APPENDIX B

AN EXAMPLE RISK CALCULATION

CONTENTS

FIGURES

TABLES

B.1 Introduction

In this appendix, which is adapted from Reference B.1, an example calculation is followed through the entire analysis from the initiating event in the accident frequency analysis through to the offsite risk. This discussion has been prepared for the reader seeking detailed information on how the risk calculations were performed. It is assumed that the reader is familiar with nuclear power plants in general and with severe accident risk analysis in particular. Since the accident frequency analysis is generally more familiar to the PRA community, and the accident frequency analyses performed for NUREG-1150 have fewer novel features than the other analyses, the discussion of the accident frequency analysis in this appendix is abbreviated. Thus, even though the accident frequency analysis requires a level of effort comparable to that required for the other analyses, the discussion of the risk calculation from the identification of the initiating event through the definition of the plant damage state (PDS) does not reflect that fact.

The example selected for this discussion is a fast station blackout (SBO) accident for Surry. This accident, denoted TMLB'* in the Reactor Safety Study, is estimated to be one of the more likely accidents and is of historical interest. Surry was chosen because the accident progression event tree (APET) for Surry is simpler than the APETs for the other plants.

The PDS designation for the fast SBO accident is TRRR-RSR. (The PDS nomenclature is explained in Section B.2.3.) This PDS has the third highest mean core damage frequency (MCDF) at Surry. Several accident sequences comprise this PDS; the one chosen for this example is T1S-QS-L, which has the highest frequency of the sequences in TRRR-RSR. (This sequence is defined in detail in Section B.2.1.) PDS TRRR-RSR is the only PDS in PDS group 3, fast SBOs. The example will be followed through the APET to accident progression bin (APB) GFA-CAC-ABA-DA. (The APB nomenclature is explained in Section B.3.4.) For the observation chosen, this bin is the most likely to have both vessel breach (VB) and containment failure (CF). The computation of the source term for this bin will be followed through the source term analysis, and this source term will then be grouped with other similar source terms in the partitioning process.

Finally, offsite consequences will be determined for the subgroup to which the source term for GFA-CAC-ABA-DA was assigned, and the results of all the analyses will be combined to obtain the measures of risk.

To determine the uncertainty in risk, the accident frequency analysis, the accident progression analysis, and the source term analysis were performed many times, with different values for the important parameters each time. A sample of 200 observations was used for the Surry analysis. The Latin hypercube sampling method, a stratified Monte Carlo method, was used. In this example, one sample member or observation, Observation 4, will be followed all the way through the risk analysis. It was chosen because it was the median observation for early fatality risk (in the analysis in which 100 percent evacuation was assumed).

B.2 Accident Frequency Analysis

The accident frequency analysis determines the expected frequencies for the many different types of core damage accidents that can occur. This appendix is not intended to present methods, as those are summarized in Appendix A and presented in detail in Reference B.2. Nevertheless, many aspects of the methods will become apparent in this discussion. Section B.2.1 is an overview of the accident frequency analysis, and Section B.2.2 contains a description of the accident sequence. Section B.2.3 describes the quantification of the cut set, and Section B.2.4 discusses how the accident sequences are grouped into PDSs.

B.2.1 Overview of Accident Frequency Analysis

Development of the chronology and frequency of the accident sequences involves many tasks or constituent analyses. These include:

*TMLB' was defined in the Reactor Safety Study as a transient loss of offsite power (T) with failure of the power conversion system (M) and the auxiliary feedwater system (L), and failure of the emergency ac power system with no recovery of offsite ac power in 1 to 3 hours (B').

- Initiating event analysis, including determination of the system success criteria,
- Event tree analysis, including accident sequence delineation,
- Systems analysis, including fault tree construction,
- Dependent and subtle failure analysis,
- Human reliability analysis,
- Data base analysis, including development of the data base,
- Elicitation of expert judgment,
- Accident sequence quantification, including recovery actions,
- Grouping of the accident sequences into PDSs, and
- Uncertainty analysis.

These tasks are performed approximately in the order given above. The quantification and the assignment of the sequences to PDSs are performed several times in iterative fashion as the information available evolves and the requirements of the subsequent analyses change.

An accident sequence is a particular accident defined by the initiating event and failures of the systems required to respond to the initiator. Sequences are defined by specifying what systems fail to respond to the initiator. In the accident frequency analysis, models (event trees, fault trees) are constructed for all the important safety systems in the plant (usually at the pump and valve level of detail). Failure rates for equipment such as pumps and valves are developed from failure data specific to the plant being analyzed and from generic nuclear power plant data bases. The models and the failure rates are used by the computer program that calculates the thousands of possible failure combinations, denoted as cut sets, that lead to core damage.

Each cut set consists of the initiator and the specific hardware or operator failures that produce the system failures. The initiator and the failures are often referred to as "events." For example, a water injection system could fail because the pump failed to start or because the normally closed, motor-operated discharge valve failed to open. Cut sets that include the pump failure and cut sets that include the valve failure, but are otherwise identical, occur in the same accident sequence since the pump and valve failures have the same effect on a system level.

The accident sequence followed for this example is T1S-QS-L, which is the highest frequency sequence that contributes to PDS TRRR-RSR. This sequence is the most probable of several sequences that involve station blackout and early failure of the auxiliary feedwater system (AFWS). The mean frequency for TRRR-RSR is 4.8E-6/reactor year, and T1S-QS-L contributes about 75 percent of that. For Observation 4, the frequency of TRRR-RSR is 4.8E-7/reactor year, and the frequency of T1S-QS-L is 2.4E-7/reactor year. (It is purely coincidental that the frequency of TRRR-RSR for Observation 4 is one-tenth of the average frequency over all 200 observations.)

Sequence T1S-QS-L is comprised of 216 cut sets. The cut set with the highest frequency, consisting of nine events, is given in Table B.1. The cut set equation for T1S-QS-L is:

T1S-QS-L = (IE-T1) * (OEP-DGN-FS-DG01) * (/DGN-FTO) * (OEP-DGN-FS-DG03) *

(NRAC-1HR) * (REC-XHE-FO-DGEN) * (NOTQ) * (QS-SBO) * (AFW-XHE-FO-CST2)

+ ...... (215 other cut sets)

The frequency of each cut set varies from observation to observation because the probabilities of some of the events are sampled from distributions. For Observation 4, the frequency of the cut set in Table B.1 is

3.4E–8/reactor year. This cut set defines one group of specific failures that cause the accident, which will be followed through the entire analysis in this appendix. Each event listed in Table B.1 is discussed in some detail in Section B.2.3 below.

Table B.1 Most likely cut set in Surry sequence T1S-QS-L quantification for observation 4.

Event	Quantification	Description
IE-T1	0.0994	Initiating Event: LOSP
OEP-DGN-FS-DG01	0.0133	DG 1 fails to start
/DGN-FTO	0.966	Success of DG 2
OEP-DGN-FS-DG03	0.0133	DG 3 fails to start
NRAC–1HR	0.44	Failure to restore offsite electric power within 1 h
REC-XHE-FO-DGEN	0.90	Failure to restore a DG to operation within 1 h
NOTQ	0.973	RCS PORVs successfully reclose during SBO
QS-SBO	0.0675	Stuck-open SRV in the secondary system
AFW-XHE-FO-CST2	0.0762	Failure of operator to open the manual valve from the AFW pump suction to CST2
Entire cut set	3.4E–8	Frequency (per year) for Observation 4

Figure B.1 shows the event tree for T1S—station blackout at Unit 1. Three of the paths through this tree lead to core damage situations that are in PDS TRRR-RSR. Accident sequence 19, T1S-QS-L, is the most likely of these three. The logical expression for this sequence, according to the column headings or top events, is:

T1S-QS-L = T1S * NRAC-HALFHOUR * /Q * QS * L,

where /Q indicates not-Q, or success. System success states like /Q are sometimes omitted during quantification if the state results from a single event since the success value is very close to 1.0. T1 is a loss of offsite power (LOSP) initiator, and the "S" in T1S indicates that it is followed by failure of the emergency ac power system (EACPS). Failure of EACPS, although not shown explicitly in Figure B.1, is determined by a fault tree, and

T1S = T1 * Failure of EACPS,

where failure of EACPS is failure of diesel generator (DG) 1 and DG 3, or, failure of DG 1 and DG 2. (Failure of only DG 2 and DG 3 implies success of DG 1, which is not SBO for Unit 1. If DG 2 fails, it is assumed that DG 3 is assigned to Unit 2. Failure of all three DGs is included in a different sequence.) Note that T1 appears as IE-T1 in the cut set; the SBO is implied by events OEP-DGN-FS-DG01 and OEP-DGN-FS-DG03.

The cut set considered in this appendix has the failure of DG 1 and DG 3. A simplified depiction of the fault tree for DG 1 is shown in Figure B.2. The fault tree for DG 3 is similar. The heavier line in Figure B.2 indicates the failure, OEP-DGN-FS-DG01, in the cut set of interest. The other failures are included in other cut sets. (Figs. B.2 and B.3 are illustrative only and do not provide an accurate representation of the complete fault trees. The complete fault trees are given in Appendix B.2 to Ref. B.3.)

In Figure B.1, the cut set of interest is part of sequence 19, T1S-QS-L, which is shown by the heavier line. The first top event is the initiator, discussed above. The second top event concerns the recovery of offsite ac power within 30 minutes (NRAC-HALFHOUR). In the event of a loss of auxiliary feedwater, core uncovery will occur in approximately 60 minutes. A 30-minute time delay for reestablishing support systems (including canal water level) was assumed from the time that ac power was restored to the time that feed and bleed could be established. Thus, ac power must be recovered within 30 minutes in order to mitigate failure of auxiliary feedwater.

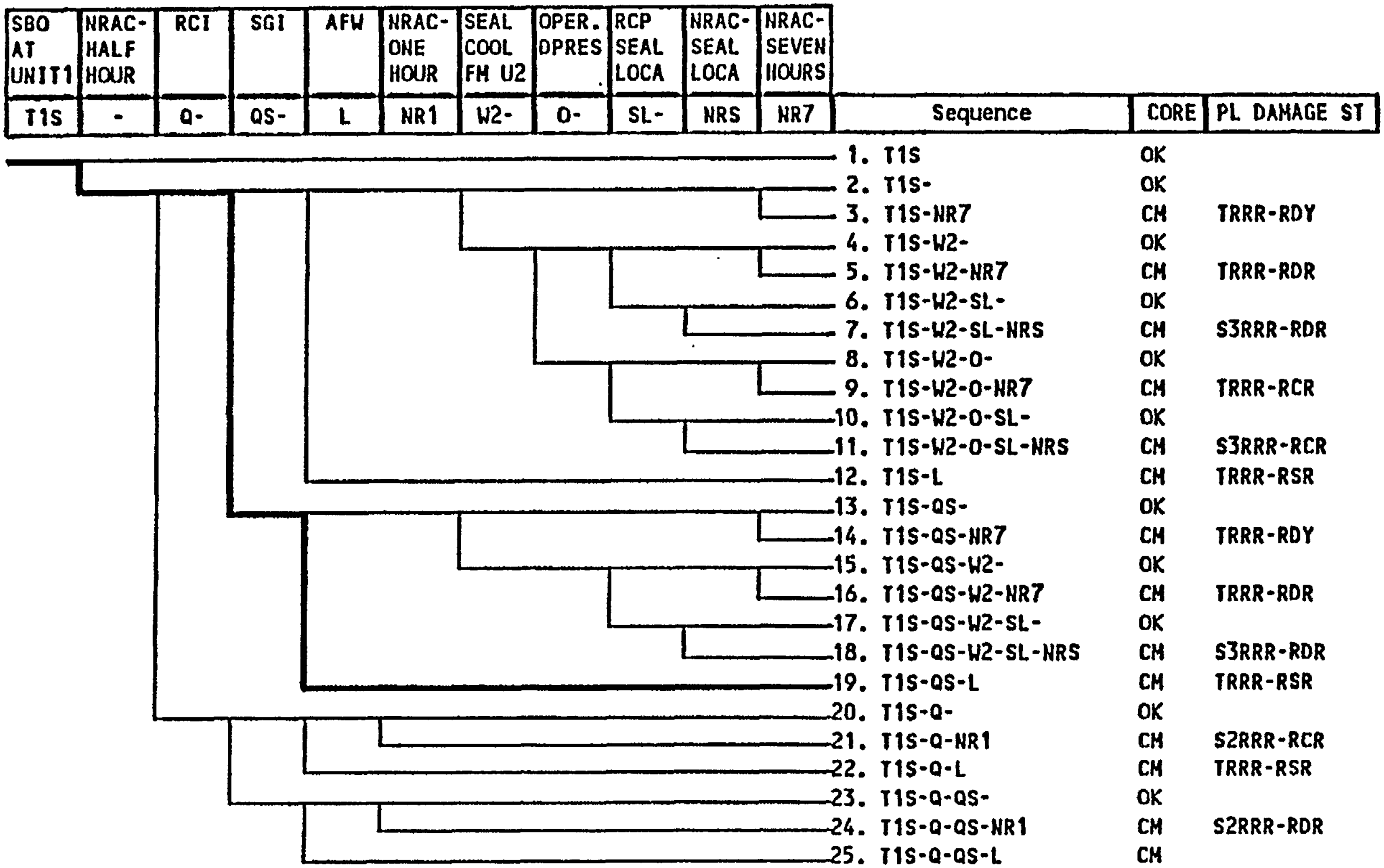


Figure B.1 Event tree for T1S-SBO at Surry Unit 1. (This figure is adapted from Section 4.4 of Ref. B.3. No PDS assignment is indicated for sequence 25 because the sequence frequency fell below the cutoff value.)

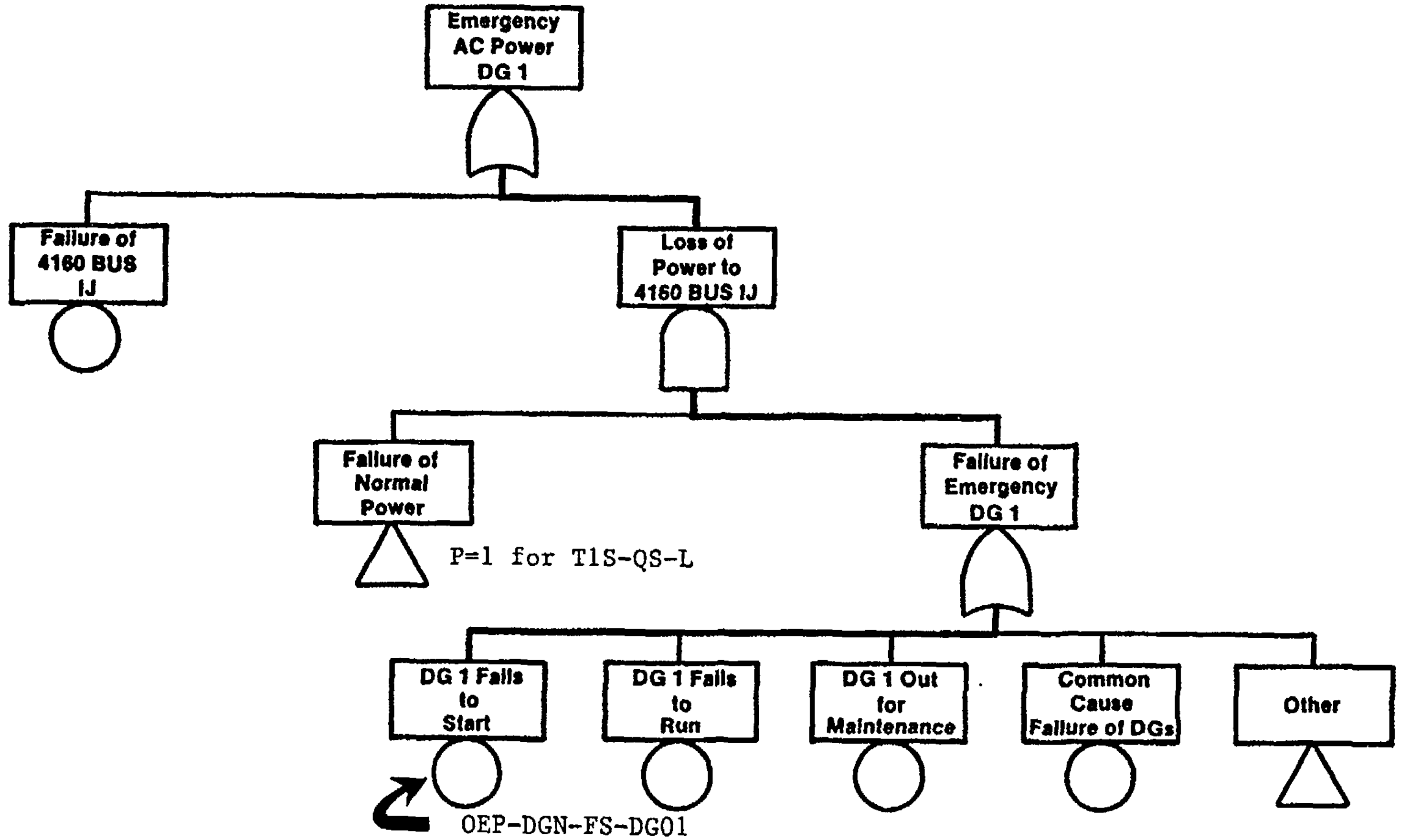


Figure B.2 Reduced fault tree for DG 1 at Surry Unit 1. (This figure is a greatly simplified version of the fault tree given in Appendix B.2 of Ref. B.3. P = 1 indicates that the failure probability is 1.0.)

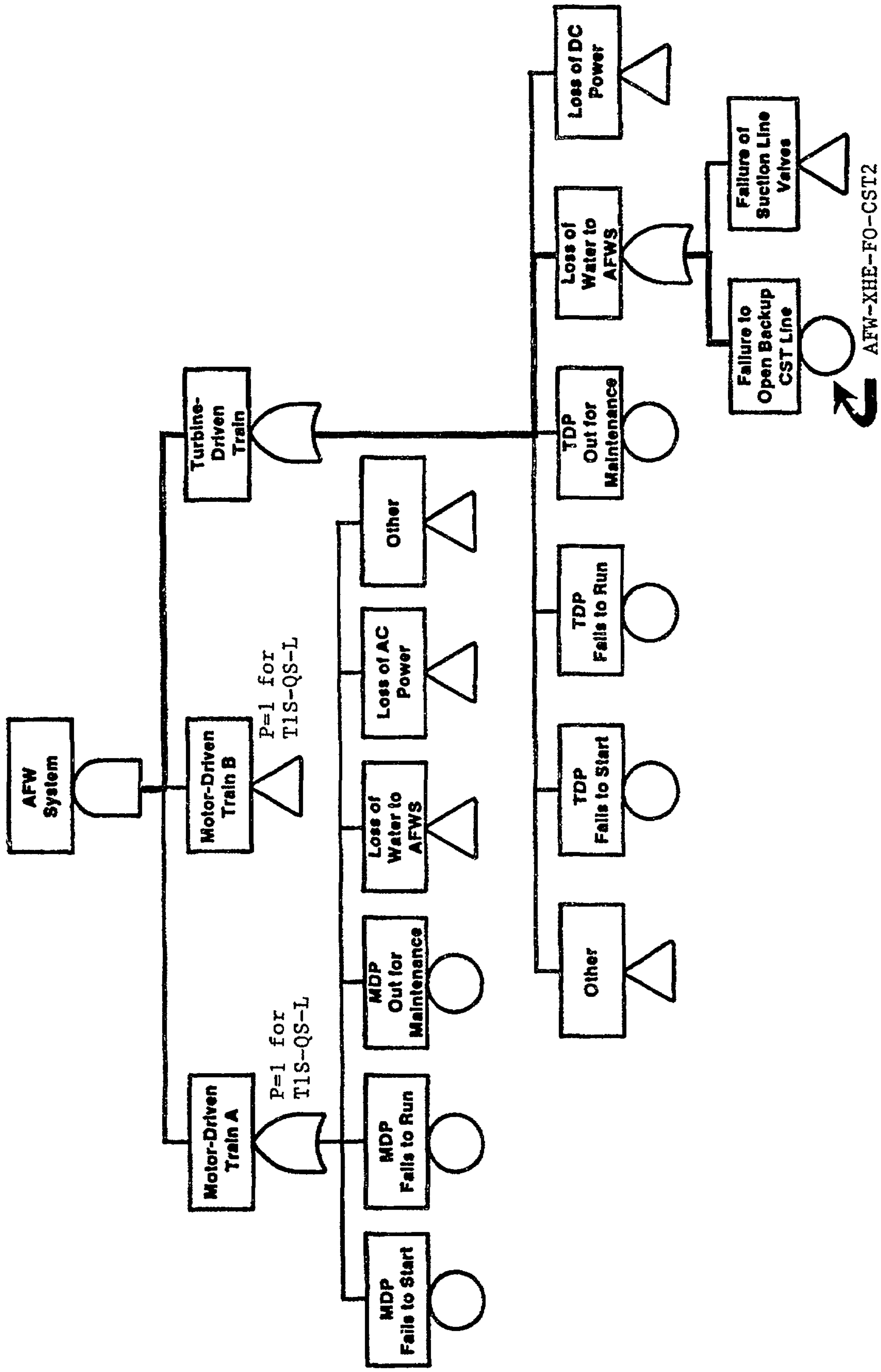


Figure B.3 Reduced fault tree for AFWS at Surry Unit 1. (This figure is a greatly simplified version of the fault tree given in Appendix B.2 of Ref. B.3. P = 1 indicates that the failure probability is 1.0.)

The third top event is RCI, failure of reactor coolant system integrity, Event Q. In the cut set being followed, the success branch is taken here, i.e., the PORVs cycle correctly and do not stick open. The fourth top event in Figure B.1 is SGI, failure of steam generator (secondary side) integrity. In this cut set, a relief valve on the main steam line sticks open, Event QS, so the failure branch is taken. Failure of QS causes rapid depressurization of the steam generator and rapid depletion of condensate water.

The fifth top event, L, is failure of the AFWS. In the cut set listed in Table B.1, the steam-turbine-driven AFWS runs until the condensate storage tank (CST) is depleted at about 60 minutes after the start of the accident. The AFWS fails at that time because the operators fail to switch the pump suction to the backup CST. This failure is event AFW-XHE-FO-CST2. Thus, the appropriate power recovery time for this cut set is NRAC–1HR, which replaces NRAC-HALFHOUR in the cut set, although the failure (lower) branch for NRAC-HALFHOUR is indicated on the event tree. None of the subsequent top events are applicable since the failures that have already occurred are sufficient to cause core damage, so there are no further branches on the path to sequence 19.

Figure B.3 is a simplified fault tree for the AFWS. The motor-driven AFW trains, of course, require ac electrical power and are not available for this accident. The heavier line in Figure B.3 indicates the failure that occurs in the cut set considered for this example. The other failures are included in other cut sets.

A general description of this accident sequence follows in the next section. More detail on the methods used in the accident frequency analysis may be found in Reference B.2. Details of the specific analysis for Surry may be found in Reference B.3.

B.2.2 Description of Accident Sequence

An LOSP initiator, IE-T1, starts this transient accident by tripping the reactor and the main steam turbine. The DG assigned to Unit 1, DG 1, fails to start, OEP-DGN-FS-DG01. DG 3 also fails to start, OEP-DGN-FS-DG03. (DG 2 is dedicated to Unit 2.) The event /DGN-FTO indicates that DG 2 is successfully powering Unit 2. The failure to start of DG 1 and DG 3 causes a complete failure of ac power at Unit 1. However, dc power is available from the Unit 1 batteries until they are depleted (in roughly 4 hours).

The pressure boundary of the reactor coolant system (RCS) is intact, so loss of water from the RCS is not an immediate problem. However, all the systems capable of injecting water into the RCS depend on pumps driven by ac electric motors. Thus, if decay heat cannot be removed from the RCS, the pressure and temperature of the water in the RCS will increase to the point where water will flow out through the PORVs, and there will be no way to replace this lost water.

Heat removal after shutdown is normally accomplished by the auxiliary feedwater system (AFWS). Surry's AFWS has three trains: two of these trains have pumps driven by ac electric motors, and these trains are unavailable due to the SBO. The only means of heat removal in a blackout situation is the steam-turbine-driven AFW train. In the accident defined by the cut set in Table B.1, the steam-turbine-driven AFW train is initially available as steam is being generated in the steam generators (SGs) to drive the steam turbine, and dc power is available for control purposes. The initiating LOSP causes the main steam isolation valves (MSIVs) to close, preventing the steam being generated in the SGs that is not needed by the AFWS turbine from flowing to the main condenser. The normal means of venting excess steam from the secondary system is through the atmospheric dump valves (ADVs), but in this sequence they are failed in the closed position because of the loss of 120 v ac power. Thus, pressure relief takes place through one or more of the secondary system safety-relief valves (SRVs).

In this accident sequence, at least one of the secondary system SRVs fails to reclose, which causes water to be lost at a significant rate from the secondary system. This is event QS in Figure B.1; it is denoted QS-SBO in the cut set. The AFWS initially draws from the 100,000-gallon condensate storage tank (CST). With an SRV stuck open, the AFWS will draw from the CST at 1,000 to 1,500 gpm to replace the water lost through the SRV, thus depleting the CST in 1.0 to 1.5 hours. A 300,000-gallon backup water supply (CST2) is available, but the AFWS cannot draw from this tank unless a manual valve is opened. In this cut set, the operators fail to open this valve, and the AFWS fails. This human error is event AFW-XHE-FO-CST2. There are two recovery actions in this cut set. One is the failure to restore offsite power within 1 hour (NRAC–1HR), and the other is the failure to recover a failed DG

(REC-XHE-FO-DGEN). In the path to sequence 19 shown in Figure B.1, the failure to recover offsite power is NRAC-HALFHOUR. In this particular cut set, the time to failure of the AFWS is longer than in the majority of cut sets in sequence T1S-QS-L, and this failure is replaced by NRAC–1HR.

With the failure of the turbine-driven AFW train, and no ac power to run the motor-driven AFW trains, the reactor coolant system (RCS) heats up until the pressure forces steam through the PORV(s). Water loss through the PORV(s) continues, with the PORV(s) cycling open and closed, until enough water has been lost to reduce the liquid water level below the top of active fuel (TAF). The PORVs do not stick open; this is event NOTQ. Without electric power, there is no way to replace the water lost from the RCS. The uncovering of the TAF (UTAF) marks the transition of the accident from the accident frequency analysis to the accident progression analysis. The onset of core degradation follows shortly after the UTAF.

B.2.3 Quantification of Cut Set

Table B.1 gives the specific cut set being considered in this example and shows the quantification of each event in the cut set for Observation 4. A discussion of how each quantification was derived follows.

IE-T1 is the initiating event: LOSP. The frequency of this initiating event was sampled from a distribution. The quantification for Observation 4 is 0.0994. This value is above the mean value of 0.077. The distribution for LOSP was derived from Surry station historical experience, using the methods in Reference B.4. This analysis uses Bayesian models for both the frequency of LOSP and the time to recovery of offsite power. Utility data from 63 LOSP incidents was analyzed to develop a composite offsite power model that combined the effects of failures of the grid, events at the plant (e.g., switchyard problems), and severe weather. The model can be adjusted to reflect specific switchyard design.

OEP-DGN-FS-DG01 is the failure of DG 1 to start. The probability of this event was sampled from a distribution. The quantification for Observation 4 is 0.0133. This value is slightly below the mean value of 0.022. The distribution for this event was derived from the Surry plant records of DG operation for 1980 to 1988. In this period, there were 484 attempts to start the DGs and 19 failures. Eight of these failures were ignored since they occurred during maintenance. A lognormal distribution with an error factor of three was used to model the uncertainty in this event. The error factor was based on a very narrow chi squared uncertainty interval.

/DGN-FTO indicates that DG 2 has started and is supplying power to Unit 2. Thus, DG 3, the "swing" DG at Surry, may be aligned to supply power to Unit 1. The Surry station consists of two units. Emergency power is supplied by three DGs; DG 1 can supply power only to Unit 1, DG 2 can supply power only to Unit 2, and DG 3 can be aligned to supply power to either unit. If DG 2 starts and runs initially, DG 3 is not required for Unit 2. The probability of this event was sampled from a distribution. The quantification for Observation 4 is 0.966, which is almost equal to the mean value (0.97) of this distribution. The distribution was developed from Surry plant data on DG operation at Unit 2 in a manner similar to that for the previous event.

OEP-DGN-FS-DG03 is the failure of DG 3 to start. The quantification for Observation 4 is 0.0133, the same as for OEP-DGN-FS-DG01 above. The same distribution was used for both DG 1 and DG 3, and the sampling was fully correlated.

NRAC–1HR is the failure to restore offsite power within 1 hour. Initially, the probability of this event was sampled from a distribution obtained using the offsite power recovery methods of Reference B.4. As the uncertainty in this event proved to be only a small contributor to the uncertainty in the core damage frequency in the uncertainty analysis performed for the accident frequency analysis alone, it was not sampled in the integrated analysis. For the integrated analysis, NRAC–1HR was set to the mean value of the distribution, 0.44, for every observation in the sample.

REC-XHE-FO-DGEN is the failure to restore a DG to operation within 1 hour. The probability of this event was sampled from the distribution for this operation that appears in the Accident Sequence Evaluation Program (ASEP) generic data base (Ref. B.2). The uncertainty in this event was not a significant contributor to the uncertainty in the core damage frequency. It was not sampled in the integrated analysis, and REC-XHE-FO-DGEN was set to the mean value of the distribution, 0.90, for every observation in the sample.

NOTQ indicates that the RCS PORV(s) successfully reclose during SBO. Event Q is the failure of the RCS PORV(s) to reclose in an SBO sequence, so NOTQ is success. The probability of this event was sampled from a distribution in the stand-alone version of the accident frequency analysis. Because the uncertainty in NOTQ was not a significant contributor to the uncertainty in the core damage frequency, NOTQ was set to 0.973, the complement of the mean value of the distribution for Event Q, for the integrated analysis. The distribution for Event Q was taken from the ASEP generic data base (Ref. B.2).

QS-SBO is the failure of a PORV or SRV in the secondary system to reclose after opening one or more times. For an SBO, the PORVs on the secondary side, also known as the atmospheric dump valves, are not operable, so it is the SRVs that open. The probability of this event was sampled from a distribution. The quantification for Observation 4 is 0.0675, which is considerably less than the mean value (0.27) of this distribution. The distribution for QS-SBO was determined from the ASEP generic data base (Ref. B.2). This analysis considered the number of times an SRV may be expected to open, and the rate at which the SRVs at Surry are expected to fail to reclose (Ref. B.3).

AFW-XHE-FO-CST2 is the failure of the operator to open the manual valves to the auxiliary condensate storage tank, CST2. This action is necessary to provide a supply of water for the AFWS after the primary condensate storage tank is depleted. The probability of this event was sampled from a distribution derived using a standard method for estimating human reliability. AFW-XHE-FO-CST2 is the failure to successfully complete a step-by-step operation following well-designed emergency operating procedures with a moderate level of stress. The method used is presented in Reference B.2, and detailed results may be found in Reference B.3. The quantification for Observation 4 is 0.0762, which is slightly above the mean value (0.065) of this distribution.

B.2.4 Accident Sequence and PDS

The cut set gives specific hardware faults and operator failures. In determining the general nature of the accident, however, many cut sets are essentially equivalent. These cut sets are grouped together in an accident sequence. For example, consider the cut set described above. In the description of the accident, it would have made little difference whether there was no ac power because DG 1 was out of service for maintenance (see Fig. B.2) or whether DG 1 failed to start as in the cut set in Table B.1. The fault is different, and the possibilities for recovery may be different, but the result is the same on a system level. Thus, both cut sets occur in accident sequence T1S-QS-L, along with many other cut sets that also result in the same combination of system failures. In the example, the important development for defining the accident is that DG 1 has failed. Exactly how it failed must be known to determine the probability of failure but is rarely important in determining how the accident progresses after UTAF.

The accident frequency analysis results in many significant accident sequences, typically dozens and perhaps a hundred or so. As the accident progression analysis is a complex and lengthy process, accident sequences that will progress in a similar fashion are grouped together into plant damage states (PDSs). That is, sequences with similar times to UTAF, similar plant conditions at UTAF, and that are expected to progress similarly after UTAF, are grouped together in a PDS. Figure B.1 shows the three sequences that are placed together in PDS TRRR-RSR. They are T1S-QS-L, T1S-L, and T1S-Q-L. (A fourth sequence, T1S-Q-QS-L, sequence 25, would have been placed in TRRR-RSR but was eliminated because of its low frequency.) T1S-QS-L is by far the most likely of these accident sequences and has been described above. Sequence T1S-L is similar to T1S-QS-L but has the AFWS failing at the very start of the accident because of failures in the steam-turbine-driven AFW train itself (such as fail to start, fail to run, etc.). In T1S-Q-L, which is much less probable than either T1S-QS-L or T1S-L, an RCS PORV sticks open, and there is no way to replace the water lost through this valve.

The process of assigning accident sequences to PDSs forms the interface between the accident frequency analysis and the accident progression analysis. The characteristics that define the PDSs are determined by the accident progression analysts based on the information needed in the APET. These characteristics are carefully reviewed with the staff that performs the accident progression analysis to ensure that all situations are included, that the definitions are clear, and that there are no ambiguous cases. Then, every cut set is examined to determine its appropriate PDS. This often requires an iteration through the event tree and fault tree analyses since assignment to the proper PDS may require information, for example, about the containment spray systems, that was not needed to determine the core damage frequency. Thus, it is possible that the cut sets that form a single accident sequence might be separated into two (or more) different PDSs, although this never occurs in the Surry analysis.

The seven letters that make up the Surry PDS indicator denote characteristics of the plant condition when the water level falls below the TAF and consideration of the accident passes from the accident frequency analysis to the accident progression analysis. For PDS TRRR-RSR, each character in the PDS designation is explained below. Recoverable means the system is not operating but can operate if ac power is recovered.

T – RCS is intact at the onset of core damage;

R – Emergency core cooling is recoverable;

R – Containment heat removal is recoverable;

R – ac power can be recovered from offsite sources;

R – The contents of the refueling water storage tank (RWST) have not been injected into the containment but can be injected if ac power is recovered;

S – The steam-turbine-driven AFWS failed at, or shortly after, the start of the accident; the electric-motor-driven AFWS is recoverable; and

R – Cooling for the reactor coolant pump (RCP) seals is recoverable.

A more complete description of the PDS nomenclature may be found in Reference B.1. The assignment of sequences to PDSs is discussed in Reference B.3. For internal initiators at Surry, 25 PDSs were above the cutoff frequency of 1.0E–7/reactor year for the accident progression analysis. They were placed in seven PDS groups based on the initiating events. The seven PDS groups for internal initiators at Surry, in order by decreasing mean core damage frequency, are:

1. Slow SBO;
2. Loss-of-coolant accidents (LOCAs);
3. Fast SBO;
4. Event V (interfacing-system LOCA);
5. Transients;
6. ATWS (failure to scram the reactor); and
7. Steam generator tube ruptures.

The example being followed here goes to the third PDS group, Fast SBO, which consists of only a single PDS, TRRR-RSR.

B.3 Accident Progression Analysis

The accident progression analysis considers the core degradation process and the response of the containment and other safety systems to the events that accompany core degradation. Of particular interest is whether the containment remains intact, since this determines the magnitude of the fission product release in many accidents. In the analyses conducted for NUREG–1150, the accident progression analysis is performed by use of a large event tree. While a simple event tree like that shown in Figure B.1 can be easily illustrated and evaluated with a hand calculator, the event trees used for the accident progression analysis are too large to be depicted in a figure and have so many paths through them that they can only be evaluated by a computer program.

B.3.1 Introduction

The APET for Surry consists of 71 questions. Many of these questions are not of particular interest for PDS TRRR-RSR; therefore, only about half the questions are listed in Table B.2 and shown in Figure B.4.

Table B.2 Selected questions in Surry APET.

Question	Branch Taken or Parameter Defined	Source of Quantification	Meaning of Branch or Parameter
1. RCS Integrity at UTAF?	Br.6	PDS Def.	RCS intact—water loss is through cycling PORVs
8. Status of ac Power?	Br.2	PDS Def.	Will be available when offsite power recovered
10. Heat Removal from SGs?	Br.2	PDS Def.	Will be available when offsite power recovered
12. Cooling for RCP Seals?	Br.2	PDS Def.	Will be available when offsite power recovered
13. Initial Cont. Condition?	Br.3	Acc.Freq.	Containment intact
15. RCS Pressure at UTAF?	Br.1	Summary	RCS is at system setpoint pressure (2500 psia)
16. PORVs Stick Open?	Br.2	Internal	PORVs do not stick open
17. T-I RCP Seal Failure?	Br.1	Acc.Freq.	RCP seals fail
19. T-I SGTR?	Br.2	Experts	No steam generator tube rupture
20. T-I Hot Leg Failure?	Br.2	Experts	No hot leg or surge line failure
21. AC Power Early?	Br.2	Distrb.	Offsite ac power is not recovered before VB
23. RCS Pressure at VB?	Br.3	Internal	The RCS is at intermediate pressure (200 to 600 psia)
28. Cont. Pressure before VB?	Par.1	Summary	The containment is at 26 psia just before VB
29. Time of Accm. Discharge?	Br.2	Summary	The accumulators discharge during core melt
30. Fr. Zr Oxidized In-Ves.?	Par.2	Experts	0.866 of the Zr is oxidized in-vessel
31. Amt. Zr Oxidized In-Ves.?	Br.1	Summary	A high fraction of Zr is oxidized in-vessel
32. Water in Cavity at VB?	Br.2	Summary	The reactor cavity is dry at VB
33. Fr. Core Released at VB?	Par.3	Experts	0.544 of the core is released at VB
34. Amt. Core Released at VB?	Br.1	Summary	A high fraction of the core is released at VB
35. Alpha-Mode Failure?	Br.2	Experts	There is no alpha-mode failure
36. Type of Vessel Breach?	Br.1	Experts	High-pressure melt ejection occurs at VB
38. Size of Hole in Vessel?	Br.1	Internal	The hole in the vessel is large
39. Pressure Rise at VB?	Par.4	Experts	The pressure rise at VB is 56.8 psig
41. Ex-Vessel Steam Explosion?	Br.2	Internal	There is no ex-vessel steam explosion at VB

Table B.2 (Continued)

Question	Branch Taken or Parameter Defined	Source of Quantification	Meaning of Branch or Parameter
42. Cont. Failure Pressure?	Par.7	Experts	The containment failure pressure is 148.4 psig
	Par.8		The LHS number for failure mode is 0.808
43. Containment Failure?	Br.4	Calc.	The containment does not fail at VB
45. AC Power Late?	Br.1	Distrb.	Offsite ac power is recovered during early CCI
46. Late Sprays?	Br.1	Summary	Containment sprays are recovered during early CCI
49. How much H_2 Burns at VB?	Par.8	Internal	0.30 of the hydrogen burns at VB
50. Late Ignition?	Br.1	Experts	Ignition occurs during early CCI
	Par.9	Internal	95 percent of the hydrogen burns if ignition occurs
	Par.10	Internal	The pressure rise scale factor is 1.12
51. Late Burn? Pressure Rise?	Br.1	Calc.	Hydrogen combustion occurs during early CCI
	Par.11	Calc.	The load pressure is 100.2 psia
52. Containment Failure?	Br.4	Calc.	The containment does not fail during early CCI
53. Amount of Core in CCI?	Br.2	Internal	A medium amount of the core is involved in CCI
54. Is Debris Bed Coolable?	Br.1	Internal	The debris bed is coolable if water is available
55. Does Prompt CCI Occur?	Br.1	Summary	Prompt CCI occurs
62. Very Late Ignition?	Br.2	Experts	Ignition does not occur during or after late CCI
68. Basemat Meltthrough?	Br.1	Internal	The basemat eventually melts through
71. Final Cont. Condition?	Br.3	Summary	The only containment failure is basemat meltthrough

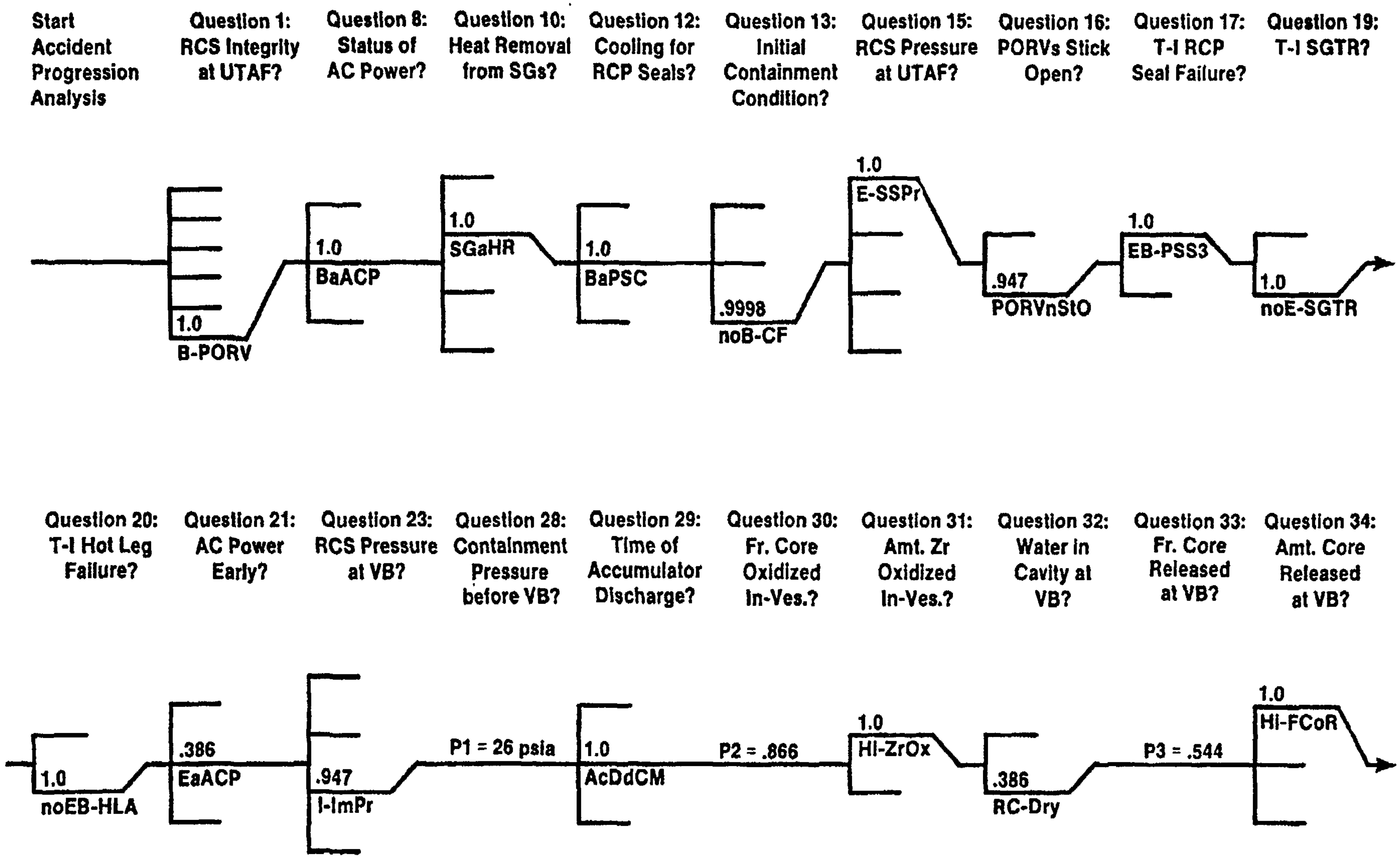


Figure B.4, Sheet 1 Simplified diagram of first part of Surry accident progression event tree. (The complete tree is too large to be depicted graphically. The complete tree is listed and discussed in Appendix A to Ref. B.1.)

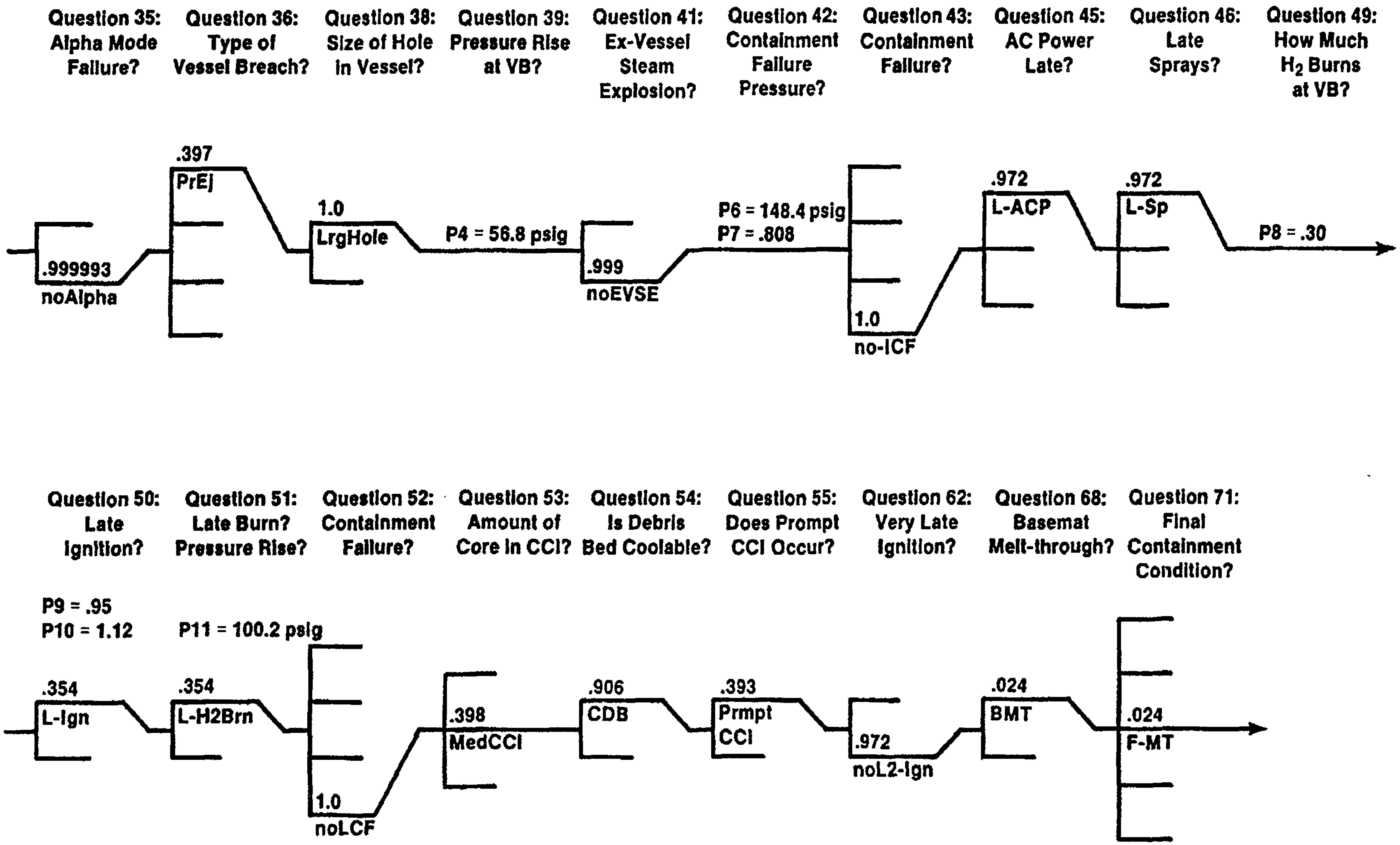


Figure B.4, Sheet 2 Simplified diagram of second part of Surry accident progression event tree.

A full listing of the questions in the Surry APET and detailed discussions of them may be found in Appendix A to Reference B.1. A discussion of how the event trees are defined and evaluated may be found in the methodology discussion in Reference B.5. Many of the branching ratios and parameter values used were determined by expert panels. More detail on this subject may be found in Part I and Part VIII of Reference B.6. EVNTRE, the computer code used to evaluate the APET, is documented in Reference B.7.

Figure B.4 shows the 38 questions displayed and discussed for this example. Only the path chosen for this example is followed from beginning to end in this figure. That is, at each question, only the branch chosen for this example continues on to the next question. In the complete evaluation of the APET for Observation 4 for PDS group 3, many of the branches shown as ending in Figure B.4 do terminate because they have zero probability.

However, many other branches shown as ending in Figure B.4 have nonzero probability and do propagate to the end of the tree. They are undeveloped in Figure B.4 because of space limitations.

In Figure B.4, which is best read in conjunction with Table B.2, the probability of the branch taken is shown above the line. It is the probability of that branch for the entire question and may have contributions from paths other than the one followed for this example. That is, all paths through the APET pass through every question. The probability of a particular branch in Figure B.4 reflects all paths, not just the one being followed in this example, and thus may be different from the probability for this path. Below the line in Figure B.4 is the branch mnemonic abbreviation. This is a succinct way of referring to each branch in the tree, and it is useful to have this information when relating this abbreviated Surry APET to the complete APET listed in Appendix A to Reference B.1.

The complete APET contains case structure, which is not shown in Figure B.4. By defining different cases for a question, different branch probabilities may be defined that depend on the branches taken at previous questions. For example, the branch taken at Question 15, RCS Pressure at UTAF, depends upon the RCS Integrity at UTAF, Question 1. This dependency is implemented by defining a number of cases. Case 2 is the system setpoint pressure (2500 psia) case for Question 15. One of the applicability conditions for Case 2 is that there be no break in the RCS at UTAF, i.e., that Branch 6 was taken at Question 1. For Case 2, the probability for the first branch, system setpoint pressure, is 1.0. Only the total branch probability for the path of interest can be shown in Figure B.4. There is no way to show branching probabilities as functions of the case structure for each question in a compact plot of the APET such as this.

As discussed above, for Observation 4, the accident frequency analysis determined that PDS TRRR-RSR had a frequency of 4.8E–7/reactor year. As PDS group 3 consists solely of TRRR-RSR, the frequency of group 3 is also 4.8E–7/reactor year for Observation 4. The APET is evaluated without regard to this frequency, and the result is a conditional probability for each path given the occurrence of PDS group 3. There are too many paths through the APET for us to be able to keep and treat each path individually. Therefore, paths that are similar as far as the release of fission products and risk are placed together in accident progression bins (APBs or just "bins") as explained in Section B.3.4. For the bin that results from the path followed in this example, denoted GFA-CAC-ABA-DA, the conditional probability is 0.017. The absolute frequency of this bin from PDS group 3 is the product of these two values, or 8.1E–9/reactor year.

Table B.2 lists the 38 questions shown in Figure B.4. These are the most important questions for following TRRR-RSR through the APET. The question is often given in abbreviated form to avoid using two lines. The "Branch Taken or Parameter Defined" column gives the branch taken at that question for the path being followed through the APET. If a parameter is defined in the question, the parameter number is given. The "Source of Quantification" column gives the source of the branch probability or the distribution for the parameter value for this question. PDS Def. means that the branch taken is determined by the definition of the PDS. Acc. Freq. means that the split between the branches at this question was determined in the accident frequency analysis. "Summary" indicates that the branch taken at this question is determined solely by the branches taken at previous questions. "Internal" means that the split between the branches, or the parameter value, was determined by the NUREG–1150 team of analysts, usually with assistance from other experts in various national laboratories. "Distrb." means that the probability of offsite power recovery was determined from distributions of power recovery as a function of time prepared for each reactor site. "Experts" indicates that the sampling is from a distribution determined by one of the expert panels that considered the most important issues for risk.

A discussion of each question follows in Section B.3.2. An expanded discussion of a few questions that were quantified by panels of experts follows in Section B.3.3. Finally, the binning of the results of the evaluation of the APET is discussed in Section B.3.4.

B.3.2 Discussion of APET Questions

Question 1. RCS Integrity at UTAF?

This question defines the state of the RCS at the start of the accident progression analysis. UTAF indicates the uncovering of the TAF, which is the nominal starting point for this analysis. The first character in the PDS definition, "T", indicates that TRRR-RSR has no failures of the RCS pressure boundary. Branch 6 is chosen; the water loss is through the cycling PORVs.

Question 8. Status of ac Power?

Branch 2 is chosen as indicated by the fourth character in the PDS definition. This is the "available" state, and it indicates that ac power will be available throughout the plant if offsite power is recovered after UTAF. The accident frequency analysis concluded that recovery of power from the diesel generators was of negligible probability. Recovery of offsite power in time to prevent core damage was considered by the accident frequency analysis. Recovery of offsite power after the ostensible onset of core damage but before vessel failure is more likely than not for TRRR-RSR. Recovery of power would allow the high-pressure injection system (HPIS) and the containment sprays to operate as these are also in the available state at UTAF. (The questions concerning emergency core cooling system (ECCS) and spray states are not listed in the interest of brevity.)

Question 10. Heat Removal from SGs?

As determined by the sixth letter of the PDS indicator, Branch 2 is chosen. This branch indicates that the steam-turbine-driven AFWS is failed, but the electric-motor-driven AFWS is available to operate when power is restored.

Question 12. Cooling for RCP Seals?

The last character of the PDS definition indicates that the accident frequency analysis concluded that there would be no cooling water flow to the RCP seals unless ac power was recovered. Thus, Branch 2 is taken.

Question 13. Initial Containment Condition?

The Surry containment is maintained below atmospheric pressure, at about 10 psia, during operation. The accident frequency analysis concluded that the probability of a pre-existing leak is negligible and that the probability of an isolation failure at the start of the accident was 0.0002. The more likely branch, no containment failure (Branch 3), is followed in this example.

Question 15. RCS Pressure at UTAF?

This question summarizes the information in the previous questions to determine the RCS pressure at the onset of core damage. As there is no break in the pressure boundary and no heat removal by the AFWS, the only water loss mechanism is the cycling PORVs: the RCS must be at the setpoint pressure of the PORVs, about 2500 psia. This pressure range is indicated by Branch 1.

Question 16. PORVs Stick Open?

After the core degradation process has proceeded for some time, the PORVs will be passing hydrogen and superheated steam and will be operating at temperatures well in excess of those for which they were designed. Based on the rate at which PORVs fail to reclose at normal operating conditions, the number of cycles expected, and allowing for degraded performance at high temperatures, failure of the PORVs was estimated to be of indeterminate probability. As there was no information available on PORV performance at temperatures considerably above the design temperature, a uniform probability distribution from 0.0 to 1.0 was used for this question. That is, the probability that the PORVs will stick open is equally likely to be anywhere between 0.0 and 1.0. In Observation 4, the value for PORV failure is 0.0528. This example follows the more likely branch, Branch 2, and the PORVs reclose.

Question 17. Temperature-Induced (T-I) RCP Seal Failure?

In normal operation, the seals around the shafts of the reactor coolant pumps (RCPs) are kept from overheating by a flow of relatively cool water. If this cooling flow is not available, the seal material may become too hot and fail. Failure of the RCP seals is important in both the accident frequency analysis and the accident progression analysis. In the accident frequency analysis, whether the seals fail, and when they fail, determines the time to UTAF and the RCS pressure at UTAF. In the accident progression analysis, if the seals have not failed before UTAF or whether the seals fail after UTAF may determine the RCS pressure when the vessel fails. The containment loads at VB are strongly dependent on the RCS pressure at that time.

As part of the accident frequency analysis, an expert panel was convened specifically to consider the failure of RCP seals. One of their conclusions was that the seals must be deprived of cooling for some time before failure is likely. In TRRR-RSR, UTAF occurs fast enough that the probability of RCP seal failure calculated in the accident frequency analysis was negligible. That is, by the time the seals have been without cooling long enough to have a significant chance of failure, the water level has dropped below the TAF and the consideration of the accident has passed to the accident progression analysis. In the accident sequence chosen for this example, then, seal failure only occurs in the accident progression analysis.

In the accident frequency analysis, the question of RCP seal failure is sampled zero-one; that is, in some observations a seal-failure branch has a probability of 1.0, and in other observations the no-seal-failure branch has a probability of 1.0. The accident progression analysis samples RCP seal failure the same way for consistency. For the entire sample, the probability of seal failure for this case where the RCS is at setpoint pressure (2500 psia) is 0.71. That is, of the 200 observations, 142 have seal failure and 58 have no seal failure. In Observation 4, the seals fail, so Branch 1 is taken. More discussion on the matter of RCP seal failure may be found in Section B.3.3 and in Reference B.8.

Question 19. Temperature-Induced (T-I) Steam Generator Tube Rupture (SGTR)?

After some period of core melt, the gases leaving the core region are expected to be quite hot. If these gases heat the steam generator (SG) tubes sufficiently, failure of the tubes may be possible. The expert panel that considered this issue concluded that T-I SGTR was possible but very unlikely if the RCS was at PORV setpoint pressure, and not possible if the system was at less than setpoint pressure (Ref. B.6). The failure of the RCP seals has reduced the RCS pressure below the setpoint of the PORVs, so, for Observation 4, there is no possibility of T-I SGTR, and Branch 2 is taken.

Question 20. Temperature-Induced Hot Leg Failure?

The very hot gases leaving the core region during melt may also heat the hot leg or the surge line to temperatures where failure is possible. The experts considered this failure much more likely than T-I SGTR, but only if the RCS was at, or near, the PORV setpoint pressure (Ref. B.6). The failure of the RCP seals has reduced the RCS pressure considerably below the setpoint of the PORVs, so, for Observation 4, there is no possibility of T-I hot leg or surge line failure. Branch 2 is taken.

Question 21. AC Power Early?

This question determines whether offsite power is recovered in time to restore coolant injection to the core before vessel failure. Distributions giving the probability of offsite power recovery as a function of time for the Surry plant are sampled to obtain the values used in this question (Ref. B.4). The times marking the beginning and the end of the time period considered were determined by considering the rate at which this accident progresses and the nature of the plant. For PDS TRRR-RSR, case 2 of this question is applicable; the time period is 0.5 to 2.0 hours after the start of the accident (LOSP). The average value for power recovery in this period for this case is 0.565. The value in Observation 4 is slightly above average at 0.614. If power is recovered during this period, it is likely that vessel breach will not occur. Because an example that proceeds to vessel breach is desirable, the less likely branch is chosen at this question. Branch 2 indicates that offsite power is not available in the plant during this period but may still be recovered in the future.

Question 23. RCS Pressure at VB?

This question determines the pressure in the RCS, including the vessel, just before the vessel fails. For the cases with large breaks in the RCS or with no breaks in the RCS, this pressure is well known. For cases with small (S2) or very small (S3) breaks, the pressure at VB depends upon the time between core slump and VB and the rate at which the pressure decays away following the steam spike at core slump. The RCP seal failure may be of large S3 or small S2 size although all are classed as S3 breaks in this analysis. Taking the range of break sizes and the likely delay between core slump and vessel breach into account, it was estimated that it was equally likely that the RCS pressure at VB would be in the High range, the Intermediate range, or the Low range (Ref. B.6). This question is sampled zero-one. In Observation 4, the Intermediate range is selected. Therefore, all of the accident, except the 5.3 percent with the PORVs stuck open, goes to Branch 3.

Question 28. Containment Pressure before VB?

The total pressure in the containment just after vessel breach consists of the baseline pressure before breach plus the pressure rise associated with the events at VB. (The pressure rise at VB is considered in Questions 39 and 40.) The containment pressure before VB is a function of spray operation and the magnitude of the blowdown from the RCS. The path followed in this example has no sprays and no large break. The results of detailed mechanistic simulation codes indicate that the containment atmospheric pressure will be around 26 psia in this case. Parameter 1 is set to 26 in this question. As the RCS pressure was above the accumulator setpoint when the core uncovered, and is below the setpoint (due to the RCP seal failure) at VB, the accumulators must have discharged during the core melt. Branch 2 is chosen.

Question 30. Fraction of Zr Oxidized In-Vessel?

The fraction of the Zr oxidized in the vessel before VB determines the rate of the core degradation process and temperatures of the gases leaving the core region. The amount of unoxidized Zr in the core debris leaving the vessel is also important in determining the nature of the core-concrete interaction (CCI). The expert panel provided distributions for this parameter for cases that depended upon the RCS pressure and the time of accumulator discharge (Ref. B.6). The path followed here has setpoint pressure in the RCS at the start of core melt and accumulator discharge during core melt. Observation 4 contains the value 0.866 for parameter 2 for this case. The median value for this distribution is 0.45; the value in Observation 4 is the 91st percentile value. As the fraction of Zr oxidized in the vessel is related to the temperature of the gas leaving the core by a known physical mechanism, the value for this parameter is as rank correlated with the probability of T-I hot leg failure as possible.

Question 31. Amount of Zr Oxidized In-Vessel?

The expert panel that considered containment loads at vessel breach gave distributions for two discrete levels of in-vessel Zr oxidation. Therefore, the oxidation fractions obtained from a continuous distribution in the previous question must be sorted into two ranges or classes. This is accomplished by Question 31; the fraction 0.40 divides the fraction of Zr oxidized in-vessel into High and Low ranges. The value of parameter 2 selected from the experts' distribution in the previous question, 0.866, falls in the High range; Branch 1 is taken.

Question 32. Water in Reactor Cavity at VB?

At Surry, the cavity is not connected to the containment sumps at a low level. The only way to get an appreciable amount of water in the cavity before VB is for the sprays to operate. As there is no electric power to operate the spray pumps in this blackout accident, the cavity is dry at VB in the path followed in this example. This is indicated by Branch 2.

Question 33. Fraction of Core Released from Vessel at Breach?

The expert panel provided a distribution for the amount of the core ejected promptly when the vessel fails (Ref. B.6). This is the fraction of the core that can be redistributed in the containment by the subsequent gas blowdown in a direct containment heating event. Observation 4 contains the value 0.544 for parameter 3. This is the 92nd percentile value. The median value is 0.27.

Question 34. Amount of Core Released from Vessel at Breach?

This question sorts the parameter values obtained from the experts' distribution in the previous question into three classes. The fraction 0.40 divides the High range from the Medium range for the fraction of core released at VB. The value of parameter 3 selected from the experts' distribution in the previous question falls in the High range; Branch 1 is chosen.

Question 35. Alpha-Mode Failure?

An alpha-mode failure is a steam explosion (fuel-coolant interaction) in the vessel that fails the vessel in such a way that a missile fails the containment pressure boundary as well. The distribution for this failure mode was constructed from the individual distributions contained in the Steam Explosion Review Group report (Ref. B.9) modified and updated as explained in Reference B.6. The alpha-mode failure probability in Observation 4 is 0.00011. This is considerably less than the mean value. It is so low that alpha-mode failures are truncated within the tree and do not appear in the results. The path selected for this example follows the more probable branch, Branch 2.

Question 36. Type of Vessel Breach?

This question determines the way in which the vessel fails. The possible failure modes are pressurized ejection, gravity pour, or gross bottom head failure. A panel of experts considered the relative likelihood of these possible failure modes (Ref. B.6). Their aggregate conclusion is sampled zero-one. The mode selected in Observation 4 is pressurized ejection (also denoted high-pressure melt ejection). For the whole sample, this failure mode is selected 60 percent of the time for the case where the vessel is at a high or intermediate pressure. Branch 1 indicates pressurized ejection upon vessel breach.

Question 38. Size of Hole in Vessel?

The experts who considered the loading of the containment at vessel breach gave pressure rise distributions that depend upon the size of the hole in the vessel. Hole size was also to have been determined by the experts, but no usable results were obtained. The hole size question was considered by a national laboratory expert in this field (Ref. B.6). He concluded that a small hole (nominal size = 0.1 m^2) was much more likely than a large hole (nominal size = 2.0 m^2). This question is sampled zero-one. Only 10 percent of the time is the large hole branch, Branch 1, selected as it was in Observation 4.

Question 39. Pressure Rise at VB?

The magnitude of the pressure rise in containment that accompanies vessel breach was determined by a panel of experts (Ref. B.6). In defining their distributions, the experts took into account all the pressure rise mechanisms, including vessel blowdown, steam generation, hydrogen burns, ex-vessel steam explosions, and direct containment heating. The pressure rise at vessel breach is treated in two questions, 39 and 40, in the Surry APET because the experts considering this issue defined so many cases. The large hole cases are considered in Question 39. The applicable case for the path being followed in this example is case 11: large hole, high fraction of the core ejected at breach, RCS at intermediate pressure, and dry cavity. For Observation 4, the 34th percentile value, 56.8 psig, was selected for this case. Parameter 4 is set to this value. This issue is discussed further in Section B.3.3.

Question 41. Ex-Vessel Steam Explosion?

This question determines whether a significant steam explosion occurs when the hot core debris falls into water in the reactor cavity upon vessel breach. In the path for this example, the cavity is dry, so there is no steam explosion, which is indicated by Branch 2.

Question 42. Containment Failure Pressure?

Two sampled variables are determined in this question. The first is the failure pressure of the containment. It is sampled from a distribution provided by structural experts who considered the Surry containment specifically. The other value is a random number between 0.0 and 1.0 that is used to determine the mode of failure if the containment fails. The value for the failure pressure in Observation 4 is 148.4 psig. This is the 93rd percentile value. The mean and the median failure pressures are around 126 psig. The random number selected for determining the mode of failure is 0.808 for Observation 4. Thus, in this question, parameter 6 is assigned a value of 148.4 psig and parameter 7 is assigned a value of 0.808. This issue is discussed further in Section B.3.3 and in Reference B.6.

Question 43. Containment Failure and Type of Failure?

This question determines if the containment fails shortly after vessel breach, and, if it fails, the mode of failure. This calculation is done in a FORTRAN "user function," which is evaluated at this question in the APET. Failure is determined by comparing the load pressure with the failure pressure (Refs. B.5 and B.6). In the user function, the failure pressure is converted to absolute pressure (163.1 psia) and the load pressure is calculated by summing the baseline containment pressure (parameter 1, see Question 28), 26 psia, and the pressure rise at VB (parameter 4, see Question 39), 56.8 psi. The load pressure, 82.8 psia, is less than the failure pressure so there is no containment failure at vessel breach in Observation 4. No containment failure is indicated by Branch 4.

Question 45. AC Power Late?

This question determines whether offsite power is recovered after vessel breach and during the initial period of CCI. The same basic distributions sampled in Question 21 are sampled again to obtain the probability of power recovery in this period. The average value for power recovery in this period for this case is 0.888. The value in Observation 4 is slightly above average at 0.927. This is the probability that power is recovered in this period if it was not recovered in the previous period, and it applies only to the fraction, 0.386, that did not have power recovered in the previous period. The most likely branch, Branch 1, is taken here; the path being followed in this example thus has power recovery at this point.

Question 46. Late Sprays?

As the sprays were available to operate at the start of the accident (Question 6, not discussed in the interest of brevity), they operate now that power has been restored throughout the plant. Branch 1 is selected for the path of interest.

Question 49. How Much Hydrogen Burns at Vessel Breach?

The restoration of power means that the sprays will begin to operate in the containment and that ignition sources will probably be present. The sprays will condense most of the steam in the containment and may convert the atmosphere from one that was inert because of the high steam concentration to one that is flammable. To determine the hydrogen concentration in the containment atmosphere during this period, the fraction of the available hydrogen burned at VB must be known. For the path of interest, pressurized ejection at VB with no sprays operating (the sprays were recovered after VB), there is a good chance that all or most of the the containment would have been effectively inert at VB because of the steam concentration. It was estimated internally that, on the average, 30 percent of the hydrogen produced in-vessel would burn at VB. Thus, parameter 8 is set equal to 0.30.

Question 50. Late Ignition?

This question determines the likelihood of ignition and sets the values of two parameters. The experts who considered ignition concluded that, if electric power were available, ignition was almost ensured in a matter of seconds or minutes, given that the atmosphere was flammable. In the path of interest, due to power recovery and the de-inerting of the containment, ignition is essentially ensured. Parameter 9 is the conversion ratio for hydrogen combustion, i.e., the fraction of the hydrogen that burns if there is ignition. The Surry containment is fairly open, and steam condensation due to the spray action is expected to make it well mixed at this time. The conversion factor is estimated to be 0.95, and parameter 9 is set to this value. Parameter 10 is the scale factor applied to the adiabatic pressure rise. A distribution was obtained for this value internally. The value for Observation 4 is 1.12, the 91st percentile value, and parameter 10 is set to this value. (Values of the scale factor greater than 1.0 account for the possibility that local flame acceleration will result in pressures greater than those calculated for deflagrations using the adiabatic assumptions. Global detonations were not considered at Surry.)

Question 51. Late Burn? Pressure Rise?

In this question, a FORTRAN "user function" is evaluated to determine if the containment atmosphere is flammable and, if it is, the total pressure that results from the ensuing deflagration. The amount of hydrogen in the containment is computed from the fraction of the Zr oxidized before vessel failure (parameter 2, see Question 30) and the fraction of the existing hydrogen that burned at vessel failure (parameter 8, see Question 49). This assumes that the ignition takes place before CCI or early in the CCI, i.e., before any appreciable amount of hydrogen has been generated by the CCI. The fraction of the hydrogen available that is consumed in the deflagration is given by the conversion ratio, parameter 9, read in the previous question. The baseline pressure is determined from the masses of the different gas species in the containment assuming a 50 percent steam mole concentration. The pressure rise calculated with the adiabatic assumptions is multiplied by the scale factor (parameter 10, Question 50) to obtain the final load pressure. For Observation 4 and the path of interest, 253 kg-moles of hydrogen burned resulting in an adiabatic pressure rise of 64.7 psia. The scaled pressure rise is 72.6 psia, and the total load pressure is 100.2 psia. Parameter 11 is set to this value.

Question 52. Containment Failure and Type of Failure?

This question determines if the containment fails several hours after vessel breach. If CCI occurs, failure at this time would be during the initial portion of CCI. This is designated the "Late" period. If the containment fails, the mode of failure is determined. This calculation is done in a FORTRAN "user function" as in Question 43. Failure is determined by comparing the load pressure with the failure pressure (parameter 6, see Question 42). The failure pressure is 163.1 psia. The load pressure is 100.2 psia, so there is no late containment failure for Observation 4. This is indicated by Branch 4.

Question 53. Amount of Core in CCI?

This question determines the amount of core available for CCI, should it take place. The path being followed has pressurized ejection at VB and a large fraction of the core ejected from the vessel. Pressurized ejection means that a substantial portion of the core material was widely distributed throughout the containment. For this case, it was estimated that between 30 and 70 percent of the core would be available to participate in CCI. This is the Medium range for CCI, indicated by Branch 2.

Question 54. Is Debris Bed Coolable?

This question determines if the core debris in the reactor cavity will be coolable, assuming that water is available. The path being followed has pressurized ejection at VB, so a substantial portion of the core material was widely distributed throughout the containment, and this portion of the core debris is likely to be coolable. It was internally estimated that, for this case, the probability of the debris in the cavity being in a coolable configuration is 80 percent (Ref. B.6). Note that for the debris to actually be cooled, in addition to the debris being in a coolable configuration, water must be present in the cavity at vessel breach and must be continuously replenished thereafter. This question only determines whether the debris configuration is coolable. The most likely branch, Branch 1, is followed for the example path, indicating that the debris bed configuration is potentially coolable. In the path being followed, the reactor cavity is dry at vessel breach, so whether the debris bed is coolable is a moot point.

Question 55. Does Prompt CCI Occur?

The reactor cavity is dry at vessel breach since the sprays did not operate before VB, so CCI begins promptly. While the sprays are recovered in the period following VB, they may not start to operate until some time after vessel breach. It was internally concluded that if the cavity was dry at VB, the debris would heat up and form a noncoolable configuration, and that, even if water was provided at some later time, the debris would remain noncoolable. Thus, prompt CCI occurs, and Branch 1 is chosen.

Question 62. Very Late Ignition?

Ignition leading to a significant hydrogen burn does not occur during the late portion of CCI, or after CCI, in the path being followed through the Surry APET for this example. Ignition occurred in the previous period and ac power has been available since that time. As an ignition source has been present since the late burn, any hydrogen that accumulates after the burn will burn off whenever a flammable concentration is reached. Burns at the lower flammable concentration limit will not threaten the Surry containment. Therefore, Branch 2, no ignition, is taken at this question.

Question 68. Basemat Meltthrough?

The path of interest has a medium amount of the core involved in CCI and the sprays start after VB and operate continuously thereafter. As the basemat at Surry is 10 feet thick, eventual penetration of the basemat by the CCI was internally judged to be only 5 percent probable for this case (Ref. B.6). Branch 1 is followed at this question. Although this branch indicates basemat meltthrough and is less probable than the other branch, it is taken because the source term and risk analyses are not of much interest if there is no failure of the containment.

Question 71. Final Containment Condition?

This is the final question in the Surry APET; it summarizes the condition of the containment a day or more after the start of the accident. Only the most severe failure is considered, that is, if the containment failed at vessel breach, a later basemat meltthrough would be ignored. In the path followed through the APET, there were no aboveground failures, so Branch 3 is selected, indicating basemat meltthrough.

B.3.3 Quantification of APET Questions by Expert Judgment

This section contains detailed quantification of three questions in the APET that were considered by the expert panels. The first is Question 17: probability of RCP seal failure. The second is Question 39: pressure rise in the containment at VB. The last is Question 42: containment failure pressure.

Temperature-Induced RCP Seal Failure

Question 17 determines whether there is a temperature-induced failure of the RCP seals. This failure mechanism is considered in the accident frequency analysis as well as in the accident progression analysis as it is important to both. The panel of experts that considered RCP seal failure was convened as part of the accident frequency analysis, and the results of that panel were used here as well. These experts concluded that the seal degradation depended primarily on the amount of time the seals had spent at elevated temperatures. For fast SBO accidents such as TRRR-RSR, the seal failure would not occur before UTAF. (It could, however, occur after UTAF.) Thus, for the accident sequence and PDS considered in this example, RCP seal failure is primarily of interest in the accident progression analysis.

The RCP seal is designed to allow a small amount of leakage (3 gpm) of primary coolant water during normal operation. The purpose of the leakage is to cool the shaft of the pump. This leak rate is well within the capacity of the normal makeup system. During an SBO with loss of the AFWS, there is no heat removal from the RCS and no cooling flow to the RCP seals. As the temperature and pressure of the reactor coolant system rise, the ability of the RCP seals to control leakage at acceptable levels determines whether the integrity of the RCS will be maintained. Significant leakage of RCS water through the seals will hasten the uncovering of the core and reduce the time available for restoration of ac power and core cooling.

The RCP seal is a complex multistage labyrinth seal that uses elastomer o-rings and free-floating seal plates. The integrity of the o-rings and the stability of the plates depend on the pressure in the RCS and the temperature of the water passing through the seal. Should the RCP seals fail, the size of the leak and the time of failure are functions of the combination of o-ring and seal plate failures in the seal assembly.

In the operating history of Westinghouse reactors, there has never been a seal failure caused by loss of seal cooling. However, there have been six incidents where seal cooling has been lost in U. S. Westinghouse reactors. In each case, the loss of seal cooling lasted less than 1 hour, which is the minimum time considered necessary to degrade the seal o-rings. While instability of the seal plates could occur at any time after the loss of seal cooling, this phenomenon has not been observed in any of the incidents to date. The o-ring material has been tested by both the Idaho National Engineering Laboratory (INEL) (Refs. B.10 and B.11) and the French national electrical utility, EDF. These tests showed that the o-ring material can be degraded when subjected to off-normal temperatures and pressures.

Both Westinghouse (Ref. B.12) and Atomic Energy of Canada Ltd. (AECL) (the latter under contract to the NRC) have performed extensive analyses of the performance of the RCP seal assemblies under off-normal conditions. Neither these tests or analyses, nor the incidents to date, have provided sufficient data for a quantitative probability model of RCP seal leak rate as a function of time after loss of cooling on which all parties can agree. Furthermore, the analyses by Westinghouse and AECL are proprietary. For these reasons, the resolution of this issue was delegated to a separate panel of three experts who were familiar with the problem and who had access to this proprietary information.

The three experts on this panel were:

Michael Hitchler, Westinghouse,
Jerry Jackson, U.S. Nuclear Regulatory Commission, and
David Rhodes, Atomic Energy of Canada Limited.

Which expert provided each distribution is not identified. The experts are described below as A, B, and C, which is not necessarily the order given above. They were asked to determine the probability of failure of the Westinghouse RCP shaft seals and corresponding leak rates under SBO conditions. More detail on this issue may be found in Reference B.8.

With the approval of the panel, the issue of RCP seal failures was decomposed into two questions:

1. What is the likelihood of the various combinations of o-ring and seal plate failures in a single RCP, and what is the resulting leak rate for each combination of failures?

2. What correlation, if any, exists between pumps for each combination of similar o-ring and seal plate failures?

The first question simplified the issue by focusing attention on the specific leak paths that might develop in a single pump. The second question expanded the scope of the panel's analysis to develop total leak rates for all of the RCPs. (Surry has three pumps.)

In resolving the first question, the experts agreed to develop a single event tree that would represent the set of all possible failure combinations and their corresponding leak rates for a single pump. A consensus was reached on the expected leak rate assigned to each set of failures. Each expert assigned his own probabilities to the various events of the event tree to arrive at his own estimate of single pump leak rate probabilities. To resolve the second question, each expert gave his judgments regarding the correlation of failures of event tree events between pumps. Then the experts' correlation elicitations were used to extend each expert's single pump model to obtain leak rates and their probabilities for all three pumps.

The single pump event tree is shown in Figure B.5. The probabilities on the tree are those for Expert A. It should be noted that some of the event probabilities on the tree are shown as functions of time. The experts concluded that degradation of o-ring elastomer material is dependent on the length of time that the o-rings are exposed to uncooled RCS water. The extension of Expert A's elicitation to a three-pump model is shown in Figures B.6 and B.7. Figure B.7 is the continuation of Figure B.6; it shows the various failure combinations for the first stage seal plates of the three pumps, based on Expert A's elicitation on the correlation of first stage seal failures. The five outcomes on Figure B.6 are passed on to Figure B.7, where the first stage o-ring and second stage component failures are shown. The result is 16 possible outcomes on Figure B.7, each with a time-dependent probability. Similar trees were developed for Experts B and C, but each expert's tree was unique because of differences in their elicitations.

To illustrate the method used, the path to outcome 5 in Figure B.7 will be followed. Expert A concluded that this was the most likely outcome; the path starts on Figure B.6, where the upper branch taken at top event B1 indicates success; that is, the first stage seal ring of pump 1 does not fail. The first stage seal rings also do not fail for pumps 2 and 3, so transfer path 1 is reached on Figure B.6. Transfer path 1 is the top entry path on Figure B.7, and Expert A concluded that this was the most likely transfer path (probability = 0.951). In the path to outcome 5 in Figure B.7, the first stage o-ring fails, so the lower branch is taken at

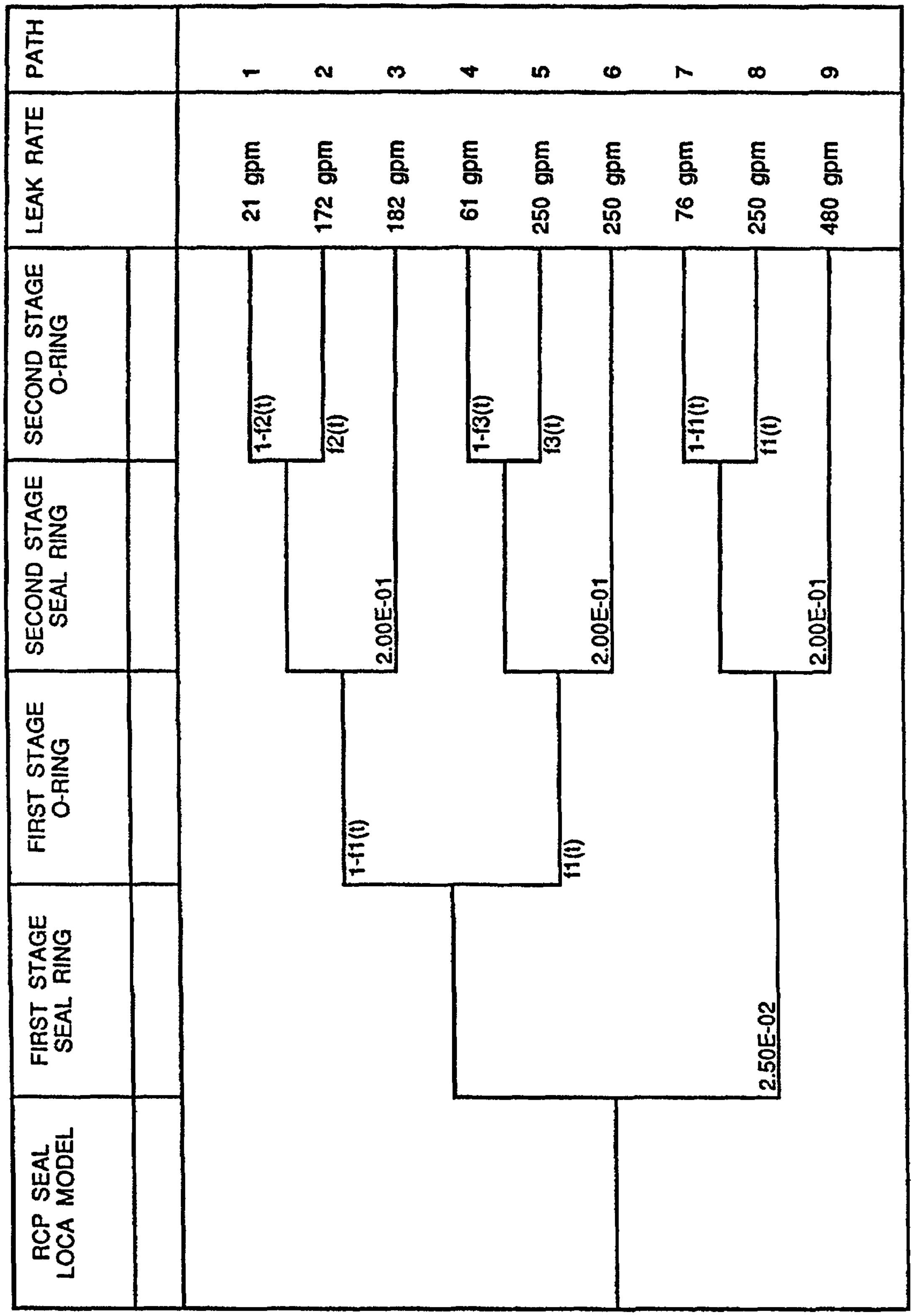


Figure B.5 Event tree used by all three experts in determining the probabilities of different leak rates for a single reactor coolant pump. The branch fractions shown are for Expert A. (This figure is adapted from Section C.4 of Ref. B.8.)

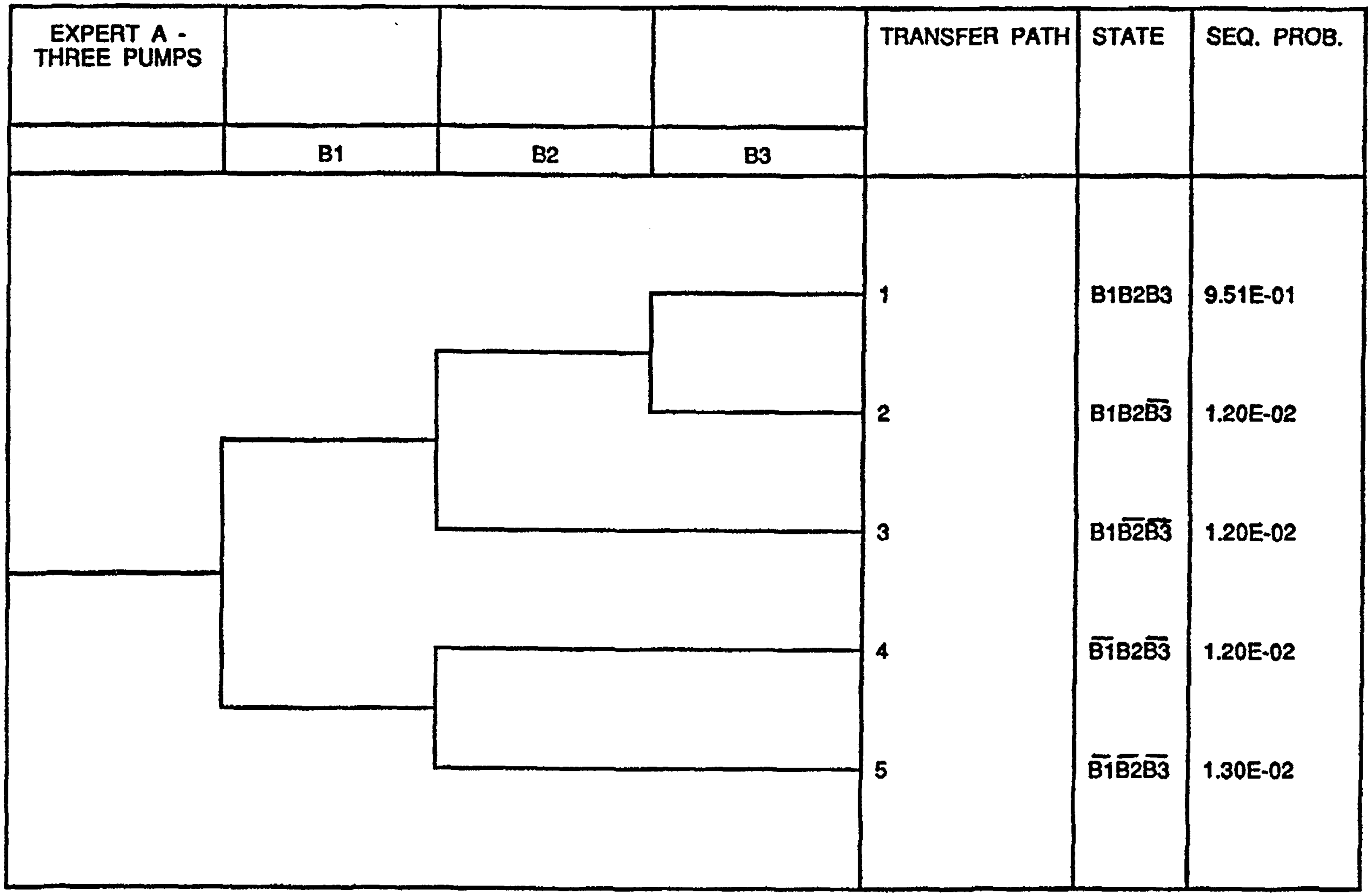


Figure B.6 First part of the event tree used by Expert A in determining the probabilities of different leak rates for all three reactor coolant pumps. The transfer paths indicate the entry point on the second part of this event tree. The top events concern failure of the first stage seal rings; B1 for pump 1, B2 for pump 2, and B3 for pump 3. In the State column, Bi indicates no failure of the seal rings, and $\overline{Bi}$ indicates failure of the seal rings, for pump i. (This figure is adapted from Section C.4 of Ref. B.8.)

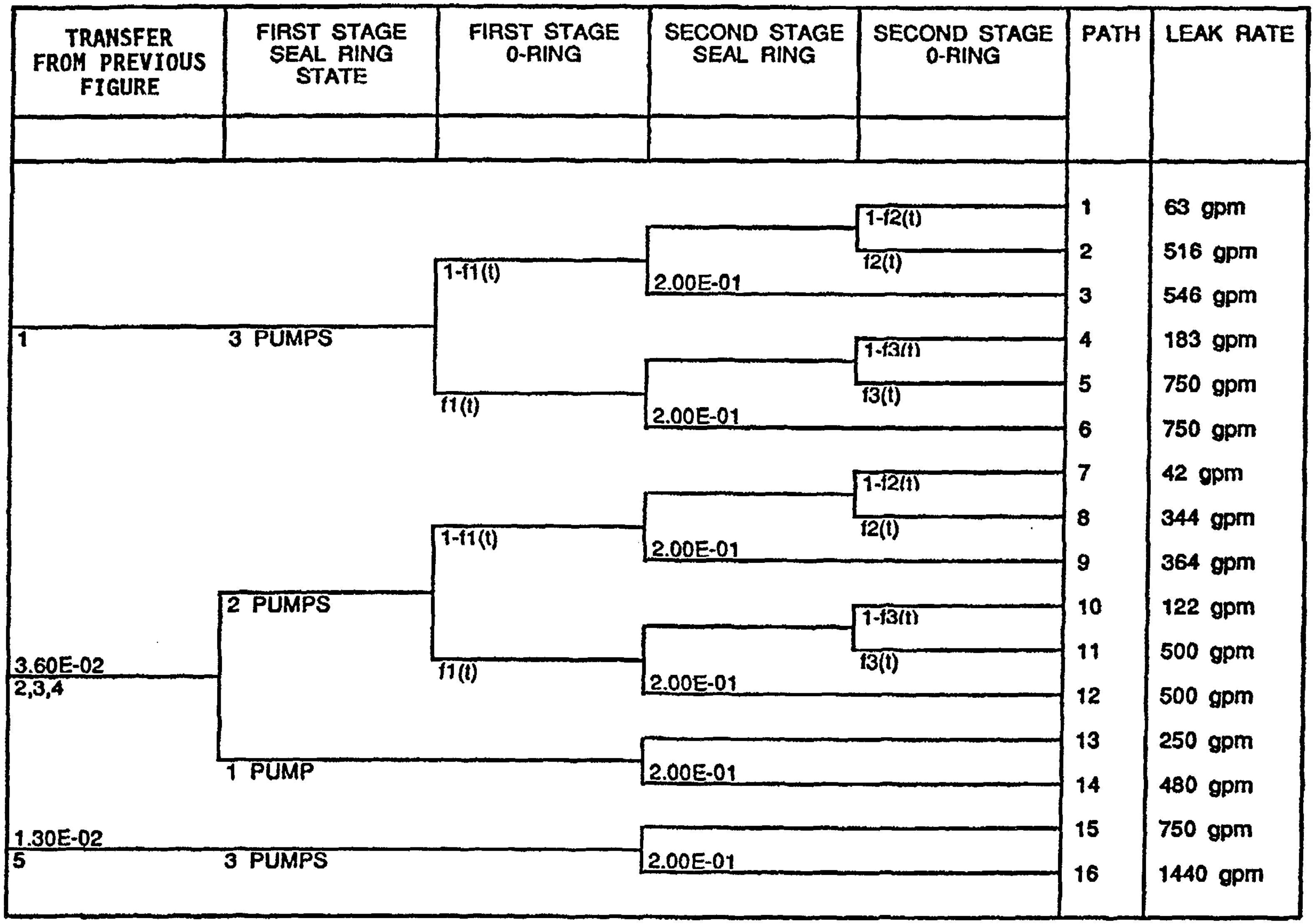


Figure B.7 Second part of the event tree used by Expert A in determining the probabilities of different leak rates for all three reactor coolant pumps. The transfer paths indicate the exit point from the first part of this event tree. (This figure is adapted from Section C.4 of Ref. B.8.)

top event "First Stage O-Ring." For the probability of this branch, Expert A developed a time-dependent model, denoted f1(t) on Figure B.7. Expert A was of the opinion that if the o-ring failed in the seal for one pump, they would fail on the other pumps as well, so the lower path at top event "First Stage O-Ring" represents the failure of the o-rings in all three pumps. At the next top event, the second stage seal rings do not fail, so the upper branch is taken. Expert A assigned a probability of 0.80 to this branch. At the final top event, the second stage o-ring fails in all three pumps. Expert A represented the probability of this failure by another function of time, denoted f3(t). Outcome 5 on Figure B.7 is a 250-gpm leak in all three pumps, for 750 gpm total. The probability of this outcome is a function of time, rising from zero at 1 hour after the loss of core cooling to 0.76 at 2.5 hours.

Experts A and C had fairly similar models for the single pump fault tree. Both treated failure of the first stage o-rings as a step function of time. Experts A and C concluded that failure would be virtually certain by 1.5 hours and 2.0 hours, respectively. Both reasoned that the first stage seal plates would be very reliable, but that integrity of the seals would be compromised by high probability failures of the first and second stage o-rings and second stage seal plates. Experts A and C judged that the likelihood of a second stage failure was somewhat dependent on the status of the first stage as first stage failure could compromise the ability of the second stage to succeed. Expert B's model was considerably more optimistic than those of Experts A and C. He also concluded that the probability of o-ring failure would be a function of time, but with a maximum value of 0.15 for the first stage and 0.50 for the second stage. His probability for seal plate failure was similar to those of Experts A and C, but he did not think that the second stage was dependent on the status of the first stage.

The most significant difference between Expert B and Experts A and C is the failure of.the o-rings as a function of time. Expert B thought that the o-rings would degrade slowly, and, by 4 hours after loss of cooling to the RCS, the RCS would have been depressurized by the operators. He believed that the o-rings would not fail in the depressurized environment. Experts A and C were of the opinion that the degradation of the o-rings would be so rapid that the question of depressurization within 4 hours was moot.

With respect to the correlation of o-ring and seal plate faults between pumps, Expert C's elicitation was the most simplistic. He concluded that similar components would behave similarly in different pumps. Thus, his three-pump leak rate model was exactly the same as his single-pump model, except that the leak rates of the single-pump model are multiplied by three. Experts A and B had significantly more complex elicitations for correlation of faults between pumps. Both had similar models for the correlation of first stage seal plate failures.

They both judged that the first stage seal plates could fail independently of each other, but they agreed to a simplifying assumption that, should similar components in any two pumps fail, the third pump would experience the same failure. Thus, Expert B's model for first stage seal plate failures is the same as that of Expert A in Figure B.6. The probabilities for several of the five outcomes for the first stage seal plate failure tend to be somewhat lower for Expert B than for Expert A. However, both models show the first outcome (all three first stage seals succeed) to be the most dominant outcome by far.

For the second stage, Experts A and B both concluded that the second stage o-rings would all fail in the same manner. But Expert A concluded that the second stage seal plates would all fail in the same manner, while Expert B judged that the second stage seal plates would fail independently.

The final RCP seal LOCA leak rates were calculated by averaging the leak rate probabilities of the three experts for various time intervals. Each expert's leak rate probabilities were given equal weight with respect to the others. The results are shown in Table B.3. (The o-rings in the RCP seals can be made from two types of material. The new material is much more resistant to degradation at high temperatures. The experts considered both types of material. All the pressurized water reactors (PWRs) considered for NUREG-1150 had o-rings made of the old material when these analyses were performed. Table B.3 shows only the results for seals with o-rings composed of the old, less heat-resistant material.)

The entries in the table give the probability of having the total leak rate shown at the times listed. Values in parentheses denote the probabilities that apply if the RCS is not depressurized.

Table B.3 Aggregate results for RCP seal failure with existing o-ring material.

Leak Rate (gpm)	1.5 (h)	2.5 (h)	3.5 (h)	4.5 (h)	5.5 (h)
63	0.31	0.29	0.27	0.27(0.26)	0.27(0.24)
183 to 224	0.15	0.04	0.05	0.05(0.06)	0.05(0.08)
372	0.008	0.005	0.005	0.004	0.003
516 to 546	0.0004	0.0003	0.0003	0.0003	0.0003
602 to 614	0.001	0.0	0.0	0.0	0.0
750	0.53	0.66	0.66	0.66	0.66
1440	0.004	0.004	0.004	0.004	0.004

The time dependence shown in Table B.3 could not be incorporated directly into the accident frequency analysis. Instead, eight RCP seal states were defined, and Table B.3 was used to derive probabilities for these states. Some of the less likely leak rates were combined with similar leak rates. The result for the Surry accident frequency analysis was:

Seal State	Probability	Total Leak Rate and the Time Seals Fail
1.	0.29	Design leakage (no failure)
2.	0.014	183 gpm at 90 minutes
3.	0.53	750 gpm at 90 minutes
4.	0.0043	1440 gpm at 90 minutes
5.	0.016	183 gpm at 150 minutes
6.	0.13	467 gpm at 150 minutes
7.	0.0040	561 gpm at 150 minutes
8.	0.016	183 gpm at 210 minutes

In the accident frequency analysis, each of these eight RCP seal states was considered separately as the different flow rates and different times of failure led to UTAF at different times. This level of detail could not be accommodated in the accident progression analysis. The APET considered only two RCP seal states: failed and not failed. Based on the results of the expert panel given above, the failed state has a probability of 71 percent. The failed state was designated as an S3 break (less than 2-in. diameter) even though the most likely flow rate, 750 gpm total, is in the lower end of the range of flows of the S2 breaks (0.5-in. to 2-in. diameter). This assignment, initiated in the accident frequency analysis, keeps the RCP seal failures separate from the stuck-open PORV cases, since the latter were all classified as S2 breaks and avoids having to split the RCP seal failures between the two break sizes.

As mentioned in the discussion of Question 17, the accident frequency analysis sampled this issue in the zero-one manner, i.e., there were eight states for the RCP seals: seven failure states and one design leakage (no failure) state. In each observation, one of these states was assigned a probability of 1.0 and the other seven were assigned a probability of 0.0. The relative frequency of each state in the entire sample corresponded to the aggregate distribution of the experts, e.g., 29 percent of the observations had the design leakage state with a probability of unity. The accident progression analysis samples RCP seal failure the same way for consistency except that there are only two states. The sample for the accident progression analysis consists of 200 observations, so 142 observations had the failure state selected and 58 had the no-failure state selected.

Pressure Rise at Vessel Breach

Questions 39 and 40 determine the pressure rise at VB in the Surry APET. Two questions are required because of the number of cases to be considered. Vessel failure usually causes the pressure to rise in the containment, sometimes dramatically. A number of mechanisms may contribute to this pressure rise: vessel blowdown, steam generation by the expelled debris, hydrogen combustion, ex-vessel steam explosions, and direct containment heating (DCH). The expert panel convened to consider the containment loads at VB concluded that the contributions of each of these mechanisms were generally not separable. Thus, the distributions for pressure rise provided by the experts include the contributions from all the pressure rise mechanisms. RCS blowdown and DCH cause significant loads to the containment only if the RCS pressure is 10 to 20 atmospheres or more above that of the containment at vessel breach.

After some discussion with the panel, the following case structure for Surry was adopted:

Case	RCS Pressure (psia)	Cavity Water	Sprays Operating
1	2000 to 2500	Full	Yes
1a	2000 to 2500	Half	Yes
1b	2000 to 2500	Dry	No
1c	2000 to 2500	Full	No
3	500 to 1000	Full	Yes
3a	500 to 1000	Half	Yes
3b	500 to 1000	Dry	No
4	15 to 200	Half	Yes

The panel defined eight subcases by considering the following variations (nominal values in parentheses):

Zr Oxidation—High (60 percent) and Low (25 percent),

Melt Fraction Ejected—High (75 percent) and Low (33 percent), and

Initial Hole Size—Large (2 m^2) and Small (0.1 m^2).

As there were eight cases, eight subcases for each meant that each expert provided 64 distributions for pressure rise at VB for Surry. Four members of the containment loadings expert panel considered the pressure rise at Surry. They were:

Kenneth Bergeron, Sandia National Laboratories,
Theodore Ginsberg, Brookhaven National Laboratory,
James Metcalf, Stone and Webster, and
Alfred Torri, Pickard, Lowe, and Garrick.

Expert A approached the problem by using the available CONTAIN (Refs. B.13 and B.14), MAAP (Ref. B.15), and Surtsey results (Refs. B.16 through B.19) to assess pressure rise distributions for three base cases. His base cases were chosen to represent the most severe pressure rises for the three different RCS pressure levels analyzed and were 1b, 3b, and 4. The low Zr oxidation, large hole, and large fraction ejected subcase was used for each base case.

For the middle portions of his base case distributions, Expert A placed the most reliance on the CONTAIN results as reported in NUREG/CR-4896 (Ref. B.20) and some subsequent calculations (Refs. B.21 through B.23). He obtained the extreme values from energy balance calculations. Using a PC spreadsheet program, he then adjusted these base cases for the effects of hole size, the amount of core ejected, and the fraction of Zr oxidized in-vessel to get values for the other 61 subcases.

Expert B also based his "best estimates" on CONTAIN calculations and on scaled experiments. The case for the 500 to 1,000 psia pressure range was taken as a base, and the cumulative distribution function (CDF) for that case was modified to obtain the CDFs for other cases. Expert B concluded that the presence of water in the cavity could either enhance or reduce the pressure, so the median for the wet cavity cases was kept the same as for the dry cavity cases, but the distribution was stretched at both ends. On the low side, an overabundance of water might reduce pressure by two bars. On the high side, calculations indicate the possibility of increasing pressure by one bar. Expert B took his high extreme values from a one cell adiabatic equilibrium code he had written to analyze Zion and Surry. While calculating the low side of the distribution, he considered phenomena that might reduce pressure, such as larger drop diameter or faster trapping.

Dependence on the extent of Zr oxidation, the VB area (hole size), and the fraction of melt ejected was also considered by Expert B. CONTAIN calculations (Refs. B.20 through B.23) have indicated that there is little dependence on previous Zr oxidation, probably because of oxidation starvation in the cavity. The effect of greater hole area is to give higher pressure rises across the entire distribution because the gas would exit with higher velocity. The effect of fraction of core ejected was handled by scaling the base case ratio of final to initial pressure.

Expert C used HMC calculations (Ref. B.24), CONTAIN calculations (Refs. B.20 through B.23), and MAAP calculations (Ref. B.15). He tabulated the cases described in the issue description and applied the code results that appeared to be the most applicable to each case. He was forced to modify the code results in many instances to account for differences between the initial conditions in the code calculations and the case under consideration. Expert C used the HMC calculations for the several cases in which there was water in the cavity and considered the highest pressures calculated by HMC to be the upper bounds of his distributions. The pressure rise without direct heating formed his lower bound.

Expert C relied on CONTAIN and MAAP results in cases in which the cavity was dry. CONTAIN calculations with unconditional hydrogen burn and default burn were averaged and used for the upper part of his distributions, while the MAAP results were used for the lower part of his distributions. Although he believed the CONTAIN calculations to be consistently above the median, he considered the results quite credible. From CONTAIN sensitivity calculations, Expert C was able to estimate the effects of changes in initial conditions, and using these estimates he obtained distributions for the subcases for which no HMC, CONTAIN, or MAAP results were directly applicable.

Expert D used CONTAIN results (Refs. B.20 through B.23) as the basis for his analysis because CONTAIN is currently the only code that has a DCH model. For his base case, he took the high-pressure case with a large fraction of melt ejected (75 percent) and a small initial hole. No further definition of the base case was necessary because Expert D was of the opinion that the effects of co-dispersed water should not be included and that the fraction of the Zr oxidized in-vessel was not particularly important. (CONTAIN runs in which the in-vessel oxidation was varied showed small differences in pressure rise.) To obtain his distribution for this base case, he started from results of the 18-node Surry model with unconditional hydrogen burn as defined in References B.21 and B.22. Expert D adjusted these results to account for alternate particle sizes, an alternate trapping model, and the effect of the thin steel in the containment on peak pressure.

For the small hole cases, Expert D adjusted the CONTAIN pressures upward somewhat since there is the possibility that more than one penetration may fail at or about the same time. For the cases with the sprays operating, he reduced the pressures about 1.5 to 2.5 bars below the pressures in the equivalent cases without the sprays operating. Expert D concluded that changing the particle size assumed in CONTAIN could only decrease the pressure rise. If the particle size assumed in the CONTAIN calculations (1.0 mm) is increased, the pressure rise will decrease because the material in the center of the particle will not have reacted before the particle is quenched. If the particle size is decreased from that assumed in CONTAIN, there is a negligible effect since all the metal in the particle is already reacting. CONTAIN assumes that the core debris distributed throughout the containment during the blowdown phase of the DCH process is homogeneous. Expert D expects the entrained material to be richer in oxides than a homogeneous mixture, which would decrease the pressure rise somewhat. He also pointed out that, when DCH occurs, only a very small portion of the hydrogen pre-existing in the containment or produced during the high-pressure melt ejection (HPME) can be expected to remain unburned after the event is over.

Results for all 64 subcases may be found in Reference B.6. Statistical tests on the 64 subcases showed that many of them could be combined, that the differentiation made on the fraction of Zr oxidized in-vessel could be dropped, and that all the subcases for the low-pressure case (case 4) could be consolidated. The result of the statistical analysis was that there were 13 distinct cases for Surry.

However, the dividing point between high and low fraction ejected used by the expert panel on containment loads, 50 percent ejected, was very near the high end of the aggregate distribution given for fraction ejected by the in-vessel experts. As defined by the loads panel, the high-fraction-ejected subcase has the fraction ejected greater than 50 percent with a 75 percent nominal value and the low-fraction-ejected subcase has the fraction ejected less than 50 percent with a 33 percent nominal value. The aggregate distribution from the in-vessel panel for core fraction ejected has a maximum value of 60 percent, and the probability that the fraction ejected will exceed 50 percent is only about 11 percent.

Not wishing to place 89 percent of the samples in the low-fraction-ejected subcase of the loads panel, and as the "high-low" division was more coarse than necessary, the core-fraction-ejected distribution of the in-vessel panel was divided into three ranges: 0 to 20 percent, 20 to 40 percent, and 40 to 60 percent. The pressure rise distributions from the loads panel were then adjusted to provide pressure rises for these three ranges.

For the 0 to 20 percent ejected range, the average of the low-fraction-ejected results and the case 4 results (RCS pressure < 200 psia) were used. As the low-fraction-ejected case had 33 percent (nominally) ejected, and case 4 had, in effect, no core ejected at high pressure, this appeared to be appropriate. For the 20 to 40 percent ejected range, the low-fraction-ejected results from the loads panel were be used directly since the nominal value used by the loads experts was 33 percent ejected. For the 40 to 60 percent ejected range, the loads low-fraction-ejected distributions and high-fraction-ejected distributions were averaged. The average of the nominal fractions ejected is 54 percent, which is reasonably close to the center of this range. This treatment of the distributions was discussed with, and approved by, a member of the containment loadings expert panel.

This expands the number of cases for Surry from 13 to 19. Plots of the aggregate distributions for these 19 cases are contained in Reference B.6.

For the example being followed through Observation 4, the path through the APET went to case 11 of Question 39. This case has intermediate pressure in the RCS at VB, dry cavity, large hole, and high (40 to 60 percent) core fraction ejected at breach. This is case 3b of the loads panel. The statistical analysis found no significant differences between the expert's results for cases 1, 1a, and 3b. As explained above, the 40 to 60 percent ejected distribution is the mean of the loads panel low-fraction-ejected aggregate distribution and high-fraction-ejected aggregate distribution. Figure B.8 shows the distributions of the four experts and the aggregate for case 3b, large hole, for both core fractions ejected. Also included in Figure B.8 is a plot showing the two aggregates for case 1/1a/3b, large hole, and the aggregate for case 4, as received from the loads panel, and the three aggregate distributions derived therefrom for the three ranges of the in-vessel panel distribution for core fraction ejected. The distribution for 40 to 60 percent ejected was used in the sampling process to obtain the value of 56.8 psig, the 34th percentile value, used for Question 39, case 11, in Observation 4.

Containment Failure Pressure

The value for the containment failure pressure is determined in Question 42. The Surry containment is a cylinder with a hemispherical dome roof. Both the cylinder and the dome are constructed of reinforced concrete. The foundation is a reinforced concrete slab. The containment is lined with welded 0.25-inch plate steel. The containment is maintained below ambient atmospheric pressure, at about 10 psia, during operation. The design pressure is 45 psig. The free volume is about 1,850,000 cm^3. A section through this containment is shown in Figure B.9.

A panel of structural experts was convened to determine the loads that would cause containment failure at Surry and the other plants. As the probability of a global detonation in the Surry containment was considered to be quite small, only static loads were treated for Surry. Such loads would result from the pressure rise that accompanies VB or a deflagration. Typical pressure rise times would be on the order of a few seconds, which is longer than the containment response time.

Four members of the structural expert panel considered failure pressure and failure mode for the Surry containment. They were:

Joseph Rashid, ANATECH Research Corp.,
Richard Toland, United Engineers and Constructors,

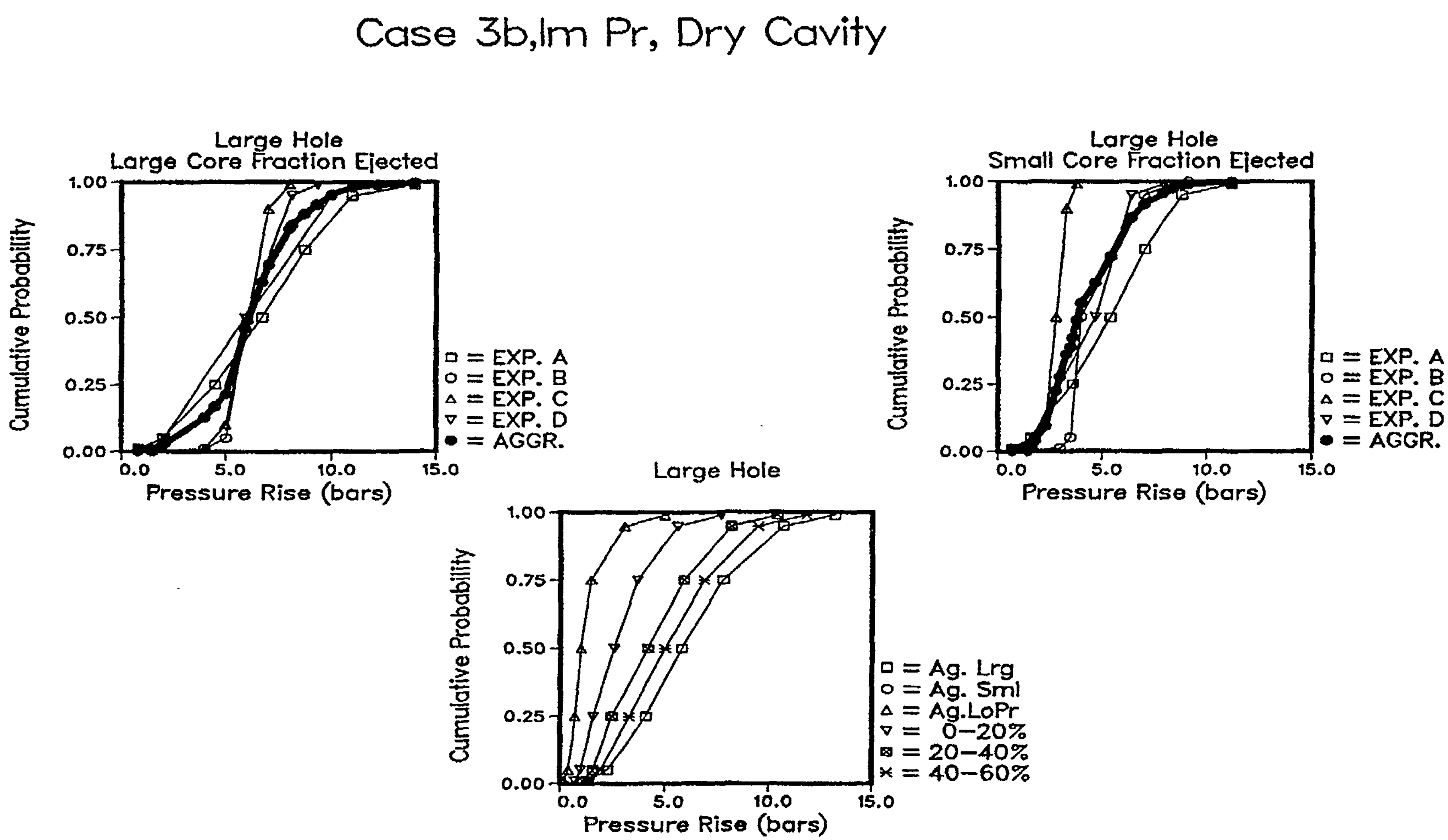


Figure B.8 Results of expert elicitation for pressure rise at vessel breach for Surry. [The pressure rise is shown for the case where the RCS is at intermediate pressure (200–600 psia), the cavity is dry, and the sprays are not operating when the vessel fails by the formation of a large hole. In the top two plots, the first four distributions are those given by the experts and the fifth plot is the aggregate distribution. The lower plot shows the aggregate distributions provided by the experts and the distributions actually used in the APET. The first curve is the large-core-fraction-ejected aggregate from the upper left plot, and the second curve is the small-fraction-ejected aggregate from the upper right plot. The third curve is the aggregate for low pressure (less than 200 psia) in the RCS at VB. The last three curves are the actual distributions used in the APET evaluation. The distribution for 20–40 percent ejected is coincident with the second curve (small fraction ejected).]

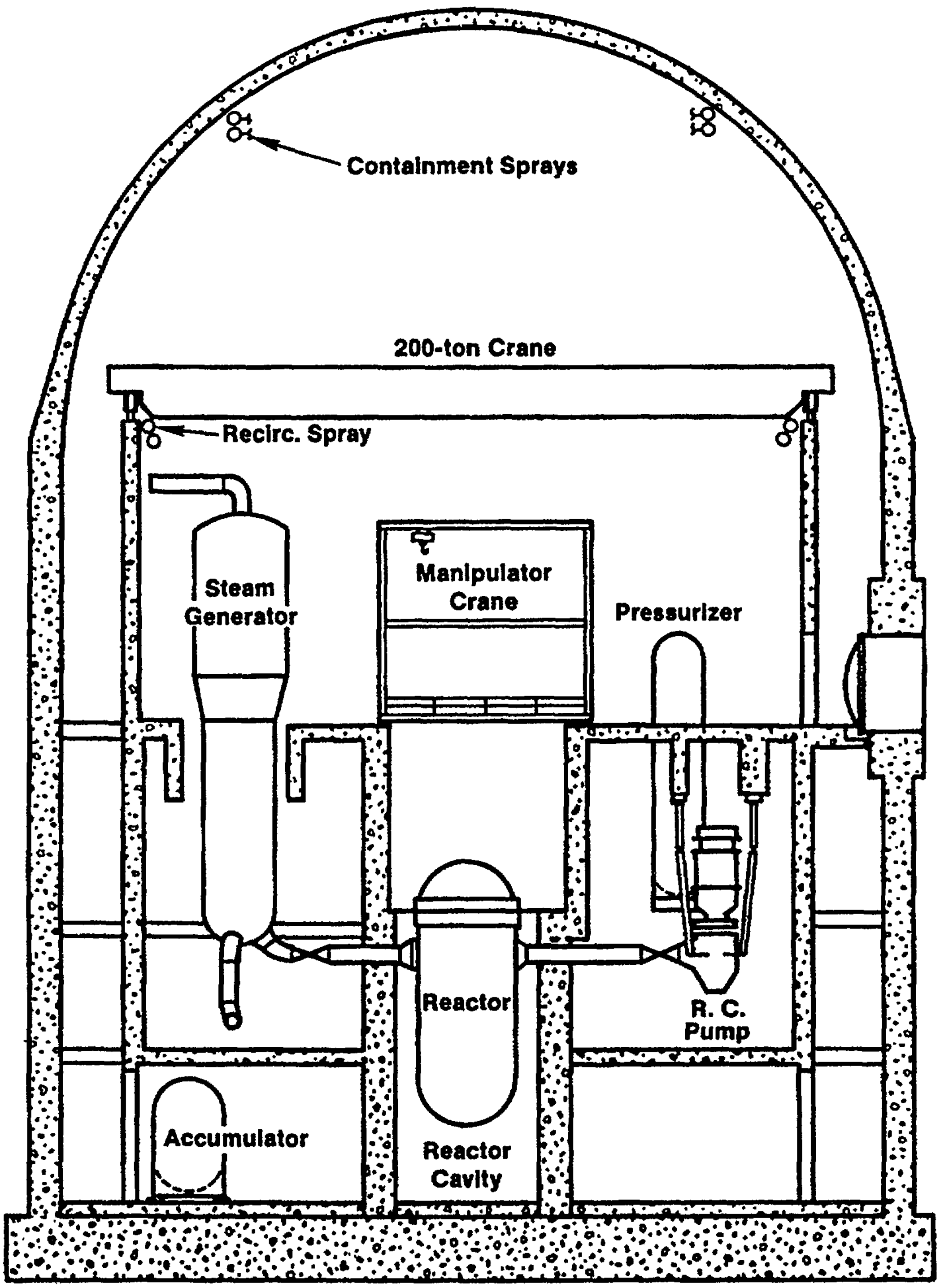


Figure B.9 Simplified schematic of Surry containment.

Adolph Walser, Sargent and Lundy, and
J. Randall Weatherby, Sandia National Laboratories.

They did not differentiate on the basis of failure location since any failure location except shear at the basemat-cylinder junction would result in a direct path to the outside. The reinforcing and concrete details in this junction area were such that three of the four experts ruled out failure in this location. (The fourth expert did not specify failure location explicitly.)

The experts treating Surry did not perform any extensive new calculations. They reviewed the previous detailed calculations and the drawings of the containment, including reinforcing details, penetrations, and hatches and airlocks. Their experience allowed them to judge how comprehensive the previous analyses had been and, when there were conflicting results, which result was more likely to be correct.

Expert A based his conclusions on previous analyses of the Indian Point containment (Ref. B.25), the Surry containment (Ref. B.26), and the drawings of the Surry containment structure. He considered four failure modes: hoop failure in the cylinder, hoop failure in the dome, shear failure at the cylinder-basemat junction, and penetration failure. Meridional failure in the dome will be similar to the hoop failure and was not considered explicitly.

On the basis of the detailed drawings and some brief calculations, Expert A concluded that the cylinder-basemat junction was a very strong region and ruled out failure at this location. He looked briefly at the equipment hatch, personnel airlock, pipe penetrations, and electrical penetrations and concluded that they were sufficiently similar to those at Zion that failures at these locations were of relatively low probability. At low and medium stress levels, with the liner taken into account, Expert A concluded that the dome is stronger than the cylinder. However, the way the rebar was placed at the top of the dome led Expert A to question whether the dome would be stronger than the cylinder at high stress levels.

For the cylinder, the hoop stress can adequately be calculated by hand. In this manner, Expert A concluded that general yield of the rebar would occur at 119 psig, which agrees with the Stone & Webster analysis (Ref. B.26). This is the lowest pressure for which Expert A would expect to find any chance of failure; at this pressure the cylinder wall has moved out 2 inches. Expert A then calculated that 2 percent hoop strain corresponded to 150 psig, including the effects of strain hardening of the rebar. At this level of strain, he concluded that liner tear is certain at discontinuities such as those around penetrations and stiffener plates. Further, concrete cracking at 2 percent general strain will have removed much of the liner support. At 2 percent strain, the cylinder wall has moved out 16 inches.

In summary, Expert A concluded that the containment would fail between 120 and 150 psig and that the probability density of failure was uniform in that range. His median value was 135 psig.

Expert B based his analysis on the Stone & Webster study of the Surry containment (Ref. B.26), studies of other plants such as Indian Point 2 and 3 (Ref. B.25), Seabrook (Ref. B.27), and the test of the 1/6-scale model at Sandia (Ref. B.28). Expert B's hoop membrane stress analysis showed that there would be general yielding of the shell and rebar at 120 psig, and that rebar that just met the minimum requirements would fail at 144 psig. If all the rebar were of average strength, the rebar would fail at 166 psig.

Based on the reference analyses and this information, Expert B placed his median failure pressure at 120 psig and his upper bound at 165 psig. He placed his lower bound at 70 psig. This took into account the possibility of faulty rebar joints or liner tears due to stress concentrations around openings.

Expert C based his conclusions on an analysis of the mid-section of the cylindrical portion of the containment. His study of the drawings and the results of other analyses led him to conclude that this was the weakest portion of the containment. His conclusions about the leak failure mode and liner tear are largely based on the 1/6-scale model test at Sandia (Ref. B.28). Once a liner tear has developed, it is difficult to see how it could be kept from expanding with a continued increase in pressure.

Expert C concluded that failure was most likely in the 135 to 147 psig range, and he placed 70 percent of his probability there. He placed 10 percent of his probability below 135 psig to allow for his uncertainty about the actual rebar properties.

Expert D's analysis led him to conclude that a leak was certain to develop by 130 psig. At this pressure the rebar has yielded considerably and reached a strain of about 1 percent. He would expect leaks to develop because of dislocation at discontinuities (Ref. B.29). There is no possibility of a leak developing at pressures below 75 psig. This value was obtained by hoop membrane stress analysis assuming that the liner is at its yield stress of 35,000 psi. If the liner and the hoop reinforcement are both at their respective yield stress, which is 55,000 psi for the reinforcement and 35,000 psi for the liner, the pressure would be 110 psig. Expert D took 110 psig to be his median value for leaks. He noted that the specified minimum yield strength is 55,000 psi for the reinforcement and 35,000 psi for the liner.

Expert D took the lower threshold for rupture to be 140 psig, which was determined by a local effects analysis of the discontinuity at the basemat-cylinder junction (Ref. B.30). He expected that a crack would open at this junction for a substantial portion of the circumference. Although the crack might be very small, it would be long enough to depressurize the containment in less than 2 hours. He concluded that rupture was certain when the main reinforcement reaches its specified minimum ultimate strength. For the Surry containment, Expert D considered catastrophic rupture to be impossible.

Figure B.10 shows the distributions of the four experts and the aggregate distribution for total cumulative failure probability. Experts A and C concluded that there is little or no chance of failure by 120 psig, while Expert D concluded that failure is almost certain by 120 psig. The aggregate distribution for the failure pressure of the Surry containment was formed by weighting equally the individual distributions of the four structural experts who considered this issue.

From the information provided by the experts, aggregate distributions were also obtained for the mode of containment failure. Because the containment did not fail in this example, the question of the mode of failure is not discussed here. The results of the experts' elicitations on the mode of failure may be found in Reference B.6, and the method used to determine the mode of failure in the APET is discussed in References B.1 and B.5.

For use in Question 42, a value for the containment failure pressure is obtained from the aggregate distribution by a random sampling process. The value for the failure pressure in Observation 4 is 148.4 psig. This is the 93rd percentile value. The mean and the median failure pressures are around 127 psig.

B.3.4 Binning Results of APET

There are so many paths through the APET that they cannot all be considered individually in the source term analysis. The results of evaluating the APET are therefore condensed into accident progression bins (APBs) or just bins. The computer code, EVNTRE, that evaluates the APET places the paths through the tree in the bins as it evaluates them. At Surry, each bin is defined by 11 characteristics of the path taken through the event tree. (For the summary discussions contained in Volume 1 of NUREG-1150, these detailed bin definitions were collapsed into a smaller set.) The bin definition provides sufficient information for the algorithm used in the source term calculation. The binning method provides the link between the accident progression analysis and the source term analysis, which calculates the fission product release.

The computer input file that contains the binning instructions is referred to as the "binner." It is listed and discussed in detail in Reference B.1. A discussion of the binning process may be found in the methodology discussion in Reference B.5.

In computer files, the bin is represented as an unbroken string of 11 letters. For presentation here, hyphens have been inserted every three characters to make the bin more readable. A given letter in a given position has a definite meaning. For example, the first characteristic primarily concerns the time of containment failure. If the first character in the bin designator is a "C", containment failure before VB is indicated.

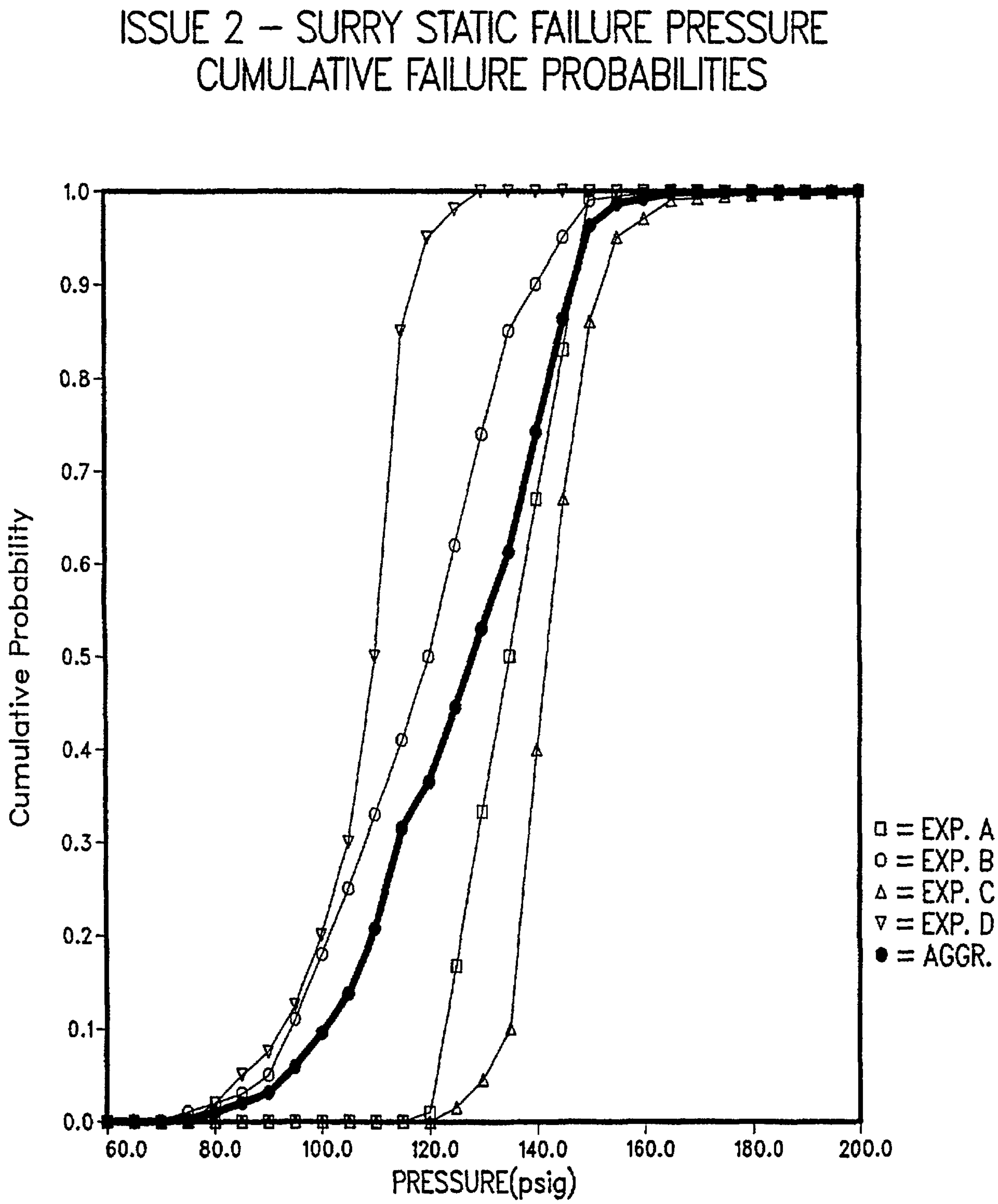


Figure B.10 Results of expert elicitation for static failure pressure of Surry containment. (The first four curves are the distributions of the four experts, and the fifth curve is the aggregate distribution.)

For PDS group 3, Observation 4 produced 22 bins. These resulted from all the paths that remained above the cutoff probability (1.0E-7). For example, the alpha-mode probability was so low in Observation 4 that all the alpha-mode paths were truncated and there are no bins with alpha-mode failures of the containment. The most probable bin (0.55) in Observation 4 is HDC-CFC-DBD-FA, which has no VB and no containment failure. It results from offsite ac power recovery before the core degradation process had gone too far.

Bin GFA-CAC-ABA-DA results from the path followed through the tree in this example for Observation 4. It is the most likely (0.017) bin for Observation 4, which has both VB and containment failure. Basemat meltthrough occurred a day or more after the start of the accident. Containment failure in this time period is indicated by the character "G" in the first position. The other ten characteristics are defined in a similar manner. For bin GFA-CAC-ABA-DA, each character in the bin designation has the following meaning:

G - Containment failure in the final period;

F - Sprays only in the Late and Very Late periods;

A - Prompt CCI, dry cavity;

C - Intermediate pressure in the RCS at VB;

A - High-pressure melt ejection (HPME) occurred at VB;

C - No steam generator tube rupture;

A - A large fraction of the core was available for CCI;

B - A high fraction of the Zr was oxidized in-vessel;

A - High amount of core in HPME;

D - Basemat meltthrough; and

A - One effective hole in the RCS after VB.

The binning follows directly from the path through the APET with one exception. At Question 53, the amount of core in CCI was determined to be medium (Branch 2). The binning above shows that the fraction of the core involved in the CCI is large. The reason for this is that the computer code that performs the source term analysis, SURSOR, subtracts the amount of the core involved in HPME from the total passed to it. To avoid subtracting this amount twice, whenever HPME occurs, the amount of the core involved in CCI is set to Large in the binner.

It is common to keep more information in the binner than that actually used in the source term code. The reason is so that the results of the accident progression analysis can be examined in more detail. By reducing the amount of information passed on to the source term analysis in a "rebinning" step, the amount of source term calculation time can be reduced. Thus, the APBs from an evaluation of the APET by EVNTRE are processed or rebinned by a small computer program, PSTEVNT (Ref. B.31) before the source term analysis.

SURSOR does not distinguish between the various after-VB containment failure times. So PSTEVNT combines the "Very Late" and "Final" containment failure times. The result is that the indicator for failure in the Final period is changed from a "G" in the 1st character to an "F". Bin characteristics 2 through 9 and characteristic 11 are unchanged by the processing with PSTEVNT. The other change is in the 10th character. SURSOR treats BMT in the same manner as it treats a leak in the final period, so Leak, "C", and BMT,"D", are combined and appear as "C". SURSOR also determines whether a bypass of the containment has occurred directly from character 1 ("A" or "B" for Event V) and from character 6 ("C" for no SGTR), so Bypass ("E") and NoCF ("F") are combined as "D" in the rebinner.

(At one time BMT was considered separately from final leaks in SURSOR; the releases of inert gases and organic iodine from BMT were lower than those from a late leak. It turned out to be very difficult to determine, with any certainty at all, just how much lower than the final leak releases the BMT releases should be. As the BMT releases were not expected to be substantial contributors to risk, in the interest of simplicity, the BMT releases were conservatively assumed to be equivalent to the final leak releases.)

Thus the "rebinned" bin equivalent to GFA-CAC-ABA-DA is FFA-CAC-ABA-CA. The meaning of each rebinned character is:

F - Containment failure in the Very Late or Final period;

F - Sprays only in the Late and Very Late periods;

A - Prompt CCI, dry cavity;

C - Intermediate pressure in the RCS at VB;

A - HPME occurred at VB;

C - No steam generator tube rupture;

A - A large fraction of the core was available for CCI;

B - A high fraction of the Zr was oxidized in-vessel;

A - High amount of core in HPME;

C - Leak or basemat meltthrough; and

A - One effective hole in the RCS after VB.

As mentioned in the introduction to this section, for Observation 4 the conditional probability of bin FFA-CAC-ABA-CA is 0.017 (given that PDS group 3 has occurred) and the absolute frequency is 8.1E–9/reactor year. For Observation 4, PDS group 3 is not the only group to produce this bin when the APET is evaluated. Group 1, slow SBO, also produces this bin. For Observation 4, the frequency of PDS group 1 is 9.3E–6/reactor year, and the conditional probability of APB FFA-CAC-ABA-CA is 2.6E–3, so the absolute frequency is 2.4E–8/reactor year. In the source term calculation, there is no point in calculating a source term twice for FFA-CAC-ABA-CA for Observation 4. Therefore, the bins resulting from the seven PDS groups for internal initiators are combined to produce a master bin list for each observation. In producing the master bin list, FFA-CAC-ABA-CA from group 3 is combined with FFA-CAC-ABA-CA from group 1; the total frequency for FFA-CAC-ABA-CA is 3.2E–8/reactor year for Observation 4.

B.4 Source Term Analysis

The source term is the information passed to the next analysis so that the offsite consequences can be calculated for each group of accident progression bins. The source term for a given bin consists of the release fractions for the nine radionuclide groups for the early release and for the late release, and additional information about the timing of the releases, the energy associated with the releases, and the height of the releases.

The source term analysis is performed by a relatively small computer code: SURSOR. The aim of this code is not to calculate the behavior of the fission products from their chemical and physical properties and the flow and temperature conditions in the reactor and the containment. Instead, the purpose is to represent the results of the more detailed codes that do consider these quantities. The release fractions are calculated in SURSOR using a limited number of factors. Many of these factors were considered by a panel of experts. Collectively, they provided distributions for these factors, and the value used in any particular observation is determined by a sampling process. The sampling process used is Latin hypercube sampling (LHS) (Ref. B.32); it is a stratified Monte Carlo method and is more efficient than straightforward Monte Carlo sampling.

The 60 radionuclides (also referred to as isotopes or fission products) considered in the consequence calculation are not dealt with individually in the source term calculation. Some different elements behave similarly enough both chemically and physically in the release path that they can be considered together. The 60 isotopes are placed in nine radionuclide classes as shown in Table B.4. It is these nine classes that are treated individually in the source term analysis. A more complete discussion of the source term analysis, and of SURSOR in particular, may be found in Reference B.33. The methods on which SURSOR is based are presented in Reference B.5, and the source term issues considered by the expert panels are described more fully in Part IV of Reference B.6.

The example being followed has led to accident progression bin FFA-CAC-ABA-CA for Observation 4. The total absolute frequency for this APB is 3.2E-8/reactor year, which comes from PDS group 1 and PDS group 3. The path followed to this point came through PDS group 3, Fast SBO.

Table B.4 Isotopes in each radionuclide release class.

Release Class	Isotopes Included
1. Inert Gases	Kr-85, Kr-85M, Kr-87, Kr-88, Xe-133, Xe-135
2. Iodine	I-131, I-132, I-133, I-134, I-135
3. Cesium	Rb-86, Cs-134, Cs-136, Cs-137
4. Tellurium	Sb-127, Sb-129, Te-127, Te-127M, Te-129, Te-129M, Te-131, Te-132
5. Strontium	Sr-89, Sr-90, Sr-91, Sr-92
6. Ruthenium	Co-58, Co-60, Mo-99, Tc-99M, Ru-103, Ru-105, Ru-106, Rh-105
7. Lanthanum	Y-90, Y-91, Y-92, Y-93, Zr-95, Zr-97, Nb-95, La-140, La-141, La-142, Pr-143, Nd-147, Am-241, Cm-242, Cm-244
8. Cerium	Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241
9. Barium	Ba-139, Ba-140

B.4.1 Equation for Release Fraction for Iodine

In this example of a complete calculation, only the computation of the release fraction for iodine will be presented in detail. The releases of the other fission products are calculated in an analogous fashion. The total release is calculated in two parts as if the containment failed before, at, or a few tens of minutes after vessel breach. The early release occurs before, at, or within a few tens of minutes of vessel breach. The late release occurs more than a few tens of minutes, typically several hours, after vessel breach. In general, the early release is due to fission products that escape from the fuel while the core is still in the RCS, i.e., before vessel breach (VB), and is often referred to as the RCS release. The late release is largely due to fission products that escape from the fuel during the CCI, i.e., after VB, and is referred to as the CCI release. For situations where the containment fails many hours after VB, the "early" release equation is still used, but the release is better termed the RCS release, and after both releases are calculated in SURSOR, both releases are combined into the late release and the early release is set to zero. The "late" release includes not only fission products released from the core during CCI, but also material released from the fuel before VB that deposits in the RCS or the containment and then is revolatilized after VB.

The early or RCS iodine release is calculated from the following equation:

$$ST = [FCOR * FVES * FCONV / DFE] + DST.$$

And the late or CCI iodine release is calculated from:

$$STL = [(1 - FCOR) * FPART * FCCI * FCONC / DFL] + FLATE + LATEI.$$

In these equations, some terms that pertain only to steam generator tube ruptures (SGTRs) have been omitted since bin FFA-CAC-ABA-CA has no SGTR. The meaning of the terms is as follows:

ST = fraction of the core iodine in the RCS release to the environment;

FCOR = fraction of the iodine in the core released to the vessel before VB;

FVES = fraction of the iodine released to the vessel that is subsequently released to the containment;

FCONV = fraction of the iodine in the containment from the RCS release that is released from the containment in the absence of any mitigating effects;

DFE = decontamination factor for RCS releases (sprays, etc.);

DST = fraction of core iodine released to the environment due to direct containment heating at vessel breach;

STL = fraction of the core iodine in the late release to the environment;

FPART = fraction of the core that participates in the CCI;

FCCI = fraction of the iodine from CCI released to the containment;

FCONC = fraction of the iodine in the containment from the CCI release that is released from the containment in the absence of any mitigating effects;

DFL = decontamination factor for late releases (sprays, etc.);

FLATE = fraction of core iodine remaining in the RCS that is revolatilized and released late in the accident; and

LATEI = fraction of core iodine remaining in the containment that is converted to volatile forms and released late in the accident.

Like ST and STL, DST, FLATE, and LATEI are expressed as fractions of the initial core inventory. DST, FLATE, and LATEI are not independent of the other factors in the equations given above. Complete expressions for these three terms and an expanded discussion of them may be found in Reference B.18.

Some of these factors are determined directly by sampling from distributions provided by the expert panels. Others are derived from such values, and still others were determined internally. In Section B.4.2, each factor in the equation above will be discussed briefly, and the source of the value used for each factor will be given. In Section B.4.3, three of the factors are discussed in more detail.

For Observation 4, the following values were used in the equation for the RCS iodine release for bin FFA-CAC-ABA-CA:

FCOR = 0.98
FCONV = 1.0E−6
DST = 0.0
FVES = 0.86
DFE = 34.0
resulting in ST = 2.5E−8.

ST is a very small fraction of the original core inventory of iodine because the containment failure takes place many hours after VB and there is a long time for natural and engineered removal process to operate.

For Observation 4, the following values were used in the equation for the late or CCI iodine release for bin FFA-CAC-ABA-CA:

1 - FCOR = 0.02
FCCI = 1.0

DFL = 82.2
LATEI = 0.0044
FPART = 0.57
FCONC = 1.2E-4
FLATE = 7.2E-9
resulting in STL = 0.0044

Containment failure occurs a long time after most of the radionuclides have been released from the fuel during CCI, so there is a long period in which the aerosol removal processes operate. Thus, the CCI release of iodine in nonvolatile form is very small, and the total late iodine release is almost all due to the late formation of volatile iodine in the containment.

B.4.2 Discussion of Source Term Factors

As most of the parameters in the source term equations were determined by sampling from distributions provided by a panel of experts, Part VI of Reference B.6 is not cited for each of the parameters. The parameters not determined by expert panels are discussed in References B.1, B.6, and B.33.

The values for many of the parameters defined above are obtained from distributions when SURSOR is evaluated, most of which were provided by experts. They determined distributions for the nine radionuclide release classes defined in Table B.4. Only the distributions for iodine (class 2) are discussed here, but distributions exist for the other eight classes as well (Ref. B.6). These distributions are not necessarily discrete. While the experts provided separate distributions for all nine classes for FCOR, for other factors, for example, they stated that classes 5 through 9 should be considered together as an aerosol class. Note that the distributions for the nine radionuclide classes are assumed to be completely correlated. That is, a single LHS number is obtained for each factor in the source term equation, and it applies to the distributions for all nine radionuclide classes. For example, in Observation 4 the LHS number for FCOR is 0.828. That means the 82.8th percentile value is chosen from the iodine distribution, the cesium distribution, the tellurium distribution, etc., for FCOR.

FCOR is the fraction of the fission products released from the core to the vessel before vessel failure. The value used in each observation is obtained directly from the experts' aggregate distribution. There are separate distributions for each fission product group (inert gases, iodine, cesium, etc.) for high and low Zr oxidation in-vessel. Each distribution takes the form of a curve that relates the values of FCOR to a cumulative probability. A value of FCOR is obtained in the following manner: the LHS program (Ref. B.32) selects a number between zero and 1.0 that is the cumulative probability. Using this value, the value of FCOR is obtained from the experts' aggregate cumulative probability distribution. The LHS number in Observation 4 for FCOR is 0.828, and the corresponding FCOR value for iodine is 0.98. For Observation 4, then, almost all the iodine is released from the core to the vessel before breach. FCOR is discussed in more detail in Section B.4.3.

FVES is the fraction of the fission products released to the vessel that is subsequently released to the containment before or at vessel failure. As for FCOR, the value used in each observation is obtained directly from the experts' aggregate distribution, and there are separate distributions for each fission product group. The LHS number in Observation 4 for FVES is 0.931. The corresponding value in the experts' aggregate distribution for FVES for iodine is 0.86. So, in this example, most of the iodine in the vessel before breach is released from the vessel to the containment.

FCONV is the fraction of the fission products in the containment from the RCS release that is released from the containment in the absence of mitigating factors such as sprays. The expert panel provided distributions for FCONV for four cases, each of which applies to all species except the noble gases. These cases apply to containment failure at or before VB, or within a few hours of VB. (There is a fifth distribution that applies to Event V.) None of these distributions is used in the path followed for this example since containment failure happens a day or more after the start of the accident. Because of the long time period for the engineered and natural removal processes to reduce the concentration of the fission products in the containment atmosphere, the fraction of the fission products released before or at VB remaining airborne at the time of containment failure is very small. This fraction was estimated internally to be 1.0E−6, and FCONV is set to that value for final period releases. (The particular value of 1.0E−6 is not important; any very small value would be satisfactory.) This value is used whether the release is due to aboveground failure or basemat meltthrough. FCONV is discussed in more detail in Section B.4.3.

DFE is the decontamination factor for early releases. For APB FFA-CAC-ABA-CA, the containment sprays are the only mechanisms that contribute to DFE. The expert panel concluded that the distributions used for the spray decontamination factors (DFs) were less important to determining offsite risk and the uncertainty in risk than whether the sprays were operating and other factors, so the spray DF distributions were determined internally. There are two spray distributions that apply to the fission products released from the RCS before or at VB: the first applies when the containment fails before or at VB and the RCS is at high pressure at VB; and the second applies when the containment fails after VB or when the containment fails at VB but the RCS is at low pressure. Each distribution applies to all species except the noble gases. The LHS number for the spray distribution for Observation 4 was 0.928. Using the distribution for late containment failure, a spray DF value of 3.4 is obtained. For failures of the containment in the final period, the value from the distribution is multiplied by 10 to account for the very long period that the sprays have to wash particulate material out of the containment atmosphere. Thus, DFE is increased from 3.4 to 34.

DST is the fission product release (in fraction of the original core inventory) from the fine core debris particles that are rapidly spread throughout the containment in a direct containment heating (DCH) event at VB. The experts provided distributions for the fractions of the fission products that are released from the portion of the core involved in DCH for VB at high pressure (1,000 to 2,500 psia) and for VB at intermediate pressure (200 to 1,000 psia). There are separate distributions for each fission product group (inert gases, iodine, cesium, etc.). However, neither the high-pressure nor the low-pressure set of distributions was used in calculating the source term for FFA-CAC-ABA-CA because the containment failure occurs so long after VB. It was internally estimated that the amount of fission products from DCH remaining in the atmosphere many hours after VB would be negligible, so DST is set to zero for this APB.

FPART is the fraction of the core that participates in the CCI. Bin FFA-CAC-ABA-CA has a "large" fraction, nominally 40 percent, of the core participating in HPME. As 5 percent of the core is estimated to remain in the vessel indefinitely, the fraction participating in DCH is 0.95 * 0.40 = 0.38. The fraction of the core available to participate in CCI is thus 0.95 - 0.38 = 0.57.

FCCI is the fraction of the fission products present in the core material at the start of CCI that is released to the containment during CCI. The experts provided distributions for four cases that depended upon the fraction of the Zr oxidized in-vessel and the presence or absence of water over the core debris during CCI. There are separate distributions for each fission product group. For the path being followed in this example, bin FFA-CAC-ABA-CA indicates that a large fraction of the Zr was oxidized in-vessel before VB and that the cavity was dry at the start of CCI. However, for iodine, the case is immaterial since all the iodine remaining in the core debris is released during CCI for any case and for every point on the distribution. Thus, FCCI is 1.0.

FCONC is the fraction of the fission products released to the containment from the CCI that is released from the containment. The expert panel provided distributions for FCONC for five cases. There are separate distributions for each fission product group (inert gases, iodine, cesium, etc.). None of these cases applies directly to the situation for APB FFA-CAC-ABA-CA since this bin has containment failure in the final period (after 24 hours). Since containment failure occurs many hours after most of the fission products have been released from CCI, only a very small fraction of these fission products will still be in the containment atmosphere at the time of containment failure. This fraction was estimated internally to be on the order of 1.0E−4. The exact value is determined by using the FCONC distribution for case 3, rupture before the onset of CCI. The ratio of the LHS value from the distribution to the median value times 1.0E−4 is the value of FCONC used for final period containment failure. In Observation 4, the LHS number for determining FCONC is 0.777. The iodine value of FCONC for this point on the FCONC, case 3, for the CDF is 0.78, and for the median value of the distribution is 0.63. Thus, FCONC is set to 0.78/0.63 * 1.0E−4 = 1.2E−4. This value is used whether the release is due to aboveground failure or basemat meltthrough.

DFL is the decontamination factor for late releases. At Surry, DFL can be due to either the containment sprays or a pool of water over the core debris during CCI. Since the CCI began in a dry cavity, only the spray DF applies for bin FFA-CAC-ABA-CA. The procedure used to obtain the spray DF for the CCI release for final period CF is similar to that used to obtain the value for DFE (discussed above). There is only one distribution for the spray DF for the CCI release, and it applies to all species except the noble gases. The same LHS number (0.928) is used for all the spray distributions, giving a CCI spray DF value of 8.2. As for DFE, because the containment fails in the final period, the value from the distribution is multiplied by 10 to account for the very long time the sprays have to wash particulate material out of the containment atmosphere. Thus, DFL is 82.

FLATE accounts for the release of iodine from the RCS late in the accident. Like DST, it is a fraction of the original core inventory. Iodine that had been deposited in the RCS before VB may revert to a volatile form after the vessel fails and make its way to the environment. This term considers only revolatilization from the RCS; revolatilization from the containment is considered in the next term. The experts provided distributions for the fraction of the radionuclides remaining in the RCS that are revolatilized. The amount remaining in the RCS is a function of FCOR, FVES, and other terms and is calculated in SURSOR (Ref. B.33). The experts concluded that whether there was effective natural circulation through the vessel was important in determining the amount of revolatilization. Thus, there are two cases: one large hole in the RCS and two large holes in the RCS.

The experts provided separate distributions only for iodine, cesium, and tellurium. (Revolatilization is not possible for the inert gases as they would deposit, and it is negligible for radionuclide classes 5 through 9.) For accident progression bin FFA-CAC-ABA-CA, the last character indicates that there is only one effective hole in the RCS: the hole formed in the bottom head when the vessel failed. The other failure of the RCS pressure boundary is the RCP seals, and the path through them is considered too tortuous to allow effective natural circulation to develop. The LHS number for late revolatilization of Observation 4 is 0.412, and the corresponding value for iodine from the experts' distribution for iodine is 0.033. This number is applied to the fraction of the iodine remaining in the RCS, which is small, and then the FCONC value for tellurium is applied to that value to determine how much of the iodine that is revolatilized from the RCS escapes from the containment. (The Te value for FCONC is considered to be generally appropriate for revolatilized material since it, like Te, is slowly released over a long time period.) The resulting value for revolatilized iodine that escapes from the containment is very low, 7.2E–9.

LATEI accounts for iodine in the containment that may assume a volatile form and may be released late in the accident. The volatile forms are typically organic iodides such as methyl iodide, but are not limited to organic forms. The primary source of this iodine is the water in the reactor cavity and the containment sumps (which are separate at Surry). This term is added to the late release only for radionuclide class 2, iodine. The experts provided only one distribution. The LHS number for late revolatilization in Observation 4 is 0.055, and the corresponding value for iodine from the experts' distribution is 0.0051. This number is applied to the fraction of the iodine remaining in the containment. Based on the values of FCOR, FVES, FCCI, and other factors, the fraction of the original core iodine still in the containment and available to assume a volatile form was determined to be 85 percent for Observation 4. Applying the release fraction obtained from the experts' distribution to this gives a late revolatilization iodine release fraction of 0.0044. LATEI is discussed in more detail, and the expression used to calculate it is given in Section B.4.3.

While the total iodine release is small compared to a case where the containment fails at VB or is bypassed from the start (such as Event V), the iodine release is very large compared to the other radionuclide classes except inert gases (see Section B.4.4). This relatively large release fraction for iodine is entirely due to the LATEI term. Even though the release point for basemat meltthrough is underground, no allowance is made for attenuation or decontamination of the late iodine release represented by the LATEI term. For the example considered, the very slow passage of the gases through wet soil with a low driving pressure would undoubtedly result in some reduction in the late iodine release. This reduction could be quite large. Although giving no credit for removal in the wet soil is conservative, it is unimportant for the sample as a whole. Other observations and other modes of containment failure dominate risk. For the mode of containment failure in this example, basemat meltthrough, however, the release of late organic iodine is a major contributor to risk, and the risk from this release may be overestimated by the neglect of iodine removal in the wet soil. Even with this conservative estimate of the late iodine release, the total iodine release and the risk therefrom are very small compared to the releases and risks from accidents and pathways in which the containment fails at or before vessel breach, or where the containment is bypassed.

It could be argued that, since BMT is so much more likely than early CF, overstating the BMT release results in a significant overestimate of the population dose and latent cancer fatality estimates. However, bypass accidents (V or SGTR) are twice as likely at Surry as nonbypass accidents that lead to BMT. As the iodine releases from the bypass accidents are more than an order of magnitude higher than BMT iodine releases, this argument is not valid.

B.4.3 Quantification of Source Term Factors by Experts

In this section, the quantification of three factors in the source term equation that were considered by the expert panel is presented in more detail. The eight issues considered by the source term expert panel are:

1. FCOR and FVES
2. Ice Condenser DF (not applicable to Surry)
3. FLATE
4. FCCI
5. FCONV and FCONC
6. LATEI (not used for PWRs)
7. Reactor Building DF (not applicable to PWRs)
8. DCH Releases (DST)

Three of these issues are not applicable to Surry. Of the eight factors in the iodine equation for Surry, only three are discussed here. More extensive documentation of all the issues may be found in Part IV of Reference B.6. The source term factors chosen for discussion here are FCOR, FCONV, and LATEI. The consideration for FVES is similar to that for FCOR, only there are more cases. The consideration for FCONC is similar to that for FCONV, except that the experts provided a distribution for each fission product group for FCONC and they did not for FCONV. Of the remaining factors, LATEI made the largest contribution to the iodine example considered above.

B.4.3.1 FCOR

FCOR is the fraction of the fission products released from the core to the vessel before vessel failure. Four members of the source term expert panel provided distributions for FCOR:

Peter Bieniarz, Risk Management Associates,
Robert Henry, Fauske and Associates, Inc.,
Thomas Kress, Oak Ridge National Laboratory, and
Dana Powers, Sandia National Laboratories.

Two of these four experts concluded that there were no significant differences between PWRs and BWRs as far as FCOR was concerned, and each provided one distribution that applied to both types of reactors. The other two experts provided separate PWR and BWR distributions and further subdivided this issue by providing different distributions for high Zr oxidation in-vessel and low Zr oxidation in-vessel.

Expert A based his analysis for FCOR upon the experimental work done on the release of fission products from fuel (Refs. B.34 and B.35). He concluded that the results for cesium could be well represented by an equation similar to the diffusion equation and that the constants in the solution could be determined from the data. He obtained release rates for the other fission products by considering their "relative volatilities." The results of applying this method of calculating release rates appeared to him to agree reasonably well with experiments. Expert A then wrote a simple computer program to vary the temperature rise with time over a range of reasonable scenarios and keep track of the amount of each fission product released. Expert A provided FCOR distributions for both high and low Zr oxidation in-vessel for both types of reactors.

Expert B based his conclusions for FCOR on a large number of MAAP (Ref. B.15) calculations for various accident scenarios. He also relied on Reference B.36 and the evidence from TMI-2 (Refs. B.37 and B.38). The MAAP results served as the basis for his conclusions, but he included uncertainty for phenomena not modeled in MAAP and phenomena that MAAP currently does not treat in sufficient detail. For example, Expert B thought that MAAP sometimes overestimated the releases of certain nuclide groups because the process of core collapse imposes physical limitations on other processes that MAAP does not consider adequately at this time. He also concluded that neither the reactor type nor the amount of Zr oxidation in core had a significant effect on FCOR, so he provided a single distribution for FCOR.

Expert C reasoned that even if the dependency of the fission product release rates on temperature were much better known, the release rates, and thus FCOR, could not be much better predicted because the variations of the temperatures in the core by time and location are so crudely known at this time, especially after the onset of relocation. The extent of metal oxidation is also a significant uncertainty. Relocation not only changes the surface to volume ratio, but it alters the hydrogen-steam ratio, which in turn affects the diffusion and transport rates of the fission products. Thus, the current models, which largely depend upon Arrhenius-type equations, have definite limitations. For example, the STCP (Ref. B.39) tends to overpredict FCOR because it poorly treats the formation of eutectics and the gradual relocation of the core. Expert C provided separate FCOR distributions for high and low Zr oxidation in-vessel for both types of reactors.

Expert D did not consider the amount of Zr oxidation in-vessel or the type of reactor to be important for FCOR; he provided one distribution for FCOR. Expert D was of the opinion that all the noble or inert gases (Xe and Kr) would escape from the fuel. For tellurium, he concluded that the data were so ambiguous and conflicting that he could not support any particular distribution. He thus specified that a uniform distribution between zero and one be used. For the other seven radionuclide groups, he provided nonuniform distributions. His conclusions were based on a set of experimental work that he has performed or was performed by others (Refs. B.40 and B.41). He made use of several small computer programs to manipulate the experimental results to obtain release fractions for different pressures and temperatures.

The aggregate distributions for the two PWR cases are shown in Figures B.11 and B.12, as are the distributions of each of the four experts who considered this issue. The estimated fraction released depends strongly on the volatility of the fission products, as might be expected. The differences between I and Cs are not great. The differences between the less volatile fission products Ba, Sr, Ru, La, and Ce are small. Note that the differences between the high-Zr-oxidation case and the low-Zr-oxidation case are small compared to the differences between the experts. Furthermore, the differences between the radionuclide classes are often less than the differences between the experts for a given class. This is indicative of the uncertainty in the source term area.

B.4.3.2 FCONV

This issue concerns the fraction of radionuclides released to the containment atmosphere from the vessel before it fails or at failure that is subsequently released to the environment if the containment fails. FCONV may be defined by the equation:

FCONV = mVout/mVin

where:

mVin = mass (kg) of a radionuclide (or radionuclide class) released from the vessel to the containment atmosphere at or before vessel breach (VB); and

mVout = mass (kg) of a radionuclide (or radionuclide group) released from the vessel to the containment atmosphere at or before VB that is subsequently released from containment.

That is, FCONV is the fraction of mVin that is released to the environment when the containment fails.

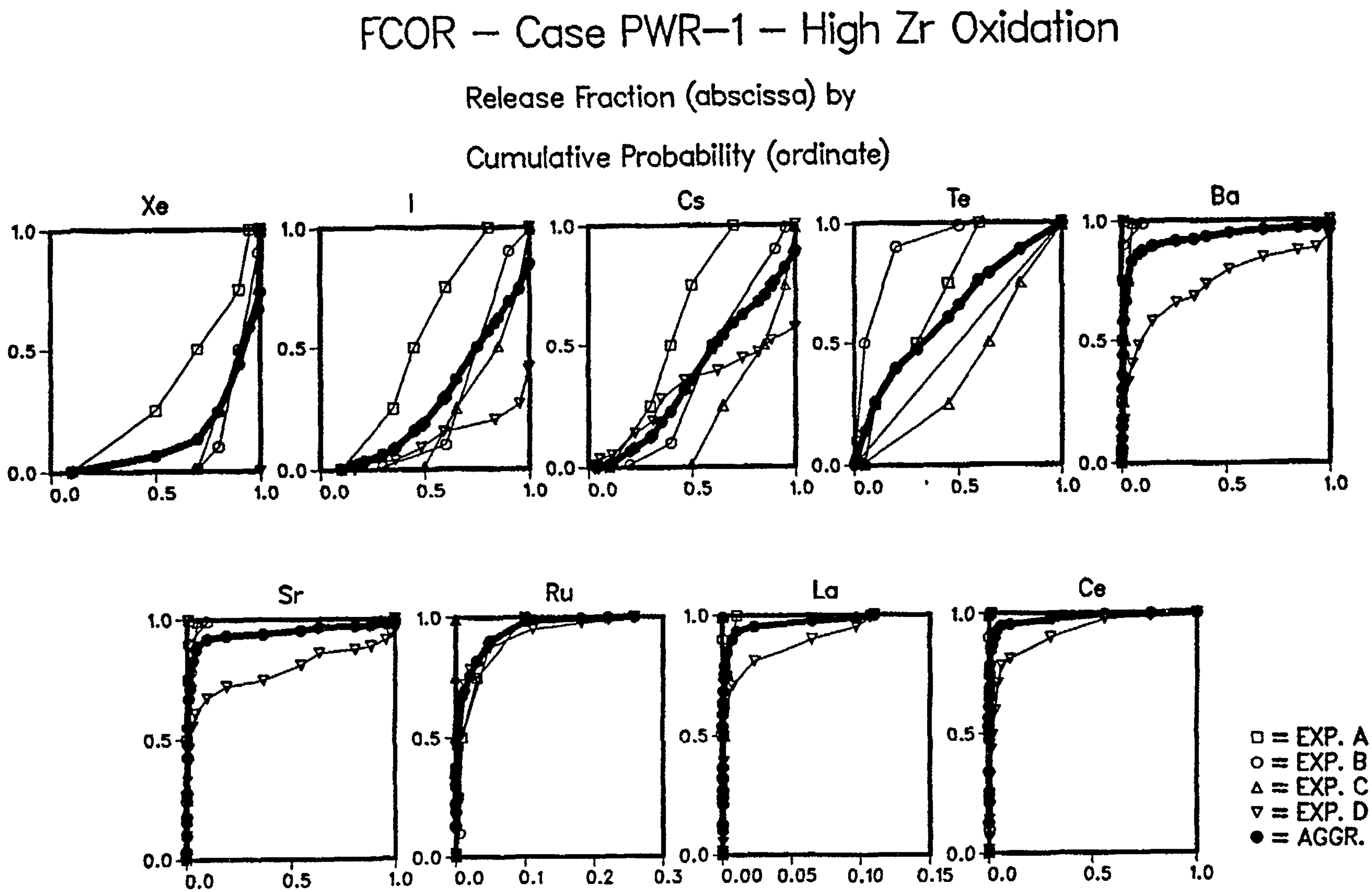


Figure B.11 Results of expert elicitation for FCOR, fraction of the fission products released from core to vessel for the nine radionuclide groups. (This figure shows the results when a high fraction of the core Zr is oxidized in-vessel. In each plot, the first four curves are the distributions of the experts, and the fifth curve is the aggregate distribution.)

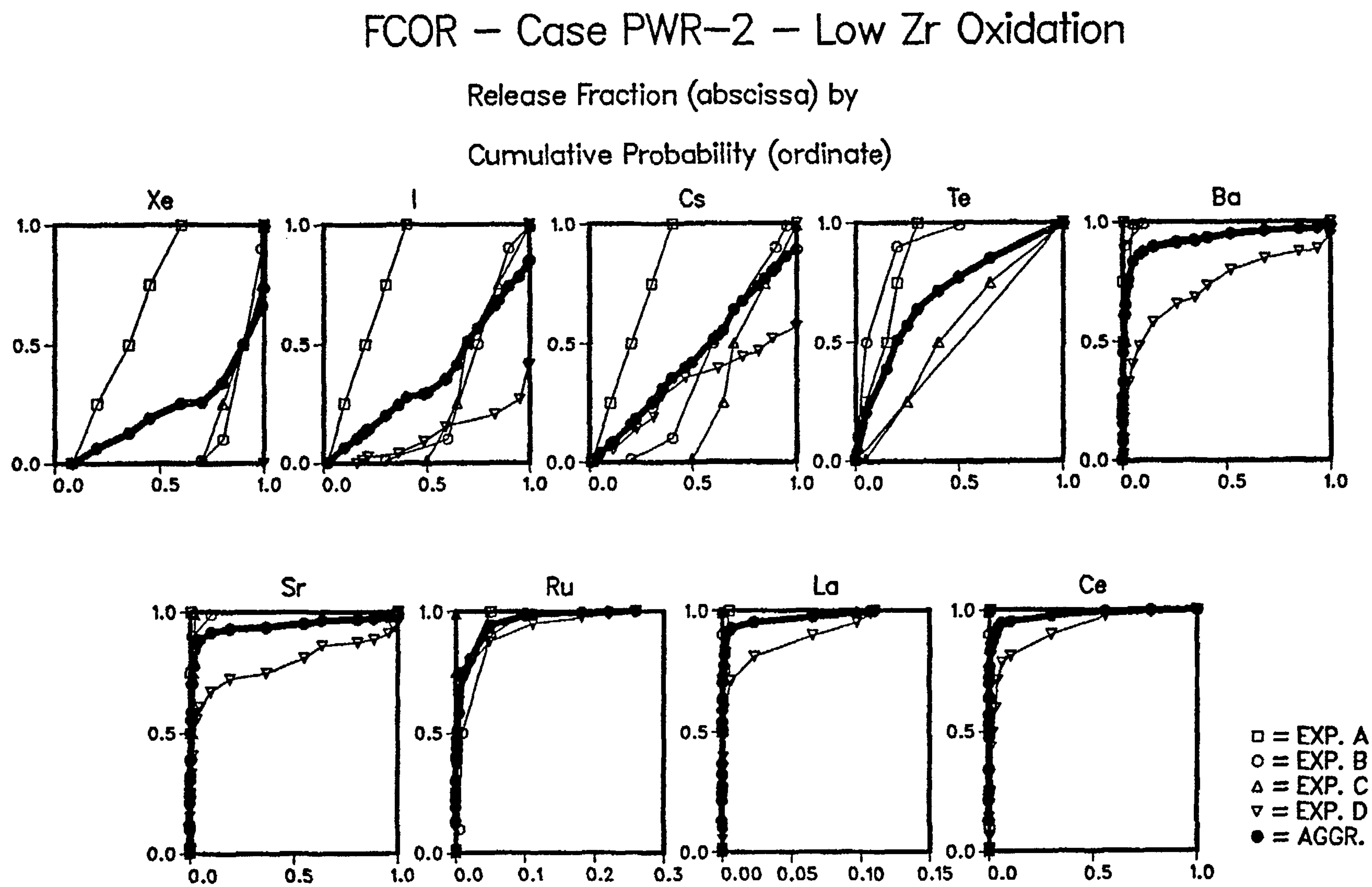


Figure B.12 Results of expert elicitation for FCOR, fraction of fission products released from core to vessel for the nine radionuclide groups. (This figure shows the results when a low fraction of the core Zr is oxidized in-vessel. In each plot, the first four curves are the distributions of the experts, and the fifth curve is the aggregate distribution.)

Three cases were defined for FCONV for the large, dry PWR containments:

1. Early containment failure, leak
2. Early containment failure, rupture
3. Late containment failure, rupture

Early containment failure means at or before vessel breach, and "late" means at least 3.5 hours and nominally 6 hours after VB. A "leak" is a failure of the containment that results in leakage significantly larger than design leakage but is small enough so that the containment does not depressurize in less than 2 hours. The nominal leak is a hole with an area of 0.1 ft^2. A "rupture" is a containment failure sufficient to depressurize the containment in less than 2 hours; the nominal hole size is 7 ft^2. The releases from late leaks were deemed to be low enough that the value of FCONV for late leaks could be derived from these three cases without significantly affecting risk.

FCONV is defined to be the release fraction from containment excluding the effects of engineered safety features such as containment sprays. One expert, however, concluded that one of the principal removal mechanisms, aerosol agglomeration, depended upon the humidity of the atmosphere for case 1. While the humidity may depend on whether the sprays are operating, aerosol removal from the atmosphere by the sprays is considered separately.

Five members of the source term panel considered FCONV:

Andrzej Drozd, Stone and Webster,
James Gieseke, Battelle Columbus Division,
Thomas Kress, Oak Ridge National Laboratory,
Y. H. (Ben) Liu, University of Minnesota, and
David Williams, Sandia National Laboratories.

They all concluded that the inert gases would be completely released and that all the other radionuclides would behave as aerosols. Thus, their distributions for FCONV apply to fission product classes 2 through 9.

Expert A obtained his estimates of the event timing and, thus, residence times from References B.42 and B.43. Expert A concluded that the most important factor in determining FCONV was the residence time of the aerosols in the containment; the longer the time between the formation of the aerosols and the failure of the containment, the smaller the release. Because of the opposing effects of the dynamic shape factor on coagulation and settling, the uncertainty in the dynamic shape factor has little effect on the fraction released. Expert A did not distinguish between the volatile fission products and the refractory groups because he concluded that a significant fraction of the volatiles is released from the fuel prior to VB and deposit on the surfaces of the reactor coolant system. The refractory fission products are released from the fuel at a slower rate and a significant fraction is released after VB and has a direct pathway to the containment. Thus both the volatile and nonvolatile species have similar release rates during the times of interest. He also stated that the aerosol concentration in the containment dropped dramatically in 1 to 2 hours and did not change much after that. The atmospheric humidity has little effect; high humidity makes particles more compact. The compact particles settle out faster but do not agglomerate as fast.

Expert B used NAUA (Ref. B.44) calculations done in conjunction with STCP calculations (Refs. B.42, B.43, and B.45) as a basis for his results, obtaining values directly from NAUA computer output as well as from published reports. For practical considerations, only Xe, I, Cs, and Te were considered for FCONV, and these were deemed to be applicable to all the fission product groups. Other sources consulted by Expert B are the Brookhaven National Laboratory uncertainty study (Ref. B.46), an Electric Power Research Institute (EPRI) calculation for Peach Bottom (Ref. B.47), the recent CONTAIN calculations (Refs. B.20 through B.23), the MELCOR analysis of Peach Bottom (Ref. B.48), and other MELCOR calculations (Refs. B.49 and B.50).

Expert B intended his distributions to include uncertainties from:

1. Surface area (deposition area or compartment height),
2. Natural circulation,
3. Hygroscopic nature of aerosols (primarily I and Cs groups),
4. Particle shape factors (not a big effect),
5. H_2 burn, and
6. Residence time.

Expert C examined the available code calculations relevant to aerosol and fission product behavior in, and release from, the containment. In most cases, these calculations were performed with the STCP (Ref. B.39) or CONTAIN (Ref. B.14) codes. He developed base distributions for FCONV from the code results and then modified them for the effects of factors not considered by the codes. The scale factors applied to the base distributions took into account factors such as aerosol agglomeration, aerosol source strength, timing, shape factors, and containment volume.

Finally, Expert C modified the resulting distributions if they were greatly different from his intuitive expectations.

Expert D considered calculations performed in the GREST exercise (Ref. B.51), by Sandia with MELCOR (Ref. B.48), and by the ANS (Ref. B.52). He concluded that the factors affecting the value of FCONV include:

1. Aerosol characteristics (shape factors, distribution, density);
2. Residence time (size and time of containment failure);
3. Whether the containment was open or divided into many compartments;
4. The effective height of the containment;
5. Thermodynamic state of the atmosphere (superheated or condensing); and
6. Hygroscopic nature of the aerosols.

Expert D noted that the ANS parametric study showed a decrease in aerosol concentration by a factor of 10 in 2 hours and that both the ANS parametric study and KfK DEMONA experiments (Ref. B.53) showed that the existence of many compartments in the containment reduced the release by about a factor of 1.6. Expert D pointed out that the LACE experiments (Ref. B.54) show that, if the hygroscopic effect is present, it can be dominant. A hydrogen burn, by decreasing the residence time and reducing the condensation in the atmosphere, can increase the release fraction FCONV by a factor of 2.

Expert E used an EPRI/FAI aerosol behavior algorithm (Ref. B.55) to perform an independent uncertainty analysis for this issue. He directly varied the aerosol source rates and the aerosol form factors (gamma and chi). To study the impact of the timing and mode of containment failure, he varied the containment failure time and leak rates, assuming choked flowthrough holes from 0.1 ft^2 to 7 ft^2. He also considered pre-existing leakage, steam condensation onto walls, and the impact of pool flashing in his calculations.

Expert E assumed that the aerosols were released directly from the reactor vessel and obtained his aerosol form factors from the QUASAR (Ref. B.46) and QUEST (Ref. B.56) studies. He concluded that the timing and mode of containment failure is the major source of uncertainty. Because it affects agglomeration, the level of turbulence in containment is also an important uncertainty.

The distributions of the five experts who considered this issue are shown in Figure B.13. Case 1 is divided into wet and dry subcases because one of the experts concluded that the release fraction depended on the

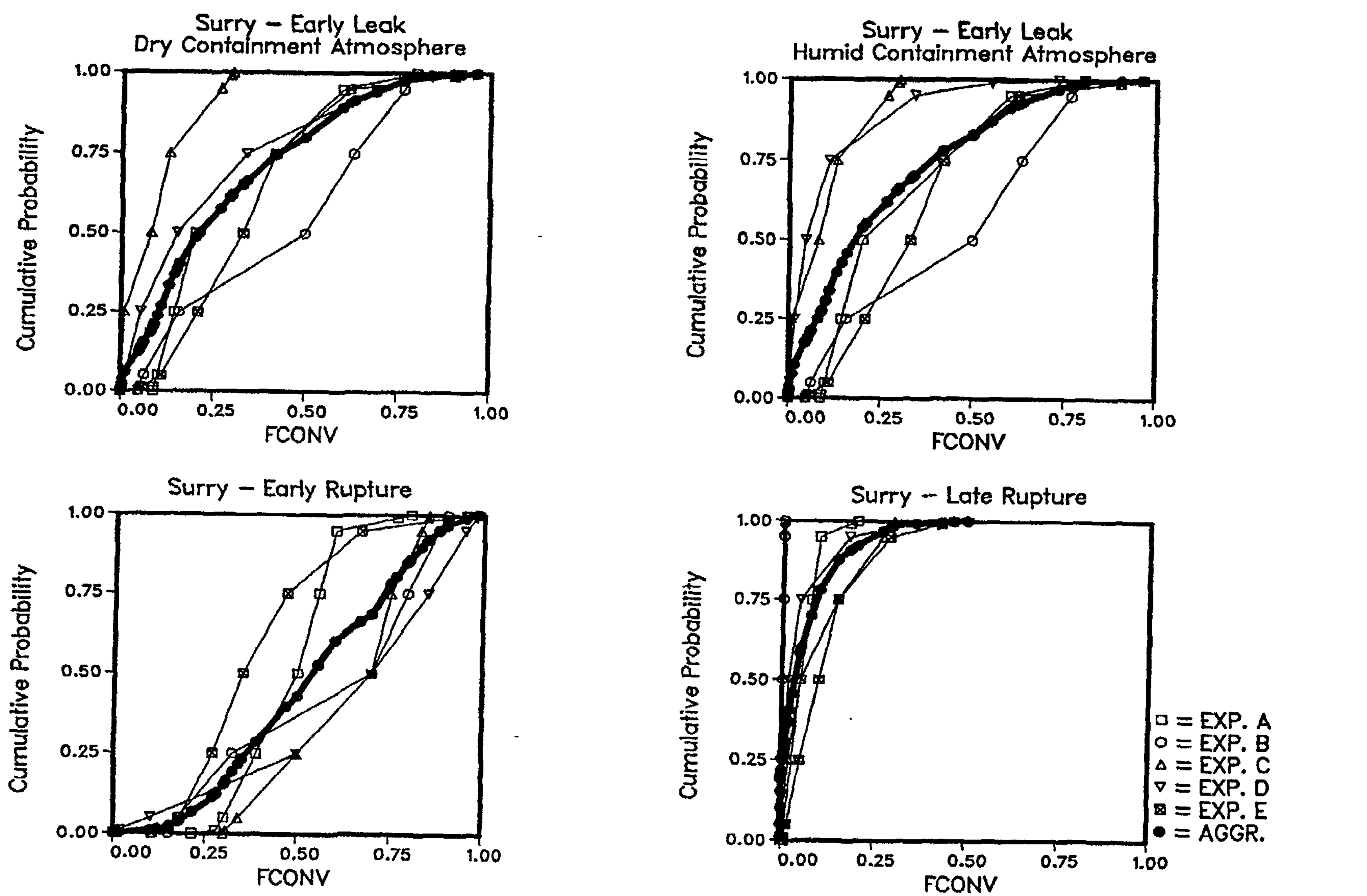


Figure B.13 Results of expert elicitation for FCONV, fraction of fission products in containment from RCS release that is released to environment. (In each plot, the first five curves are the distributions of the experts, and the sixth curve is the aggregate distribution.)

humidity for this case. The differences between the wet and dry atmosphere subcases are small compared with the differences between the experts, so this distinction was dropped. The aggregate distributions are also shown on Figure B.13. The differences between the experts are large compared to the differences between cases 1 and 2. As explained in the discussion of FCONV in Section B.4.2, none of these four distributions was used for bin FFA-CAC-ABA-CA since the containment failure time was so late.

B.4.3.3 LATEI

The question of interest for this issue is how much of the iodine in the containment late in the accident assumes a volatile form (typically organic) and is released to the environment. This volatile iodine is assumed to be unaffected by all removal mechanisms (pool scrubbing, sprays, deposition, etc.). The release fraction determined in this issue applies to almost all the iodine released from the fuel and retained in the containment. The bulk of this iodine is expected to be in aqueous solution, so the issue was specifically framed as release from water pools.

The late release of volatile iodine was deemed to be much more important for BWRs than for PWRs because the BWR design often results in most of the iodine released during core degradation being transported to and retained in the suppression pool. Therefore, the panel of experts was asked only about BWRs directly. They were asked to consider the release of volatile iodine from a BWR suppression pool following containment failure and from water in the pedestal region beneath the reactor pressure vessel (RPV) during CCI.

For the release of volatile iodine from the suppression pool after the containment has failed, two cases were defined: (1) the pool remains subcooled, and (2) the pool is at the saturation temperature. In case 1, considerable surface evaporation is expected but no bulk boiling. In case 2, substantial flashing of the pool would accompany containment failure.

For the release of volatile iodine from water that overlies the core debris in the RPV pedestal, there are also two cases: (1) the drywell is flooded at the time of VB and the entire CCI takes place beneath a pool at least a few feet deep; and (2) the RPV pedestal area contains some water at the time of VB but most of this water is boiled away during CCI.

The results of the expert elicitation on this issue are contained in detail in Part IV of Reference B.6. They are not summarized here because the source term calculation for Surry did not use the results of any of the BWR cases. The PWR situation is somewhat different since the bulk of the iodine is expected to be contained in solution in the sump water. The sump water does not play the same role in heat removal that the suppression pool does in the BWR, and the sump water at Surry is separate from the water in the reactor cavity. Thus, none of the BWR cases is directly applicable although the subcooled suppression pool case is the most applicable. Instead of using this BWR case, the distribution obtained specifically for PWRs in the first draft of NUREG-1150 (Ref. B.57) was used. This is discussed further in Part VI of Reference B.6.

The equation used to calculate the late release of iodine in volatile form is:

$$LATEI = XLATE * [\{FCOR * FVES + (1 - FCOR) * FPART * FCCI\} - ST - STL + FLATE]$$

where XLATE is the fraction of the iodine in the containment late in the accident that assumes a volatile form and is released to the environment. The other terms have been defined above. The term in brackets [ ] is the fraction of the initial core inventory that is in the containment at late times. FCOR * FVES is the RCS release to the containment, and ST is the RCS release from the containment. Similarly, (1 - FCOR) * FPART * FCCI is the CCI release to the containment, and STL is the CCI release from the containment. The FLATE iodine is not considered amenable to this release mechanism because its residence time in the containment is short.

Figure B.14 displays the four aggregate distributions obtained for late volatile iodine release fraction, XLATE, for the BWR cases described above and the distribution for XLATE used for Surry. The range of release fractions used for Surry is the same as for the most applicable BWR case—subcooled suppression pool. The details of the distribution used for Surry are not particularly important as the risk is

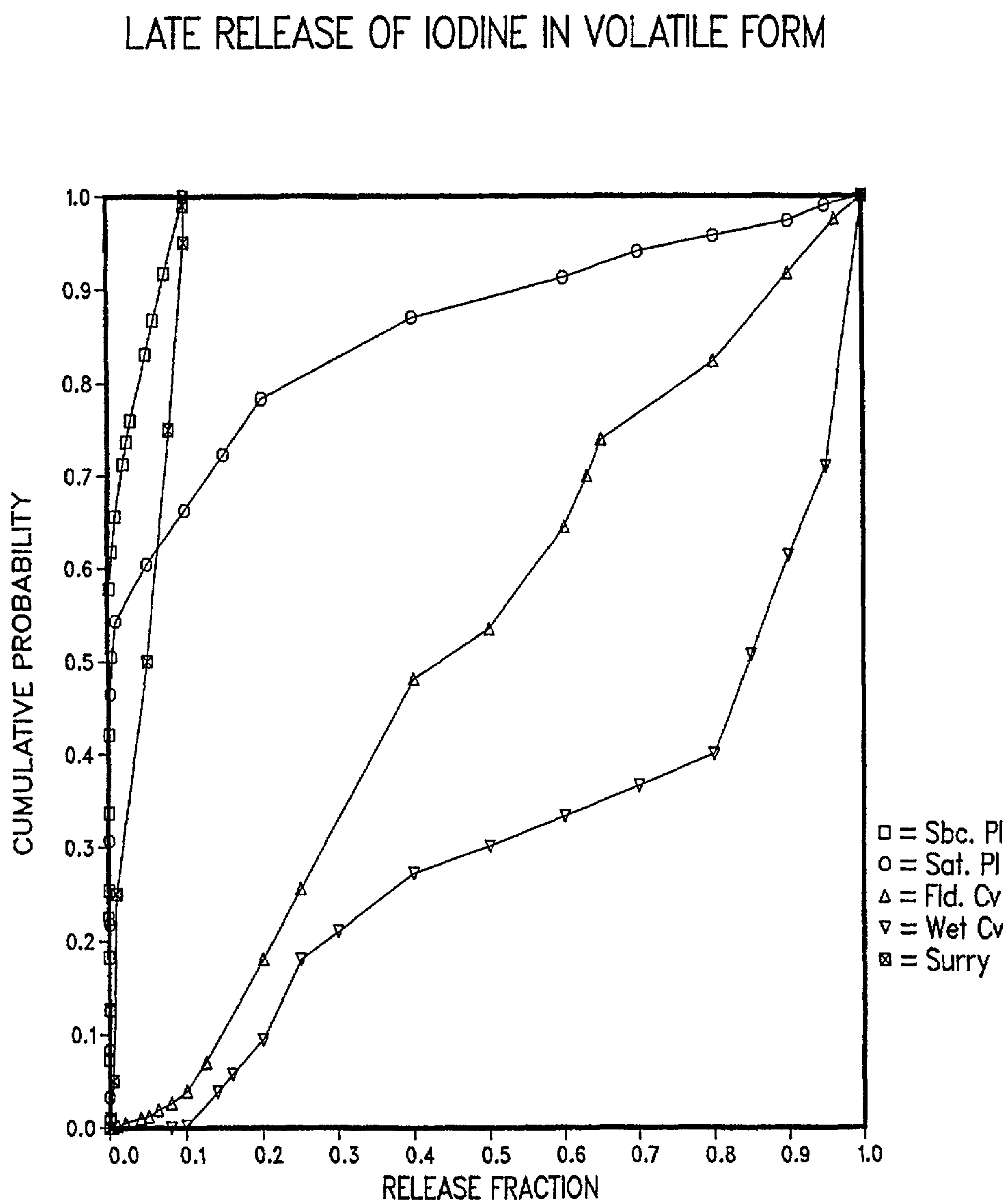


Figure B.14 Distributions for late release of iodine from containment in volatile form. (The first four curves are the aggregate distributions for BWRs: release from a subcooled pool, release from a saturated pool, release from a flooded cavity, and release from a wet cavity. The fifth curve is the distribution used for Surry.)

dominated by accident scenarios in which the path from the reactor vessel to the atmosphere is quite direct (Event V and SGTRs). As this volatile release of iodine occurs late in the accident, its contribution to early fatality risk is negligible. In the accidents that contribute the most to the latent cancer fatality risk, there is little iodine remaining in the containment at late times to be released by this mechanism.

B.4.4 Releases for All Fission Products

'. release that commences a day or more after the onset of core damage or 10 hours or more after VB would be expected to have very small releases, and such is observed to be the case here. The iodine release is dominated by volatile (mostly organic) species that form late in the accident. When the release is so late in the accident, there is only one release, and the distinction between an RCS or early release and a CCI or late release is not kept. The RCS release is put in the late release with the CCI release, and the early release is set to zero. Thus, the complete early and late release fractions for bin FFA-CAC-ABA-CA are:

Fission Products	Early Release	Late Release	Total Release
Xe, Kr	0.0	1.0	1.0
I	0.0	4.4E-3	4.4E-3
Cs, Rb	0.0	8.6E-8	8.6E-8
Te, Sc, Sb	0.0	2.3E-7	2.3E-7
Ba	0.0	2.8E-7	2.8E-7
Sr	0.0	1.2E-9	1.2E-9
Ru, etc.	0.0	3.0E-8	3.0E-8
La, etc.	0.0	3.1E-8	3.1E-8
Ce, Np, Pu	0.0	2.0E-7	2.0E-7

SURSOR also provides the times and energies associated with the early and late releases, the release elevation, and the time that a general emergency is declared. For bin FFA-CAC-ABA-CA, the times for the early release are irrelevant. The other parameters are:

TW = time warning is given = 6.1 h

T2 = start of late release = 36 h

DT2 = duration of late release = 6 h

E2 = energy release rate = 3600 W

ELEV = height of release = 10 m

If BMT releases had been calculated separately, the release height would have been zero. Since BMT and leak releases are treated together, a height more appropriate to a leak above ground is used.

B.5 Partitioning of Source Terms

The accident progression analysis and the source term analysis, each performed once for the 200 observations that constitute the sample, produced 18,591 source terms. This is far too many to be able to perform a consequence analysis for each, so a reduction step is performed before the consequence analysis. This step is called partitioning. Partitioning is performed for all the observations in the sample together.

B.5.1 Introduction

Partitioning is a grouping of the source terms based on the radiological potential of each source term to cause adverse effects on humans. The factors used in partitioning are those most important for the magnitude of the risk: the release fractions for the early release, the release fractions for the late release, and the time between start of the evacuation in the surrounding region and the start of the first release. It is difficult to take each of the nine radionuclide groups into account separately when grouping the source terms, so "effect weights" are determined for each release. These provide a means of considering all the fission products as if they were just one. The partitioning process consists of grouping the source terms together based on these effect weights. Each source term group is then further subdivided based on evacuation timing relative to the start of the release. For releases that have the possibility of causing early fatalities as well as latent cancer fatalities, the grouping is two-dimensional and is based on the source term's potential to cause both types of fatalities. In this analysis for Surry, there were 6,820 source terms with nonzero early fatality weights (EFWs) and nonzero chronic fatality weights (CFWs). For releases that have the possibility of causing only latent cancer fatalities, the grouping is one-dimensional; 11,771 source terms were of this nature.

The partitioning process is carried out by a computer code—PARTITION (Ref. B.58). The process is an interactive one, with the user choosing the number of cells or divisions to be used and selecting those grids that have so few points that they may be combined with a neighboring cell.

B.5.2 Effects Weights

The early fatality weight (EFW) of a source term is a measure of the radiological potential of a source term to cause early fatalities in the absence of any mitigating effects except relocation. The chronic fatality weight (CFW) of a source term is, similarly, a measure of the radiological potential of a source term to cause latent cancer fatalities.

The calculation of the EFW has two parts. First, the releases of the 59 radionuclides other than I-131 are converted into equivalent amounts of I-131. Then the total release, expressed in an equivalent amount of I-131, is used to estimate the number of resultant early fatalities.

The isotope conversion factor used to determine the equivalent amount of I-131 for each isotope is based on the bone marrow dose from the three pathways that give an acute dose: cloudshine, groundshine, and inhalation.

The cloudshine or immersion dose results from being surrounded by air containing radioactive molecules. The inhalation dose comes from breathing the contaminated air. Radioactive molecules are absorbed into the body from the air in the lungs. The groundshine dose comes from standing or walking on ground on which radioactive particles have been deposited. The cloudshine and groundshine doses are immediate, as the body receives beta or gamma radiation from radionuclides that decay outside the body. The inhalation dose is received some time later when the radionuclides absorbed from the air decay inside the body.

The conversion factor is computed from an equation that is presented and derived in Reference B.5. It depends upon the dose factor for each pathway, the shielding factor for each pathway, the breathing rate, and the deposition velocity. For each pathway and organ of the body, the dose factor is a constant that depends upon the type of radiation emitted by the isotope, the energy of that radiation, and, for the inhalation dose factor, the selective transport of the isotope to, and absorption in, the specific organ. For cloudshine, the dose factor relates the dose rate to the concentration in the air. For groundshine, the dose factor relates the dose rate to the concentration on the ground. For inhalation, the dose factor relates the dose rate to the amount of the isotope inhaled. The shielding factor accounts for the fact that some of the time the people will be indoors and will be partially shielded by the building. The deposition velocity measures the rate at which solid particles are deposited from the plume.

Table B.5 lists the dose factors used in calculating the isotope conversion factors and the isotope conversion factor itself for 11 representative isotopes. Complete information about the calculation of effects weights, with the values of the parameters used for all 60 isotopes, may be found in Reference B.5. The groundshine dose factor is the factor for an exposure of 8 hours; radioactive decay during this time is accounted for in computing this factor. The applicable concentration is the concentration at the beginning of the period. The inhalation dose factor used is the acute inhalation factor. The cloudshine and groundshine dose factors for Sr-90 are zero because it produces no gamma radiation. The groundshine dose factor for Kr-88 is zero because it is not deposited.

Table B.5 Partitioning parameters and results.

	Dose Factors						
Isotope	Cloud-shine (1E-15 Sv-m³/Bq-s)	Ground-shine (8 h) (10^{-12} Sv-m²/Bq)	Inhal-ation (Acute) (10^{-9} Sv/Bq)	Half-Life	Reactor Inventory (10E+18 Bq)	EF Isotope Conv. Factor	($ELCF_k$ + $CLCF_k$) /R_k (LCF/ 10E+15 Bq)
Co–60	99.6	50.3	0.40	5.3 yr	0.025	6.6	59.2
Kr–88	116.0	0.0	0.0004	2.8 h	2.9	2.0	0.0003
Sr–90	0.0	0.0	1.7	28 yr	0.19	4.3	118.
Zr–95	29.2	0.17	0.28	66 d	5.5	2.5	2.1
Ru–106	8.1	4.6	0.087	369 d	1.0	0.71	5.2
Te–132	7.6	35.3	0.25	3.2 d	4.7	3.4	0.13
I–131	14.5	8.7	0.035	8.0 d	3.2	1.0	0.29
Cs–137	22.2	12.6	0.56	30 yr	0.24	2.8	114.0
Ba–140	7.1	7.3	0.47	13 d	6.2	1.9	0.53
La–140	94.8	44.2	0.14	40 h	6.4	5.4	0.083
Pu–239	0.0017	0.0014	2.4	24000 yr	0.0008	6.0	1565.0

Each isotope is converted into an equivalent amount of I–131 by the equation:

$$EQN_k = CF_k \; I_k \; STN_k \; \exp[\,-\lambda_k\,(TN + DTN/2)],$$

where:

N = 1 for the early release and N = 2 for the late release,

EQN_k = the equivalent amount of I–131 for isotope k for release N,

CF_k = the isotope conversion factor for isotope k,

I_k = the inventory of isotope k at the start of the accident,

STN_k = the release fraction of isotope k for release N,

λ_k = the decay constant for isotope k,

TN = the time of the start of release N, and

DTN = the duration of release N.

Release fractions are determined in the source term calculation for nine radionuclide groups; each isotope is assigned to one of these groups. The total I–131 equivalent release is then found from:

$$EQ = \Sigma_k \, EQ1_k + \Sigma_k \, EQ2_k.$$

The relationship between the size of the equivalent release and the number of early fatalities is nonlinear because of the threshold effect and is complicated by the variability of the weather and the uneven distribution of the population around the site. The last factors are treated by using the weather-averaged, mean, or expected value for early fatalities as calculated by MACCS (Ref. B.59), using the actual site weather and demographic data. The effects of release magnitude were determined by making complete MACCS runs for I–131 releases of different sizes. The result is shown in Figure B.15. The MACCS

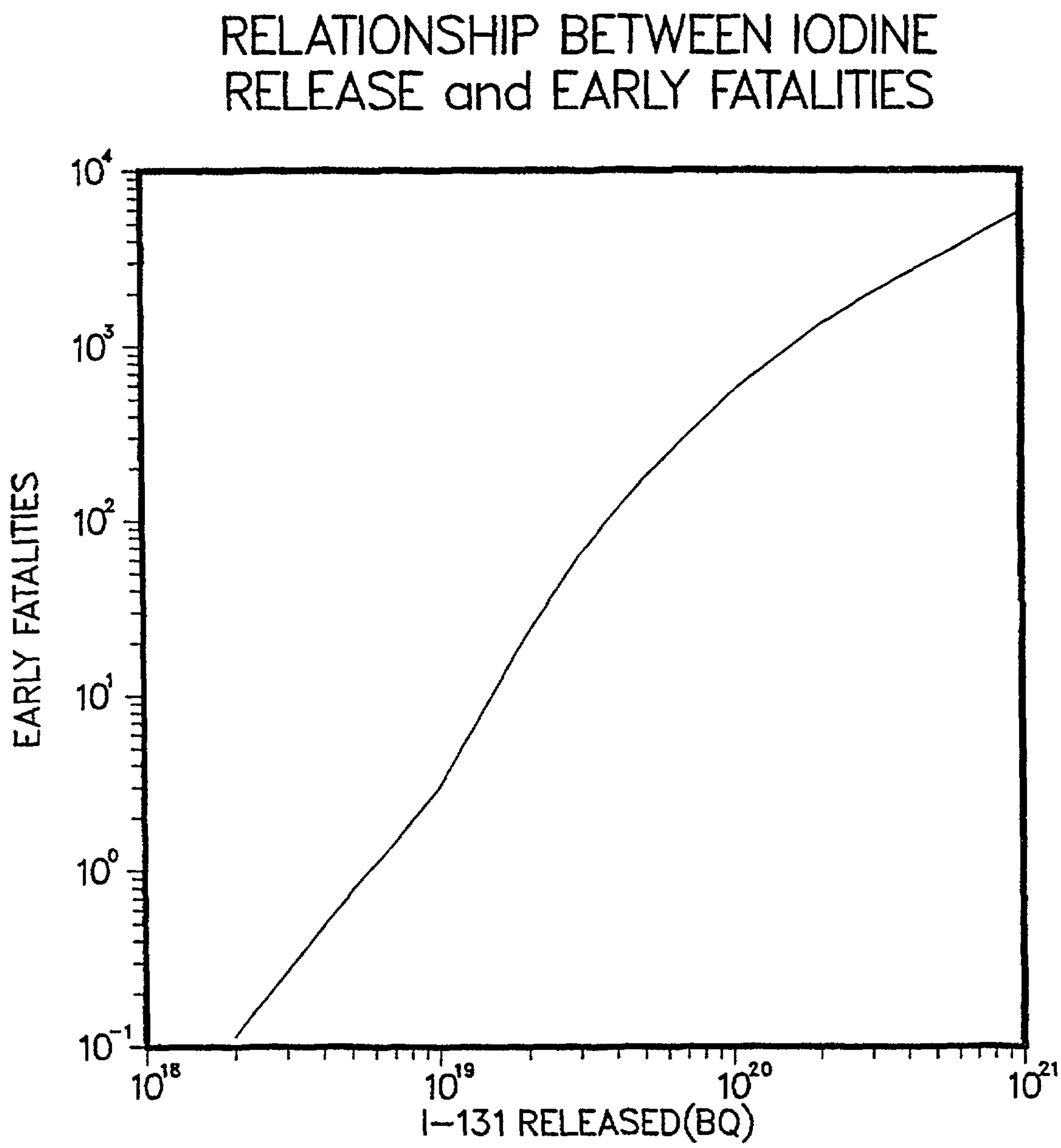


Figure B.15 Relationship between I-131 release and mean early fatalities used in determining early effects weights for partitioning.

calculations assumed an instantaneous release at ground level, with no evacuation or sheltering. No immediate mitigative action was used because the purpose of the EFW is to measure the radiological potential for early fatalities. The relocation criteria normally used for MACCS were not changed, however.

In summary, the procedure in calculating the EFW for a release is to compute the I-131 equivalent for each isotope for both the early and late release and then to total these values to obtain a total I-131 equivalent release. The curve relating the total equivalent release to mean early fatalities, Figure B.15, is then used to find the number of early fatalities, which is the early fatality weight, EFW. Releases with EQs less than 2E+18 Bq are assigned an early fatality weight of zero.

The method used to determine the chronic fatality weight is different. Because of the nearly linear relationship between the amount of an isotope released and the number of latent cancer fatalities, there is no need to convert the amount released for each isotope to an equivalent amount of a reference radionuclide. Instead, MACCS was used to determine the relationship between the amount released and the mean number of latent cancer fatalities for each site, using the weather and population data appropriate for that site. For each isotope, a MACCS run was made in which only that isotope was released. The entire inventory of the isotope in the reactor was released for the inert gases and 10 percent of the entire inventory for the other radioisotopes. The latent cancer fatalities due to the dose obtained in the first 7 days and those due to the chronic dose (received after the first 7 days) were obtained from the MACCS output. Since these fatalities are approximately linear with the amount released, the ratio of their total number to the amount released gives a reliable measure of such fatalities per unit release in the absence of any mitigating actions. This method is not applicable to early fatalities because of the threshold effect.

The chronic fatality weight for each isotope is given by the equation

$$CFW_k = I_k\,(ST1_k + ST2_k)(ELCF_k + CLCF_k)/R_k$$

and the total chronic fatality weight is then

$$CFW = \Sigma_k\, CFW_k,$$

where the summation is over all isotopes. I_k and STN_k have been defined above, and

CFW_k = the chronic fatality weight, in latent cancer fatalities, for a release of an amount $I_k\,(ST1_k + ST2_k)$ of isotope k;

$ELCF_k$ = the number of latent cancer fatalities due to early exposure from a release of an amount R_k of isotope k;

$CLCF_k$ = the number of latent cancer fatalities due to late exposure from a release of an amount R_k of isotope k; and

R_k = the amount of isotope k released in the MACCS calculation used to determine $ELCF_k$ and $CLCF_k$.

The early exposure is that obtained in the first 7 days after the accident and the late exposure is the exposure obtained after the first 7 days. The distinction is made because slightly different methods of these two periods are used in MACCS. Table B.5 lists values of the ratio $(ELCF_k + CLCF_k)/R_k$ for 11 representative isotopes. A total of 60 radionuclides are used in the consequence calculation. A complete listing of the conversion factors and CFWs for each may be found in Reference B.5.

Table B.5 also lists the half-life of the isotopes, and the reactor inventory at the time the accident starts. The half-life, the inventory, the EF isotope conversion factor, and the specific CFW, together with a rough idea of the release fractions for each radionuclide group, can be used to assess roughly the importance of the isotope to early and late offsite health effects. The 11 isotopes listed are all fairly important for at least one of the two consequence measures as this was one of the criteria for their selection. However, Kr-88 is clearly of little importance for chronic fatalities because the half-life and the specific CFW are low. Pu-239, on the other hand, can be seen to be of less importance than other isotopes for early fatalities because the inventory is very low and the isotope conversion factor is about the same size as that for more common fission products. The release fractions for Pu-239 also tend to be low.

The early and chronic fatality weights described here are not used in calculating consequences; they are only used in partitioning the source terms into groups of similar source terms. An average source term for each subgroup is used to calculate the offsite consequences with MACCS.

The process described in this section is applied to the source term for bin FFA-CAC-ABA-CA in Observation 4; the result is EFW = 0.0 and CFW = 2.7. The EFW is zero for this bin because the release is so low. Evacuation is not taken into account in determining the EFW, so the fact that the evacuation is complete before the FFA-CAC-ABA-CA release starts has no effect on the EFW although it may be very important in computing the actual consequences.

B.5.3 Partitioning Process and Results

The partitioning process divides the "space" defined by the logarithm of the effect weights into a number of cells. For the source terms for which both EFW and CFW are nonzero, this produces a rectangular grid on a two-dimensional plot. For the source terms for which the EFW is zero, the "space" to be divided is one-dimensional; that is, partitioning involves defining cells based on CFW alone. In the Surry analysis, there are 11,771 source terms with EFW = 0.0. The source term for bin FFA-CAC-ABA-CA is one of these.

For the 11,771 source terms with EFW = 0.0, the range of $\log_{10}$(CFW) is from −4.2 to 3.7. This range was divided into six groups or cells. The results of the initial division was:

Group		1		2		3		4		5		6	
$\log_{10}$(CFW)	−4.2		−2.9		−1.6		−0.2		+1.1		+2.4		+3.7
Count		349		1553		2892		1599		2438		2940	
% Weighted Freq.		7.9		33.2		49.3		2.5		4.9		2.1	

The second line gives the values of $\log_{10}$(CFW) that divide the range into six cells or groups. The third line gives the number of source terms in each group, and the last line gives the percentage of the weighted frequency in that group. (The weighted frequency of a bin is the absolute frequency of the bin, or the PDS frequency multiplied by the bin probability for the observation that applies.) The sixth group has slightly more source terms in it than the third group, but the frequencies of the source terms in the sixth group are very low. Thus the percentage of the frequency in group 3 is 25 times as high as that in group 6. The source term for bin FFA-CAC-ABA-CA has CFW = 2.7, and $\log_{10}(2.7)$ = 0.43, so the source term for bin FFA-CAC-ABA-CA goes into group 4. Group 4 includes all the source terms that have EFW = 0.0 and for which $\log_{10}$(CFW) lies between −0.2 and +1.1. Source terms are placed in this group if they meet these criteria no matter what PDS or APB they represent. Although the number of source terms in group 4 is fairly high at 1,599, the frequency of these source terms on the whole is fairly low, and the fraction of the frequency in group 4 is only 2.5 percent.

It is not worth making separate calculations for groups that have a small fraction of the weighted frequencies. Based on the information given in the table immediately above, it was decided that three CFW groups would be sufficient. Groups 1, 4, and 6 were eliminated, and the source terms in those groups were pooled with the neighboring groups. Groups 4 and 6 were pooled with groups 3 and 5 because fractions of the weighted frequencies in groups 4 and 6 were so low. The frequency of group 1 is higher than that of group 5, but the consequences of group 1 are very low, so the absorption of the source terms of group 1 into group 2 has a negligible effect on the risk.

The final partitioning is:

Group		1		2		3		4		5		6	
$\log_{10}$(CFW)	−4.2		−2.9		−1.6		−0.2		+1.1		+2.4		+3.7
Count		0		1902		3223		0		6646		0	
% Weighted Freq.		0		41.1		49.8		0		9.1		0	

The $\log_{10}$(CFW) for bin FFA-CAC-ABA-CA is just above the center value for group 4, so it was placed in group 5. The average values of the release fractions and the release characteristics for group 5 at the end of partitioning include the source terms originally in groups 4 and 6 that have been absorbed into group 5, not just the source terms originally in group 5.

Group 5 in the partitioning process for source terms with zero EF weight becomes source term 17 when the partitioning for source terms with nonzero EF weights are considered and empty groups are eliminated. The mean frequency of this group is 3.4E−6/reactor year, and the conditional probability of this group, with respect to the source terms with EFW = 0.0, is 0.084.

Each source term group is subdivided into three source term subgroups (STGs) on the basis of evacuation timing:

STG1 (early evacuation): Evacuation starts at least 30 minutes before the release begins.

STG2 (synchronous evacuation): Evacuation starts between 30 minutes before and 1 hour after the release begins.

STG3 (late evacuation): Evacuation starts 1 or more hours after the release begins.

For source term 17, STG1 has 94 percent of the source terms, and STG2 has 6 percent of the source terms. There are no source terms in STG3. As would be expected from the very late release time for the source term for bin FFA-CAC-ABA-CA for Observation 4, this source term goes to STG1.

Consequence calculations are not done for source term group 17 as a whole; they are done for the subgroups since the timing differs markedly between the subgroups. The mean source terms for each subgroup form the basis for each consequence calculation. The properties of subgroup 1 of source term 17 are given in Table B.6. This information is used by the computer code MACCS (Ref. B.59) to calculate the offsite consequences. Although the source term for bin FFA-CAC-ABA-CA of Observation 4 had a zero early release, some of the other source terms placed in STG1 of source term 17 in the partitioning process had nonzero early releases, thus the mean early release fractions for STG1 are nonzero. The representation of the source term for FFA-CAC-ABA-CA by a source term with a nonzero early release does introduce a slight distortion. However, the early release is small, and it occurs at 13.3 hours, which is well after the time at which the warning to evacuate is given, 6.9 hours. The small early release and the fact that the evacuation is completed before the release commences mean that there are no early fatalities from the early release for STG1. As the latent cancer fatalities and population dose do not depend on the release timing to any significant degree, the error introduced by the nonzero early release is negligible.

The results of partitioning are contained in two files. One file contains all the source terms that MACCS will use to calculate consequences. STG1 of source term 17 is the 49th source term in this file and is referred to as SUR−49 for the consequence calculation. The second output file indicates the STG in which each bin of each observation was placed in the partitioning process. For bin FFA-CAC-ABA-CA of Observation 4, this was STG1 of source term 17, now known as SUR−49.

Table B.6 Properties of source term 17, subgroup 1.

Property	Minimum Value	Maximum Value	Frequency Weighted Mean
Release Height	10	10	10
Warning Time	2.2E+4	3.6E+4	2.5E+4
Start Early Release	4.7E+4	5.1E+4	4.8E+4
Duration Early Release	0.0	3.6E+3	3.3E+2
Energy Early Release	0.0	7.0E+8	9.2E+5
ERF Xe, Kr	0.0	1.0E 0	1.4E-1
ERF I	0.0	1.5E-1	7.3E-3
ERF Cs, Rb	0.0	1.1E-1	5.4E-3
ERF Te, Sc, Sb	0.0	2.9E-2	1.2E-3
ERF Ba	0.0	1.4E-2	1.2E-4
ERF Sr	0.0	2.4E-3	2.3E-5
ERF Ru, etc.	0.0	1.1E-3	6.6E-6
ERF La, etc.	0.0	5.2E-3	2.8E-5
ERF Ce, Np, Pu	0.0	1.4E-2	1.4E-4
Start Late Release	4.7E+4	1.3E+5	1.1E+5
Duration Late Release	1.0E+1	2.2E+4	1.2E+4
Energy Late Release	0.0	7.0E+8	9.2E+5
LRF Xe, Kr	0.0	1.0E 0	8.1E-1
LRF I	5.0E-6	1.3E-1	4.0E-2
LRF Cs, Rb	0.0	5.0E-2	3.9E-4
LRF Te, Sc, Sb	3.4E-11	9.6E-2	2.7E-4
LRF Ba	6.3E-14	1.7E-2	4.9E-5
LRF Sr	1.0E-18	1.4E-3	2.7E-6
LRF Ru, etc.	5.2E-18	1.6E-3	4.2E-6
LRF La, etc.	5.2E-18	1.7E-3	6.5E-6
LRF Ce, Np, Pu	1.6E-13	1.4E-2	4.2E-5

Note: ERF means early release fraction, LRF means late release fraction, the release height is in meters, the energy of the release is in watts, and all times are in seconds.

B.6 Consequence Calculation

The computer code MACCS (Ref. B.59) is used to determine the consequences of a release of fission products from the damaged reactor. Consequences are the offsite results of the accident expressed in societal terms; for example, number of early fatalities or the risk of latent cancer fatality to the population within 10 miles of the plant. A separate MACCS calculation is performed for the mean source term associated with each STG.

B.6.1 Description of Consequence Calculation

The consequence calculation is an extensive calculation. The inventory of fission products in the reactor at the time of the accident and the release fractions for each radionuclide class are used to calculate the amount released for each of the 60 isotopes considered by MACCS. Then, for a large number of weather situations, the transport and dispersion of these radionuclides in the air downstream from the plant is calculated. The amounts deposited on the ground are computed for each distance downwind. Doses are computed for a hypothetical human at each distance from immersion in the contaminated air, from breathing the contaminated air, from the exposure due to radioactive material deposited on the ground, and from drinking water and eating food contaminated by deposited radioactive particles. The first three of these dose pathways result in immediate doses that can cause health effects within a few hours or days of the release. These are called acute effects. In addition to the three pathways that cause acute effects, long-term exposure from contaminated ground and ingestion also contributes to latent cancer fatalities and other delayed effects. These are known as chronic effects. Doses to an individual are converted to population doses using population data. Doses and health effects are calculated for nine organs of the human body.

The consequence calculation requires a large amount of supporting data. Files of weather data and demographic data specific to the plant region are required. Information is needed on land use (crops grown or dairy use) and land value in the surrounding area. Dose factors are used to relate the dose to each organ to the concentration of each of the isotopes as explained in Section B.5.2.

To assess the effects of different weather on the consequences, the complete transport, deposition, and dose calculation is repeated over 1,000 times for each source term. For each of 16 wind directions, the consequence calculation is performed for about 130 different weather situations. Weather data from the specific plant being modeled are used. The wind direction determines the population over which the plume from the accident passes. The atmospheric stability is also important as it determines the amount of dispersion in the plume downwind from the plant. Deposition is much more rapid when it is raining than when it is not. Each weather sequence contains information about how the wind direction, stability, and precipitation changes from hour to hour. The consequence calculation also computes the effects of the evacuation of the population from the immediate area around the plant. More information on the consequence calculation may be found in Reference B.59.

B.6.2 Results of Consequence Calculation

As discussed in Section B.5.3, the source term for bin FFA-CAC-ABA-CA for Observation 4 was included in group SUR-49 in the partitioning process. The eight consequence measures used in the risk computation, with the mean or expected values computed for partition group SUR-49, are:

Early Fatalities	0.0
Early Injuries	4.2E-6
Latent Cancer Fatalities	1.1E+2
Population Dose—50 miles	2.7E+5 person-rem
Population Dose—region	6.9E+5 person-rem
Economic Cost (dollars)	1.8E+8
Individual Early Fatality Risk—1 mile	0.0
Individual Latent Cancer Fatality Risk—10 miles	7.6E-5

The distribution for latent cancer fatalities is shown in Figure B.16. Note that these consequences assume that the release SUR-49 occurred.

The population doses are the effective dose equivalent whole body doses. The individual early fatality risk is the risk to an average individual within 1 mile of the site boundary. It is calculated using the actual population distribution within 1 mile of the site boundary. For SUR-49, there are no early fatalities and the early fatality risk within 1 mile is zero as well. This is due to the very small magnitude of the early release. As 0.5 percent of the population is assumed not to evacuate when told to do so, neither of these measures would be zero if the release were large enough to cause early fatalities. The individual latent cancer fatality risk is the risk to an average individual within 10 miles of the plant, excluding the food pathways. The low total release fractions for SUR-49 result in a low value for this risk measure also.

After the MACCS calculations are completed, the results for each weather trial (wind direction and weather sequence) for each STG are used in constructing complementary cumulative distribution functions, as discussed below in Section B.7.3. Averages over the weather trials determine the expected consequence values for each STG. These are extracted and assembled into a matrix for use in determining distributions of expected values of risk.

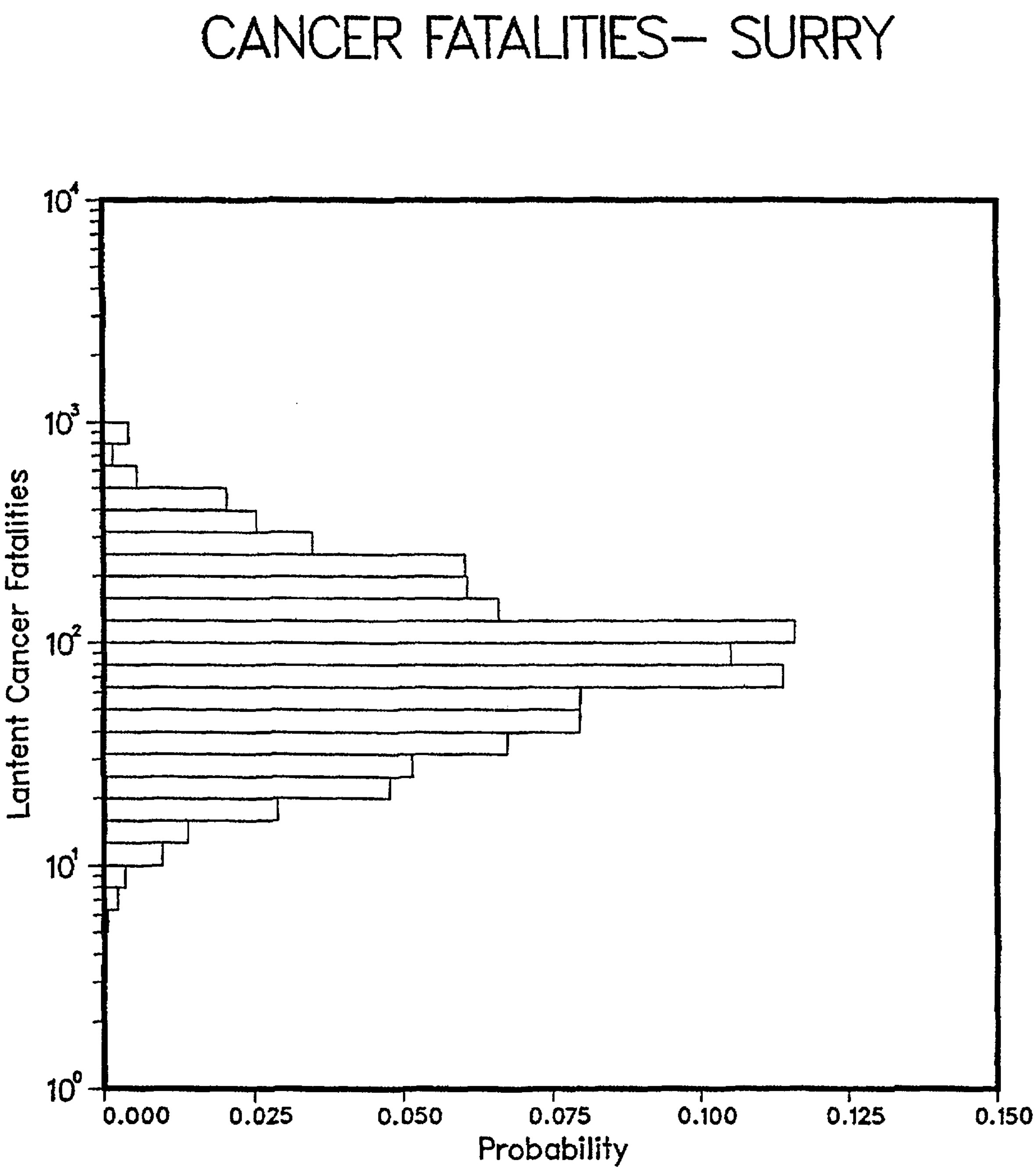


Figure B.16 Distribution of latent cancer fatalities computed for STG SUR-49.

B.7 Computation of Risk

The risk measures that form the final result of a PRA are formed by merging the results of all the preceding analyses. Section B.7.1 introduces the concepts and terms. Mean risk is discussed in Section B.7.2, and displays that show the variability in risk due to the weather and other variables are discussed in Section B.7.3.

B.7.1 Introduction

The final step in a complete risk calculation is putting together the results of all the steps described above to produce a measure of risk. The accident progression analysis and the source term analysis were carried out on conditional bases. That is, the accident progression analysis was performed for each PDS group for each observation without regard for the frequency of that PDS group. Similarly, for each observation in the sample, the source terms for each bin were calculated without taking the bin frequency into account. The partitioning process, however, used the absolute frequency of each source term in determining which groups of source terms could be combined into neighboring groups and in determining the mean values used to represent each STG. To accomplish this, for each observation, the absolute frequency of each PDS group was obtained from the accident frequency analysis, and the conditional probability of the bins was obtained from the accident progression analysis. The consequence analysis was performed for each source term subgroup without regard for its absolute or relative frequency.

Risk is generally displayed in three forms: mean risk, histograms of risk, and complementary cumulative distribution functions (CCDFs). For each sample member and for each consequence measure, the results are averaged over the weather conditions to obtain an expected value. If, for one consequence measure, the average of the 200 expected values is taken, the result is actually the mean value of the expected risk, or the mean value of the weather-averaged mean risk. It is usually referred to as just the mean risk. The mean values of expected risk do provide single numbers (for each consequence measure) that characterize the risk. However, they provide no information about the variability or uncertainty in the risk. Histograms of expected risk display the variation in weather-averaged expected risk among the members of the sample. Mean risk is discussed in Section B.7.2 below.

To display the variability in risk due to the weather, CCDFs are used. While a separate CCDF is formed for each observation, not all 200 are usually plotted at once. Instead, statistical measures of the sample of CCDFs are plotted. CCDFs are presented and explained in Section B.7.3.

B.7.2 Calculation and Display of Mean Risk

As described in Section B.6, a separate consequence calculation is performed with MACCS for each source term subgroup, and the result was a large number of consequences. They are represented by C_{lvwk}, which is the value of consequence measure l for wind direction v, weather sequence w, and source term subgroup k. When a frequency-weighted average is formed over the wind direction and the weather sequences, a set of mean or expected consequence measures, C_{lk}, is formed. As there are 52 source term subgroups, 52 expected values are computed for each of the eight consequence measures.

The expected risk for measure l and observation n can be expressed in terms of the results of the individual analyses by:

$$\mathrm{Risk}_{ln} = \Sigma_h\, \Sigma_i\, \Sigma_j\, \Sigma_k\, f_n(\mathrm{IE}_h)\; P_n\,(\mathrm{IE}_h \rightarrow \mathrm{PDS}_i)\; P_n\,(\mathrm{PDS}_i \rightarrow \mathrm{APB}_j)\; P_n\,(\mathrm{APB}_j \rightarrow \mathrm{STG}_k)\; C_{lk},$$

where:

Risk_{ln} = expected risk for consequence measure l for observation n (consequences/reactor year);

$f_n(\mathrm{IE}_h)$ = frequency (per reactor year) of initiating event h for observation n;

$P_n\,(\mathrm{IE}_h \rightarrow \mathrm{PDS}_i)$ = probability that initiating event h leads to PDS_i for observation n;

$P_n\ (PDS_i \rightarrow APB_j)$ = probability that PDS_i leads to APB_j for observation n;

$P_n\ (APB_j \rightarrow STG_k)$ = probability that APB_j was partitioned into source term subgroup k for observation n; and

C_{lk} = expected value of consequence measure l for source term subgroup k.

All the terms in the equation change from observation to observation except C_{lk}. The distribution for initiating event frequency is part of the data used in the calculations performed in the accident frequency analysis. The summation over all the different initiating events is performed in the accident frequency analysis, and the output of the accident frequency analysis is actually

$$f_n(PDS_i) = \Sigma_h f_n(IE_h)\ P_n(IE_h \rightarrow PDS_i),$$

the frequency for each PDS group for each observation.

$P_n\ (PDS_i \rightarrow APB_j)$ is the result of the accident progression analysis and the evaluation of the accident progression event tree. It consists of a list of APBs for each PDS group, with a probability for each, for each observation. The APB probability is conditional on the occurrence of the PDS group.

$P_n\ (APB_j \rightarrow STG_k)$ represents the combined result from the source term calculation and the partitioning process. For each observation, a source term is computed for each APB in that observation. The source terms for all 200 observations in the sample are placed in source term subgroups in the partitioning process. The outcome of the source term analysis and partitioning is the STG to which each APB in each observation is assigned and a mean source term for each STG. $P_n\ (APB_j \rightarrow STG_k)$ is an element of the matrix that contains this information; the matrix element for given values of j and k is 1.0 if APB_j was placed in STG_k and 0.0 otherwise.

$Risk_{ln}$ is the expected risk for each observation. What is reported as the mean risk for measure l is really the expected or mean value of expected risk:

$$Risk_l = \Sigma_n Risk_{ln}/n_{LHS}$$

where n_{LHS} is the number of observations in the LHS (200 for Surry). That is, the expected or mean value of risk is the average of 200 values of weather-averaged expected risk.

The distribution of the 200 values for the mean latent cancer fatality (LCF) risk are displayed in the form of a histogram. Such a histogram for LCFs is shown in Figure B.17 for 99.5 percent evacuation. The quantiles are obtained by ordering the 200 observations by the value of the independent variable. The 5th and 95th percentiles, the mean, and the median are shown. Not shown are the extreme values (Observation 123, 7.8E–5 LCFs/reactor year and Observation 76, 7.5E–2 LCFs/reactor year) since they are not indicative of the sample as a whole. The mean for Observation 4 is 5.0E–3 LCFs/reactor year, which is near the 75th percentile and is well above the median.

The equation for risk given above provides the means for determining the risk due to certain subsets or groups of events and the contribution of these subsets to mean risk. For example, say the contributions of the APBs were desired. The risk due to APB_k for observation n is:

$$Risk_{lnj} = \Sigma_h \Sigma_i \Sigma_k\ f_n(IE_h)\ P_n(IE_h \rightarrow PDS_i)\ P_n(PDS_i \rightarrow APB_j)\ P_n(APB_j \rightarrow STG_k)\ C_{lk}.$$

For each observation, the fractional contribution of APB_j is:

$$Risk_{lnj}\ /Risk_{ln}.$$

The risk and the fractional contributions for combinations of APBs can be determined in an analogous manner.

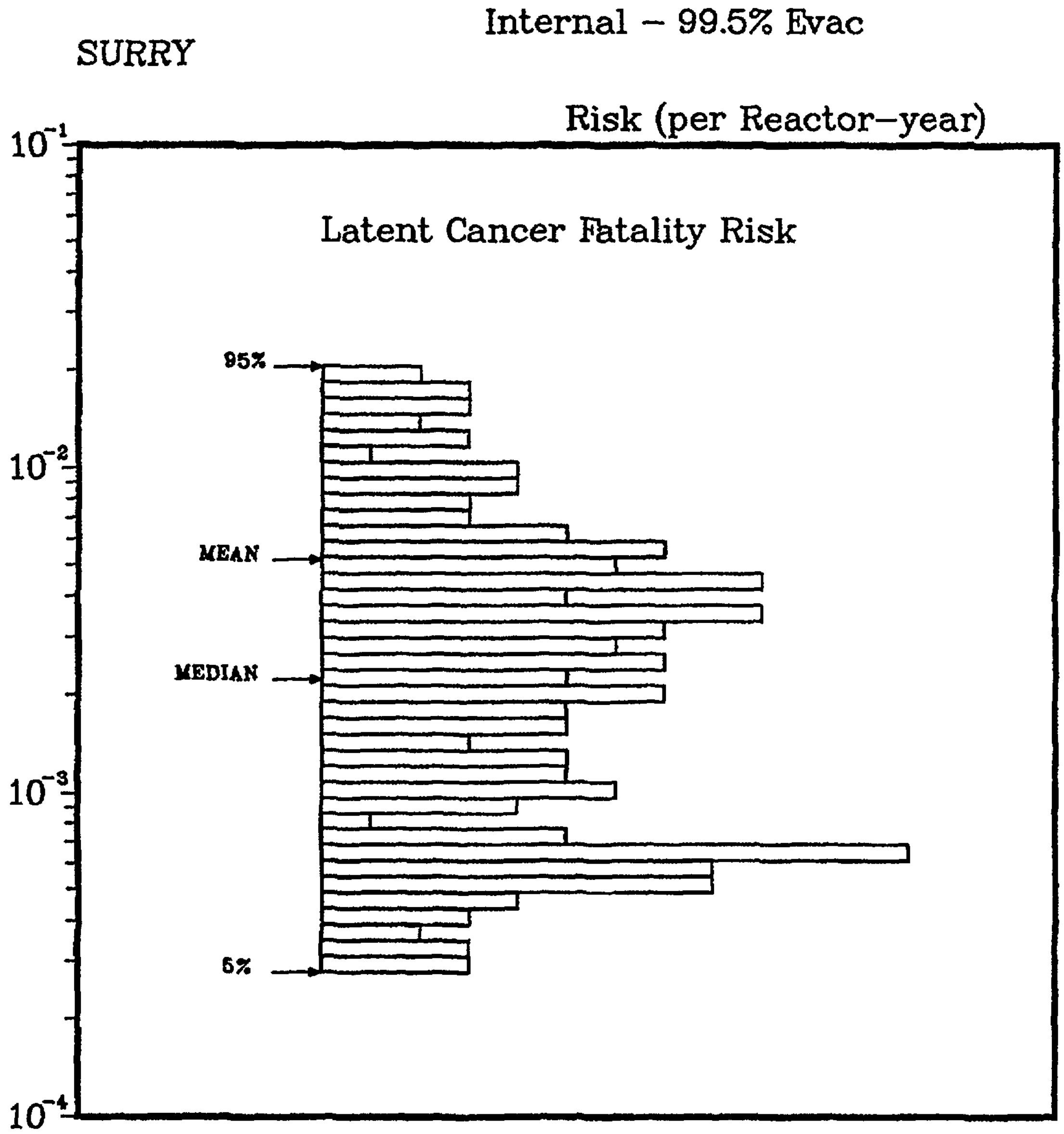


Figure B.17 Distribution of expected (weather-averaged) latent cancer fatality risk for Surry.

For the sample as a whole, the mean risk from APB_k is:

$$(\Sigma_n \mathrm{Risk}_{lnj})/\mathrm{n_{LHS}}.$$

Fractional contributions to mean risk for the entire sample can be determined in two ways. The expression

$$[\Sigma_n(\mathrm{Risk}_{lnj})/\mathrm{Risk}_{\mathrm{ln}}]/\mathrm{n_{LHS}}$$

is termed the mean fractional contribution to risk for APB_j. The expression

$$[(\Sigma_n\ \mathrm{Risk}_{lnj})/\mathrm{n_{LHS}}]/\mathrm{Risk}_l$$

is termed the fractional contribution to mean risk for APB_j. Figure 3.14 of the main report shows the fractional contribution to mean risk for groups of APBs. Because the distributions for risk often have very long tails and the mean is then determined by the few observations with the highest risk, the fractional contribution to mean risk and the mean fractional contribution to risk can be quite different. There is no consensus that one method of calculating the contribution to mean risk is preferable to the other.

The summations in the expression for $\mathrm{Risk}_{\mathrm{ln}}$, and the computations necessary to obtain the statistical measures, are performed by a computer program named RISQUE. A description of this code may be found in an appendix to Reference B.5.

B.7.3 Calculation and Display of CCDFs

As already indicated, the output from MACCS is a large number of consequences represented by C_{lvwk}, the magnitude of consequence measure l for wind direction v, weather sequence w, and source term subgroup k. When the results for each weather trial (combination of wind direction and weather sequence) are kept separate and the probability of each weather trial is taken into account, a complementary cumulative distribution function (CCDF) may be obtained for each STG. Figure B.18 displays the latent cancer fatality CCDFs for STG SUR-49 and for all 52 STGs. The CCDF for a single STG displays the effects of the variability of the weather and relates the magnitude of the consequence to the probability that it will be exceeded. (The cumulative distribution function (CDF) displays the probability that a certain value will not be exceeded. The CCDF displays the probability that a certain value will be exceeded. CCDFs are usually shown for historical reasons.) The CCDFs in Figure B.18 have conditional probability on the ordinate: each curve displays the latent cancer fatality results conditional on the occurrence of a release. For example, the top plot in the figure displays the results of the consequence analysis conditional on the SUR-49 release.

To understand how the CCDF for a single STG is formed, let the subscript u represent a combination of wind direction and weather sequence; then C_{lvwk} may be written C_{luk}. Each combination of wind direction and weather sequence is denoted a weather trial. C_{luk} is now the value of consequence measure l for weather trial u and source term subgroup k. The CCDF for consequence measure l and STG_k is formed from a set of results of the form:

$$(C_{luk}, P_u) \qquad u = 1, 2, ..., \mathrm{nWT},$$

where:

P_u = probability that weather trial u will occur, and

nWT = number of weather trials (about 2,500).

The set (C_{luk}, P_u) is ordered on the consequence measure (i.e., $C_{luk} < C_{l,\ u+l,\ k}$). The results of the consequence evaluation include both C_{luk} and P_u.

Since it is the complementary cumulative probability that is plotted in Figure B.18, this figure results from plotting the set $(C_{luk}, 1 - CP_u)$ where CP_u is the cumulative probability defined by:

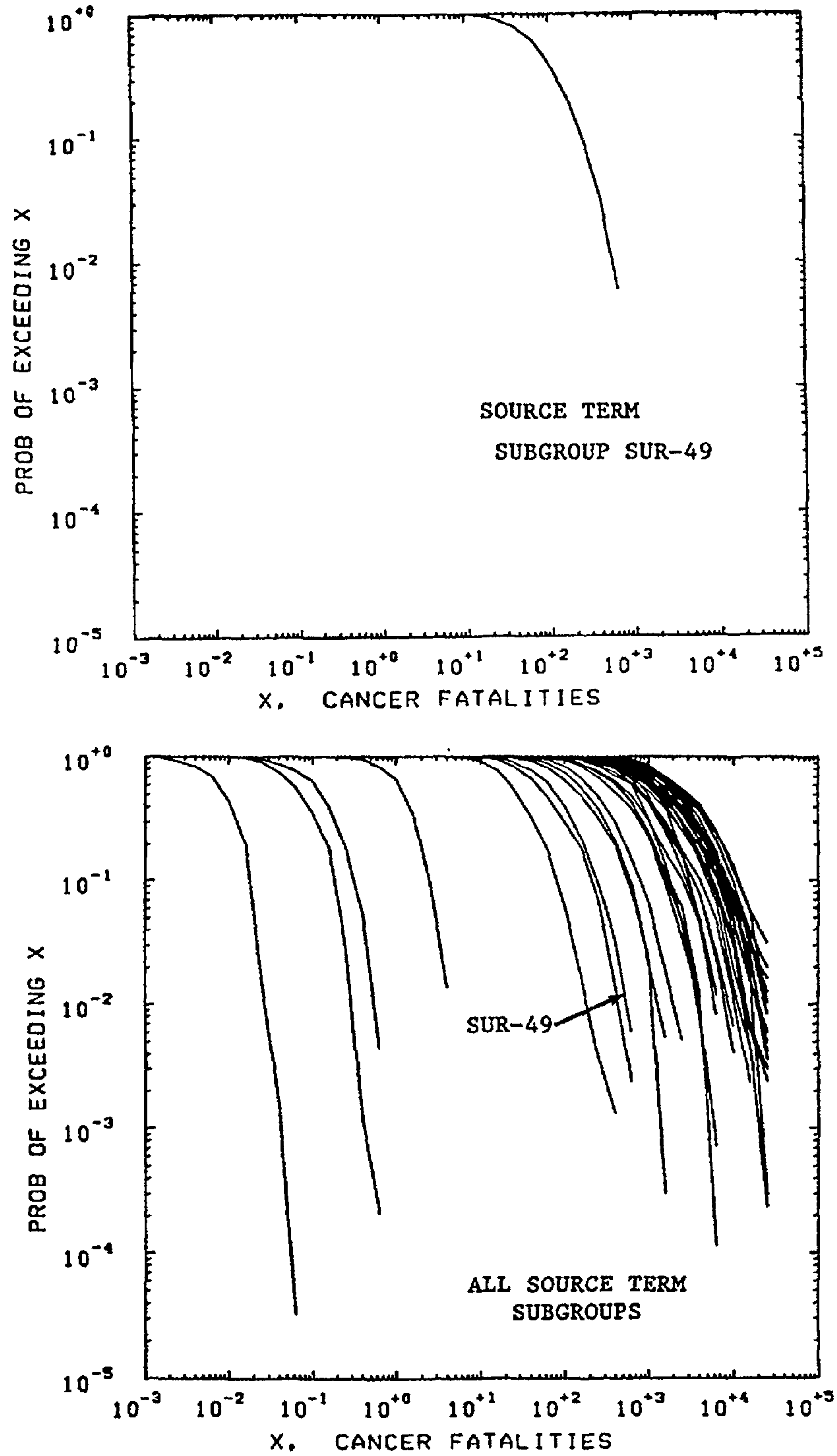


Figure B.18 CCDFs for latent cancer fatalities for STG SUR-49 and for all 52 STGs.

$$CP_u = \sum_{q=1}^{u} P_q.$$

Figure B.18 shows the probability of exceeding a given consequence. For instance, the top plot shows that the conditional probability of exceeding 100 LCFs is about 0.10, given the occurrence of SUR-49.

The next step is to form the CCDF for each of the n_{LHS} (200) observations that comprise the sample. This is done by considering a large number of values on the abscissa in the lower plot in Figure B.18. Let X represent a value of a consequence measure; in this figure X would be a number of latent cancer fatalities. For each STG_k, the ordinate (i.e., Y) of the point (X,Y) on the curve defined by $(C_{luk}, 1 - CP_u)$ gives the probability of exceeding X consequences. Let this probability be represented by $P_{xc_k}(X)$. The next factor needed is f_{nk}, the frequency of STG_k for observation n. The equation for f_{nk} is analogous to the equation for $Risk_{ln}$:

$$f_{nk} = \Sigma_i \Sigma_j \, f_n(PDS_i) \, P_n(PDS_i \rightarrow APB_j) \, P_n(APB_j \rightarrow STG_k),$$

where all the terms have already been defined in Section B.7.2.

So for one STG_k, the contribution to the CCDF for observation n is the set of points $(X, P_{xc_k}(X) f_{nk})$, where the product $P_{xc_k}(X) f_{nk}$ represents the frequency for observation n that STG_k will result and that X will be exceeded. When all the STG_k are considered, it may be seen that the set of points that defines the CCDF for observation n becomes

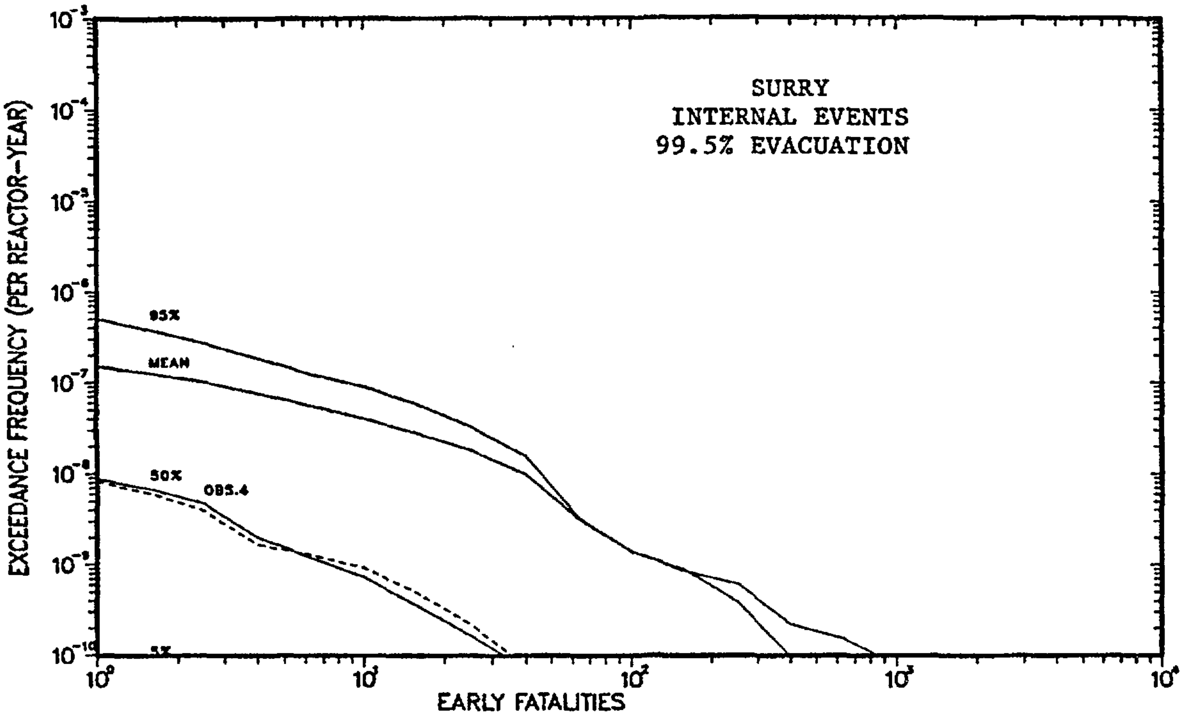
$$(X, \Sigma_k [P_{xc_k}(X) f_{nk}]),$$

where the summation is over all the source term subgroups. As there are 200 observations in the sample, 200 such CCDFs are calculated. The dashed lines on the plots in Figure B.19 are the CCDFs for Observation 4 for early fatalities (EFs) and LCFs.

While the CCDFs for all 200 observations for consequence measure l could be placed on one plot, the result is cluttered and hard to read. Instead, four statistical measures of the set of 200 curves are plotted. The four measures of the sample are the 5th percentile, the 95th percentile, the median, and the mean. They are shown on Figure B.19. These four curves do not represent specific observations. Instead, each point on these curves is a statistical measure of the sample at that value of the consequence. For example, consider the mean curve for latent cancer fatalities in Figure B.19. For 100 LCFs, there are 200 values of the exceedance frequency, one for each observation in the sample. As each observation is as likely as the next, these 200 values are summed and divided by 200 to determine the mean for 100 LCFs. This process is repeated for 105 LCFs, 110 LCFs, and so on. In this way, the points that comprise the mean curve are obtained. The result does not coincide with any one of the 200 observations. The 5th and 95th percentiles are obtained in an analogous manner. Since the 200 values for a given LCF value are ordered, the 5th percentile value is the value of the 10th observation.

The CCDF for each observation displays the effects of the variability of the weather, through the term P_u, as well as some variability due to the sampling process, through the term f_{nk}. The shape of the CCDF for a single observation depends upon the relative frequency of each STG in that observation, which in turn depends on all the sampled factors in the accident frequency analysis, the accident progression analysis, and the source term analysis. As the CCDFs for each STG are the same for all the observations in the sample, the differences among the CCDFs for the 200 observations are due to the different frequencies for each STG in each observation.

The CCDF can be read in a number of ways. Consider the median curve for LCF in Figure B.19; it shows that the median frequency of exceeding one to ten LCFs is between 3E-6/reactor year and 4E-6/reactor year; the median frequency of exceeding 100 LCFs is about 2E-6/reactor year; the median frequency of exceeding 1,000 LCFs is 5E-7/reactor year; and the median frequency of exceeding 10,000 LCFs is 5E-9/reactor year. The same interpretation is valid for the other measures of the sample of 200 CCDFs. Another way to read a CCDF is to consider one particular value of the consequence. Take 1,000 LCFs for



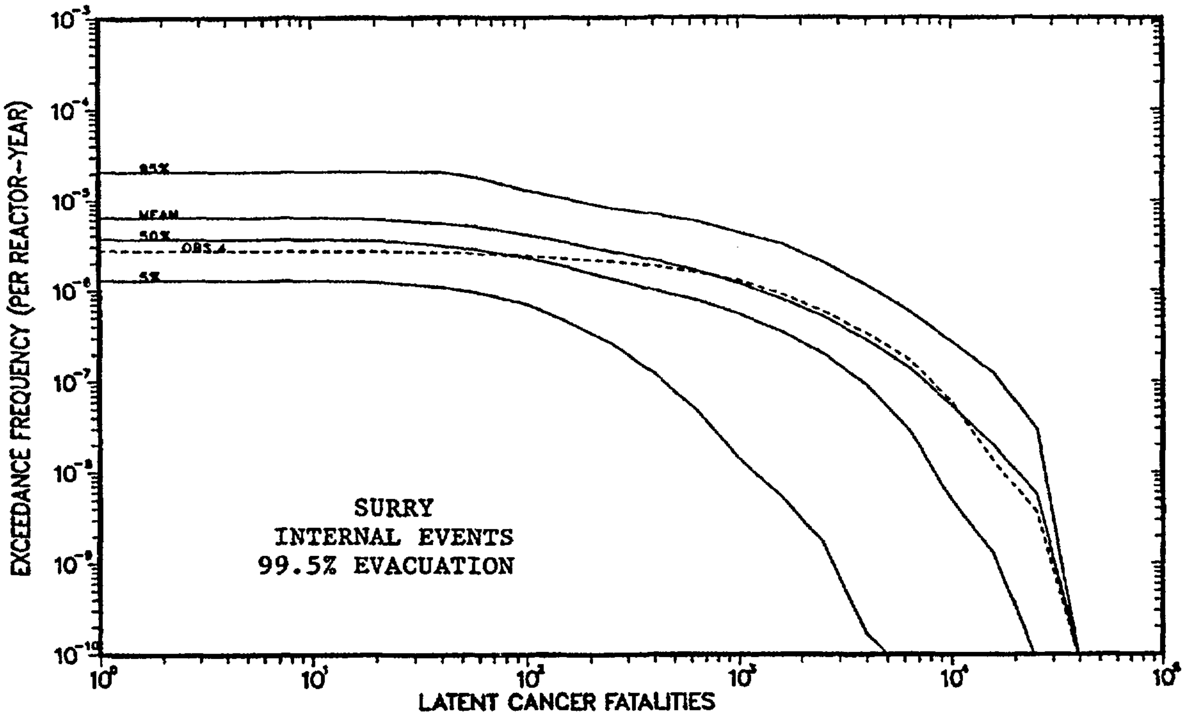


Figure B.19 Computed curves showing four statistical measures of 200 CCDFs for Surry for early fatalities and latent cancer fatalities. (The CCDF for Observation 4 is also shown.)

example: in 95 percent of the observations, the frequency that 1,000 LCFs would be exceeded is less than 4E–6/reactor year; while in half of the observations, the frequency that 1,000 LCFs would be exceeded is less than 5E–7/reactor year.

Because of the wide range of the consequences, the CCDFs are plotted with a logarithmic scale on both axes. The distributions typically have long "tails" in both directions. For distributions like this, the mean value often reflects just a few high observations. Such is the case for the early fatalities in Figure B.19. Above about 150 EFs, the mean value of the sample exceeds the 95th percentile. This means that the mean in this portion of the plot is determined by fewer than 10 observations out of the 200 in the sample.

B.8 Summary

In this section, a specific accident at Surry has been followed all the way through a complete analysis, from the initiating event in the accident frequency analysis through to the offsite risk values computed by the consequence analysis. The example selected was a fast SBO accident; the sequence is denoted T1S1-QS-L, and the PDS is TRRR-RSR. To determine the uncertainty in risk, a sample of 200 observations was used for the Surry analysis. Observation 4 was followed in this example, and specific numbers for the important events and parameters for this observation were given above.

This accident is initiated by the LOSP. The emergency DG dedicated to Unit 1 (DG1) fails to start, and the swing DG (DG3) also fails to start, so all ac power is lost, which is designated a station blackout. The only means of heat removal from the reactor core in a station blackout situation is the steam-turbine-driven (STD) AFW train, which operates as designed initially. However, a secondary system safety relief valve fails to close, and the water lost out this valve fails the AFWS in about an hour. Offsite power is not restored by this time, and the plant staff fails to get one of the two DGs capable of supplying power to Unit 1 to start. With the failure of the STD-AFW train, and no ac power to run the motor-driven AFW trains, there is no heat removal from the reactor. The reactor coolant system (RCS) heats up until the pressure forces open the PORVs. Water (steam) loss through the PORVs continues, with the PORVs cycling open and closed, until enough water has been lost to reduce the liquid water level below the top of active fuel. Without electric power, there is no way to replace the water lost from the RCS. After the water level has dropped some distance below the top of active fuel, the fuel above the water level heats up sufficiently to degrade.

The accident frequency analysis determined the frequency of the initiating events and the probabilities of the other failures that led to this state. The frequency of the initiating event for Observation 4 is 0.099/reactor year, and the frequency of the cut set defined in Table B.1 is 3.4E–8/reactor year. This cut set was placed with many others in sequence T1S1-QS-L, which has a frequency, for Observation 4, of 2.4E–7/reactor year. Sequence T1S1-QS-L was grouped with other fast station blackout sequences in plant damage state TRRR-RSR, which has a frequency of 4.8E–7/reactor year for Observation 4. As TRRR-RSR is the only PDS in PDS group 3, this is also the group frequency.

The core melt process and the response of the containment were treated in the accident progression analysis. Although the most likely outcome for TRRR-RSR was termination of the accident without failure of either the reactor vessel or the containment due to the recovery of offsite power, the path followed through the APET took the branch in which power was not restored in time to avert vessel breach. The RCS was at the PORV setpoint pressure at the start of core degradation since there were no breaks in the system. However, the path through the event tree took the branch in which the RCP seals failed, and the vessel failed with the RCS at intermediate pressure. While high-pressure melt ejection and direct containment heating accompanied the failure of the vessel, the pressure rise in the containment was insufficient to fail the containment. However, the core material attacked the concrete floor of the reactor cavity, and the containment failed by basemat meltthrough many hours after the start of the accident. This path through the event tree is designated accident progression bin FFA-CAC-ABA-CA. For the observation chosen, this bin is the most likely bin that has both vessel breach (VB) and containment failure (CF).

For Observation 4, the evaluation of the APET for PDS group 3 resulted in a conditional probability of 0.017 for bin FFA-CAC-ABA-CA. Bin FFA-CAC-ABA-CA was also generated in evaluating the APET for PDS group 1. The absolute frequency of this bin for both PDS groups, for Observation 4, is 3.2E–8/reactor year. Only a small portion of this frequency came from the cut set listed in Table B.1.

In the source term analysis, the release fractions, the timing of the release, and the energy and height of the release were determined for bin FFA-CAC-ABA-CA. These values define the source term. In the partitioning process, the source term for bin FFA-CAC-ABA-CA of Observation 4 was grouped with other similar releases into source term group SUR–49. In the consequence analysis, the offsite consequences of the mean source term for SUR–49 were computed. SUR–49 has such small release fractions that it caused no early fatalities. The total release caused (in this calculation) 113 latent cancer fatalities when averaged over all weather conditions.

Finally, the results of the four constituent analyses were combined to determine risk. For the consequences averaged with respect to the weather variability, when all PDS groups and APBs are considered, the contribution of Observation 4 was slightly below the median for early fatalities and near the mean value for latent cancer fatalities. Although bin FFA-CAC-ABA-CA was the most likely bin in Observation 4 that had both vessel breach and containment failure, it was a relatively improbable bin in the entire sample, and its contribution to the risk measures was very small. The position of the CCDFs for Observation 4 is shown in Figure B.19.

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