of the entire APET resulting in a conditional probability of the APBs for each of the PDSs. As noted in Section 2, this is

P(APB)j/PDSi = conditional probability of APB bin j given PDSi.

Although the EVNTRE code can be used directly for producing the binned results, a post processor, POSTSM, was used to do the actually sorting into bins for this study. In this manner, the output of the EVNTRE processing of the tree can be kept somewhat generalized (in terms of a large number of unsorted bins) and the POSTSM code can do the final manipulation of the results. The advantage of this process is that the analysts can adjust the bins or create reduced bins with POSTSM alone, without the need to rerun the entire EVNTRE input.

Table 4-1

4.10 Type of Results Obtained From The Analysis

The chief output accioent progression and containment analysis is the matrix of APB probabilities for each PDS that is one the elements of the risk equation, as discussed in Section 2. There are actually a set of these matrices, one for each sample member. However, this analysis also results in many insights concerning accident progression, even without the completion of the risk calculat'on. The reports for each plant include intermediate results for this task, and insights developed in completion of this part of the analysis are also provided.

The mean frequencies of the APBs are provided for each plant. The main report listing of these results is limited to the APBs that account for about 90% of the frequency for each PDS with significant frequency. This is followed by a presentation of the mean frequencies of the PDS given core damage. In other words, the bins for each PDS are weighted by the frequency of the PDS and are added together. A discussion of the insights of this part of the process is also provided.

Following the presentation of mean results, some of the uncertainty information is provided. The key parameters of the uncertainty distributions for the probabilities of the PDS conditional on core damage are provided. In addition, the uncertainty distributions for some of the important or interestingPDS are also provided. These are selected based on the outcomes of the individual analyses. (This information is available for all PDS, but is too voluminous for the primary presentation of results.)

As described above, there is often a ;.ed to describe the results in terms of reduced bin characteristics. The results for each plant are reduced to these higher-level bins in each report. For example, each plant has been considered in terms of the ultimate containment outcome and timing as the only two bin characteristics. In addition, the results were used to determine the most important events in the tree, and a reduced tree illustrating only the critical branches is displayed. The insights obtained from these intermediate results are discussed.

[To be completed after Section 2 terminology is final and after additional interaction with the NRC .o see what results are to be included.]

Note: this section has not yet had the references added. The following are lists that may be used. These references require some additional formatting.

4.11 References

1. U.S. Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants, NUREG-1150, December 1988.

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

3. A. S. Benjamin et al., Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Surry Power Station, Unit 1, Sandia National Laboratories, Albuquerque, NM, NUREG/CR-4551, Vol. 1 (Draft), February 1987.

4. A. S. Benjamin et al., Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Sequoyah Power Station, Unit 1, Sandia National Laboratories, Albuquerque, NM, NUREG/CR-4551, Vol. 2 (Draft), February 1987.

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6. C. N. Amos et al., Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Grand Gulf Power Station, Unit 2, Sandia National Laboratories, Albuquerque, NM, NUREG/CR-4551, Vol. 4 (Draft), April 1987.

7. Author, Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Zion Unit 1, Brookhaven National Laboratory, Upton, NY, NUREG/CR-4551, Vol. 4 (Draft), April 1987.

8. Responses to comments on draft 4551 NEED REFERENCE

RADIOLOGICAL SOURCE TERM ANALYSIS

A radiological source term is the fraction of radiological material available in the reactor at the beginning of the accident which is released to the environment, along with information as to the timing, energy, and location of release point. The radiological source terms for NUREG-1150 divide the radiological material into nine source term groups. The elements comprising a group all have similar chemical, physical, and physiological properties. The principal element of each group is a surrogate for all the elements within the group. The source term groups used in the present study are shown in Table 5-1.

Table 5-1
Radiological Source Term Groups

Group	Principal Element	Elements Included
1	Noble Gases	Xe, Kr
2	Iodine	I, Br
3	Cesium	Cs, Rb
4	Tellurium	Te, Sb, Se
5	Strontium	Sr
6	Ruthenium	Ru, Rh, Pd, Mo, Tc
7	Lanthanum	La, Zr, Nd, Eu, Nb, Pm, Sm, Y
8	Cerium	Ce, Pu, Np
9	Barium	Ba

Because the thermal, physical, and mechanical conditions are different for each accident sequence, the fractions of material released from the core and ultimately released to the environment are different for each accident. The objective of the radiological source term analysis is to provide a radiological source term for each accident progression bin (Section 4).

5.1 Background

PRAs have traditionally calculated the radiological source terms for a very small number of representative accident sequences using large computer codes or suites of codes. The codes compute conditions within the core, and the interaction of the core with the reactor system and working fluid during the progression of the accident. The codes calculate the quantity of radioactive material released from the fuel and reactor structure, the interaction with the structure, and eventual release from the reactor vessel (RV) or reactor coolant system (RCS). If the fuel debris interacts with concrete after leaving the RV or RCS, there is an opportunity for material to be released during the interaction. There is also an opportunity for engineered safety features (ESFs) or natural processes to remove some of the material from the containment atmosphere. The containment may leak or rupture as a result of pressure buildup or violent events during the accident, and some of the material may be released to the atmosphere.

Some of the codes or suites of codes used in estimating the accident progression and the release of radioactive material to the environment are the Modular Accident Analysis Program (MAAP),[5.1] developed by the U.S. nuclear industry, the Source Term Code Package (STCP),[5.2] developed by Battelle Columbus Division (BCD) of Battelle Pacific Northwest Laboratory (BPNL) for the NRC, and MELCOR,[5.3] which is currently under development by SNL for the NRC. Each of these code suites calculates somewhat different fractions of radioactive material released because of differing models, approximations, and assumptions.

In addition to code suites which calculate the entire accident progression, special purpose codes have been developed for a more detailed calculation of specific phases of the accident. For example, the MELPROG[5.4] code provides a detailed mechanistic calculation of conditions within the RCS before vessel failure, and CONTAIN[5.5] makes detailed and mechanistic calculations of conditions within containment. Because of the detail involved, the special purpose codes are often heavy users of computer time and resources and are thus not well suited to the calculation of many accident sequences; in addition, the special purpose codes focus on a small area and are not intended for an overall look at the entire accident. However, the special purpose codes are used for comparison and checking the results of the more general, faster running accident progression codes.

The NRC decided early in the NUREG-1150 analysis to use the STCP as a basis for source term estimates. The reasons for the choice of the STCP were availability (MAAP was not available to the NRC or its contractors, and MELCOR had not yet been fully developed), relatively low consumption of human and computer resources (MELPROG and CONTAIN require more computer time and more detailed modeling), familiarity (BCD had considerable experience in running the codes), and previous knowledge of both the capabilities and limitations of the codes.

The STCP uses the MARCH2/MERGE[5.6,5.7] codes for calculating the thermal hydraulic conditions within the RCS and containment, CORSOR[5.8] for calculating the release of radionuclides from the fuel in the RCS, TRAP/MELT[5.9] for transport and deposition within the RCS, CORCON[5.10] for the core-concrete interactions (CCI), VANESA[5.11] for the release of radionuclides in CCI and their possible recapture by overlying water, and NAUA[5.12] for transport and deposition within and release from containment.

The STCP accounts for many phenomenological and data uncertainties. Some of the uncertainties can be investigated by using alternate models provided within the codes, and for others alternative input data sets are available. If the STCP were to be used for investigating uncertainties, a separate STCP run would be required for each model or data uncertainty. Schedule and funding constraints prohibited the use of the STCP for many multiple runs for each accident. In addition, there are many phenomena for which the entire possible range of uncertainty is not included in the models in the STCP. For other phenomena no alternative models are provided. The STCP does not have any model at all for some phenomena; an example is revolatilization of material initially deposited within the RCS. Furthermore, the source terms calculated by alternate runs of the STCP would not indicate which of the alternative source terms was more likely.

Another reason for finding some alternative to the STCP was the large number of accident progression bins. It was not possible to determine which accident progression bins were most important until after the entire risk analysis had been completed. The STCP could not be used for all accident progression bins because of the heavy usage of human and computer resources. An attempt to determine the most important bins in advance, and determine the uncertainty in those by the STCP would have been practically impossible; some important sequences would have been omitted and the power of the STCP would have been wasted on some sequences of trivial importance. It is also likely that some unimportant sequences for one set of phenomenological or data assumptions will be more important if another set of assumptions is chosen.

In order to account for uncertainties and to be able to provide source terms for a large number of accident progression bins, an alternative to the STCP was required. The substitute model was required to calculate source terms which could closely mimic those calculated by the STCP or any other suite of source term codes, be very fast running, be easily understood, be capable of representing uncertainty even for phenomena not included in the STCP, and be operable in a sampling mode. The solution adopted was a parametric model, in which each parameter stands for a module of the STCP. If the parameters are properly chosen, the parametric model would give source terms nearly the same as those calculated by the STCP for the same sequence. In addition to parameters adjusted a posteriori from STCP runs, other parameters represent phenomena not considered by the STCP. The model adjusts the values of parameters for different accident conditions, and can thus extrapolate STCP results to other sequences. An implicit assumption of this operation is that coupling between parameters either does not exist, or can be adequately represented by the parametric model. Each of the parameters of the model can be varied across its expected uncertainty range. If the coupling between parameters is not important, the source terms calculated will then reflect the range of uncertainty in source terms due to the uncertainty in the parameter.

It is important to understand that the parameters represent a sort of response function for a physical process, that is, the radioactive material output from a process for a given amount of material input to the process. The parameters do not represent physical constants or quantities. No assumption needs to be made about the phenomena involved in the process. Because all accident progression and source term codes represent the same types of processes (although often using distinctly different models), the parametric models are not limited or necessarily closely tied to the STCP. If the parameters were given values appropriate for MAAP, for example, the parametric model would calculate source terms similar to those produced by MAAP. If the parameters were varied across the range of values for any code, then the outputs would represent the source terms obtainable from all possible combinations of codes.

It is also important to understand that the parametric models are unable by themselves to avoid impossible or unrealistic combinations of parameters.

Improper combinations must be avoided by scrupulous attention of the user to the allowable combinations and avoidance of forbidden combinations.

The parametric models developed for the draft report were quite simple. The values of the parameters were very closely tied to the STCP; parameter values for some sequences were simply "hard-wired" into the model. The expert panel that chose the range of parameters and their distribution also tied their estimates very closely to the STCP. The experts themselves were from the NRC and its contractors and were familiar with the STCP but were less familiar with other codes such as MAAP. Consequently, the output of the parametric models used in the draft report tended to represent variability or uncertainty in what could be expected from the STCP if other phenomena had been represented and other models included.

In contrast, the expert panels for the present study were recruited from the NRC, the DOE national laboratories, nuclear industry, universities, and independent contractors. Many of the experts were familiar with methods other than the STCP. The experts were not expected to follow the STCP, nor were the parameters necessarily to be tied to the STCP. Each expert was encouraged to develop an individual method for estimating the range and distribution of the parameters. The parametric models and their source term outputs as used in the present study are thus not specifically linked to the STCP. Some combinations of parameters would give STCP-like results, but other combinations would give MAAP-like results. Still other combinations would give source terms different from those calculated by any current codes, but which in the opinion of the experts were possible, or at least could not be ruled out.

5.2 Interface With the Accident Progression and Containment Analysis

The accident progression bins described in Section 4 are the principal vehicle for passing information from the sequence and containment analyses to the source term parametric models. The description of the accident progression bins is necessarily highly plant-specific. Some characteristics which are common to all plants are the timing and nature of containment failure, the operation of ESFs, conditions within the RCS, the degree of zirconium oxidation, the presence or absence of water, and conditions under which CCI occurs.

The number of possible attributes for the accident progression bin and the number of possible outcomes under each attribute is only limited by the complexity of the containment event trees. The containment event trees used in this study are very complex, and an enormous number--many millions--of accident progression bins could be produced. However, many of the theoretically possible accident progression bins would be expected to give identical or very similar source terms. In order to reduce the number of source terms to a tractable number, the number of attributes and outcomes was limited to those which could be expected to have a marked effect on magnitude of the source terms. A further limitation of the accident progression bins was imposed by the simplicity of the parametric models. There would be no advantage in passing fine nuances of the accident progression which could not be recognized by the parametric models. Another limitation on the number of accident progression bins was allowable because of the subsequent use of the source terms calculated by the model.

Because the source term space, which might contain a large number of individual source terms, was later to be partitioned or clustered into a relatively small number of source terms for the consequence analysis (Section 5.6), the binning was not required to represent very fine gradations of accident outcome. Also, some quantities were limited in the number of allowable outcomes because of the limitations imposed by previous panels. For example, the size of containment failure actually is a continuum over all possible hole sizes. However, the structural experts had limited their consideration to only a few possible sizes and types of failure. Also, the containment event trees only pass bins to the parametric model whose frequencies exceed some lower cutoff, which further limits the number of bins.

As a result of these limitations, the number of possible accident progression bins was reduced to a few thousand for each plant. However, this represents a marked difference from all previous PRAs, in which source terms were only calculated for a very few of the nearly infinite number of possible outcomes. It is also different from the draft version of NUREG-1150, in which source terms for some plants were only calculated for a few possible bins.

It should be recognized that even the restricted number of accident progression bins may represent some scenarios not envisioned by the members of the source term expert panels when the distributions of the model parameters were assessed. It may be that yet other previously unseen limitations or extensions should have been imposed on the parameters.

The Latin Hypercube Sampling (LHS) program selects a unique combination of model parameters for each sample member. Based on combinations of frequency and containment event tree parameters, the containment event trees pass a limited number of bins for the sample member. The parametric model then calculates a source term for each bin, using the parameter values selected for the sample member. The bins represent the uncertainty in the front-end and containment event tree, and the source terms for each bin indicate the uncertainty in source term phenomenology. A frequency for each bin has been calculated by the preceding sequence frequency and containment event tree analyses, and taken together over the entire sample the frequencies and magnitudes of the source terms represent all the uncertainty in the source terms.

5.3 Summary of Source Term Code Package Calculations

The STCP was used to calculate a few sequences for each plant. It had been hoped that the sequences chosen for calculation would be among those most important for risk. During the course of the study, it became obvious that there were many questions still to be answered, and many gaps in the calculations. Additional sequences were then calculated in an attempt to fill those gaps. Table 5-2 shows the sequences that have been calculated for each plant.

Table 5-2
Summary of Source Term Code Package Calculations

Sequence	Source	Sequence Description
	SURRY PLANT	
TLMB'	BMI-2104	Station blackout with failure of auxiliary feedwater
S2D	BMI-2104	Small break LOCA (0.5 to 2.0 in. equivalent diameter) with failure of ECCS injection.
V	BMI-2104	Interfacing systems LOCA
AB	BMI-2104	Large break LOCA with station blackout
TMLB'-β	BMI-2139	Station blackout with failure of auxiliary feedwater and failure of containment isolation
S2D	BMI-2139	
AG	BMI-2139	Large break LOCA with failure of containment heat removal, consequent containment failure and failure of recirculation due to flashing
S3B	BMI-2160	Station blackout with reactor coolant pump seal LOCA
	SEQUOYAH PLANT	
TMLB'	BMI-2104	Station blackout with failure of auxiliary feedwater
TML	BMI-2104	Transient with failure of ECCS
S2HF	BMI-2104	Small break LOCA with failure of both ECCS and containment heat removal in recirculation
S3B	BMI-2139	Blackout with reactor coolant pump seal failure
S3HF	BMI-2139	Reactor coolant pump seal failure with failure of ECCS and containment heat removal in recirculation

SEQUOYAH PLANT (Continued)

Sequence	Source	Sequence Description
TMLU	BMI-2139	Steam generator tube rupture without makeup; no other containment failure
TBA	BMI-2139	Blackout with failure of auxiliary feedwater and temperature-induced large hot-leg break
S3B	BMI-2160	Blackout with reactor coolant pump seal failure and secondary depressurization
S3H	BMI-2160	Reactor coolant pump seal failure with failure of ECCS in recirculation
S3HF	BMI-2160	Reactor coolant pump seal failure with failure of ECCS and containment heat removal in recirculation

PEACH BOTTOM PLANT
(Someone needs to check these sequences)

Sequence	Source	Sequence Description
AE	BMI-2104	Large LOCA with failure of injection
TC	BMI-2104	ATWS
TW	BMI-2104	Failure of containment heat removal
TC_1	BMI-2139	ATWS, low pressure
TC_2	BMI-2139	ATWS, high pressure
TC_3	BMI-2139	ATWS, high pressure, venting
TB_1	BMI-2139	Long-term blackout, containment fails late
TB_2	BMI-2139	Long-term blackout, containment fails at vessel breach
V	BMI-2139	Interfacing systems LOCA
TBUX	BMI-2160	Short-term blackout

GRAND GULF PLANT

Sequence	Source	Sequence Description
TC	BMI-2104	ATWS
TPI	BMI-2104	Failure of containment heat removal

GRAND GULF PLANT (Continued)

Sequence	Source	Sequence Description
TQUV	BMI-2104	Transient with failure of ECCS
S2E	BMI-2104	Small break with injection failure
TC	BMI-2139	ATWS
TB_1	BMI-2139	Long-term blackout with failure of RCIC and ADS, late containment failure
TB_2	BMI-2139	Long-term blackout with failure of RCIC and ADS, containment fails at vessel breach
TBS	BMI-2139	Fast blackout with failure of injection
TBR	BMI-2139	Fast blackout with no ESFs, containment fails late by hydrogen detonation

The STCP was being revised, extended, and improved during the time these calculations were being carried out. Consequently, the earliest calculations were performed by a version of the STCP that was less well developed than the latest calculations. An example is the in-vessel release of ruthenium. A more advanced version of the STCP gave values of ruthenium release which were orders of magnitude lower than did the earliest version. In the draft version of NUREG-1150 the parameter values were specifically linked to STCP runs. Some experts may not have been entirely familiar with all the implications of the changes made to the STCP during the course of the study. In the current study, the experts were encouraged to make use of all available information, from the STCP, MAAP, MELCOR, experiments, and any other sources, and to integrate the evolution of understanding of source term phenomenology into their distributions. The experts discussed their sources of information among themselves before being elicited on their distributions. The parameter distributions thus represent the experts' judgments as to the relative accuracy or worth of all of the sources of information available to them, including the evolution of their understanding over time.

5.4 Development of Parametric Models

Although the details of the models differ slightly from plant to plant, all of the models are of the form:

```
ST(i) = FCOR(i)*FVES(i)*FCONV/DFE +
+ FPART*(1.-FCOR(i))*FCCI(i)*FCONC(i)/DFL(i) +
+ FPME*(1.-FCOR(i))*FDCH(i)*FCONV +
+ Special terms
```

where

FCOR(i) is the fraction of nuclide i relesed from the fuel in-vessel;

FVES(i) is the fraction of nuclide i released to the RCS which is released from the RCS before or immediately after vessel breach;

FCONV is the fraction of material (except noble gases) released to the containment from the RCS which is released from the containment, not considering the effects of ESFs;

DFE is a decontamination factor for ESFs which applies to material released from the RCS;

FPART is the fraction of the entire core which participates in core-concrete interaction;

FCCI(i) is the fraction of nuclide i participating in core-concrete interaction which is released to the containment;

FCONC(i) is the fraction of nuclide i released to the containment in core-concrete interaction which is released from containment, not considering the effects of ESFs;

DFL(i) is a decontamination factor for ESFs applicable to releases of nuclide i due to core-concrete interaction;

FPME is the fraction of core involved in high pressure melt ejection;

FDCH(i) is the fraction of nuclide i in material involved in high pressure melt ejection which is released to the containment, and the special terms are plant-specific releases such as late revolatilization or late release of iodine from suppression pools. As an example of the special terms the late revolatilization term for PWPs is:

(FCOR(i)*(1.-FVES(i)) + (1.-FCOR(i))*FREM)*DLATE(i)*FCONRL(i)/DFL(i)

where

FREM is the fraction of the core remaining in the RCS after vessel breach;

DLATE(i) is the fraction of nuclide i still in the RCS following vessel breach which is later revolatilized and released to the containment;

FCONRL(i) is the fraction of late revolatilized material which would be released from containment not considering the effects of ESFs, and other quantities are as previously defined.

Another example of plant-specific customization is given by steam generator tube ruptures at PWRs. The first term in the parametric equation is modified to be:

FCOR(i)*(FISG(i)*FOSG(i) + (1-FISG(i))*FVES(i)*FCONV/DFE)

where

FSIG(i) is the fraction of nuclide i released from fuel which enters the steam generators, and

FOSG(i) is the fraction of nuclide i entering the steam generators which is then released from the steam generators to the environment. Other symbols are as previously defined.

The basic flow diagram for the most simple form of the parametric model is shown in Figure 5-1. The initial core inventory enters at the left center. At the box marked FCOR some is released from the fuel, and a fraction (1-FCOR) remains in the fuel. The fraction released from the fuel enters the box marked FVES. A fraction (1-FVES) is retained in the reactor vessel, and the fraction FVES is released. At the box marked 1/DFE ESFs remove all but the fraction 1/DFE, and at the box marked FCONV a fraction of what remains escapes to the environment. This constitutes the RCS release.

Below the box marked FCOR the fraction (1-FCOR) represents the material which was not released from the fuel in the reactor vessel. On the downward path, a fraction FPME of the fuel is ejected from the vessel at high pressure. A fraction FDCH is released to the containment atmosphere, and a fraction FCONV of that is released to the environment. This constitutes the direct heating release.

The material which was not ejected at high pressure is 1-FPME. A fraction of that, FPART, participates in core-concrete interaction. The fraction FCCI of that is released from the debris. ESFs permit a fraction 1/DFL to escape, and a fraction FCONC of that escapes to the environment. This constitutes the core-concrete release.

The material remaining in the RCS--FCOR*(1-FVES), and that which was never ejected from the vessel (not shown in Figure 5-1) is available for later revaporization.

It is necessary to point out again that no chemistry or physics is involved in the parametric equation, other than conservation of mass. All of the knowledge of chemistry and physics is involved in choosing values for the parameters such as FVES, FCOR, etc. If these parameters are correctly chosen for any scenario, then the source term calculated by the parametric model will be correct. On the other hand, the source terms are only as good as the choice of parameter values, and if the model attempts to calculate source terms for a scenario for which the parameter value is inappropriate, the source term is incorrect. The model attempts to prevent some errors of extrapolation by making the parameters scenario-dependent. That is, the values of the parameter will be different for different scenarios. For example, for PWRs the value of FVES is different depending on the RCS pressure and the type of accident. The distributions for FVES provided by the experts are different for each scenario. These scenario details are passed to the parametric model through the accident progression bin. When the model receives, for example, "high-RCS pressure, LOCA," a

Figure 5-1. Basic Flow Diagram for Parametric Models.

value for FVES is chosen from the distribution for that scenario. Consistency between the containment event trees and the parametric models is enforced by passing important values used in the containment event trees to the parametric model to be used in calculating the source terms.

The evaluation of the parameters for the draft version of NUREG-1150 was actually simplified because the parametric model was so closely linked to the STCP. Each parameter in the model can be related to a specific module in the STCP. FCOR corresponds to CORSOR, FVES to TRAP/MELT, FCCI to CORCON/VANESA and FCONV, FCONC, DFE, DFL to NAUA. The physical processes behind these modules would be included in any source term release code, but the identification between parameters and code modules might not be so transparent. For the current study the parametric model is not specifically linked to any code, so that the model parameters can be more freely chosen. However, the source term experts (even those more familiar with other codes than with the STCP) appeared to have little difficulty in relating source term phenomenology to the parametric model.

The actual evaluation of the parametric equation is trivially simple. However, assessing the distribution to be used for each scenario, choosing the decontamination factors corresponding to the actual operation of ESFs, and making certain that all the core material is correctly accounted for is more complex as well as being extremely plant specific. A set of codes called the XXXSOR codes[5.13] has been developed for this task. A distinct code had to be written for each plant, and the XXX designates which plant the code applies to. SUR represents Surry, SEQ is Sequoyah, PB is Peach Bottom, and GG is Grand Gulf.

Each of the XXXSOR codes contains the information for source term timing. Generally, this information is "hard-wired" into the code. There is a wealth of information on release timing from MAAP, MELCOR, and the STCP. The information passed in the accident progression bin is usually adequate to be able to recognize an analogous sequence which has been calculated by one of the codes. The XXXSOR code then takes approximate timing information from the tabulation of previously calculated release times. (If there is a conflict between codes, an average is used.) If the analog to the sequence being considered is inappropriate or if the choice of containment event tree parameters would have altered the timing from what was previously calculated, the XXXSOR estimates of release time could be in error. It is important to understand that just as the adequacy of source term magnitudes is dependent on proper choice of parameter values, the release timing is subject to the correct interpretation of previously calculated accident scenarios.

The accident progression bins contain scenarios for which no calculation has ever been performed, however. For example, an accident in which the debris leaving the vessel is initially cooled by water, which is not replenished and eventually boils away, would have CCI delayed relative to a sequence in which CCI is initiated promptly. However, it is not difficult to calculate approximately the time at which the water would be boiled off, which can then be related to the time at which the core would exit the vessel. Thus an approximate time for initiation of CCI can be used, even though the specific scenario may never have been calculated.

Other timing distinctions are provided by the breakdown into RCS release, CCI release, and late revolatilization release. The early and late releases have distinct components, and this distinction (along with the time of initiation and duration) for each component is part of the information provided by the XXXSOR codes.

The most important timing parameter is the warning interval--the time interval between warning the populace to evacuate and the actual release. The warning interval is highly uncertain and variable, being dependent on the ability of the plant personnel to diagnose the accident and to warn the local authorities, and the speed with which the authorities warn the populace. Because this very important time interval is so poorly known, it has not seemed appropriate to attempt a more accurate determination of other less important times.

5.5 Quantification of the Parameters in the Codes

All of the parameters were considered to be uncertain quantities (for a more extensive discussion of uncertainty see Section 7). Each parameter is represented by a probability distribution. For each sample member a specific realization of each parameter is chosen from each distribution. The sampling scheme will not repeat any specific parameter value. However, the frequency with which values in the neighborhood of other values are selected depends on the probability distribution assessed by the experts. For example, in a normal distribution more values will appear in the neighborhood of the mean than at the extremes. In fact for all of the distributions chosen by the experts, the extreme values are only sparsely represented in the sample. The values chosen by the sampling scheme are then randomly combined, unless the experts have suggested that certain values of one parameter should be combined or correlated with certain values of other parameters. For example, the sampling scheme could combine high values of FCOR with high values of FVES. However, because the extreme values of a parameter are rare in the sample, combinations of the extremes are rarer still, and most sample members will have middle values of either FCOR or FVES or both.

Distributions for the most important parameters were assessed by teams of experts. However, the number of parameters in the models and the need to determine plant- and scenario-specific distributions for most parameters made it practically impossible to have the expert teams determine the distributions for every scenario of each parameter. The parameters were prioritized according to their importance in the draft version of NUREG-1150, the degree of interest within the reactor safety committee, the adequacy of the distributions prepared for the draft version, whether the parameters were global or applied only to a single scenario, and the range of uncertainty in the parameters. The expert panel was then invited to add to or subtract from the list of parameters considered. The parameter list finally decided on by the experts was FCOR, FVES, FCONV, FCCI, FCONC, DLATE, late iodine release from the suppression pool at Grand Gulf, and reactor building decontamination factor at Peach Bottom. The list did not include every possible case for every parameter. For example, FCONV and FCONC for late leaks at PWRs were not evaluated by the expert panel, because they were expected to be so low compared to the values for early failure as to have little impact on risk. On the other hand, for some parameters the experts increased the number of cases to be considered.

A complete description of the expert elicitation process can be found in Section 7. For the purposes of this section the reader only needs to understand that the parameter distributions were the product of several experts who had different affiliations but had a common expertise and background in source term calculation. The experts often disagreed substantially with each other on the range and distribution of the parameters. These differing distributions were aggregated by averaging, so that each expert's viewpoint was represented in the aggregate distribution.

Each parameter was subdivided into cases for presentation to the expert panel. A "case" is a set of initial and boundary conditions which could be an important determinant of the value of the parameter. For example, "high prior zirconium oxidation and water present" is one case for the parameter FCCI, and "high prior zirconium oxidation and water absent" is another case. The number and description of cases varies from one parameter to another. The expert panel considerably expanded the number of cases for some parameters. The XXXSOR codes were then specifically tailored to the final case structure as approved by the expert panel, so that every case desired by the experts would be correctly included.

The expert panel was unable to consider every case for every parameter, so that values for some parameters and some cases had to be determined internally. The distributions used in the draft version of NUREG-1150 were used when it appeared that these would be adequate (the discrete distributions of the draft version were first converted to continuous distributions). In other cases, a value for one quantile (for example the median) would be suggested by a STCP or other calculation. A distribution was used which had the desired quantile value along with the distribution for some closely related parameter. In still other cases the distribution was found to be unimportant; for example, for some cases the value of the parameter was known to be very close to zero and any small value would give essentially the same risk. For these cases a constant value was chosen for the parameter.

Each accident progression bin contains all the information necessary to uniquely determine the appropriate case for each parameter. The distributions for every case of every parameter are available to the XXXSOR codes. For each sample member, the containment event tree passes the set of applicable accident progression bins and the LHS program passes the desired fractiles for each parameter. The XXXSOR code determines the appropriate case for each parameter for each accident progression bin, determines the value of the parameter from the stored distributions and the desired fractile, and evaluates the source term Equation 1. The output from the XXXSOR code consists of the fraction of the initial inventory of the nine nuclide groups listed in Table 5-1 along with the timing information (Section 5.5).

A numerical example using the distributions produced by the experts combined with an actual LHS sample member will now be given. The example is for the Surry plant. Only a single accident progression bin will be used for simplicity, but the reader should understand that each sample member would have many accident progression bins.

The example accident progression bin is DHAAACAABCA, which decodes to:

Containment failure at the time of vessel breach
Sprays never operate
Prompt, unscrubbed CCI
High pressure in RCS before vessel breach
Vessel fails by high pressure melt ejection
No steam generator tube rupture
Large amount of core in CCI
Low Zr oxidation before vessel breach
Moderate fraction of core ejected at high pressure
Containment failure is a leak
There is only one large hole in the RCS after vessel breach

The fraction of core which actually leaves the vessel is hard-wired at 95% for this plant. A "moderate" fraction of the core ejected at high pressure is taken to be the median of the in-vessel experts distribution, i.e., 0.265. However, this must be multiplied by the fraction actually leaving the vessel, so that 0.252 is the fraction involved in high pressure melt ejection and 0.050 is the fraction which never leaves the vessel. This entire amount, 0.252 + 0.050 = 0.302 does not participate in CCI. A "high" fraction in CCI is considered to be all the core available for CCI, that is 1.000--0.302 = 0.698.

The LHS sampler asks for the following fractiles to be selected:

FCOR	FVES	V-DF	FCONV	FCCI	FCONC
0.607	0.999	0.619	0.015	0.695	0.420
Spray DF	Late I	DLATE	FDCH	Pool DF	FISG
0.850	0.942	0.191	0.670	0.528	0.771

Because the accident progression bin is not a V-sequence, and there are no sprays, and there is no pool scrubbing of CCI, and there is no steam generator tube rupture, the fractiles for V-DF, Spray DF, Pool DF, and FISG are ignored. Because there are no sprays or pool scrubbing, both DFE and DFL are 1.0 for all nuclides, and can be ignored. The LHS program asks for the 60.7th percentile of the distribution for FCOR, the 99.9th percentile of the distribution for FVES, the 1.5th percentile of the distribution for FCONV, etc.

The selection of values for one parameter (FCOR) will be shown in detail. For simplicity, only the results will be shown for other parameters.

From the experts' distribution for FCOR, for low previous zirconium oxidation, the 50th and 75th percentiles are:

Group	50th Percentile	75th Percentile
NG	0.90	1.00
I	0.69	0.90
Cs	0.59	0.83
Te	0.20	0.46
Sr	0.0064	0.027
Ru	0.004	0.013
La	0.002	0.012
Ce	0.0001	0.00095
Ba	0.00015	0.0025

The fractile desired is 0.607, so by interpolation between the fractiles 0.500 and 0.750, the values for FCOR are:

Group	FCOR
NG	0.94
I	0.78
Cs	0.69
Te	0.31
Sr	0.015
Ru	0.0073
La	0.0063
Ce	0.00046
Ba	0.0012

The early release--that which takes place near the time of vessel breach--is composed of the RCS release component and the direct heating (high pressure melt ejection) component. The RCS release component is FCOR(i)*FVES(i)*FCONV. At the 99.9th percentile, the value of FVES is 1.00 for every nuclide group. The value for FCONV is the 1.5th percentile of the experts' distribution for early leak with containment dry (no sprays operate), which is .0025. The products of these values give the RCS component of the release:

Group	RCS Release
NG	0.940
I	0.0020
Cs	0.0017
Te	7.8E-04
Sr	3.8E-05
Ru	2.0E-05
La	1.6E-05
Ce	1.2E-06
Ba	3.0E-06

The direct heating component is the fraction of core involved in high pressure melt ejection times (1.0-FCOR) the 67th percentile of the experts' distribution for FDCH for a high-pressure sequence times FCONV. the products of these values are:

Group	DCH Release
NG	0.015
I	1.3E-04
Cs	1.9E-04
Te	1.1E-04
Sr	1.2E-05
Ru	3.9E-05
La	8.8E-06
Ce	8.8E-06
Ba	2.0E-05

The total early release is the sum of the RCS release and the direct heating release:

Group	Early Release
NG	0.955
I	2.1E-03
Cs	1.9E-03
Te	8.9E-04
Sr	5.0E-05
Ru	5.9E-05
La	2.5E-05
Ce	1.0E-05
Ba	2.3E-05

The late release is composed of the CCI release, the revolatilization release, and a late release of iodine by conversion of iodine remaining in containment to organic iodides. The calculation of these late releases is very similar to the calculation of early release and does not need to be detailed here. An interesting feature of this particular sample member is that FCONV is very small and the factor for conversion to volatile iodides is quite large. Most of the iodine released in vessel remains in containment, and a sizable fraction is converted to volatile iodine, which is assumed to escape containment without holdup. The result is that for this sample member, organic and other volatile iodides make up the largest part of the iodine release. This is an unusual situation which is probably unique to this sample member.

The most important points to remember for this example are that none of the parameters came from the parametric model (except for the 5% of fuel which is assumed to remain in the RCS); all of the parameters came either from the containment event tree or from the distributions given by the experts. The fractile of every distribution to be used was chosen randomly by the

LHS program. A different fractile was used for every parameter in the sample member, and also the fractiles would be different from one sample member to another.

5.6 Partitioning of the Source Terms for Consequence Analysis

(To be written by Jon Helton)

5.7 Verification of the Parametric Model

The parametric model does not represent any physical effects for conservation of mass. Therefore, beyond the rather trivial task of determining that mass is indeed conserved, verification in a strict sense is neither possible nor required. However, it is necessary to know whether the parametric model can be relied on at all. Moreover, the results of the parametric models will be extrapolated across both sequences and cases. It is necessary to know to what extent the extrapolation can be relied on.

If the experts are perfectly knowledgeable, the parametric model should be able to produce their intended source terms using their choices for the parameters. This would be a test of the ability of the parametric models to reproduce a desired source term from known or estimated values of the parameters, for a specific case. If the parametric models could be exactly extrapolated, then values found to be appropriate for one sequence would be exactly correct for another sequence. Unfortunately, the number of STCP runs for each plant was too limited to be able to perform either test separately. However a combined test of the model's ability to reproduce known or desired source terms and also the validity of extrapolation has been performed by Battelle Columbus Division (BCD).[5.14] The Surry plant was chosen for the test because more STCP runs were available for this plant than for any other. The values of the parameters were selected to match the STCP for some runs. It should be understood that the value of a parameter back calculated from one run might not match the value calculated from another run, both because the STCP can recognize differences between sequences which are not recognized by the parametric model, and because the STCP was undergoing revision during the time that the runs were being made. The conflict was usually resolved by using parameter values appropriate to the run most representative of the accident progression bins expected for the LHS study. In some cases several values were available, and the value most representative of the ensemble was used. Some of the conflicts were caused by changes to the STCP; values appropriate to the earliest version were used for some plants for which most of the STCP runs were carried out with the earlier version.

Representative results of the test study for Surry are shown in Figures 5.1 through 5.--. Generally, the parametric model predicted the releases of I, Cs, and Te quite well, with some exceptions. The releases of Ru matched STCP predictions for runs in which the earlier version of the STCP was used. The later version of the STCP predicted Ru releases several orders of magnitude lower than did the parametric model. Releases of I, Cs, and Te predicted by some runs of the STCP were lower than the predictions of the parametric model for runs in which the STCP used a mean spray drop diameter of 400 microns, but were in reasonable agreement with the STCP if a mean drop diameter of 1000 microns was used. The releases of Sr and Ba agreed with the STCP predictions in some cases and disagreed in others. The agreement for La was not as good as for other nuclides; however, the predicted releases were all quite low. The agreement for some of the more benign sequences is apparently poor. However, the source terms (other than noble gases) for these sequences are extremely low. The consequences are completely dominated by the noble gas release, which is 100% for both the parametric model and the STCP. The apparent disagreement for these very low source terms is therefore not important for risk.

The values used for the parametric model were not corrected to improve the agreement with the STCP, because the STCP values were not used in the NUREG-1150 study. For example, there would be no reason to change the value of FCOR for ruthenium, because the value used in NUREG-1150 always comes from the distribution provided by the expert panel.

An example of the back calculation of parameter values will be given for the Sequoyah plant. For this plant there were enough STCP calculations using the latest version of CORSOR so that values appropriate to the current code version could be determined. A detailed example of the back calculation of parameters from the STCP results will be given for the S3B sequence, reported in Reference 5.17. Results will be summarized for other sequences.

The initial inventory, by group (Table 4.6 of Reference 5.17) is:

Group	Total Mass (kg)
1	347
2	15.2
3	185
4	31.7
5	60.9
6	470
7	684
8	796
9	77.7

Table 5.14 of Reference 5.17 shows masses of radionuclides released from the fuel and retained in the RCS, from which values of FVES and FCOR for groups 1 to 4 can be immediately calculated:

Group	Released (kg)	Retained (kg)	FCOR	FVES
1	336.5	0.0	0.97	1.00
2	14.7	10.1	0.97	0.31
3	178.9	134.4	0.97	0.25
4	26.6	24.3	0.84	0.086

Table 5.13 of Reference 5.17 shows 356.3 kg of aerosols released and 256.6 retained, whence FVES is inferred to be 0.28 for groups 5 to 9. Table 5.15 summarizes release to containment (that is, FCOR*FVES) for all groups, from which FCOR for groups 5 to 9 is calculated to be:

Group	FCOR
5	5.6×10^{-4}
6	9.6×10^{-7}
7	1.25×10^{-7}
8	0.0
9	.0103

Table 5.15 also summarizes the release to containment from core-concrete attack [that is, (1-FCOR)*FCCI)], from which FCCI can be calculated:

Group	FCCI
2	0.18
3	0.27
4	0.22
5	0.032
6	3.4×10^{-6}
7	0.0016
8	0.0012
9	0.020

The information in Reference 5.17 is not sufficient to permit an accurate determination of FCONV AND FCONC. However, 97% of the iodine and cesium are released from the fuel before vessel breach, and 30% of the iodine and 24% of cesium are released to the containment directly from the RCS. The total release of iodine and cesium to the environment must be nearly all from the RCS component of the release, and a value of FCONV can be approximately calculated for groups 2 and 3. This value is then used as a proxy for all groups except the noble gases. FCONC for all groups can then be calculated. Table 5.33 of Reference 5.17 shows that the ice bed decontamination factor is approximately 7. During the periods when there is heavy flow of steam through the ice bed, and approximately 5. overall. A DF of 7 will be used for the RCS release and a DF of 5 for the CCI release. The calculation of FCONC depends on the difference of two numbers total release minus RCS release, which are themselves approximate. The value of DF is also the merest approximation, and the assumption that effective DF is the same for all species may be quite poor. In fact, Te and Ru are released more slowly than other radionuclides, so that the effective DF could be much lower for these if most of the ice had melted before they were completely released. With all these approximations, it does not seem proper to use a different FCONC for each group. The average of the calculated values of FCONC for groups 4 to 9 will be used (FCONC' in the table below):

Group	Release to Environment	FCONV	FCONC	FCONC'
1	1.00	1.00	1.00	1.00
2	0.02	0.50	--	0.74
3	0.017	0.50	--	0.74
4	0.0078	0.50	0.37	0.74
5	0.0063	0.50	0.97	0.74
6	2.2×10^{-7}	0.50	0.32	0.74
7	2.8×10^{-4}	0.50	0.88	0.74
8	2.3×10^{-4}	0.50	0.92	0.74
9	0.004	0.50	1.00	0.74

With these approximations, it is not surprising that the releases calculated with the parametric model do not exactly match the releases as calculated by the STCP. The small discrepancies observed below are typical, especially for groups having very low environmental releases.

Group	Release (STCP)	Release (Model)
1	1.00	1.00
2	0.02	0.022
3	0.017	0.018
4	0.0078	0.010
5	0.0063	0.0047
6	2.2×10^{-7}	5.2×10^{-7}
7	2.8×10^{-4}	2.4×10^{-4}
8	2.3×10^{-4}	1.8×10^{-4}
9	0.004	0.003

The parameter values used in SEQSOR must be applicable not only to the S3B sequence, but to any sequence whatever, although separate parameter values are allowed for low and high Zr oxidation levels. Some representative back calculated values for FCOR are:

FCOR

Group	S3HF	S3B	TMLB'	S2HF
1	0.99	0.97	0.99	0.999
2	0.97	0.97	0.99	0.999
3	0.97	0.97	0.99	0.999
4	0.84	0.84	0.27	0.84
5	0.011	5.6×10^{-4}	0.17	0.15
6	6×10^{-4}	1×10^{-8}	0.069	0.054
7	1×10^{-6}	1×10^{-7}	0.007	0.007
8	1×10^{-7}	0.0	0.0	2×10^{-4}
9		0.0103		

The STCP calculations for TMLB' and S2HF used an older version of CORSOR which appears to have overstated the release of refractories; therefore, those sequences will not be used for groups 5 to 9. The TMLB' scenario had a lower oxidation of Zr, and the value for group 4 is excluded from the average for high Zr oxidation fraction. Because of the very low releases of some of the refractories, only order of magnitude values are used. The values of FCOR used for attempting to match the STCP are:

Group	Low Zr Oxidation	High Zr Oxidation
1	1.00	1.00
2	0.99	0.99
3	0.99	0.99
4	0.99	0.99
5	0.27	0.84
6	0.006	0.006
7	1×10^{-6}	1×10^{-6}
8	1×10^{-7}	1×10^{-7}
9	0.01	0.01

Values for other parameters were back calculated and averaged in a similar manner. The Sequoyah calculations for FVES were supplemented by those for Surry in order to cover all pressure regimes.

BCD also determined the distribution of source terms using the parameter distributions provided by the experts and an LHS sample. The intent here was to determine if the source terms predicted by the STCP would fall within the distributions provided by the experts For bins in which the sprays operate, the STCP results using a 400 micron spray drop diameter were below or at the low end of the distributions from the experts. The distributions for decontamination factor due to sprays was taken from draft NUREG-1150 and had been largely based on a mean spray diameter of 1000 microns. A drop diameter of 400 microns was believed to be more appropriate, and the distributions for spray decontamination factor were widened so that the values corresponding to 400 microns fell near the center of the distribution. At the same time, the values corresponding to 1000 microns were still retained, giving the effect of a very wide distribution with great uncertainty. This change resulted in the runs using both 1000 micron and 400 micron spray diameter falling within the distribution of source terms predicted by the model.

The STCP calculation for the S3B sequence had not been available at the time the parameter values were estimated. The prediction of this sequence thus represents to some extent a "blind" test, that is, the ability of the parametric model to predict the outcome of the STCP in advance.

An older version of the XXXSOR codes (the version used in draft NUREG-1150) was tested by Brookhaven National Laboratory (BNL).[5.15] The study concluded that:

An older version of the SEQSOR code (not, however, the version used in draft NUREG-1150) was tested at SNL against STCP predictions[5.18] of source terms at Surry and Sequoyah. These calculations were not available at the time that this version of the model parameters were developed, and hence were "blind" tests.

Table 5-3
Tests of Parametric Model at SNL

Sequence	I	Cs	Te	Sr	Ru	La
			Surry S3B			
BMI-2160	0.185	0.16	0.061	0.0158	1.2×10^{-7}	8.2×10^{-7}
Model	0.22	0.22	0.083	0.062	1.5×10^{-6}	9×10^{-5}
			Sequoyah S3B*			
BMI-2160	0.38	0.32	0.11	0.12	5.4×10^{-6}	0.0094
Model	0.24	0.24	0.11	0.12	2.9×10^{-6}	0.0062
			Sequoyah S3HF			
BMI-2160	0.013	0.011	0.023	0.13	1.9×10^{-6}	0.008
Model	0.007	0.007	0.05	0.12	2.7×10^{-6}	0.0047

*Not the same S3B sequence as was used in the example of back calculation of parameters.

The tests of the parametric models show that the models can reproduce any given source term with reasonable fidelity. The parameter values can be extrapolated to other cases and sequences; however, the extrapolation may be poor if the two sequences are very different. Only a limited number of blind tests are available; these tests appear to give results as good as when the STCP outcome is known in advance. Many of the instances when the parametric models do not agree with the STCP can be explained by changes to the STCP during the course of the study. The STCP predictions fall generally within the distribution of source terms calculated from the [illegible]s' d[illegible]ibutions for the parameters.

[illegible]t be clearly understood that the parametric models do not attempt to [illegible] the STCP in any part of the NUREG-1150 study. The tests described [illegible] used the STCP calculations only for reasons of availabililty. The comparisons could have been performed using MAAP or MELCOR, or any other code system. Whether the STCP predicts source terms that are consistently higher or lower than those predicted by other code systems is irrelevant. However, the STCP was one of the sources of information used by at least some of [illegible] experts, and therefore the STCP source terms ought to fall at least [illegible] the distribution of source terms predicted by the parametric model.

The reliability of the parametric models can only be measured by their ability to predict source terms consistent with their input values. If all of the possible cases for all of the parameters had been decided by the expert panel, the parametric models would be perfectly reliable, because the experts selected their distributions with complete knowledge of the way their values would be used. The parametric models would then perfectly reflect the opinions of the experts. (Whether the experts' opinions are consistent with reality is currently untestable.) Some of the parameters, and some cases, were not decided by the experts. Furthermore, the case structure is unable to reflect every possible combination of the initial and boundary conditions. If the initial and boundary conditions closely match a case considered by the experts and only the parameters which were provided by the experts apply to the case, then the parametric models should closely match the experts' intentions.

The experts were unable to consider all the possible interactions or correlations between parameters. Some correlations are enforced; for example, the values of FCOR for low prior zirconium oxidation always accompany the values of FCCI for low zirconium oxidation. Some interactions are enforced; for example FCOR appears in the expressions for the RCS release and also for the CCI release, so that the two release components must interact. However, there are probably other correlations and interactions which were not forseen by the experts, or which were assumed to be negligible when in fact they may be quite important. The effect of unknown correlations and interactions is currently untestable.

5.8 References

(This list does not include any of Jon Helton's references)

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7.2 Types of Uncertainties

Uncertainty is defined by The American Heritage Dictionary[7.1] as: (1) the condition of being in doubt; lack of certainty; and (2) something that is uncertain. Bayes[7.2] defined uncertainty as "-------------------------------------". -----[7.3] defined uncertainty as "------------------------------".

The word "uncertain" in the sense of "not known or established; questionable, doubtful" has always been a major part of every scientific endeavor. Every experiment and every analysis has the sole purpose of reducing uncertainty. However, the nature of scientific advancement is such that research has seldom significantly reduced uncertainty; most experiments raise more questions than they answer. In fact, research often raises questions where none existed before. For example, before the Michelson-Morley experiment, there was no uncertainty about the existence of aether--virtually every scientist accepted that the velocity of light must depend on the relative velocities of the emitter and the observer and that light must travel through a medium. After the experiment, most scientists initially questioned the results and conclusions, and the uncertainty became large. Only after many replications of the experiment was the invariance of the velocity of light accepted.

During the period before the Michelson-Morley experiment, every scientist believed himself to be "objective"* in asserting that light traveled through a medium, and that the velocity increased if the source was traveling in the direction of transmission. Excellent, careful, objective analyses were carried out to prove that it was so. The fact is, that the excellent, objective scientists were <u>wrong</u>. After the experiment, subjectivity came into the argument. Whatever choice a participant made depended on his background, attitudes, and personality. His choice therefore was subjective. Those who agreed with Michelson and Morley were right (at least insofar as can be judged from our current knowledge). Subjectivity does not necessarily indicate falsehood, and objectivity does not necessarily guarantee truth. Objectivity depends on the existing state of knowledge; what is to be accepted as "true" at any time is itself a subjective choice.

Although uncertainty in science is inescapable, most scientists make a determined effort to escape from its perceived negative effects. Every analyst attempts to reduce the inherent uncertainty to manageable proportions by making simplifying assumptions about the data, initial and boundary conditions, and phenomenology. The choice of the assumptions is usually quite subjective; however, most honest analysts do not attempt to hide the subjectivity. Unfortunately, the analyst seldom knows the processes involved in the analysis well enough to be able to distinguish

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*The rule or principle of objectivity is that the results of a process should be independent of the viewpoint of the observer. The words "objectivity" and "subjectivity" often have a value-laden and emotional connotation for scientists, thereby giving objectivity a positive value and subjectivity a negative value.

clearly between assumptions that would be self-evident to any observer and those which are endemic to him or her alone. Furthermore, there is no guarantee that objective assumptions are true; an advancement in the state of knowledge may show that objectively chosen assumptions were, in fact, false.

The results of the analysis have the illusion of certainty, but only because the analyst has substituted his/her own opinions for the unknown or doubtful quantities. Another equally competent analyst might well consider a different set of data and initial conditions to be more reasonable. Uncertainty is thus not only inherent in all science, but the resolution of uncertainty is invariably idiosyncratic and subjective. Even the measurement of uncertainty is subjective. What one analyst might consider quite reasonable as representing the range of uncertainty might be totally unreasonable to others.

Uncertainty is sometimes removed by use of the term "best estimate," which denotes that some consensus is used for what is hoped will be the most likely combination of inputs and phenomenology. The "best estimate" is itself highly subjective, and is often actually dangerous for risk analysis. Severe accidents are, by their nature, rare and unexpected events. The greatest risk comes from twists of the accident that are unexpected, not from the best estimate. The so-called "best estimate" may be the most likely outcome for the consensus assumptions but will often present an overly optimistic picture of risk.

Neither analysis nor experiment is likely to remove all uncertainty, and may in fact increase it. However, one can attempt to measure the uncertainty. Because uncertainty is inherently subjective, the measurement of uncertainty must also subsume that subjectivity. There is no guarantee that a true answer will lie within the band of uncertainty which has been measured. (On the other hand, there is no guarantee that an answer arrived at objectively will be true, either.)

NUREG-1150 does not attempt to reduce uncertainty in risk analysis, nor is it an attempt to find a best estimate. This study is rather an attempt to produce an unbiased picture of uncertainty in risk. The study tries to discover the range in risk inherent in the range of plausible assumptions about phenomenology and initial and boundary conditions. The risk corresponding to the most (subjectively) plausible assumptions has a higher likelihood of being accepted by a randomly chosen expert in accident phenomena. The risk corresponding to less plausible assumptions nevertheless has some likelihood of being accepted by any expert, and may indeed be the most acceptable for some experts. Experts are sometimes wrong, and the "true" risk could lie outside the ranges found in this study.

For purposes of this study the word uncertainty has three specific and definite meanings, thus classifying uncertainty into three types: (a) phenomenological, (b) data and (c) stochastic. Each type is explained below by example.

7.2.1 Phenomenological Uncertainty

The first type of uncertainty is encountered when alternative outcomes are possible for an event, depending on which of several rules is applicable. Only one possible rule is applicable, and this rule is unknown. If the rule were known, we would be able to predict the occurrence of the event with complete confidence, because only the possible rule always applies. The event is insensitive to small variations in the initial and boundary conditions; if conditions are such that the event could occur, it will occur. The uncertainty here is with regard to the rules--the phenomenology--to be followed, and this type of uncertainty is called "phenomenological uncertainty" in this study. An example of phenomenological uncertainty is the formation of crusts on molten fuel debris. Some experts consider the formation of substantial crusts (thick enough to impede the flow of gases) to be totally certain under some conditions. Other experts consider the formation of such crusts to be impossible under the same conditions. If the opposing schools of experts could agree on the phenomenology of crust formation, there would be no uncertainty. Either the crusts would always form, or they would never form, depending on which view of crust phenomenology was mutually accepted.

Phenomenological uncertainty can be viewed as a fork in a path. Only one path leads to a correct outcome, but in the present state of uncertainty we cannot be certain which path this is. Phenomenological uncertainty could, in theory, be entirely removed by a single critical experiment.* Once the experiment indicates the correct path to be followed, any person who agrees that the experiment shows what the experimenter claims it to show would agree on the path.

Phenomenological uncertainty can be measured or expressed by neutral betting odds; the odds that a bettor would be equally willing to give or take on the correctness of one of the two possible outcomes of the uncertain proposition. The neutral betting odds can also be expressed as the bettor's subjective probability or degree of belief in the outcome. If a bettor is equally willing to give or take even odds on the outcome, he is absolutely unable to decide which of the two outcomes is more likely. This represents the state of maximum uncertainty, and can be expressed as a probability of 0.5 that an outcome will be correct.

One important point to be understood about phenomenological uncertainty is that the single probability number that represents the betting odds contains in itself all the uncertainty about the outcome. The reason is that no rational bettor is at the same time equally willing to give or take

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*Even "critical" experiments do not immediately remove phenomenological uncertainty. So many questions are raised about the applicability, conduct, limitations, and implications of important experiments that uncertainty does not begin to be narrowed until the experiment can be replicated by independent experimenters.

even odds, odds of 1:3 and odds of 3:1. If a bettor is indifferent as to whether odds are to be taken or given at some specific value of odds, i follows that he would not be indifferent at some other value of odds. Berman[7.4] has shown that attempting to attach a distribution to phenomenological uncertainty is equivalent to betting against oneself. A bettor's entire understanding of the outcome is expressed i single probability fraction. Although an individual's uncertainty is e essed by a single number, other bettors would be likely to express the uncertainty by another number. If several bettors have different probabilities for the truth of a proposition, the single probability which has the greatest chance of being acceptable to all is the arithmetic mean of all the probabilities.

Suppose that an event has possible outcomes A and B, the outcome depends on which rule is applicable. Suppose th t three observers, O1, O2, and O3 determine their neutral betting odds and hence their subjective degree of belief for the correctness of path A. O1 assigns a subjective probability of 0.1, O2 assigns a subjective probability of 0.5, and O3 assigns a subjective probability of 0.9. We now ask which of the three observers is correct. The surprising answer is--all three! The subjective uncertainty is an expression of an <u>internal state</u>--the willingness to give or take odds at a certain level--and as such is a measure of the observer rather than of the event. If the observers are not lying, each one's measure of his own internal state is equally valid. Now suppose that a critical experiment is carried out, and it is determined (for all time) that B is the only possible path. Which observer was correct before the experiment? The answer is, none of them. The only possible <u>correct</u> probability for the occurrence of outcome B is 1.0, and any observer would assign this probability after having viewed (and accepted) the experiment.

The outcome of the phenomenological uncertainty is unaffected by small changes in the initial and boundary conditions. However, some events are affected. For some events the connection to the initial and boundary conditions is so subtle or so complex that an observer is unable to explain the connection. It appears to the observer that the outcome is unpredictable and governed only by random chance. If the observer could observe a great number of trials, he might be able to estimate the frequency with which the event appears to occur, but would never be able to predict the outcome of any specific trial. An example is the tossing of a coin. Most observers have seen enough trials that they are well convinced that the frequency of heads is 0.5. However the initial and boundary conditions of coin tossing are so exceedingly complex that no observer can reliably predict the outcome of any single toss.

7.2.2 Data Uncertainty

Suppose that the observer has available limited number of trials, e.g., ten, and suppose that the observed frequency of heads is four out of ten, or 0.4. There is a possibility that another sample of trials would have given a different frequency of heads, so the observer is uncertain as to the exact long-run frequency of heads. A model or rule is available for estimating the probability of the long-run frequency from a limited number of trials, which is the binomial distribution. Tables are available for the binomial distribution. If the observer has only the limited number of trials to work from, plus the binomial tables, a probability distribution for the "true" long-run frequency can be constructed. For this example, the probability of four out of 10 is approximately 0.25 if the "true" frequency is 0.4; the probability is approximately 0.11 if the "true" frequency is 0.6; the probability is approximately 0.9 if the "true" frequency is 0.2, etc. The observer could now build a probability distribution which would show the degree of belief in various estimates of the true long-run frequency of heads. Every observer who had access to the same test and the same model would produce an identical probability distribution. If the number of tests was few, and the event was rare, the distribution would be very wide and flat. If the event was relatively common and the number of tests was large (but still finite), the distribution would be sharply peaked. Nevertheless, as long as the experimental evidence is finite, there will continue to be uncertainty as to the true long-run frequency.

The type of uncertainty described above is called data uncertainty, because it arises from the necessarily limited data available. An example is valve failure rate. Valve failures are rare, and the data are necessarily scanty. Thus, there exists uncertainty as to the "true" failure rate for any type of valve. However, if data exist for the specific type of valve, and the data are credible, there is no need to convene an expert panel to determine the probability distribution for the failure rate. All experts who might review the failure data would come up with the identical probability distribution. Although this type of uncertainty always exists (especially for rare events), it is not the subject of the uncertainty investigation for NUREG-1150. Every PRA ever conducted has dealt with this type of uncertainty.

Stochastic Uncertainty

Suppose however, that the failure rate data did not exist, or that the failure tests were poorly conducted and not very credible. An expert in the subject of valve failures might still be willing to estimate the failure rate of the valve in question by analogy to similar items of equipment or by some understanding of the likely laws of failure as they applied to this valve. The probability distribution for failure rate would now be much broader than if applicable data were available. The expert would have to account for his subjective uncertainty as to the applicability of the analogous data, or his uncertainty as to which of several competing failure laws might be most appropriate. The expert would probably internalize several such factors, along with his understanding of the variability in initial and boundary conditions, in order to arrive at his subjective probability distribution. The distribution is subjective because another expert faced with the same scanty data and lack of knowledge of the applicability of failure laws could very probably arrive at some different probability distribution. The distribution, then, is a function of the observer's viewpoint and the principle of independence does not hold. This type of uncertainty is called "subjective stochastic uncertainty," or simply "stochastic uncertainty" in NUREG-1150. Although a single critical experiment clears phenomenological uncertainty for all time, a large quantity of data is necessary to clear the stochastic uncertainty. The first experimental outcomes will be viewed by some experts as "wild" or anomalous results, and their subjective probability distributions may not be radically changed. Only after accumulative data converge unequivocally to the same "true" frequency of occurrence will the majority of experts uniformly accept the results.

The differences in the three types of uncertainty are not academic. All three types are handled differently in the sampling process. A phenomenological question with two outcomes A and B will be handled by having some sample members follow outcome A and some will follow outcome B. If the subjective probability of outcome A, as assessed by experts, is 0.10, then 10% of the sample members will follow outcome A, and only A; and 90% will follow B and only B. Sample members having outcome A may be considered as belonging to a universe in which A is the only possible outcome, and those having B belong to a universe in which only B is possible. This type of sampling is referred to as "0/1" sampling, because the probability of following a path is zero for some sample members and unity for others, but never anything in between.

On the other hand, "split-fraction" sampling is used for stochastic uncertainty. Each sample member is split between outcomes A and B, but the fraction of split between outcomes is different for every sample member. Some sample members might have a high fraction for outcome A, and others have a high fraction for outcome B. Each sample member can be considered as belonging to a universe in which the "true" long-run frequency of A is perfectly known, but the well known frequency is different for each.

Data uncertainty is also treated by split fractions. However, the distribution of the split fractions may be narrower, and the opinions of experts were not required for determining the distribution.

The division of uncertainty into types is not as clear-cut as appears from the foregoing explanations. The experts were divided in their opinion as to whether any uncertain issue should be treated as phenomenological or stochastic. Those experts who had a background in probability and statistics tended to view more issues as stochastic than did those having backgrounds in theoretical analysis. The extremes were that some experts believed all questions to be phenomenological and some believed all questions to be stochastic. The division of types is subjective and neither type is exclusively right nor wrong for any question. If some experts believed an issue to be truly phenomenological and others believed it to be truly stochastic, then the resulting aggregated distribution will be a hybrid. Sample members falling within the phenomenological part are sampled "0/1" and those falling within the stochastic part are sampled by split fractions. An example is temperature-induced large hot-leg failures in PWRs. Some experts believed that the event would either always happen or would never happen, and their uncertainty was as to which outcome would be true. Others thought that the event would sometimes happen, but under similar initial and boundary conditions might not happen, and their uncertainty was as to the frequency with which the event would occur. If a sample member falls at either end of the distribution the event will occur with probability zero or one. However, if the sample member falls in the middle of the distribution, the event will have a split fraction for occurrence.

7.3 Scope of the Uncertainty Analysis in this Program

The NUREG-1150 program attempts to show the range and distribution of risk due to uncertainty in the inputs. Some of that uncertainty is phenomenological, some is stochastic, and some is due to limited backgound of data. There are an enormous number of input points, and all are uncertain to some extent. It was thus impossible to treat all questions and issues with the same degree of thoroughness. The criteria used to select issues for detailed uncertainty analysis were:

- A high impact on risk. If an issue was highly uncertain, but variation across its entire range would not cause a large change in risk, there would be little need for a detailed treatment. The likely impact on risk was determined by the outcome seen in the draft version of NUREG-1150, by smaller scale side calculations, by the opinions of the expert panels, and by examination of previous PRAs.

- Interest within the reactor safety community. Some issues were thought not to be major determinants of uncertainty in risk, but had nevertheless been the subject of intense investigation and debate. The reason for including these issues in the analysis was that their relative unimportance could be clearly shown.

- To improve on the treatment in draft NUREG-1150. Some issues had not appeared to be important in the draft version; however, it was recognized that the treatment there was less than optimum. Such issues were included to determine whether an improved treatment would show the same relative lack of importance.

- The issue was uncertain. Even if an issue is important for the magnitude of risk, if the outcome is unquestioned there could be no impact on the uncertainty in risk.

The reader is referred to Section 7.4 for a more detailed and complete discussion of the selection of issues.

The issues meeting any of these criteria were first listed by the NUREG-1150 staff. The preliminary list of issues was presented to a panel of experts, along with reasons for their inclusion. A list of other issues was also presented, along with reasons for their exclusion. The expert panel was then asked to review the list of issues, and to add or delete issues. The expert panels were the same ones which would later be asked for quantification of the uncertain issues. An understanding of the limited time and resources available generally militated against an unwarranted or overly generous expansion of the issues.

At the same meeting, the experts were trained in the quantification of subjective probability distributions (the so-called normative training sessions), and were informed of the method and procedure for elicitation of opinions. This procedure was made as transparent as possible, so that the experts would know how their subjective distributions were to be used.

The issues selected for investigation were then debated within the expert panel. Open testimony by other interested parties was presented to the experts. A large collection of research reports bearing on the issue was given to the experts for more intensive study.

After a period of several weeks, during which the experts thoroughly familiarized themselves with the substance of the issue and in many cases carried out further research or calculations themselves, a second technical meeting was held. At this time the experts presented their views on the technical substance of the issues. Some of the experts attended other technical meetings of the panel, often without the presence of the NUREG-1150 staff.

At a final meeting, the experts described their understanding of the issue in qualitative terms, without giving any quantitative information to the other experts. After all experts had an opportunity to speak on an issue, each expert's quantitative distributions were privately elicited in the presence of a normative expert (one expert in the elicitation of opinion) and a substantive expert (one knowledgeable about the issue itself). The reader should refer to Section 7.6 for a more detailed explanation of the process of issue selection and opinion elicitation.

The procedure was too cumbersome and time consuming for more than a handful of issues in any area. Issues which were not quantified by the expert panels were then quantified internally. For some of the issues, the quantification used in the draft version of NUREG-1150 was found to be adequate. The remaining questions for internal quantification were prioritized using the same criteria as for selection of issues for the expert panels. The most important remaining questions were referred to panels of experts within SNL, who were then elicited in a similar procedure as were the external experts. Less important questions were quantified by a panel of the NUREG-1150 plant analysts, and the least important were quantified by individual analysts.

It was not necessary to undertake a new quantification for every question. For example, plant-specific data on recovery of offsite power was readily available, and was used in the containment event trees as well as in the sequence frequency analysis. Site-specific weather and demographic data were available for each of the plants studied, and these data were used in the MACCS code for consequence analysis. Some questions were administratively determined to be beyond the scope of the study, particularly the health effects models used in the MACCS code. The sequence frequency analysis used a wealth of data on component failure rates, both industry-wide and plant specific; these data were used the same as in any other PRA. For some questions there was only minimal uncertainty as to the outcome. For example, if an equipment failure had caused some engineered safety system to be inoperable at the beginning of the accident, and there was no chance of repair or replacement during the time frame of the accident, the system was considered to be inoperable throughout the accident.

7.4 Definition of Specific Variables for the Uncertainty Analysis

Those issues which were selected for quantification by the external expert panels fell into three broad classes--uncertain issues affecting primarily the sequence frequency calculation, uncertain issues affecting primarily the response of the containment and its systems, and uncertain issues affecting primarily the radiological source term. There were more issues affecting containment than for the other classes, and there was a further breakdown into issues related to the in-vessel phenomenology, containment loads, structural response, and molten core-concrete interactions. Tables 7-1 through 7-6 show the issues presented to each of the external expert panels, along with the reasons for including the issue.

Table 7-1
Issues Presented to Sequence Frequency Panel

Issue No.	Title	Reason for Inclusion

Table 7-2
Issues Presented to the In-Vessel Panel

Issue No.	Title	Reason for Inclusion
1	Temperature-induced PWR hot leg failure	Large hot leg failure could preclude direct containment heating; depressurizes RCS and precludes SGTR
2	Temperature-induced PWR SGTR	SGTR gives direct path to environment, with large release of radionuclides
3	In-vessel hydrogen production in BWRs	Hydrogen burning has potential for causing release to environemnt
4	Temperature-induced bottom head failure in BWRs	Mode of bottom head failure determines subsequent accident progression
5	In-vessel hydrogen production in PWRs	Hydrogen burning has potential for causing release to environemnt
6	Temperature-induced bottom head failures in PWRs	Mode of bottom head failure determines subsequent accident progression

Table 7-3
Issues Presented to the Containment Loads Panel

Issue No.	Title	Reason for Inclusion
1	Loads before vessel breach at Grand Gulf	Early failure of drywell or wetwell has potential for causing large source term
2	Loads due to hydrogen burn before vessel breach at Sequoyah	Early failure of containment or bypass of ice condenser has potential for causing large source term
3,4	Loads in reaccor building at LaSalle and Peach Bottom	Bypass of reactor building has potential for increasing source terms
5	Loads at vessel breach at Grand Gulf	Failure of containment at vessel breach has potential for causing large source terms
6,7	Deleted	
8	Loads at vessel breach at Sequoyah	Same as issue #5
9	Loads at vessel breach at Surry.	Same as issue #5
10	Loads at vessel breach at Zion	Same as issue #5

Table 7-4
Issues Presented to the Structural Response Panel

Issue No.	Title	Reason for Inclusion
1	Static failure pressure and mode at Zion	Containment failure is the most important determinant of source terms
2	Static failure pressure and mode at Surry	Same as issue #1
3	Static failure pressure and mode at LaSalle	Same as issue #1
4	Static failure pressure and mode at Peach Bottom	Same as issue #1
5	Reactor Building bypass at Peach Bottom	Bypass of Reactor Building has potential for allowing large release of radionuclides
6	Static failure pressure and mode at Sequoyah	Same as issue #1
7	Ice condenser failure due to detonations at Sequoyah	Failure or bypass of ice condenser has potential for large source terms
8	Drywell and wetwell failure due to detonations at Grand Gulf	Failure of drywell bypasses suppression pool. Failure of wet-well allows large release to environemnt
9	Pedestal failure due to erosion at Grand Gulf	Pedestal failure is a major factor in subsequent accident progression

Table 7-5
Issues Presented to the Molten Core-Concrete Interaction Panel

Issue No.	Title	Reason for Inclusion
1	Mark I drywell meltthrough at Peach Bottom	Drywell meltthrough bypasses suppression pool; controversial issue
2	Mark II pedestal failure timing at LaSalle	Early pedestal failure could lead to large source terms; controversial issue
3	Mark II containment failure via pedestal failure at Grand Gulf	Pedestal failure could lead to early containment failure; controversial issue

Table 7-6
Issues Presented to the Source Term Panel

Issue No.	Title	Reason for Inclusion
1	In-vessel fission product release and retention	Release and retention are major determinants of source term
2	Ice condenser DF at Sequoyah	Ice condenser is principal decontamination mechanism in blackouts
3	Revolatilization from RCS/RPV	Revolatilization could negate effects of high retention; highly uncertain issue
4	CCI release	If in-vessel release is low, CCI release could be high; uncertain issue
5	Release of RCS and CCI species from containment	Aerosol agglomeration may be major source of cleanup in blackout; highly uncertain issue
6	Late sources of iodine at Grand Gulf	Appeared as important issue in draft NUREG-1150
7	Reactor building DF at Peach Bottom	Natural decontamination processes could reduce source term; uncertain and controversial issue
8	Release during direct containment heating	Uncertain and controversial issue; direct heating is also associated with early containment failure

7.5 Sources for Uncertainty Ranges and Distributions

(FIRST PARAGRAPH TO BE DONE LATER)

REMAINDER OF SECTION FROM KEENEY & WINTERFELD

7.6 HORA & IMAN'S PAPER.

7.6.4 Elicitation Training

Training in probability assessment techniques is an integral part of the expert opinion methodology used in NUREG-1150. Each panel of experts that participated in the expert opinion process received training through a one-half day training session. This session constituted the first meeting of each panel. The training was given by consultants from the field of probability assessment and decision analysis. These trainers included Professor Steve Hora of the University of Hawaii at Hilo and Professors Detlof von Winterfeldt and Ward Edwards, both of the University of Southern California.

The purpose of training in probability assessment is to facilitate the elicitation process. Experts in various fields of science are often not trained in probability theory and the techniques of probability elicitation. The expertise possessed by the scientists and engineers on the panels is called substantive expertise and thus they are substantive experts. Expertise about probability elicitation is called normative expertise and the participants in the expert opinion process schooled in probability assessment are known as normative experts. Both substantive expertise--knowledge of the problem domain being studied--and normative expertise--knowledge of techniques for encoding beliefs into probability distributions--are required for a successful expert opinion process.

During probability training, experts are exposed to various techniques for probability elicitation and the difficulties that accompany probability elicitation. Once trained, substantive experts are better able to express their knowledge in the form of probabilities and the resulting elicitations will be of a better quality. The resulting assessments are better calibrated in the sense that they accurately reflect the expert's knowledge and uncertainty. A by-product of the training is that the experts become more comfortable with the concept of subjective probability and more confident in expressing their beliefs in terms of probability distributions.

Another benefit of training is that the time spent by the experts preparing for the issues is used more efffectively because the experts can direct their analyses to the questions that must be addressed in the elicitation sessions. Further, the elicitation sessions run smoothly since the normative and substantive experts are working with the same definitions and the same understanding of the desired product.

Training Topics. The training sessions conducted in NUREG-1150 covered several related topics. These topics included the expert opinion process itself and the need for expert opinion, the elicitation techniques for the probabilities of various types of quantities and events or phenomena, the psychological aspects of probability assessments, and the decomposition of complex issues.

Each training session began with an overview of the goals of the expert opinion process and background material on the development of that process. The process was reviewed in some detail so that the substantive experts would be aware of what would be required of them and how their elicitations would be used. Because the formalized use of expert opinion was new to many of the participants, some were initially uneasy with the concept of expert opinion and the uses that it might be put to. Gaining the confidence of these experts through familiarization with the process was essential to the success of the expert opinion effort.

There are many different types of assessments that might be required of the experts. The type of assessment depends upon the nature of the physical quantity or phenomena under study. During the training sessions, the experts were introduced to assessment instruments for continuous quantities, discrete quantities, zero-one events, and dependent events. At appropriate points in the training, the experts were asked to make assessments using the methods under discussion. Using practice assessments develops confidence and ensures that the substantive experts understand the tasks that they will be required to perform. In order to make the training more interesting and more relevant, examples were used that reflected nuclear power risk issues.

Since many of the assessments would require the development of a probability distribution for a continuous quantity, the experts were given training in both the direct assessment techniques (assessing probabilities of given intervals of values) and bisection techniques (assessing values of the variable having given cumulative probabilities) for continuous variables. Later, in the elicitation sessions, these techniques would be used interchangeably by the normative experts.

A discussion of stochastic and parametric uncertainties and how they are differentiated in an uncertainty analysis was also provided. The concept of calibration of experts and calibration functions was also introduced. However, mathematical calibration of experts was not attempted in the NUREG-1150 expert opinion process.

Psychological aspects of probability elicitation received much attention in the training because failure to recognize and deal with psychological biases can impair the quality of the resulting assessments. One of the psychological aspects discussed is the tendency to give subjective probability distributions that are too narrow and thus understate the uncertainty or, conversely, overstate knowledge. This phenomena is often called overconfidence since the effect is that expressed probability distribution expresses greater certainty than is warranted. Other psychological aspects of subjective probability assessment that were discussed include anchoring which is the tendency to assume an initial position and fail to give sufficient credit to other sources of information; representativeness which is the tendency to give too much credit to other situations that are similar in some aspects but not others; the tendency to overestimate the probabilities of rare events; and and problems with group behavior such as personality dominance. Whenever possible, examples of these difficulties were presented and the experts being trained were asked to participate in demonstrations.

At the end of the training session the participants were given an assessment training quiz containing sixteen assessment tasks using the direct and bisection methods of assessment. The participants were asked to complete the training quiz during that evening and return with the quiz completed the next morning. At that time, the results were discussed. The purpose of the training exercise was two-fold: to give the substantive experts experience with the elicitation instruments and to provide feedback on the quality of the individual's assessments. As expected, most participants found that their assessed distributions expressed overconfidence. Once aware of this tendency, it is easier for the substantive experts to correct for this bias.

Problem decomposition was the last major segment of the training session. Problem decomposition is the process of creating a model of a complex assessment that allows the experts to make a series of simpler assessments. The simpler assessments are mathematically recomposed through the model. The net result is that the resulting probability distribution is a better expression of the expert's knowledge than if the expert had been asked to make an assessment of the initial issue without the aid of a decomposition.

Training in depomposition was accomplished by presenting examples of decompositions that had been developed for the NUREG-1150 study. Several types of decompositions were shown and the process of recombining the assessments was discussed. Comments from the participants indicated that the use of problems from the nuclear safety area enhanced the value of the decomposition training.

Potential Modifications. The probability assessment training developed for NUREG-1150 appears to have been successful in developing confidence among the participants and making the assessment sessions run smoothly. While it is not possible to determine whether the training has resulted in more accurate assessments, there is good reason to believe that the training has improved the overall quality of the NUREG-1150 findings. There are two modifications to the probability assessment training that may benefit future studies. The first is that the assessment training quiz should be constructed from questions more directly related to nuclear safety. This is not an easy task, however, since the questions must have answers that are known to the creator of the quiz without being known to the participants. The second modification is that the experts should be given more direct experience with creating decompositions. This is not without cost, however, since the training could easily require an additional one-half day.

7.7 Elicitation of Expert Opinion Within the Project Staff

(I SUGGEST REMOVING THIS SECTION OR AT LEAST PUTTING IT AS A SMALL SUBSECTION ELSEWHERE. IN THE FINAL EVENT, WE DIDN'T ELICIT EXPERT OPINION FROM WITHIN THE PROJECT STAFF, PRINCIPALLY OWING TO LAMENTABLE LACK OF INTER-PERSONAL COOPERATION. EVEN THOSE ISSUES FOR WHICH AREA 5 HELP WAS REQUESTED HAVE BEEN (OR WILL BE) ELICITED IN AN INDEFENSIBLE INFORMAL MANNER. I SUGGEST WE NOT DRAW ATTENTION TO THIS GLARING GAP.)

7.8 Incorporation of the Uncertainty Information in the Risk Analysis

Each member of the expert panels produced a distribution for each case of each issue. For some issues, several dependent variables were requested, and a separate distribution was elicited for each variable. If all the experts had worked with identical case structures and if all had produced their results in the same form, the task of aggregation would have been simply a matter of taking the numerical average of all the distributions for each case. However, some experts used idiosyncratic case structures. On some issues, the experts expanded the case structure beyond what was tractable in the containment event trees (Section 4) or the XXXSOR codes (Section 5). On some issues, experts gave their results in different forms.

For the purposes of aggregation it was absolutely required that the case structure be small enough to fit into the containment event trees and XXXSOR codes and that the case structure and dependent variables be the same between experts. If the case structure was impractically large and complex, it was reduced if possible by an analysis of variance (ANOVA). The ANOVA compared the variance in the dependent variable attributable to the differences between cases and the variance attributable to the differences among experts to the unexplained variance in the dependent variable. For many issues it was found that the differences between cases were not significant compared to the differences between experts, that is, that the large and complex case structure had little effect on the dependent variable. A mathematical procedure was then used to determine which of the cases could be safely combined.

If different experts used different cases, they were first encouraged to resolve their differences; if they failed to do so it was necessary to find some common ground. The cases common to all experts were of course retained. The remaining cases were inspected, and the most important ones were retained. If an expert did not have one of these cases, but did have a closely analogous case, the analog was used for the missing case. If the expert did not have a case closely related to the missing case, then the average of the case for all other experts was used for his missing case. It was recognized that this procedure would reduce the range of uncertainty, so the substitution was resorted to as little as possible. For some issues, missing data could be filled in by interpolation or ratios of existing cases.

If the experts produced different dependent variables, some analysis was required to put all the outputs into the same form. Whenever this was done the experts involved might find the final form of their data difficult to reconcile with what had been produced in the elicitation. Therefore, analytical alteration of results was resorted to as little as possible, and attempts were made to explain the reasons for and methods of analysis to the experts.

After each of the experts' distributions was in the same format, they were aggregated by averaging. The experts' outputs were almost always in the form of cumulative distribution functions (CDFs), that is, curves or tables of the probability that the independent variable would be no greater than some specific value. The aggregation was carried out by averaging all the experts' probability values for each value of the independent variable. The aggregated results were thus also CDFs.

The LHS program then divided each CDF into bands of equal probability. If there were to be N sample members then the width of each band was 1/N. The LHS program then took one value of the independent variable corresponding to each of the N probability bands. These N values of the independent variable were then randomly combined with the N values for each of the other issues. Each sample member thus has a value for each of the issues, and each value appears once only in the entire sample. This method of sampling makes it virtually impossible that any sample member should have most of its values drawn from the extremes of the distribution. LHS sampling also ensures that every part of the probability space of every issue is represented in the sample.

Many of the issues had several cases, or sets of initial conditions. The general rule for multiple cases was that they should be correlated unless there was some good reason for not correlating. Correlation between cases means that if the LHS program selected a low value for the independent variable for one set of intial and boundary conditions, a low value was also selected for every other set of initial conditions. Correlation between cases tends to increase the apparent uncertainty.

The CETs (Section 4) and XXXSOR codes (Section 5) now have a discrete value to be used for each case of each issue (the values, of course, are different for each sample member). The applicable case is determined from the PDS, from preceding calculations, or from the APB (Section 4.8), and a single value is thus presented to the CET or XXXSOR code for analysis.

Each sample member has a unique combination of event tree inputs and source term parameters. Risk is calculated (Section 8) for each combination of parameters. The sample members all have equal probabilities of occurrence because of the way values were selected, so that each of the risk values (one for each sample member) likewise has an equal probability of occurrence. If the risks for all sample members are arranged in order of increasing risk, a distribution of risk can be determined. The sample members having the lowest risk represent the most benign combinations of phenomenology. The sample members having the highest risk represent the most severe combinations of phenomenology.

It is important to understand that a sample member does not represent an "occurrence" or "trial" with some combination of phenomenological rules. Each occurrence has a single, specific outcome. Each sample member, on the other hand, represents an average over all possible occurrences all of which follow the same phenomenological rules. The conjunction of all sample members (the sample) represents all possible occurrences following all possible combinations of phenomenological rules. Actually a finite but large set of occurrences is used as a proxy for the universe of all possible occurrences and a large but finite combination of phenomenological rules is used as a proxy for the set of all possible combinations of rules. (See Section 8.2 for a fuller discussion of this point). It is possible to consider each sample member as a PRA in its own right. The NUREG-1150 study has been particularly exacting in its analysis of all phases of the accident, particularly so with regard to containment phenomenology. The study as a whole can thus be considered to be a very large collection of carefully performed PRAs.

7.9 References

7.1 W. Morris (Ed.), The American Heritage Dictionary of the English Language, Houghton-Mifflin: Boston, MA (1981).

(NOTE: I HAVE MADE A STAB AT NOTATION FOR THE MATRIX REPRESENTATION. IT WILL PROBABLY HAVE TO BE CHANGED TO MAKE IT COMPATIBLE WITH SECTION 2)

8. RISK CALCULATION

8.1 Calculation of a Single Risk Result

This section will present a detailed explanation of the matrix representation of risk which was introduced in Section 2. As discussed before risk is the expected value (that is, the average) of consequences. The average value of the product of frequency times consequences is very nearly the same as the product of their averages, that is, it is possible to multiply the average frequency (averaged over all data) times the average consequences (averaged over all weather), with very nearly the same results as the average of the products of all possible values of frequency times all possible values of consequences.

The TEMAC code calculates the frequencies of cutsets. The cutsets are first grouped into PDSs; the frequency of a PDS (in occurrences per year) is the sum of the frequencies of all the cutsets comprising the PDS.

Consider the frequencies (in events per year) of all PDSs. Let f_i be the mean frequency of PDS "i". Suppose that there are n PDSs which have been determined to have frequencies above some lower cut-off value. The frequencies of these n PDSs can be arranged as a vector (F):

$$\begin{matrix} f_1 \\ f_2 \\ f_3 \\ . \\ . \\ . \\ f_n . \end{matrix}$$

Now consider all possible APBs. Each of these has some relative frequency, p. Let p_{ji} be the relative frequency of APB j, given the occurrence of PDS i. Then the frequency (in occurrences per year) of APB j is the row vector (A), where the frequency of APB "j" is:

$$a_j = \sum_i f_i p_{ji}$$

or in matrix notation:

(A) = [P](F). If there are m possible APBs, then [P] is matrix of m rows and n columns, and (A) is a vector of length m, each element of which is the absolute frequency of a single APB. To every APB, j, there corresponds a unique source term (ST), s_j, and to every ST there corresponds a mean consequence in consequence measure "k". The risk, that is, the expected value of consequences in measure k, is approximately the sum over all source terms of the mean consequence corresponding to each source term times the frequency of the source term. If $c_j = c(s_j)$ is the consequence corresponding to the source term from APB j, then the risk in measure k is:

$R_k = (A)^T(C)$, where the column vector (C) has as its elements the mean consequence corresponding to each source term. In terms of the previously calculated PDS frequencies and APB relative frequencies, the risk is:

$Rk = ([P](F))^T(C)$.

In many past PRAs the number of PDSs and APBs has been very limited. In a "best estimate" calculation, there is generally only one possible APB for each PDS (because, by the nature of best estimate calculations, the only outcome considered is the most likely one). It is entirely practical to calculate risk by hand. Even if multiple accident progression bins are allowed, hand calculation is not difficult if the number of PDSs and APBs is small. For example, consider four PDSs and six APBs. For the sake of this numerical example, suppose that the frequencies of the four PDSs have been found to be 1 x 10-5, 5 x 10-6, 8 x 10-6, and 1 x 10-6. Suppose that a simple CET has given the following relative frequencies for the APBs (the table below is precisely the matrix [P]):

PDS	Bin 1	Bin 2	Bin 3	Bin 4	Bin 5	Bin 6
1	0.1	0.0	0.1	0.7	0.05	0.05
2	0.0	0.2	0.0	0.8	0.0	0.0
3	0.05	0.1	0.05	0.5	0.2	0.1
4	0.1	0.0	0.0	0.0	0.9	0.0

It is a simple matter, well within the practicality of hand calculation, to find that the frequencies of the APBs are 1.05 x 10-5, 1.8 x 10-6, 1.4 x 10-6, 1.5 x 10-5, 3 x 10-6, and 8.5 x 10-7.

If the source term and consequence calculations have been carried out for each of these six bins, and the mean number of early fatalities (given the occurrence of each bin) have been found to be 1.5, 5., 7., 0.1, 10., and 5., respectively, it is also a matter of simple hand calculation to determine that the risk of early fatality is 7.03 x 10-5 per year.

For NUREG-1150 the number of source terms and hence of consequence calculations is staggeringly high. It would be impracticable to use MACCS (which is an intensive user of computer resources) for each and every source term, many of which would probably be quite similar to each other. Partitioning (Section 5.6) is resorted to in order to keep the number of MACCS calculations reasonably low. The entire source term space, containing several hundred source terms, is partitioned into a smaller number of surrogate source terms. Each of the surrogates is a proxy for all the source terms within its neighborhood. The matrix equation is modified by the inclusion of an m x r matrix, [Q], where r is the number of surrogate source terms. Each element of [Q], q_{ji} contains 1.0 if source term j is to be assigned to surrogate i, or zero otherwise. The modified risk equation is:

Rk = ([F][P])^T[Q](C). The consequence vector, (C) is now a vector of length r.

In the preceding simple example, source terms 2 and 6 have the same consequence, and could be grouped together. Source terms 3 and 5 have approximately the same consequence, and could be grouped together, with a consequence which is the average of the two. The number of consequence calculations would thus be reduced from six to four. (It should be pointed out that the reduction achieved in this simple example would hardly be considered worthwhile). The matrix [Q] has the form

$$\begin{matrix} 1 & 0 & 0 & 0 \\ 0 & 1 & 0 & 0 \\ 0 & 0 & 1 & 0 \\ 0 & 0 & 0 & 1 \\ 0 & 0 & 1 & 0 \\ 0 & 1 & 0 & 0 \end{matrix}$$

The consequences for the partitioned source terms are 1.5, 5., 8.5 (the average of consequences for source terms 3 and 5), and 0.1. If the partitioned source terms are used, the risk becomes 6.79 x 10-5 per year, which is not very different from the result calculated without partitioning.

The calculation of PDS frequencies, APB relative frequencies, source terms, and consequences may have been very complex and time consuming. However, in this simple example, the calculation of risk was completely straight forward and simple.

If there are more PDSs, of the order of 10 to 50, and hundreds or thousands of possible APBs, the difficulties of bookkeeping in hand calculation become nearly insuperable, and computer calculation is required. Simplification on the basis of neglecting the less important terms is seldom possible; the relative importance of each APB and PDS is unknown until the calculation has been completed.

For NUREG-1150, PDS frequencies are calculated by the TEMAC code, the APB relative frequencies are calculated by EVNTRE, and the source terms are calculated by XXXSOR. Partitioning is done by the PARTITION code, and consequences are calculated by MACCS. The matrix arithmetic (risk calculation) is done by PRAMIS. Each of these codes is discussed in Appendix A. Appendix B gives an example showing how each of these codes is used for an actual risk calculation.

The outputs of each major code--TEMAC, EVENTRE, and XXXSOR--is more detailed than is required for the following code. The computational task would be practically impossible if every output were to be used in the risk calculation. For example, TEMAC calculates the frequency of each sequence, or combination of initiating event and plant faults. However, the CET would not recognize the differences between many sequences. The event trees calculate relative frequencies for a vast number of accident progression pathways, but the XXXSOR codes recognize only a limited number of distinct accident progression pathways. For practical considerations of computation, some means of limiting the outputs was necessary. The information transfer between codes had to be limited to that which was usable, at the same time ensuring that essential information was passed to the following code.

Accident sequences which present similar initial conditions to the CET were gathered into PDSs (Section 3.1). Each PDS is recognizable by the CETs as unique and distinct. The CETs then calculate the relative frequencies of all accident progression pathways for each PDS. These are then gathered into APBs (Section 4.6) for input to the XXXSOR codes. The XXXSOR codes calculate a source term for each APB; however, the number of source terms is too great for practical consequence calculations. As explained above and in Section 5.6, the large volume of source terms is reduced to a workable set by partitioning.

The reduction and gathering of information at each interface is accomplished by post-processors, each of which is a code in its own right. The TEMAC post-processor (____________________), reduces cutsets to PDSs. The EVNTRE post-processor (____________________) reduces accident progression pathways to APBs. PARTITION acts as the post-processor to XXXSOR.

8.2 Risk Calculations Supporting the Uncertainty Analysis

The single risk calculation described in Section 8.1 is small part of the effort required for NUREG-1150. The major part of the study involved the determination of uncertainty in risk.

Section 7 described uncertainty and the types of uncertainty. Each of the inputs required for the calculation of risk is to some degree uncertain. For example, the PDS frequencies are uncertain because of data, phenomenological, and stochastic uncertainties. The APB relative frequencies are uncertain because of phenomenological, stochastic, and to a lesser degree data uncertainties. The source terms are uncertain because of poorly understood phenomenology. It is usually impossible to estimate the effect of uncertainty in any phenomenon in advance because of the complexity of the calculation. It is especially difficult to estimate the effects of uncertainty in several phenomena when all are acting simultaneously.

One possible solution to the difficult problem of estimating uncertainty is to perform bounding calculations, that is, to determine the risk if all uncertain quantities were at their most optimistic levels and then at their most pessimistic levels. Unfortunately, for many quantities it is difficult or impossible to determine _a priori_ which is the most pessimistic direction. Many phenomena act in different directions at different times in the accident, and phenomena which appear to be severe can actually have a benign outcome. For example, burning hydrogen early in the accident might appear to be pessimistic, because if an earlier containment failure is caused, the consequences would generally be higher. However, burning hydrogen early means that less hydrogen and oxygen are available later. At a later time the initial pressure could be higher and the total quantity of hydrogen could be greater, so that there might be a greater probability of failing containment if hydrogen is saved for a later burn. Even more to the point, if all quantities simultaneously are set to their most optimistic and pessimistic levels, the bounds are usually so large as to be nearly meaningless. If it is believed improbable that any single quantity should be at the extreme of its possible range, it is certainly not credible that all quantities should simultaneously be at their extremes.

NUREG-1150 uses sampling to assess uncertainty. The principle is simple; for each observation a value is randomly chosen for each uncertain quantity somewhere in its range; for each uncertain phenomenon a phenomenological "rule" is chosen, and for each uncertain frequency a specific frequency is chosen. The sampling is done in such a way that the most likely regions of the range for each quantity are sampled more often than the less likely regions. The method of Monte Carlo sampling is well known, and is often used for solution of difficult physical problems. If the sampling is sufficiently large* the output of all the many calculations (in this case, each calculation gives the risk) will have a distribution. The most probable values of risk will be bunched together, and the least probable values will fall in the outlying tails. Straight Monte Carlo sampling is not practical for the risk calculations, however, because of the very large sample size required. Also, there is always a chance (which is not negligible) that some important part of the distribution will be missed. In NUREG-1150, the number of variables to be sampled is so large that it is almost certain that parts of some distributions would be missed.

The sampling scheme used for NUREG-1150 is Latin Hypercube Sampling (RON IMAN'S STUFF ABOUT LHS SHOULD GO IN HERE).

In the NUREG-1150 application, a probability distribution is formulated for each uncertain variable. Some of the distributions are subjective and some are data-based. The most important of the subjective distributions were developed by teams of outside experts (Section 7.6). Each distribution was then "sliced" into regions of equal probability, and values corresponding to the slices from each of the many independent variables were randomly combined. Each such combination is a <u>sample member</u> and the set of all sample members is the <u>sample</u>. The sample could be thought of as a numerical experiment replicated many times, each time with somewhat different initial and boundary conditions. Each sample member then corresponds to a single observation.

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*A statistical test (the Kolmogoroff-Smirnoff test) is available to determine what sample size is necessary for any desired degree of accuracy of the resulting distribution. The required sample size is very large if the output distribution is to have any hope of representing the "true" distribution.

A sample member corresponds to a set of assumptions which might have been made by an individual analyst. The experts who assessed the distributions for the uncertain variables also provided guidance on correlations between variables. Enforcing these correlations ensures that the assumptions for the variables are mutually consistent. Consistency is also enforced by carrying some variables and distributions through all phases of the analysis. That is, the same distributions for power recovery are used in the sequence frequency analysis as in the accident progression analysis. The same value for in-vessel oxidation of zirconium which is used in the accident progression analysis is also used in the source term analysis, so that it is impossible to oxidize the same material twice. Consistency is further enforced by the case structure (Section 4.5) in which the outcome of one phase automatically becomes the initial condition for the next phase. Consistency is also tested, but cannot be strictly enforced, by a methodical and assiduous search for inconsistencies. For at least some recognized expert, the assumptions are reasonable or at least cannot be ruled out. Furthermore, the sampling scheme selects more sample members from the part of each distribution to which the experts gave highest credence, and fewer sample members from the parts of each distribution which the experts considered to be less likely. The result of this sampling scheme is that the majority of the sample members represent the assumptions that the majority of the experts would consider most likely to be correct.

The risk for each sample member is calculated as described in Section 8.1. Because the assumptions and input data were different for each sample member, the risk will be different for each sample member. If a large number of critical experiments could be conducted, and the outcomes showed that the true phenomenology was consistent with the assumptions used for sample member 50, the uncertainty would be reduced to the narrow range imposed by data and weather uncertainties--we would know unequivocally that the risk for sample member 50 was the "true" risk. However, it is impossible (and will probably remain forever impossible) to conduct all the necessary critical experiments. Because of the method of sampling each sample member is equally likely, and thus each represents an equally valid opinion of the expert community. The sample members at the extremes of risk are as good as those in the middle. However, the region in which the most sample members are found is the region where the majority of experts believe the risk is most likely to lie.

A sample member does <u>not</u> represent a trial, that is a throw of the dice. Each sample member represents the ensemble of all possible trials which could be made subject to the rules selected for that sample member. For any throw of a dice, there is no uncertainty as to the outcome of the throw, which is a single, clearly defined number. A single throw either does or does not result in a six, or one, or any other number. However, if a large number of throws in considered, there is only a frequency with which six will be thrown; if the dice are fair, the frequency of a six will be 0.167. Likewise, for any sample member there is only a frequency with which power is recovered at any time. In any specific accident (that is, throw of the die) power either is or is not recovered at that time. The sample member represents the ensemble of all possible accidents--those in which the power is recovered and those in which it is not recovered--and the split fraction for power recovery shows the fraction of the accidents in which power is recovered. The frequency of power recovery is uncertain, and each sample member has a different split-fraction for power recovery, based on sampling from the distribution for power recovery frequency. For phenomenological uncertainties, each sample member follows one and only one of the possible rules for outcome. If the phenomenon is containment failure at vessel breach, then each sample member will have either catastrophic rupture, or ordinary rupture, or a small leak, or no failure at all, and these outcomes are exclusive. The choice of one outcome rules out all the others. The reason is that the experts were uncertain about the phenomenology of containment failure. The majority of experts might believe, for example, that at a certain pressure leak was the most likely failure mechanism, although more severe failures could not be absolutely ruled out. However, if leaks did indeed occur, then they would be expected to occur consistently.*

The sample members are sorted in order of increasing risk, and a distribution of risk can be formed. It is necessary to clearly understand that this is not at all a probability distribution. The distribution of risk is a "belief" distribution. The statement attached to any level of the cumulative distribution function for probabilities is normally "the probability is y_i that the value of X is less than or equal to x_i", but for the risk distribution it is "the probability is y_i that an expert would <u>believe</u> the value of X to be less than or equal to x_i". As an expression of belief, it is absolutely true; the distribution could only be incorrect if the experts were lying or the NUREG-1150 staff had incorrectly applied their data. The distribution represents the opinion of a sample of the expert community about the likely values of risk. However, experts are often wrong, so that while it is <u>true</u> that the distribution represents the opinions of the experts, it may be <u>false</u> that the opinions represent reality. One should never allow oneself to think of the mean of the distribution as the mean risk; the mean only represents the opinion of an "average" expert. It is also necessary to understand that the "true" risk might not even fall within the distribution. Furthermore, the range of the distribution is that given by the experts being questioned. Another expert might have found the risk to lie completely outside the distribution found here.

After the risk values are sorted, the sets of assumptions corresponding to the extremes of risk are carefully examined. The principal question asked at this stage is whether the results are reasonable. Are they consistent with other calculated values? Did the results vary in the right direction when the input assumptions were varied? Are there sets of assumptions that seem to be clustered in any region? If so, is the clustering reasonable?

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*The experts believed that the variation in failure mode due to variability in materials and workmanship would be small relative to the large uncertainty in failure phenomenology.

A further test of reasonableness is given by examination of the average contribution to risk of each plant damage state and accident progression bin.*

For example, if a PDS had a low frequency and did not appear to be particularly serious, but had a large impact on risk, it would be a signal that something might be wrong in the calculation, and that serious checking would have to be done.

8.3 Methods for Results Evaluation

THIS SHOULD BE WRITTEN BY HELTON AND IMAN; HOPEFULLY RESTRAINED FROM MAKING THIS TOO TECHNICAL. MOST OF OUR READERS WILL NOT BE TECHNICALLY (AND ESPECIALLY NOT MATHEMATICALLY OR STATISTICALLY) SOPHISTICATED!

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*The contribution of any PDS to the total risk is different for each sample member. The average contribution to risk for a PDS is the sum of the contributions for that PDS over all sample members, divided by the total risk summed over all sample members