DRAFT LETTER REPORT

ABRIDGED RISK STUDY DURING LOW POWER / SHUTDOWN OPERATION AT SURRY

J. Jo
C. C. Lin
V. Mubayi
L. Neymotin
S. Nimnual

Safety Integration Group
Safety and Risk Evaluation Division
Department of Nuclear Energy
Brookhaven National Laboratory

May 29, 1992

Prepared for
U. S. Nuclear Regulatory Commission
FIN L-1680

Acknowledgement

The authors wish to thank C. Ryder of the NRC for his review, suggestions and editorial contributions to this report. The authors also wish to express their thanks to the members of the Source Term Advisory Group for their technical advice. The efforts of Barbara Kponou in typing and preparing this document is much appreciated by the authors.

Contents

List of Figures

List of Tables

1 INTRODUCTION

1.1 Objectives

The Office of Nuclear Regulatory Research at the U. S. Nuclear Regulatory Commission established programs to investigate postulated accidents during low power and shutdown (LP&S) operations of a PWR (Surry) and a BWR (Grand Gulf). One such program is a risk study of accident progressions and consequences.

The objective of the reported study is to make a preliminary risk determination of the progressions (Level 2 analysis) and the consequences (Level 3 analysis) of accidents during low power and shutdown operations in the Surry plant. The study was designed to obtain results for regulatory decisions that are to be made in the early summer of 1992. This letter report documents the methods, findings, and implications of the study. The sister study of the Grand Gulf plant is reported separately by the staff at Sandia National Laboratories (SNL).

1.2 Scope of the Study

The abbreviated risk analysis took place from January through April 1992. The study has been referred to as an *abridged risk* analysis. The term *abridged* means that simple event trees (about nine top event questions) were developed and used with assumptions and other approximate methods to compute rough estimates. The term *risk* in this study refers to conditional consequences (probability of the various events during the accident progressions multiplied by the consequences), given that core damage has occurred. Traditional risk estimates, computed by multiplying the conditional consequences and the frequency of the sequences leading up to core damage, could not be made at this time because the frequencies have yet to be determined in companion Level 1 and HRA studies. Uncertainty has been taken into account in a manner consistent with the detail of the abridged study.

This study investigated the possible accident progressions and the associated consequences of a single plant operating state, POS #6, mid-loop operation, where the reactor vessel head is on, the containment is closed but has no pressure retaining capability. The sister study at SNL investigated an early stage of a refueling operation. The scope of both studies is illustrated in Figure 1.1.

1.3 Methodology

The abridged process of computing conditional consequences is shown in Figure 1.2. In general, both the study reported here and the study done at SNL follow this scheme. Some differences in the details of the procedure exist and are noted at the end of Section 1.3. The process used here is an abbreviated form of the NUREG-1150 study.[1]

Accident Progression

The calculations begin with the assumption that core damage (CD) has occurred, making the consequences conditional. Given core damage, the reasonable accident progressions are delineated with the accident progression event tree (APET). Much of the delineation, particularly for the timing of key events, is based on deterministic calculations with a code used to compute source terms, such as MELCOR.[2] The likelihood of the various accident progression is reflected vis-a-vis branch point probabilities.

Branch point probabilities were assigned to reflect the likelihood of various pathways thought to exist. In large scale risk studies, the assignment can be done by groups of people knowledgeable of the severe accident issues. Here, because of resource limitations, most of all time, the assignments were done by the contractor staff. The probabilities are not as rigorous as they could be but this is one of a number of limitations of the study to be discussed. Some lack of rigor in determining the probabilities is taken into account by repeating the calculations with other possible probabilities; taken together, the repeated calculations as they were done constitute an uncertainty analysis.

Through the uncertainty analysis, distributions, instead of point values, were assigned to the branch points. The distributions are subjective but account for many possible values of the branch points. Point values are selected from the distributions with a form of Monte Carlo sampling known as Latin Hypercube Sampling (LHS).[3] After making sets of inputs, each set, consisting of point values, is assigned to the branch points and multiplied though to the ends of the APET. The calculations are repeated using the sets of inputs, building a probability distribution at the end of each pathway.

Source Terms

Having delineated accident progressions with the APET, the source terms of the progressions were calculated with a parametric code.[4] The parametric code is a mimic of the detailed source term codes; it is a collection of simple mass-balance equations, activated by the identifier representing the characteristics of the progression.

The parametric code determines source terms given the release bin and other inputs (typically various fractions, such as the inventory leaving the reactor vessel, involved in a core concrete interaction, entering the containment, and so on). Because these other variables are imprecisely known, many reasonable values can be assigned to the inputs. As in the APET calculations, distributions are assigned to the variables and sampled with LHS to form many sets of inputs values for repeated calculations. The result is a distribution of source terms for each accident progression pathway.

An internal "Source Term Advisory Group" was formed to support this study. The results of the accident progression and source term analysis were presented to and discussed with the advisory group in two meetings during the course of this analysis. The members of this group were W. T. Pratt and Hossein P. Nourbakhsh of BNL, and John Kelly and Dana Powers of SNL.

Consequences

Two sets of consequence measures were determined; an on-site dose rate (within the site boundary and designated as a *parking lot* dose rate), and off-site consequences, including early fatalities, population dose and latent cancers.

- The parking lot dose rate was computed using a recent model due to Ramsdell and a combination of the older Wilson and Regulatory Guide 1.145 models.

- Offsite consequences were computed using the MACCS code.[5]

To compute consequences a small number of source terms were randomly selected from the distributions of source terms generated with repeated use of the parametric source term code.

Uncertainty was not propagated through the consequences as it was through the APET and the source term calculations.

Conditional Risks

Conditional risk was computed by multiplying the consequences by their associated accident probability that was determined with the APET. This product of probability and consequences was computed for each accident progression pathway. The products of the pathways were summed. This process was repeated for each of the few samples of the source terms. Then, high, medium, and low results were reported.

Differences

This study differs slightly from its sister program at SNL in three ways. (1) Here, two hundred samples of uncertainty distributions were taken for source terms whereas in the SNL study one hundred samples were taken. (2) Here, nineteen samples from the source term distribution were used in consequence calculations and traced back through the APET for the probabilities needed to compute conditional risk whereas, in the SNL study, twelve samples were propagated through the APET to consequences. (3) Here, calculations of dose rates inside containment or the reactor building were not carried out since the releases were assumed to take place through the quipment hatch directly into the outside environment. At Grand Gulf, on the other hand, the release path is through the reactor building and the in-building dose rates were computed.

1.4 Strengths and Limitations of the Study

The study had strengths and limitations which are important to understand within the context of the calculations.

Limitations

- The subject of the study is one plant operating state (POS), mid-loop operation. This POS was selected for study because it was identified in a preliminary Level 1 study, known as a coarse screening analysis,[6] as potentially occurring at a relatively high frequency. Also, the POS had characteristics (e.g., reduced inventory) of interest to the staff in the Office of Nuclear Reactor Regulation at the NRC.

- The abridged study is based on the coarse screening analysis where accident sequences potentially having high frequencies, were identified. The consequences of these sequences were determined in the Level 2 and 3 abridged study reported here. The frequency is not incorporated into and merged with the Level 2 and 3 calculations to determine risk because the numerical value of the frequency estimate is believed to be approximate for such use.

- The simple APET accounts for a limited number of factors. The APET consisted of nine top event questions, compared to about seventy questions in a large scale PRA.

- The onsite dose estimates stem from simple equations yielding rough estimates.

- Variables were selected and assigned distributions for the uncertainty analysis by the contractor staff.

- Because of gaps in knowledge of the plant configuration and operator actions, assumptions were necessary. The assumptions are documented in the sections to follow. Some of the gaps will be filled with more rigorous determinations with results from detailed Level 1 and HRA studies during a follow-up Level 2 and 3 study.

Strengths

- Even with the limitations noted above, the abridged study is a systematic evaluation of severe accident progressions, accounting for some uncertainty.

- The source term analysis was reviewed by an internal advisory group.

- The contractor staff and the NRC project staff believe that the APET represents the occurrence of key events during accident progressions.

- The relationship and timings of accident progression events and factors have been determined to at least a first approximation.

- Estimates of both onsite and offsite conditional consequences were made.

The sections to follow document the abridged study of the Surry plant. The discussion above is expanded, providing important details and results.

Figure 1.1 Scope of the Current Risk Study

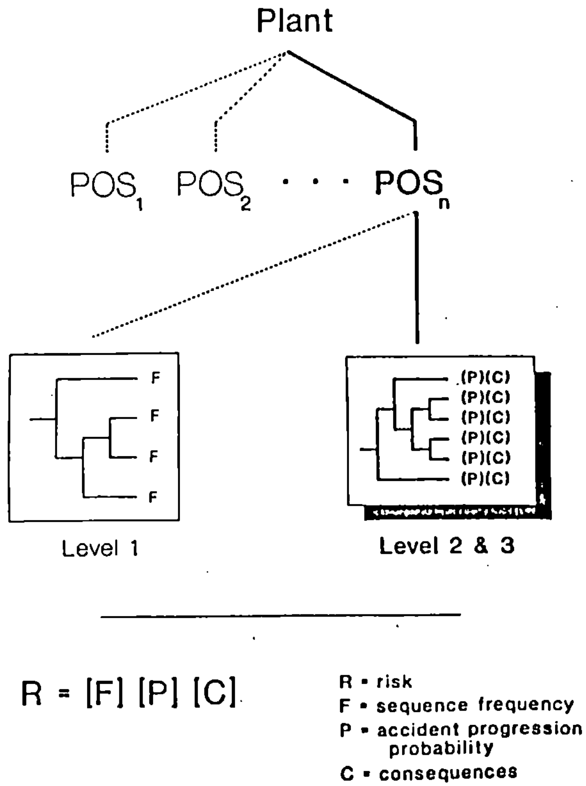
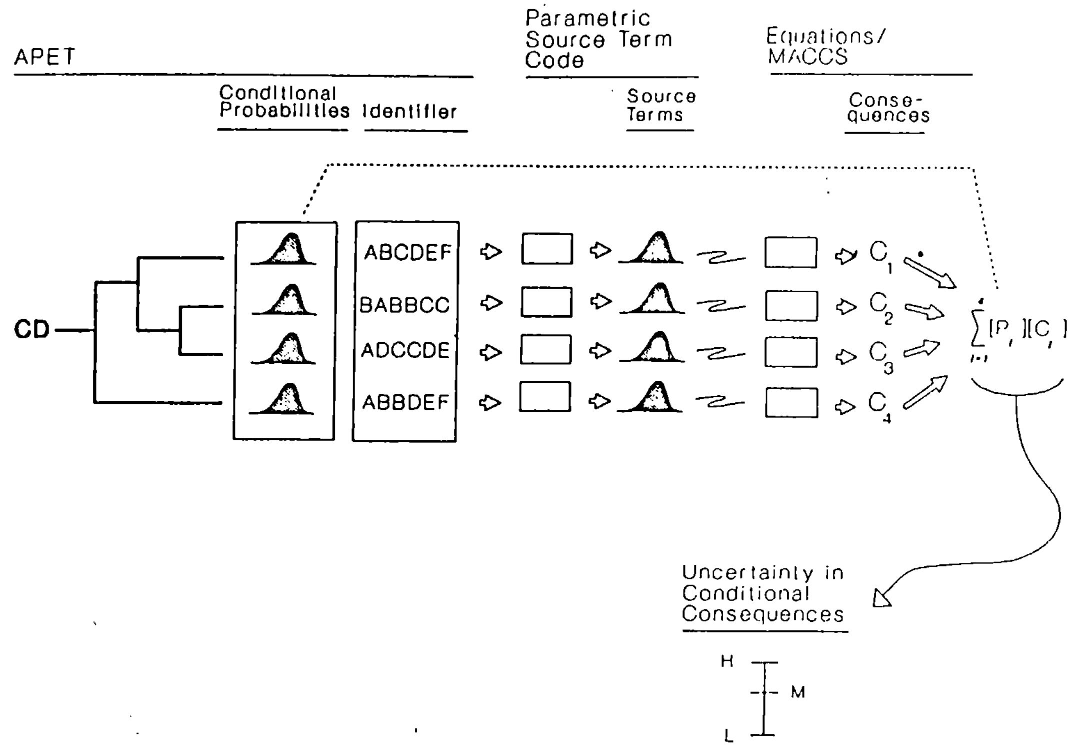


Figure 1.2 Abridged Process of Computing Conditional Consequences



2 Accident Progression Analysis

2.1 Approach

Following core damage in a severe accident, the accident progression is usually analyzed by using an Accident Progression Event Tree (APET). The APET treats the progression of an accident from the onset of core damage to the release of fission products, if any, or a successful termination of the accident. Quantification of the APET involves modelling of the physical process occurring in the vessel and containment during the various accident sequences, the availability and status of various safety equipment which could be used to mitigate the severity of the accident, and the assessment of the containment capability to retain the fission products when subjected to severe accident loads. In an APET, a series of questions are asked which represent these events and phenomena. Each path through the APET defines a unique accident progression path potentially giving rise to fission product release. The number of questions in a APET can vary depending on the details desired, and the number of relevant and important phenomena to be modelled.

To determine the extent of detail needed for the APET in the current study, extensive use was made of the results of the accident progression analysis for the Surry plant carried out for the NUREG-1150 program,[1] which was a PRA of the plant at full power. The results of the NUREG-1150 study show that the major cause of release was containment bypass followed by basemat melt-through. Early containment failure caused by various mechanisms and late containment failure resulting from gradual pressurization were either very small or negligible. This implies that the containment succeeds in retaining the fission products most of the time (except by very late basemat-meltthrough) once the containment boundary is closed. In other words, such phenomena as direct containment heating (DCH) or steam explosions were not found to be important contributors to the estimated containment failure probability and the eventual release of fission products. For accidents during low power and shutdown (LP/S) operation where the decay heat is significantly less and the reactor pressure is generally low, there are no particular reasons to believe that the containment performance would be any worse than for accidents occuring at full power.

However, as discussed in the next section, the containment does not have the capability to retain pressure during mid-loop operation at this plant. This implies that the containment is unable to contain the fission products and they will leak into the environment once they are released to the containment. This aspect simplifies the APET; since the integrity of containment is already lost at accident initiation, many questions normally needed to assess the potential for containment failure are no longer relevant. However, a number of important questions remain to be assessed in this configuration, namely the timing of recovery of core cooling capability and the availability of containment sprays.

POS 6 is characterized by relatively low decay heat levels due to the long time after shutdown that the plant enters this operating state. This low decay heat potentially increases the time available to take actions to recover core cooling capability before core uncovery. This longer time from shutdown to release also potentially reduces the fission product inventory available for release. Therefore, it is very important to determine the time of accident initiation relative to the time of shutdown. However, as shown in Table 2.1, the time to enter POS 6 after shutdown and the duration of POS 6 vary widely from one day to more than one month. Therefore, these times were selected as an uncertainty parameter to be varied in the sampling process. The Latin Hypercube Sampling method[3] was used for sampling. To determine the timing of key events in the accident progression such as core melt and vessel breach, several MELCOR calculations were performed with varying times from the time of shutdown to the time of accident initiation.

2.2 Plant Configuration

The plant configuration during the LP/S period can vary widely depending on the purpose of outage. Furthermore, a large degree of uncertainty exists for the operational state and availability of plant systems and components. Some of the examples are: number of loops isolated, size of RCS venting and availability of containment spray. For this abridged analysis, it was assumed that all the loops were isolated and the safety valves were removed for maintenance which provides a vent path from the RCS to the containment. For a more detailed analysis, these parameters may be handled as uncertainty parameters. To do that, more information will have to be available for the distribution of these parameters.

The two most important factors for determining containment response during an accident in POS 6 are the status of containment integrity and availability of sprays. Sprays are important because they are the major containment cooling system during severe accidents and can reduce the source terms by scrubbing. There is no requirement under the existing plan technical specifications to have any of the containment sprays available once the plant enters the RHR entry condition at Surry.[7] Consequently, it is possible that all of the containment sprays could be out of service and therefore would not be are available during mid-loop operation. Therefore, the spray availability was used as one of the uncertainty parameters in this study.

As a result of several discussions with the Surry personnel, it was determined that while the containment is "closed" during the mid-loop operation at Surry, closure does not ensure that the containment can contain the pressure which could be generated during the course of a severe accident and prevent release of fission products.[7] This is due primarily to the presence of a temporary restrining plug, in place of the escape tunnel, in the containment equipment hatch. This temporary plug has no overpressure capability. Therefore, the containment was assumed to leak during POS 6 for this study.

2.3 Level 1 Sequence Description

A preliminary screening analysis of the systems reliability and a characterization of the accident sequences leading to core damage for the internally initiated events were earlier performed by BNL for the Surry Unit 1 plant.[6] The major objectives of this screening analysis were to provide initial insights into any particularly vulnerable plant operational states (POSs) during low power/shutdown operations and to identify the set of major initiating events applicable to each POS. Based on this coarse screening analysis, it was determined that POS6, mid-loop operation is likely to be one of the most vulnerable plant conditions, mainly due to the reduced inventory in the RCS. The dominant causes of accidents during POS 6 are loss of RHR and loss of off-site power. Loss of RHR accident sequences occur largely due to operator errors such as over-draining, failure to maintain the level in the RCS, or failure to recognize loss of RHR accidents. Operating experience at nuclear power plants indicate a relatively high incidence of loss of RHR. For this category of accidents, the recovery probability is largely determined by the human reliability analysis (HRA). Since this HRA has a large band of uncertainty, it was also included as a uncertainty parameter. For those accidents initiated by loss-of-power, recovery from loss of power determines the probability of recovering the core cooling capability, and termination of the accident.

2.4 Event Tree Analysis

A relatively simple APET was used in this analysis to describe events in the vessel and the containment responses subsequent to core damage.

Figure 2.1 shows the containment event tree used in this analysis. The first set of questions refer to the status of containment. In this particular POS, the containment is assumed to be leaking from accident initiation. Once the status of the containment is identified, the next question asked is the timing of core cooling recovery, which determines the extent of core damage. Arrest of core degradation before failure of the vessel during a severe accident has the potential to significantly decrease the magnitude of fission product release. Therefore, the timing of recovery of core cooling capability was divided into five periods; Very early, Early, Intermediate, Late and Never (no recovery). The timing of 'Very early' extends to the point where core cooling is recovered without any core damage. 'Early' is recovery of cooling during the relatively short period after the cladding rupture of the fuel rods, but before significant core melting. 'Intermediate' is the period in which the recovery of core cooling will arrest the progress of core melt without leading to vessel breach. As a result of consultation with the source term expert panel, this intermediate period was assumed to extend until 45% core melting occurred. If core cooling is recovered during the 'Late' period (which, in this study, is defined to be more than 45% of the core melted), the vessel is assumed to be breached by the core debris. 'Never' indicates no core cooling recovery at all. Table 2.2 shows the timing of core melt progression as calculated by the MELCOR code, which was used to determine the time period available for recovery. The MELCOR calculations were performed assuming that the time of accident initiation was 24 hours after shutdown. However, as discussed earlier, this time can vary widely. Therefore, in this study, the time of accident initiation was determined by sampling from the joint distributions of the time to enter the mid-loop operation and the duration of POS 6 for each observation. Actual Surry data, which were collected for the screening level 1 analysis [4], were used to determine the distribution of these times. For the distribution of the time of accident initiation, the MELCOR-calculated timing of the core melt progression was adjusted by the decay heat to determine the time available for recovery of core cooling. The recovery probability was estimated based on the HRA recovery curve for human error,[8] the off-site power recovery curve[6] and hardware availability for each of the time periods. The hardware availability was based on the data used in the screening level 1 study.

The next questions address spray availability and whether the cavity is dry or wet, which determines the extent of core-concrete interaction (CCI). The spray availability was included as an uncertainty parameter. The outcomes of the accident sequences in the APET were classified into eight bins depending on the extent of core damage, vessel breach and spray availability as shown in Fig. 2.1.

The basic structure of the APET shown in the Figure 2.1 is sufficiently general to be applied to other POSs that occur during low power and shutdown operation. However, since it was determined that the containment during POS 6 at Surry had no pressure holding capability, the branches related to 'Closed' and 'Open' containment were not developed further in this study.

This APET was applied to each of the major cutsets leading to core damage sequences identified in the preliminary screening level 1 study. In the screening level 1 analysis, the core damage was defined to have occurred when the coolant level is decreased to the top of active fuel. However, as discussed above, the accident can still be terminated without core damage if the core cooling is recovered during the 'Very early' period. There is one possible exception to this, during the 'Very Early-Early' periods when cooling water is recovered. If the clad becomes embrittled on heat up it could fracture on quenching, releasing the gap inventory. Water could enter the ruptured fuel rods and leach out iodine from the fuel. Depending on temperature and solubility limits, the iodine would be partitioned between the water and the containment atmosphere. While this accident scenario would not be important for off-site consequences, it could have significant on-site implicatons. Due to the limited time available, quantification of the releases was not possible. In estimating the final risks conditional on core damage, only those accident sequences which were actually predicted to result in core damage were included; namely, those accident sequences which were terminated in the 'Very early' period were not included in the calculations for determining conditional risk. Figure 2.2 compares the conditional probability of core damage arrest before vessel breach for the LP/SD analysis compared with the full power analysis of NUREG-1150 at Surry. Figure 2.2 indicates that the vessel is not breached approximately half of the time given core damage for both low power and full power accidents.

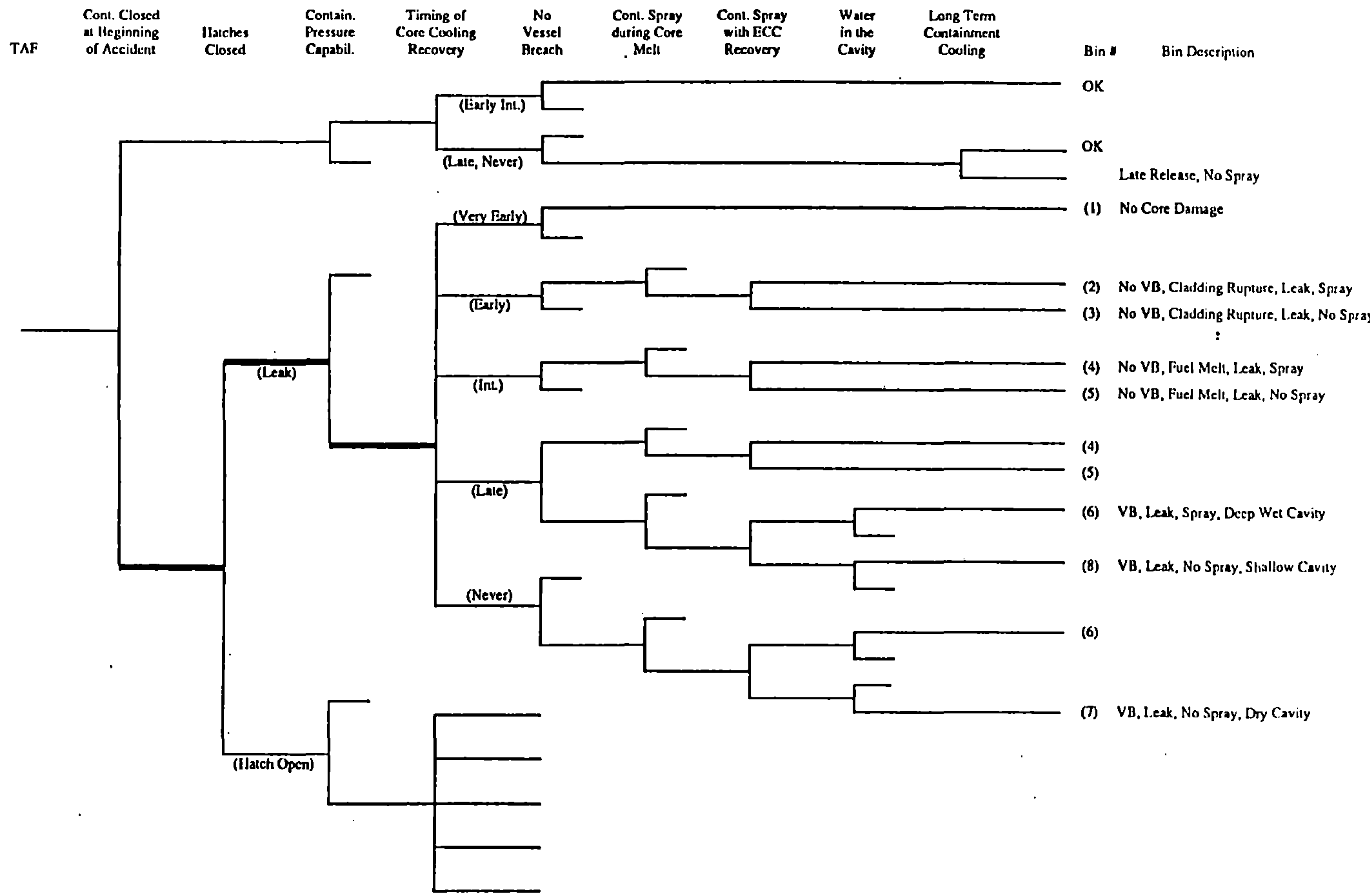


Figure 2.1 Accident Progression Event Tree for the Abridged Low Power/Shutdown Risk Analysis

Figure 2.2 Fraction of Core Damage Sequences which are Terminated Without Vessel Breach

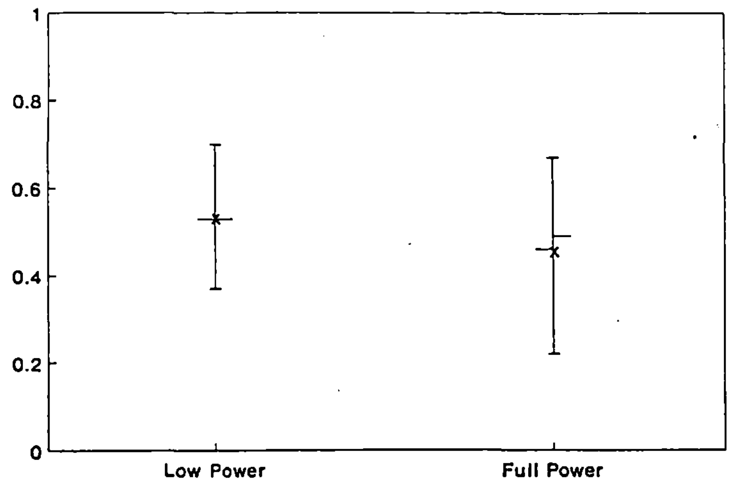


Table 2.1 **ACCIDENT TIMING**

(MELCOR CALCULATION WITH 24 HOURS FROM SHUTDOWN)

Core Uncovery:	**~90 minutes**
Cladding Rupture:	**~200 minutes**
30% Melt:	**~240 minutes**
60% Melt:	**~300 minutes**
Vessel Breach:	**~350 minutes**

3 Source Term Analysis

3.1 Approach

Early in the project it was decided that the source terms (STs) used in this abridged low power/shutdown (LP/SD) PRA should address uncertainty. Partly due to the accelerated nature of this project, it was also decided that wherever possible the NUREG-1150 distributions for ST definition would be used to calculate the LP/SD source terms. The parametric ST code, SURSOR,[4] that was developed in NUREG-1150 for Surry, was therefore used as the basis for ST definition in the present study.

Two additional efforts were taken to assure the adequacy of the source terms: The first involved comparing the calculational results from MELCOR for LP/SD with the data used in SURSOR (as well as the calculational results obtained from SURSOR); and the second involved the establishment of a Source Term Advisory Group to provide guidance, and additional information if necessary, on the modification of the SURSOR code for the present LP/SD study.

3.2 Description of Parametric Model

As discussed above, the SURSOR code, together with its associated distributions provided in NUREG-1150, was selected as the basis for ST definition in the present study. This section provides a brief discussion of the SURSOR code, its evaluation (for modification if required), and the final parametric model used in the present LP/SD study.

SURSOR is a parametric computer code used in NUREG-1150 to predict source terms for full power operation. The parameters in SURSOR were defined in NUREG-1150 by expert elicitation. A distribution, instead of a single value, was assigned to each parameter to address ST uncertainty. The Source Term Advisory Group, based on a consideration of the differences between full power and LP/SD operations, identified two parameters in SURSOR as important and possibly different (than the values used in NUREG-1150) for LP/SD source term definition. The first parameter is FCOR, which defines the fraction of the radionuclide in the core released to the vessel before vessel breach (VB), and the second parameter is FVES, which defines the fraction of the radionuclide released to the vessel that is subsequently released to the containment. The distributions of these two parameters (as defined in NUREG-1150) were compared with results from MELCOR calculations to establish their values to be used in the present study.

In SURSOR, a source term is uniquely defined by the Accident Progression Bin (APB). Eleven characteristics are used to define an APB. Table 3.2.1 shows the APBs and the attribute assigned to each of the 11 characteristics for the LP/SD APBs derived in Section 2 that would cause significant offsite FP releases. Because the containment does not have a pressure retaining capability during a LP/SD operation, even if it is closed, the two characteristics related to containment conditions, the time and size of containment failure, are assigned attributes "early" and "leak," respectively. The mode of vessel breach is assigned either as "no VB" or as a bottom head failure according to the definition of the APBs developed in Section 2. The containment spray condition is also assigned a value according to APB definition, and the mode of core-concrete interaction (CCI) is assigned a value based on core injection and containment spray recovery conditions (and thus the water available to the corium) defined in the APBs. The attributes of other characteristics are either defined based on LP/SD conditions or are not important for LP/SD conditions.

SURSOR was used to predict fission product (FP) release fractions for the five LP/SD APBs (APB-4 through APB-8) presented in Table 3.1. Two hundred sets (or observations) of release fractions were produced for each of the five bins to address ST uncertainty. Figures 3.1(a) through (d) present the ranges (5 percentile to 95 percentile) of the release fractions of the nine radionuclide categories for APBs 4 through 7, respectively. Also presented in these figures are the median (50 percentile) and mean values of the release fractions from the 200 observations, and the calculational results from the MELCOR cases that are related to the individual APBs. Figures 3.2(a) and 3.2(b) present the distributions (the range and the median value) of FCOR and FVES used in SURSOR, and the calculated values from MELCOR for three cases. As shown in Figure 3.2(a), the difference between the MELCOR calculated values for the two cases with different accident initiation time (24 and 72 hours after reactor shutdown, MELCOR Base Case and MELCOR Case 6) is not significant, and the values predicted for a core recovery case (MELCOR Case 3) are less than those predicted for the other two MELCOR cases that proceed to vessel breach. Certainly, the release fractions for the core recovery case would depend on the time of recovery.

As shown in the above figures, the MELCOR calculated values in general fall within the ranges of SURSOR predictions. Although for some radionuclide categories the MELCOR calculated values are closer to the upper ranges of the SURSOR predictions, they can be attributed to ST uncertainties, and there are no apparent phenomenological reasons that call for the modification of the SURSOR distributions. Consequently, the Source Term Advisory Group did not recommend any change to the SURSOR code for ST predictions in this abridged study for Surry.

3.3 Results

To limit the number of MACCS calculations, and at the same time to provide a range of uncertainty, 19 source terms were (randomly) selected (from the 200 STs, using the LHS sampling method) for each of the five LP/SD bins for MACCS calculations.* This, when combined with the two time parameters defined in Section 2 (associated with drained maintenance and refueling), provides 38 source terms for each bin for MACCS calculation.† Figure 3.3 shows the ranges, median and mean values, for the release fractions of all the source terms selected for MACCS calculations. Also presented in this figure, for comparison, are the values calculated by MELCOR for a core recovery case (MELCOR Case 3) and a vessel breach case (MELCOR Base Case).

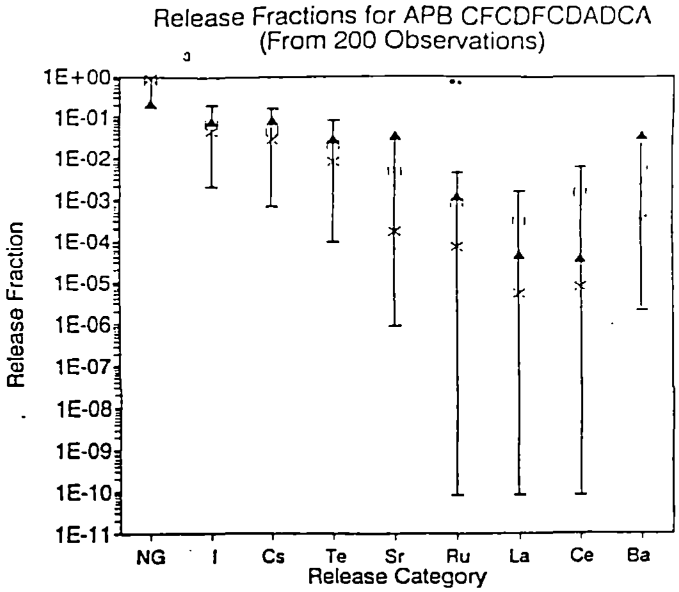
In addition to release fractions, a complete description of a source term also requires the specification of the timing, energy, and height of the release. The timing of the release affects both the radioactive decay of the inventory and the warning time for offsite emergency response (e.g., evacuation). Table 3.2 presents the mean values of the release fractions for the nine radionuclide categories, the release time (i.e., the time when FP release begins), and the release duration. Both

* This is the minimum number of STs to provide a 5% to 95% range. However, because of the low confidence level associated with the values obtained from such a small sample, they are simply referred to as the upper and lower limits of the calculations, and with no percentiles associated with them.

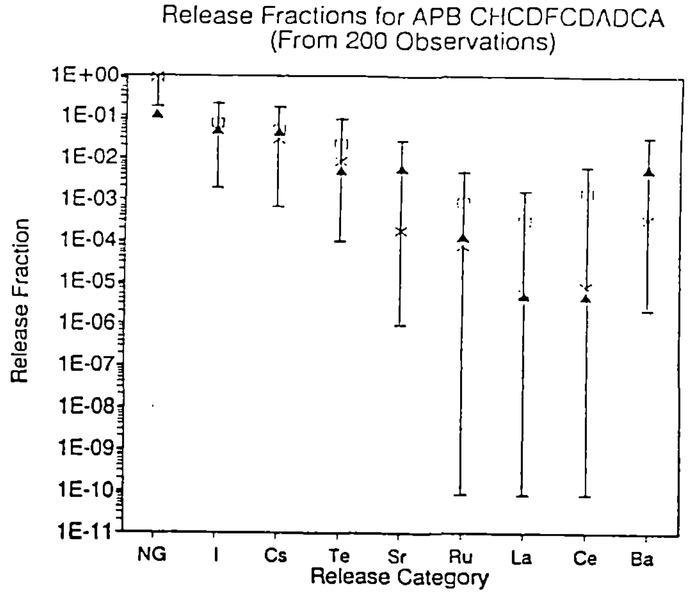
† Source terms for APB-4 through APB-8 are provided in this section for later MACCS calculations. Source terms for APB-1 through APB-3 are not provided because they are not expected to cause significant offsite consequences (Due to early recovery, there is no core damage for APB-1 and only cladding damage for APB-2 and APB-3.). It is also noted that the STs for APB-4 and APB-5 obtained in this section are the same (and thus they are combined in later MACCS calculations). This is because the containment spray, which is assumed to be recoverred after the recovery of core injection (and thus in-vessel FP releases), has negligible effect on FP releases.

the release times and the release durations presented in Table 3.2 are obtained from MELCOR calculations. Since the release time presented in Table 3.2 is measured from accident initiation, it is not suitable for the calculation of radioactivity decay after reactor shutdown, but can be used to determine the timing for emergency activities. The warning time for offsite emergency response is the time between the notification to the public, and the time of FP release (in the present study the warning time is assumed to be 60 minutes after accident initiation). The energy of the release is assumed to be 1.0E6 watts, which is a value between the high and low values used in NUREG-1150 for similar containment failure conditions, and the height of FP release is assumed to be at ground level for all bins.

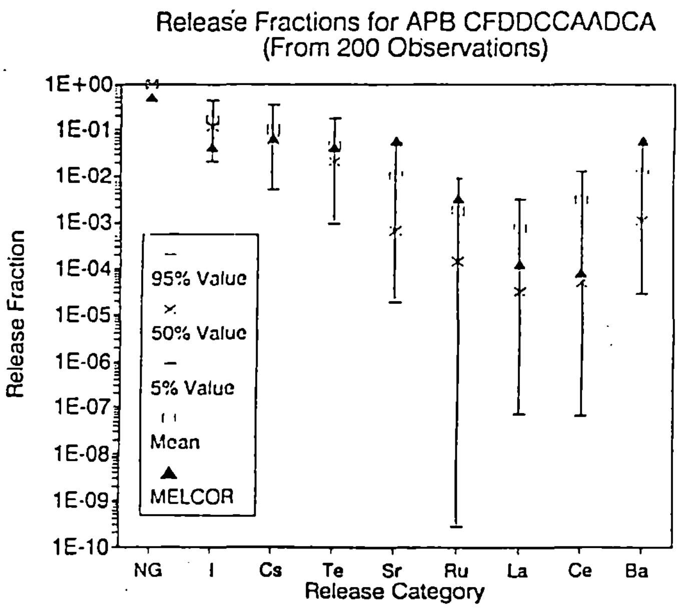
One of the most important parameters in the LP/SD source term definition, and which is not considered in a full power analysis, is the time of accident initiation from reactor shutdown. This parameter determines the inventory available for release at accident initiation. Because of its importance, it is treated as one of the uncertainty parameters in the present study (see Section 2). A randomly selected value is assigned to each source term defined in this section. Figure 3.3 shows the ranges, median, and mean values of this parameter used in the present study for both drained maintenance and refueling.



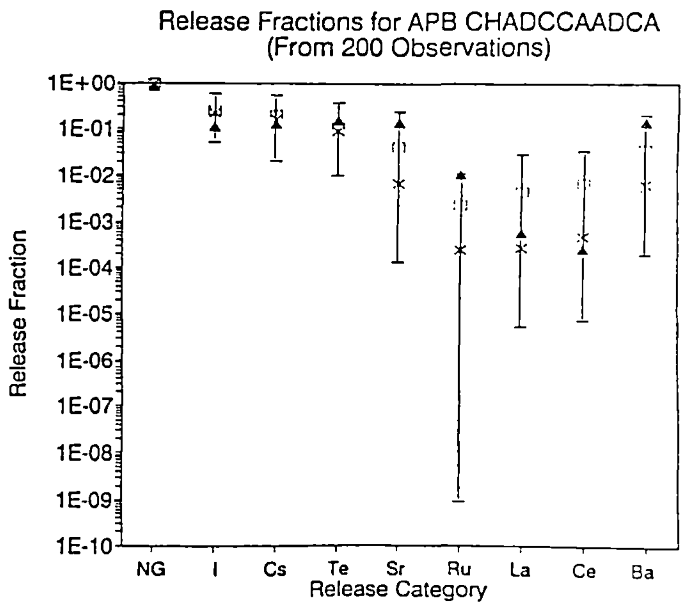
(a) APB 4



(b) APB 5

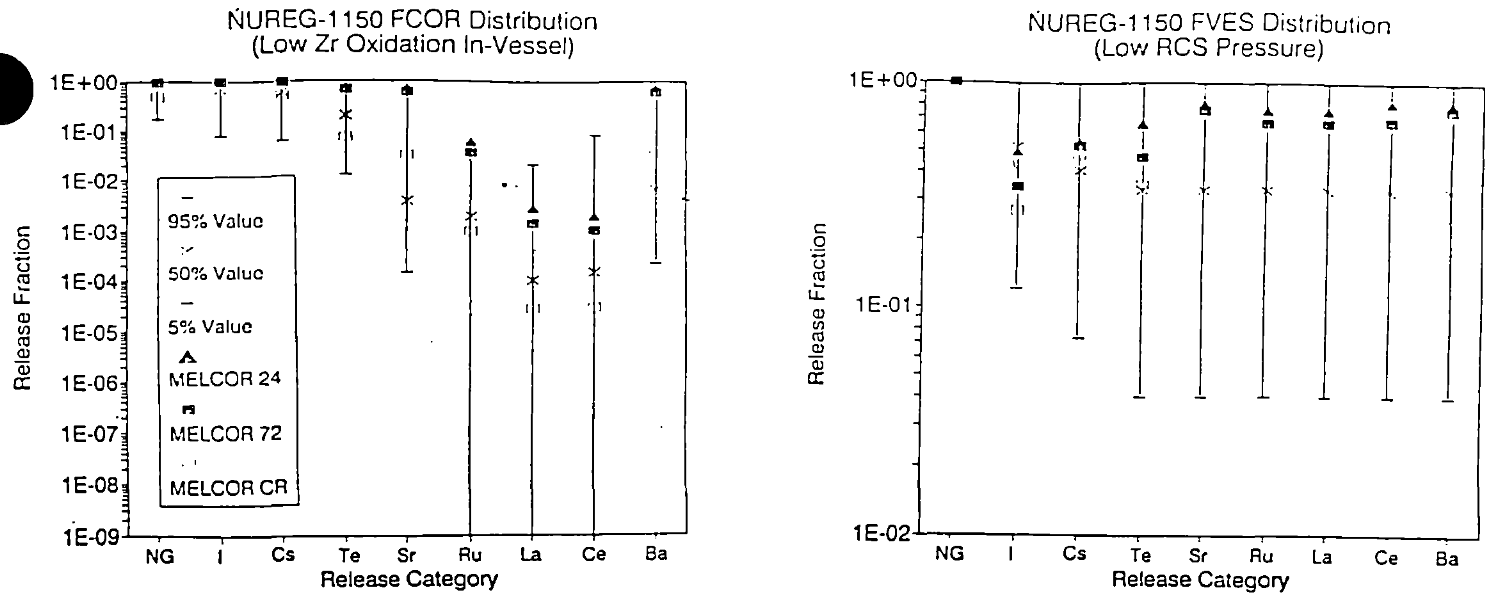


(c) APB 6



(b) APB 7

Figure 3.1 Release Fractions Calculated by SURSOR (95%, 50%, 5%, and Mean Values) and MELCOR



(a) FCOR **(b) FVES**

Figure 3.2. Comparison of Values of FCOR and FVES Used in SURSOR and Calculated by MELCOR

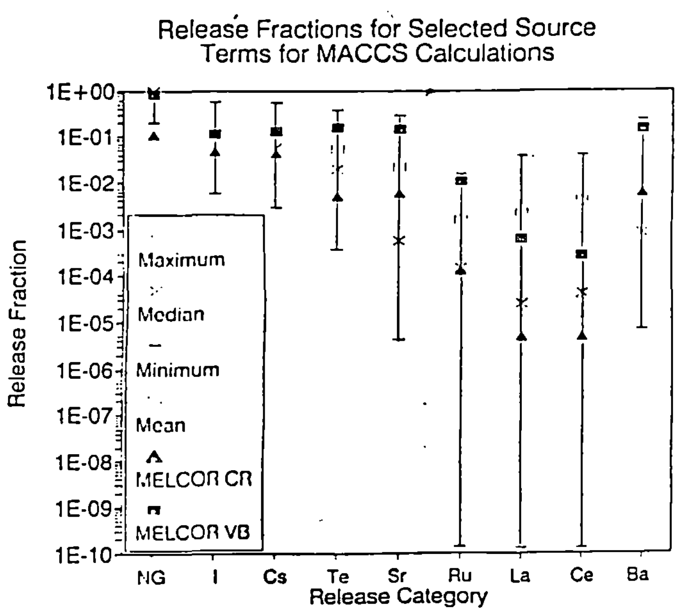


Figure 3.3 Comparison of the Release Fractions Used for MACCS Calculation (Maximum, Median, Minimum, and Mean) and Calculated by MELCOR

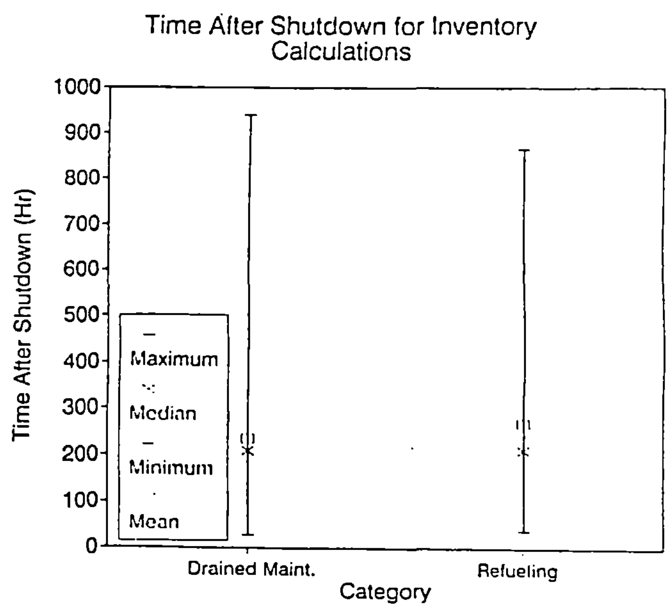


Figure 3.3.2 The Time of Accident Initiation After Reactor Shutdown for Radioactivity Inventory Calculation

Table 3.1 Low Power/Shutdown APBs for Source Term Calculations

APB #	APB ID(1)	1 CF Time	2 Spray	3 CCI Mode	4 RCS Pres	5 VB Mode	6 SGTP	7 CCI Size	8 Zr Oxide	9 HPME	10 CF Size	11 RCS Hol
4	CFCDFCDADCA	Early	Lt-to-VL	No	Low	No	No	No	Low	No	Leak	One
5	CHCDFCDADCA	Early	No	No	Low	No	No	No	Low	No	Leak	One
6	CFDDCCAADCA	Early	Lt-to-VL	Prompt Deep	Low	BtmHd	No	Large	Low	No	Leak	One
7	CHADCCAADCA	Early	No	Prompt Dry	Low	BtmHd	No	Large	Low	No	Leak	One
8	CHBDCCAADCA	Early	No	Prompt Shlw	Low	BtmHd	No	Large	Low	No	Leak	One

Note: (1) According to APB identification used in NUREG-1150.

Table 3.2 Mean Release Fractions and Timing of Release

Case No.	Mean Release Fraction									Timing of Release (Minutes)	
	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba	Release Time	Duration
4	0.702	0.064	0.047	0.019	5.89E-03	8.25E-04	3.08E-04	1.31E-03	6.12E-03	190	50
5	0.702	0.064	0.047	0.019	5.89E-03	8.25E-04	3.08E-04	1.31E-03	6.12E-03	190	50
6	1.000	0.149	0.096	0.041	1.31E-02	1.65E-03	7.76E-04	2.78E-03	1.33E-02	190	120
7	1.000	0.228	0.184	0.108	5.80E-02	2.33E-03	6.40E-03	8.60E-03	5.17E-02	190	400
8	1.000	0.182	0.127	0.072	2.53E-02	1.84E-03	2.49E-03	4.59E-03	2.35E-02	190	400

4 Consequence Analysis

4.1 On-site Consequences

4.1.1 Parking Lot Dose Rate Calculation (PLDR)

The parking lot dose rate (PLDR) has been calculated using two different models for the building wake centerline concentration. The total PLDR is calculated as a sum of the inhalation and cloud exposure dose rates based on the radionuclide concentration in the wake region of a building. A brief overview of the approach is given below.

The dose rate is calculated as a sum of the cloud inhalation dose rate, D_i^{INH}, and the cloudshine dose rate, D_i^{CLOUD} (based on the 60 radionuclides in the MACCS dosimetry routine):

$$D = \sum_{i=1}^{i=60} \left[D_i^{inh} + D_i^{cloud} \right],$$

where

$$D_i^{INH} = DFI_i \beta \frac{r_i I_i}{\tau} [...] ,$$

$$D_i^{CLOUD} = DFC_{\infty i} \frac{r_i I_i}{\tau} [...] \, Sv/s ,$$

DFI_i - inhalation dose conversion factor, Sv/Bq;

$DFC_{\infty i}$, $\frac{Sv \times m^3}{Bq \times s}$, is the semi-infinite cloud conversion factor for nuclide i;

β - breathing rate, m^3/s. In this calculations, the breathing rate $\beta = 0.000266\ m^3/s$ following the MACCS code default value;

r_i - fraction of nuclide's i inventory released over the puff duration, τ;

I_i - total inventory of nuclide i, B_q.

In the correlations shown above, [...] denotes an expression specific to a particular correlation for predicting the average concentration in a building wake (Ramsdell,[9] Wilson,[10] Reg. Guide 1.145.[11]

Centerline Concentration Calculation

The PLDR was estimated using the following wake centerline concentration models: Ramsdell,[9] Wilson,[10] and Reg. Guide 1.145.[11] Brief descriptions of each model follow.

Ramsdell Model

The Ramsdell model[9] calculates the concentration in the far-region of the wake by including the effects of the lateral and vertical diffusion due to background turbulence:

$$[...] = \frac{1}{\pi U\left[\sigma_y^2 + (KA/a^2U^2)F(T_s)\right]^{1/2} \times \left[\sigma_z^2 + (KA/a^2U^2S^2)F(T_{sv})\right]}$$

with the assumption that K = 0.5.

In the correlation above,

$$F(T) = 1 - [1 + x/(UT)]\exp[-x/UT)]\,,$$

where $T = T_s$ or T_{sv}; $T_s = A^{1/2}/u^*$, sec;

$T_{sv} = T_s$ for extremely unstable weather (Class A, Pasquill-Gifford), and $T_{sv} = T_s/2.5$ for extremely stable conditions (class G);

$S = 1$ for extremely unstable weather (Class A, Pasquill-Gifford), and $S = 2.5$ for extremely stable conditions (class G);

$u^* = aU$, friction velocity, m/s; $a = 0.4/ln(z/z_0)$; U is the average wind speed at $z = 10$ meters, m/s, and surface roughness length $z_0 = 0.1$ m; based on this, $a = 0.0869$, A - building area, m^2; σ_y and

σ_z - diffusion coefficients due to the background turbulence.

Wilson Model

The Wilson model suggests a correlation for calculating the lower limit on the dilution in the wake (which corresponds to maximum concentration in the wake):

$$D_{\min} \approx 0.11 K_e \frac{x^2}{A} ;$$

this leads to the following expression for $\frac{\chi}{Q}$:

$$\left(\frac{\chi}{Q}\right)_{\max} \equiv [...] = \frac{1}{0.11} \frac{1}{Ux^2} .$$

As recommended in Ref. 10 a multiplier of 5.0 was used for the ground level release calculations (elevation lower than 0.2 H, where H is the height of the building).

NRC Reg. Guide 1.145 Model

Equation (2) of the Reg. 1.145 model[11] is used for calculating $\frac{\chi}{Q}$:
where $\sigma_y = ax^b$ and $\sigma_z = cx^d$,
x is the distance from the source, m, and
the dispersion constants a = 0.0722, b = 0.9031, c = 0.2, and d = 0.602 for stable weather, Pasquill-Gifford Class F.

Calculation Assumptions

The scoping calculations were performed with the following set of input parameters corresponding to the Surry building and site:
distance from source, x = 10 to 300 meters,
building projected area, $A = 1500\ m^2$,
and, finally, wind speed at 10 m elevation, $U = 1.2\ m/s$.

The wind speed above was obtained by an arithmetic averaging of the wind speeds observed at the Surry site during the most stable weather conditions (Class F stability).

Results

The bounding calculations were performed for three source terms referred to as "High," "Medium," and "Low (Gap release)". The Wilson/Reg. Guide 1.145 labelled box in Fig.4.1 are based on using the Reg. Guide 1.145 prediction limited from above by the values predicted by the Wilson model.

The results for the dose rate expressed in Rem/h shown in Figures 4.1 indicate a variation in dose rate of about two orders of magnitude as a function of the source term. The "parking lot" dose rates are high and are likely to lead to non-stochastic health effects for exposed workers. In view of the relatively large number of on-site personnel during shutdown operations, these dose rates outside containment suggest careful examination of on-siteevacuation schemes to limit consequences.

4.2 Off-site Consequences

MACCS calculations of the off-site consequences have been performed for the 152 POS 6 source terms generated by using the LHS sampling[3] of the SURSOR calculation results. There were a total of nineteen sample groups (one for Drained Maintenance and one for Refueling outages) each containing four distinct sets of release fractions for the nine radionuclide groups represented in the MACCS calculations. The calculations were run using the original MACCS input files used for the NUREG-1150 Surry-1 calculations.[1] The only file which had to be changed in order to accommodate the specifics of the shutdown study's source terms was the ATMOS input file. The information on all MACCS input parameters and assumptions can be found in the NUREG-1150 report on Surry-1.[1]

The time of release for each group was determined using the LHS technique. Based on this time, the initial inventory for each source term was calculated by using a logarithmic interpolation between the two closest data points. The Surry-1 inventories for various times after shutdown were taken from.[12]

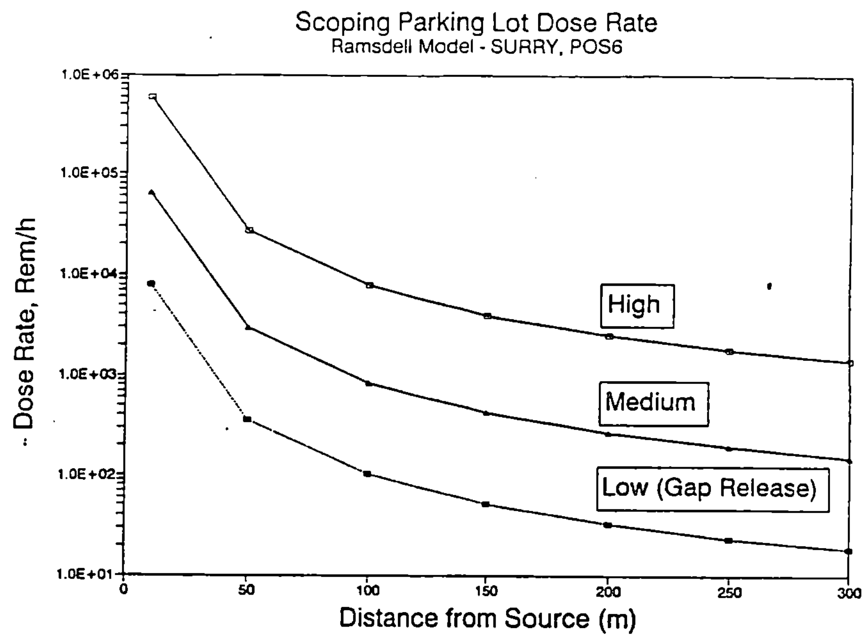
The following additional information/assumptions were used:

1. Release power: 1.0 MW (sensitivity calculations with 0.0 MW).
2. Release elevation: 28' (8.54 m).
3. Warning time: 130 minutes (130 min = 190 min - 60 min).

Figures 4.1 and 4.2 show the results for the early and latent fatalities predicted by MACCS. Bins 5 through 8 contain thirty eight data points each. The median value is shown only for Bin 7; zero values were predicted for the remaining bins.

The results of the calculations indicate that the highest number of early fatalities was predicted for Bin 7. Note that the number of early fatalities is very small (less than 10^{-2}) even for the most severe accidents involving the vessel bottom head failure.

Figure 4.1 On-Site Parking Lot Dose Rate



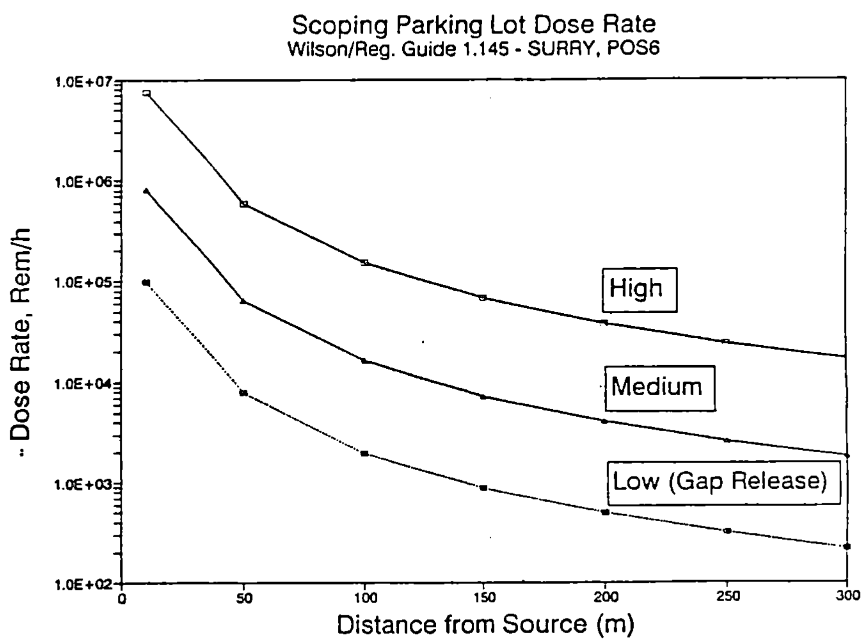
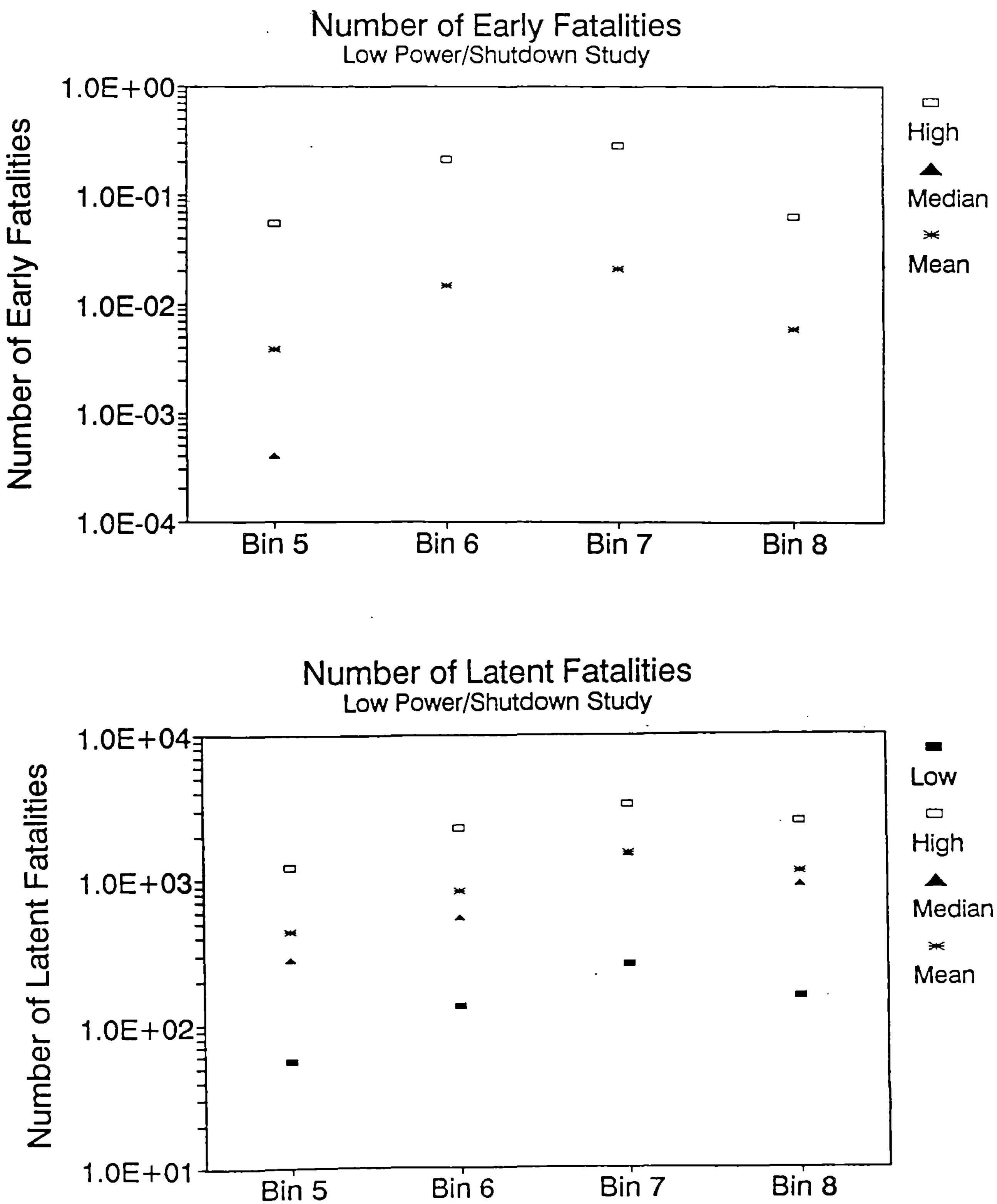


Figure 4.2 Off-Site Consequences Calculation by MACCS



5 Integrated Risks Conditional on Core Damage

Once the consequences are calculated for each of the release bins, risks are evaluated by combining the accident progression analysis, source term analysis and consequences. Uncertainty in risk is determined by assigning distributions to important variables, generating samples from these variables, and propagating each observation of the sample through the entire analysis. If the core damage frequencies of the PDS had been available from the level 1 analysis, absolute integrated risks could have been calculated for this particular POS. However, since the frequencies of the core damage accidents are not available for this study, the risk were calculated as conditional on core damage; i.e., the results presented are averaged over various accident progressions, given core damage.

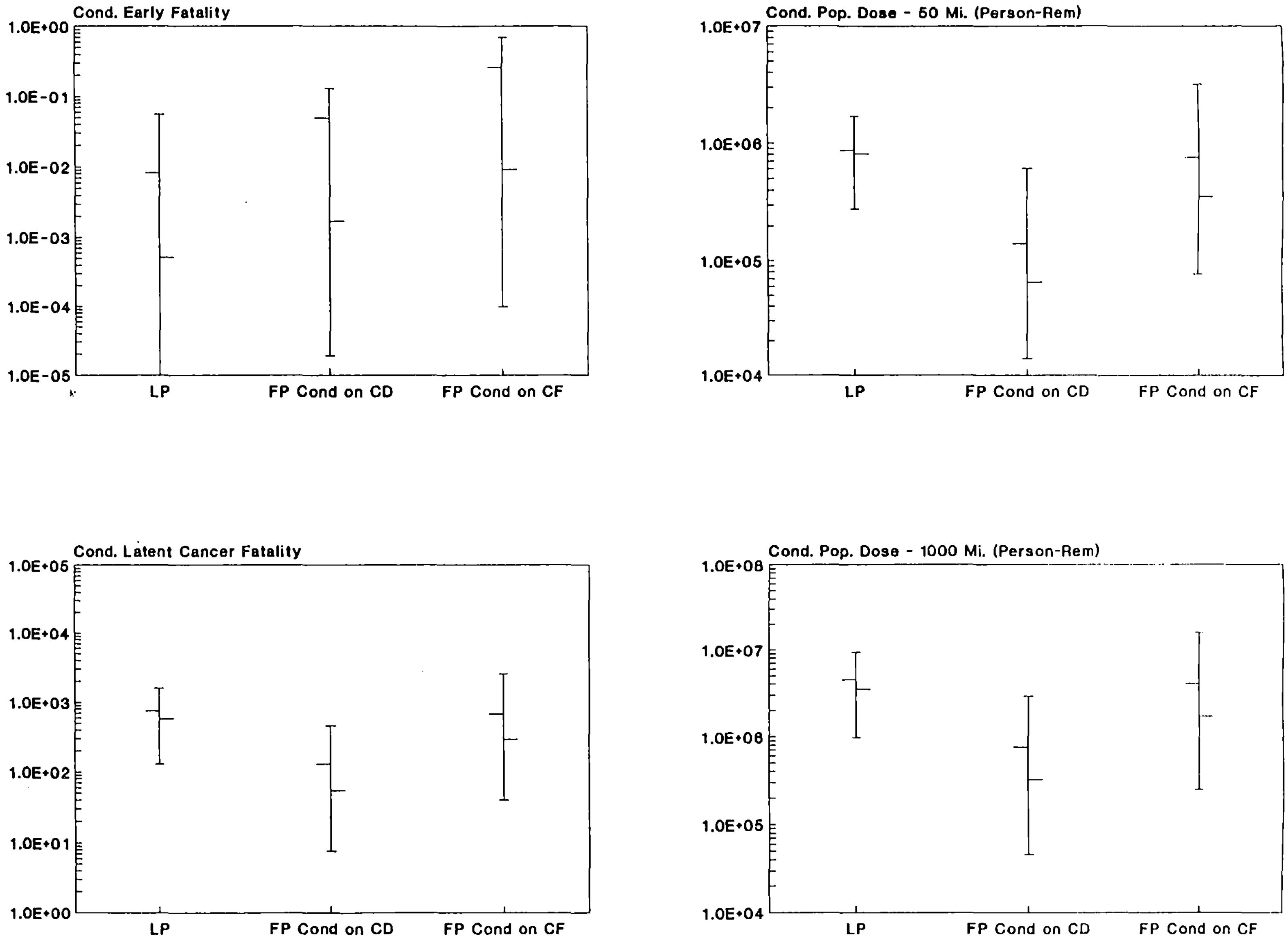
Figure 5-1 shows ranges of the four risk measures (conditional on core damage) which were calculated for the POS 6 at Surry. The risk measures presented are the early fatalities, late cancer fatalities, the population dose at 50 miles, and the dose at 1000 miles. The upper and lower bounds shown in the figures do not represent any particular statistical measures, since the number of samples was not sufficiently large to attach any statistical significance to these ranges. However, if a sufficiently large number of samples were used, these bounds are expected to asymptotically approach the 5th and 95th percentiles. Also shown in the figures for comparison are results of the same risk measures for the full power operation at Surry from the NUREG-1150 study. The NUREG-1150 results shown were converted to risks conditional on core damage and conditional on containment failure for ease of comparison.

The risk comparison shows that the early fatality risk from low power operation during POS 6 is considerably less than that of the full power operation (conditional either on core damage or on containment failure). This result is expected since the fission products have had a long time to decay and the species which have the greatest influence on the early fatalities generally have shorter half lives.

The figures also show that the latent cancer fatalities and population doses are higher than those predicted for the full power accidents conditional on core damage. However, these long term health effects are about the same for accidents conditional on containment failure. This is due to the fact that these risk measures are more affected by slow-decaying species and the longer decay time has less impact on these species. Therefore, the risks are similar once containment is failed. Since the containment is assumed to be essentially open during POS 6 of shutdown, the off-site risk of latent health effects averaged over core damage sequences is higher for POS 6 than for full power operation.

It is emphasized here again that these comparisons are conditional on core damage or containment failure, i.e., assuming the same core damage frequencies or the same containment failure probability. However, the real risk profile is determined by the product of these conditional risks with the frequencies of occurrence of the conditions giving rise to the risk. If the frequencies of core damage accidents are significantly at losw power/shutdown from those at full power, the integrated risk profiles will be dominated by those (Level 1) frequencies.

Figure 5.1 **Comparison of Risks Conditional on Core Damage**



• FP results based on NUREG-1150.

6 Insights and Conclusions

The abridged risk study, while preliminary and subject to confirmation in a number of areas needing more detailed analyses, has, nevertheless, shown that during shutdown a severe release with conditional long-term consequences approaching those of full power operation can occur. In the id-loop operation, POS 6, where the RCS inventory is less than half of full inventory, the loss of RHR can proceed rather quickly to core uncovery in less than 2 hours if corrective actions are not (or cannot be) taken. The progression of the accident beyond core uncovery and its possible mitigation depends on a number of factors. These include the timing of the recovery of core cooling, and the availability of containment sprays. In POS 6, the isolation of containment in the sense of achieving a pressure holding capability is judged to be not possible within the time frame of interest. Thus the containment is expected to leak right from the start of the release. This possibility could have significant implications for on-site habitability and, in particular, for the ability to successfully undertake necessary corrective actions.

The defense-in-depth philosophy of U.S. nuclear power plants traditionally considers three barriers to the release of fission products into the environment; the fuel pellet itself, the cladding, and the containment. During shutdown operation and especially in the mid-loop condition, no credit can be assigned to the containment as a barrier. Thus, unlike the full power case at Surry where the containment is expected to retain the fission products in over 90 percent of the releases, defense-in-depth at shutdown could be negated by the intrinsic operational condition of the plant. In this case, the only possible mitigation (apart from containment sprays whose availability is also in question) is provided by the natural decay of the radionuclide inventory, particularly the short-lived isotopes of iodine and tellurium which are primarily associated with early health effects. The off-site consequence results which show essentially no early fatalities confirm this insight. However, these results also show that the conditional long-term health effects due to the long-lived isotopes of cesium, etc could in fact be as severe, if not worse (from a risk standpoint), as the corresponding results at full power, due mainly to the fact that the containment barrier is not present. The ultimate risk significance of the conditional results reported here, however, depends on the frequencies of the accident sequences leading to core damage. These frequencies are, currently, being calculated in the Level 1 analysis. If the core damage frequency during low power/shutdown is of the same order of magnitude as at full power, then the result of this study show that probababilistic risk analysis of reactor accidents needs to be extended, in general, to cover the risk during LP/SD operation.

7 References

1. U. S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, Vols. 1 - 3, December 1990 - January 1991.
2. R. M. Summers, et. al., "MELCOR 1.8.0: A Computer Code for Nuclear Reactor Severe Accident Source Term and Risk Assessment Analyses," NUREG/CR-5531, Sandia National Laboratories, SAND90-0364, January 1991.
3. "A FORTRAN 77 Program and Users Guide for the Generation of Latin Hypercube and Random Sampling Use with Computer Codes," R.L. Iman, et al., Sandia National Laboratory, NUREG/CR-3624, March 1984.
4. H. N. Jow, W. B. Murfin and J. D. Johnson, "SXOR Codes User's Manual," NUREG/CR-5360, Sandia National Laboratories, December 1989.
5. D. I. Chanin, et. al., "MELCOR Accident Analysis Consequence Code System," Sandia National Laboratories, NUREG/CR-4691, Sandia National Laboratories, SAND86-1562, Vols. 1 - 3, February, 1990.
6. T-L. Chu, et al., "PWR Low Power and Shutdown Accident Frequencies Program: Phase 1-Coarse Screening Analysis," Draft Letter Report, November 1991. Available in the NRC Public Document Room, 2120 L Street, NW, Washington, DC.
7. A Letter from T. L. Chu of Brookhaven National Laboratory to C. Lovett of Surry, Feb. 1992.
8. "Handbook of HRA," A. D. Swain, NUREG/CR-1278, August 1983.
9. J. V. Ramsdell, Diffusion in Building Wakes for Ground-Level Releases, Atmospheric Environment, Vol. 24B, No. 3, pp. 377-388, 1990
10. D. J. Wilson, Dilution of Exhaust Gases From Building Surface Vents, ASHRAE Trans., 83 (Pt. 1), pp. 168-176, 1977.
11. U. S. Nuclear Commission Regulatory Guide 1.145, Revision 1, November 1982.
12. R. S. Denning, R. Freeman-Kelly, P. Cybulskis, and L. A. Curtis, "Source Term Calculations for Assessing Radiation Dose to Equipment," NUREG-CR-4949, July 1989.

Enclosure 3

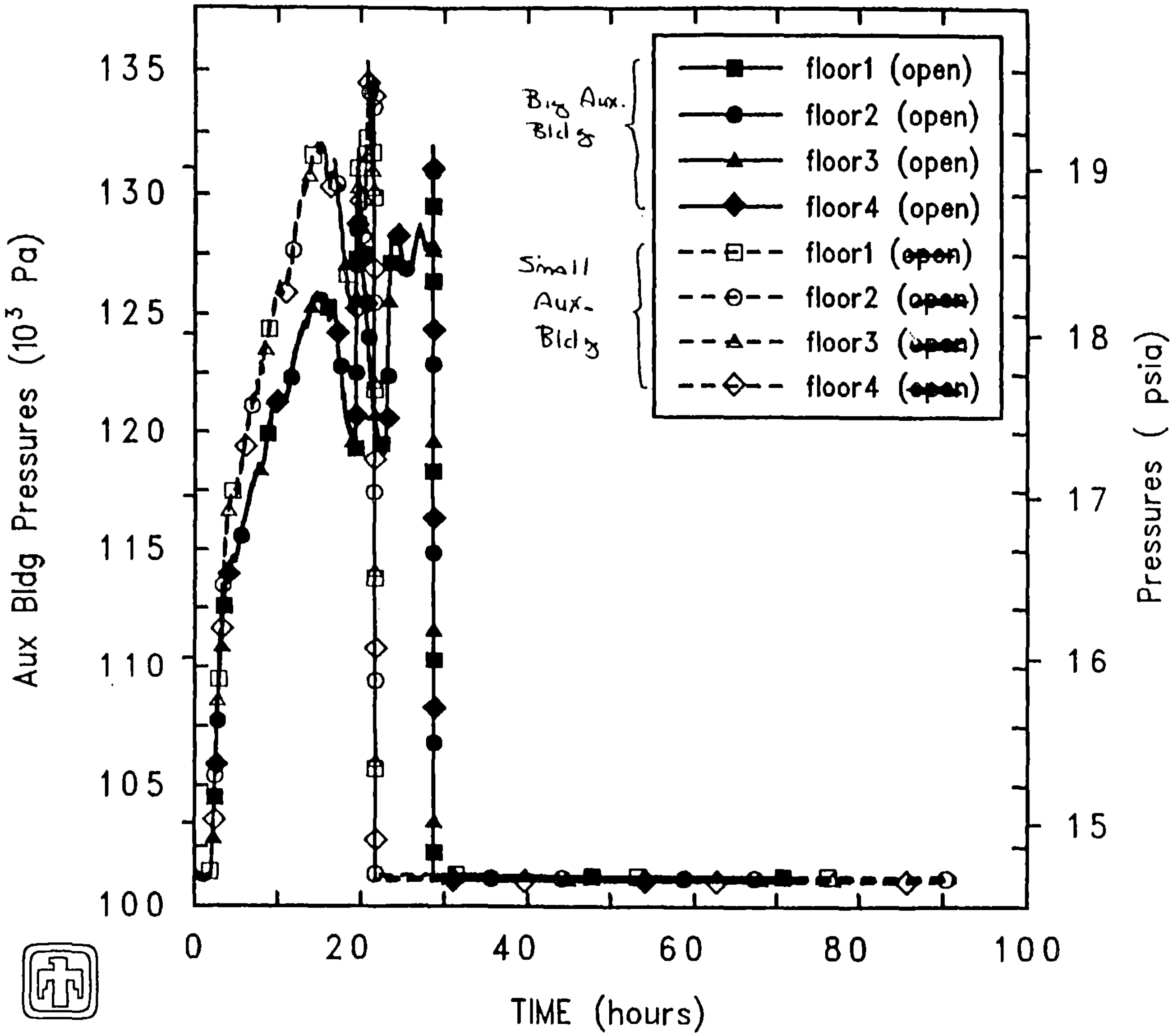
Plots of MELCOR Calculations of the Grand Gulf Plant

Accident Progression Timing

Calculation	Timing of Key Events from Initiation of Accident (hr.s)				
	Time to TAF	Core Damage	Vessel Failure	Aux. Bldg Failure	Contain. failure
PRA MODEL INPUT					
PRA Model: Containment Open	13.0	18.3	25.4	21.1	Cnt Open(2)
PRA Model: Containment Fails	13.0	19.4	28.6	No Fail. (3)	30.
MELCOR RESULTS					
Base Case (BC)-No Aux Bldg	12.7	18.3	25.4	(1)	(2)
* BC w/ Small Aux. Bldg	13.0	18.8	24.5	21.6	(2)
* BC w/ Big Aux. Bldg	13.0	18.8	28.6	28.6	(2)
* BC w/ Containment Closed	13.6	19.4	28.6	(1)	22 - 80.
* BC w/ Cnt Closed and H2 Burns	13.5	19.4	29.1	(1)	22 - 80.
BC initiated 15 days after SD	19.7	28.3	39.8	(1)	(2)

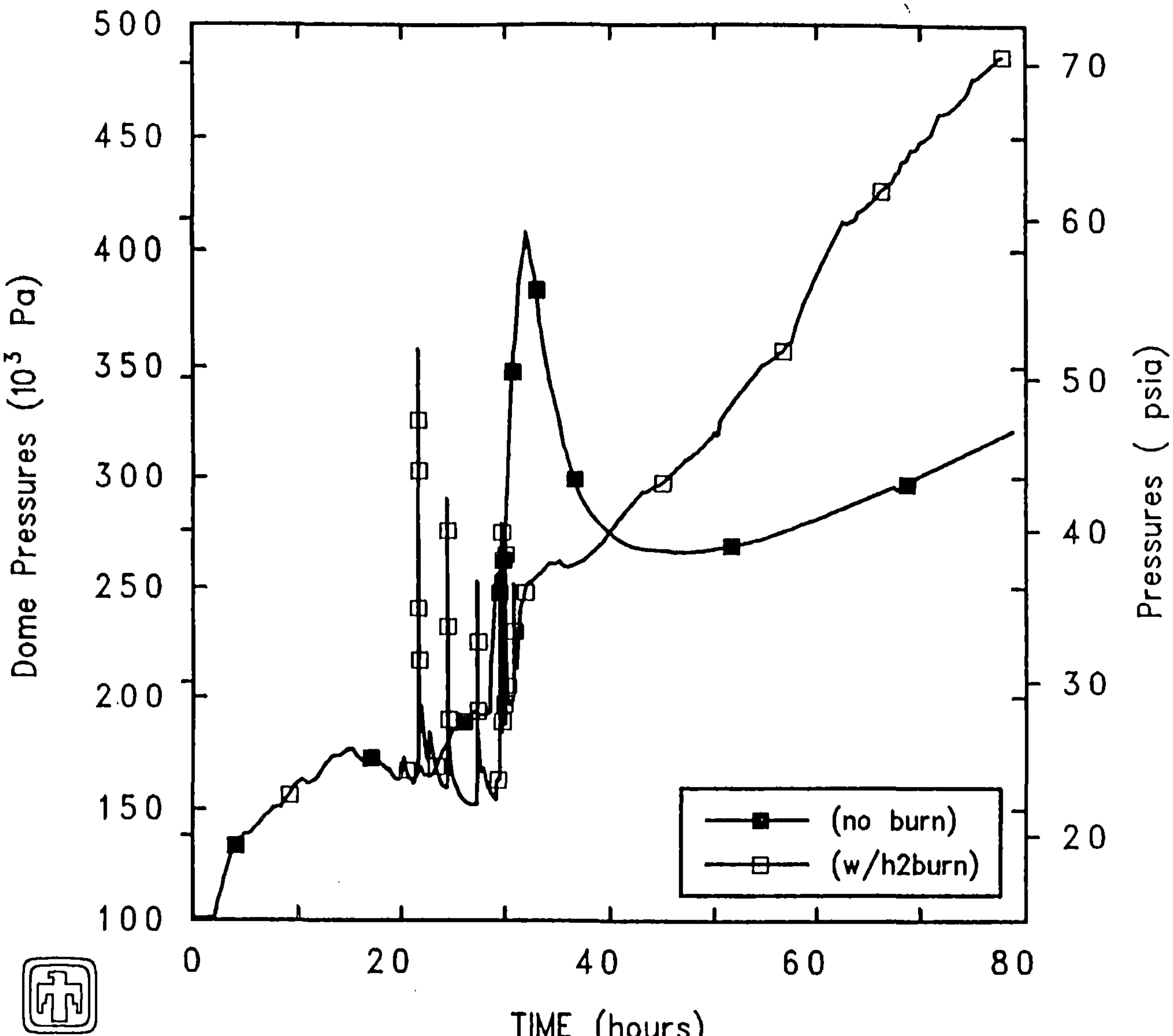
Notes:
1. Auxiliary building model not included
2. Containment is open during the accident
3. Containment failure bypasses the auxiliary building
4. MELCOR POS 6 BC Calculation:
   - Accident Initiated 4 days after shutdown
   - Containment is open (i.e, equipment hatch and both personnel locks)
   - Injection, shutdown cooling, and containment sprays are all unavailable
5. Core damage is defined as the first gap release
6. TAF = Collapsed water level at the top of the active fuel

Event Times
Grand Gulf



Grand Gulf POS 6-1a (open aux bldg)
CVCHDAN 3/22/92 07:34:05 MELCOR

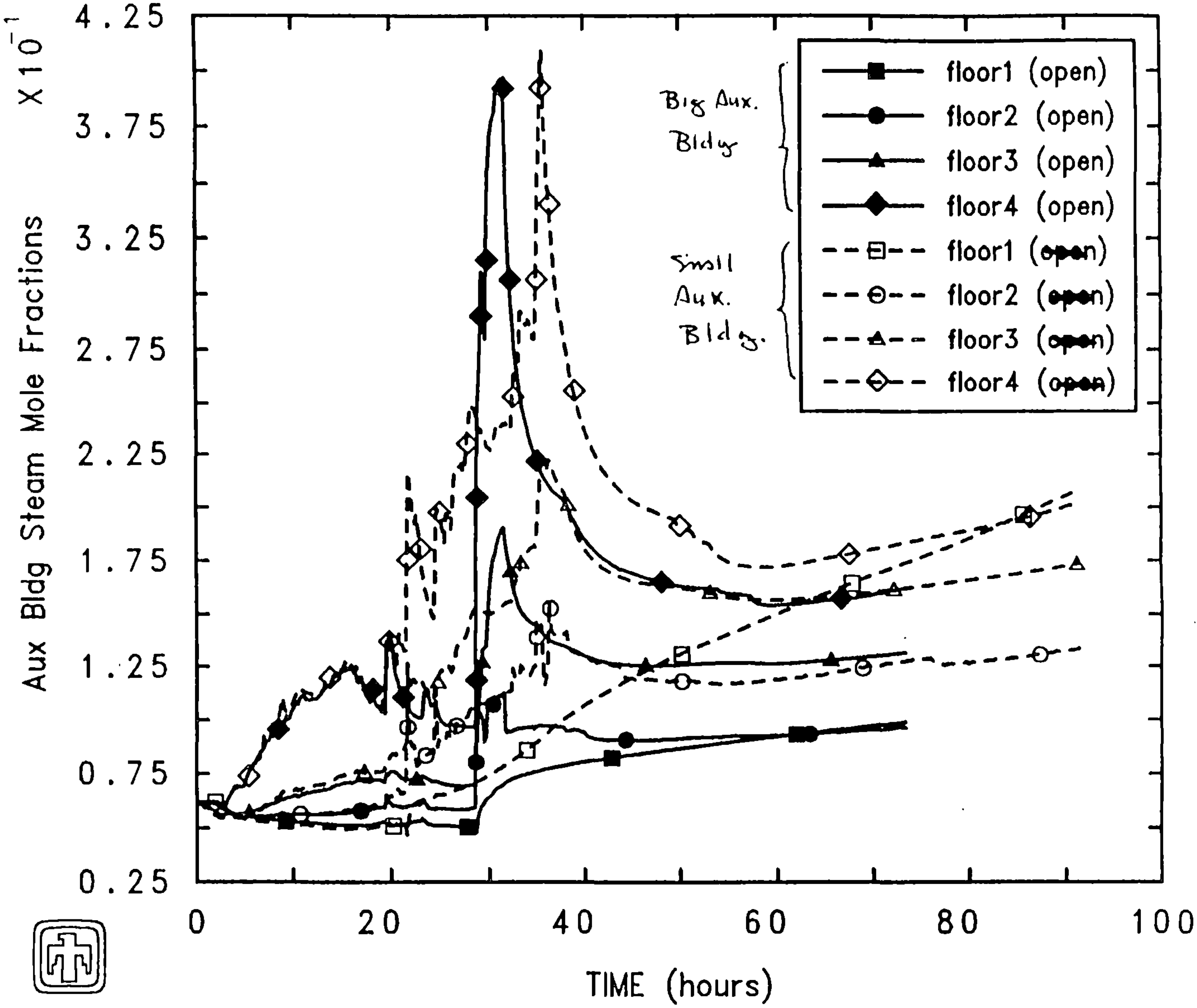
Item 2-GG



Grand Gulf POS 6-1b

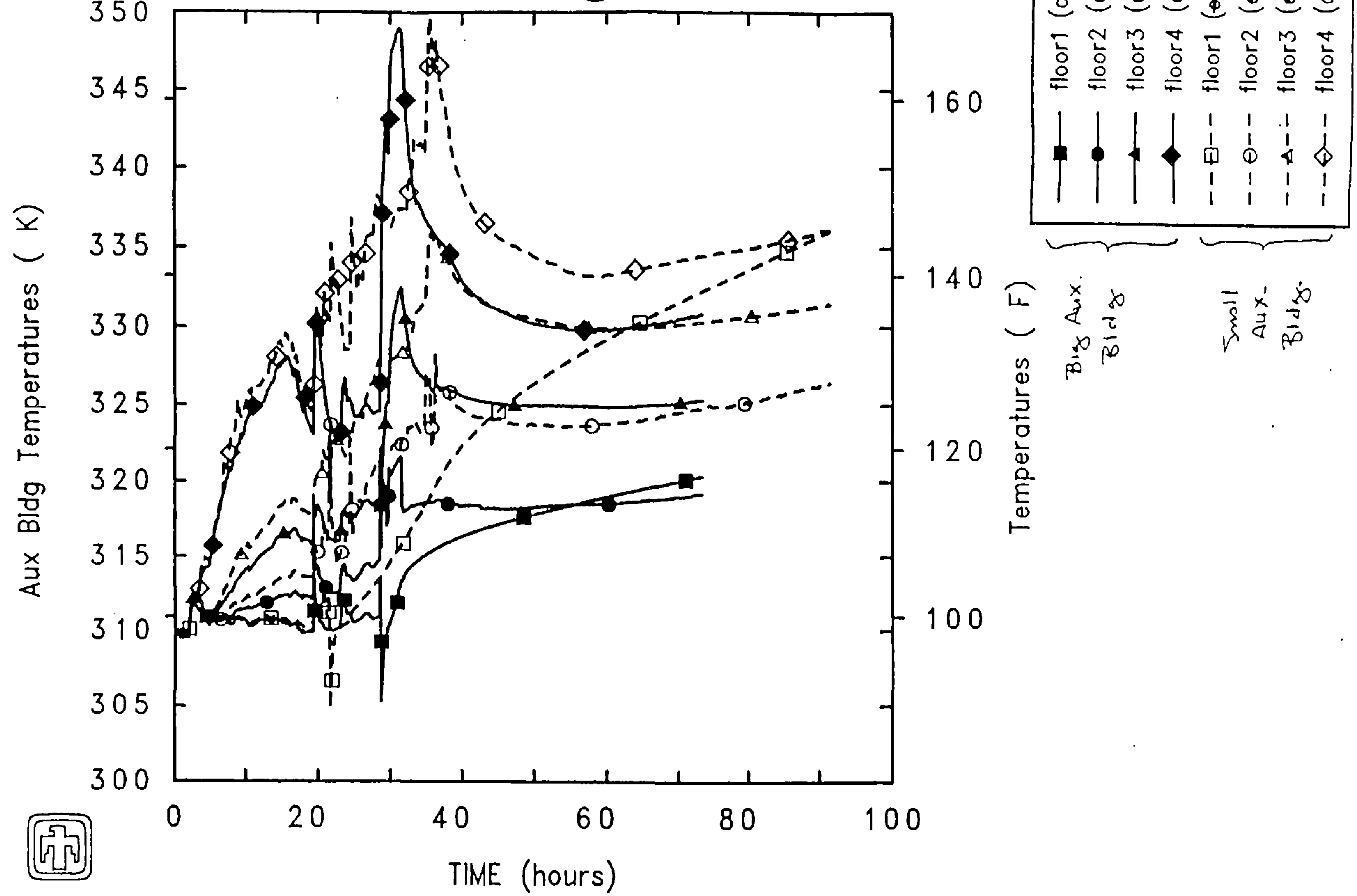
CICHDBK 3/09/92 07:34:28 MELCOR

Item 3a - GG



Grand Gulf POS 6-1a (open aux bldg)
CVCHDAN 3/22/92 07:34:05 MELCOR

Item 3b-GG



Grand Gulf POS 6-1a (open aux bldg)
CVCHDAN 3/22/92 07:34:05 MELCOR

Item 4-66

Grand Gulf MELCOR Masses and Volumes

Mass of Selected Core Components

Total Fuel Mass (UO2)	166200 Kg
Total Zircaloy Mass	81539 Kg
Total Steel Mass	87901 Kg
Total Control Poison Mass	1252 Kg

Chemical Class Masses:

Xe	463.7 Kg
Cs	268.4 Kg
Ba	207.5 Kg
I	20.9 Kg
Te	40.8 Kg
Ru	307.0 Kg
Mo	350.6 Kg
Ce	594.0 Kg
La	571.1 Kg
U	132390.0 Kg
Cd	1.4 Kg
Sn	8.6 Kg

Control Volumes

CORE

100: Lower-Plenum	108 M^3
103: Upper-:plenum/seperator	65 M^3
104: Dryer/Steam Dome	191 M^3
105: Downcomer	196 M^3
111: Channel	37 M^3
121: Bypass	31 M^3

DRYWELL

201: Drywell	6554 M^3
202: Weir Wall	1462 M^3
204: Pedestal	245 M^3

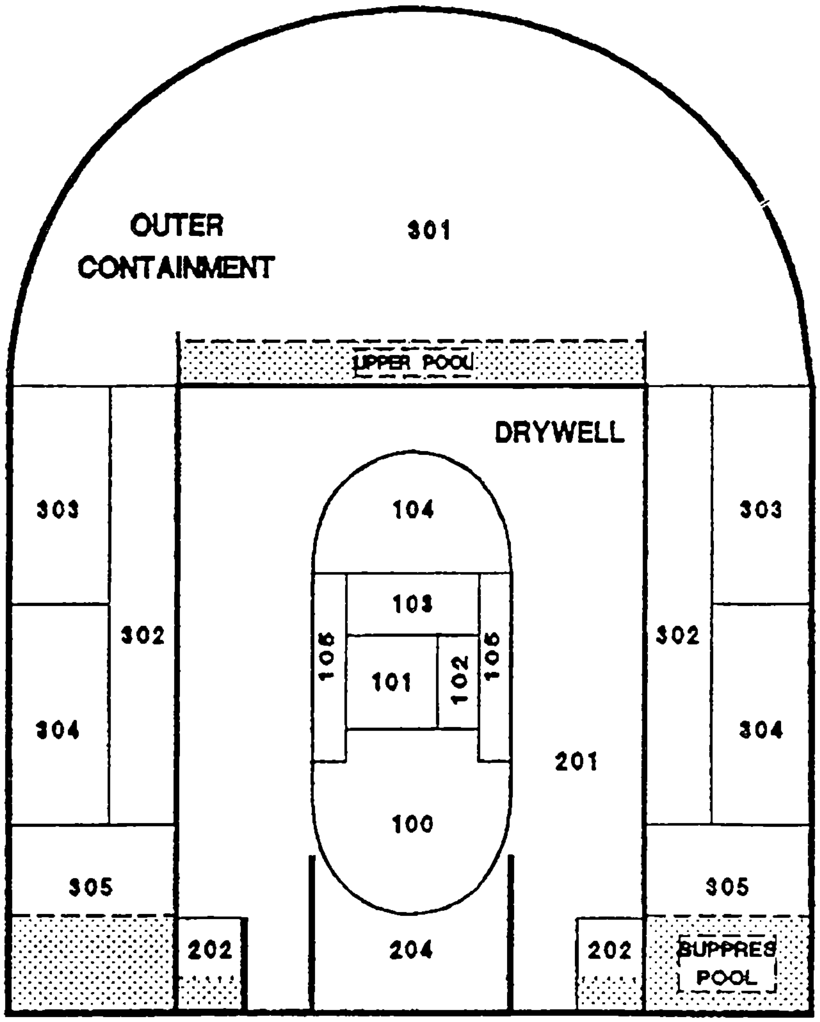
CONTAINMENT (WETWELL)

301: Dome	25461 M^3
302: Equipment Hatch	1654 M^3
303: Upper Annulus	4480 M^3
304: Lower Annulus	3278 M^3
305: Wetwell	7783 M^3

AUXILIARY BLDG	Big Aux.	Small Aux.
510: Floor 1	20570	13110 M^3
520: Floor 2	15820	11170 M^3
530: Floor 3	19840	14520 M^3
540: Floor 4	17510	15500 M^3

Item 7-GG

Containment Nodalization



Item 9-GG

Enclosure 4

Plots of MELCOR Calculations of the
Surry Plant

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

Upton, Long Island, New York 11973

(516) 282- 2337
FTS 666-
FAX: (516) 282-5730

Department of Nuclear Energy
Safety and Risk Evaluation Division
Building 130

June 3, 1992

Mr. Christopher Ryder
RES/PRAB
Mail Stop NLS-372
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Ryder:

SUBJECT: ACCIDENT PROGRESSION AND RISK ANALYSES OF PWR LOW POWER/SHUTDOWN, FIN L-1680-2

Please find attached the BNL responses for the "NRR Data Request" for the Surry Abridged Level 2/3 Analysis.

If you have any questions or need more information, please let us know.

Sincerely,

Jae Jo

J. Jo
Safety Integration Group

/bkp

cc: R. Bari (w/o attachment)
J. Boccio (w/o attachment)
R. Duffey (w/o attachment)
V. Mubayi (w/o attachment)
W. T. Pratt (w/o attachment)

ATTACHMENT

- MELCOR Calculations: General Information on the MELCOR Calculations
- Question 1: Figure 1. Mass Release Rate from RCS
  (Plots for "Energy Release Rates" are not available)
- Question 2: Figure 2. Pressure vs. Time for Open Containment
- Question 3: Figure 3. Pressure vs. Time for Closed Containment
- Question 4: Not applicable to the Surry Plant
- Question 5: Figures 4 and 5. Total Radionuclides Mass in Containment
  (Plots for "Release Rates" are not available)
- Question 6: Not applicable to the Surry Plant
- Question 7: Table 1. Mass in the Vessel at Start of the Calculation
- Question 8: Please refer to the "Chapter 7. Reference" of the draft letter report
- Question 9: Figure 6. Noding Diagram
  ~~Table 2. Heat Structure Input~~ Not in a readily useable form

MELCOR Calculations

The SURRY PWR plant is a 3 loops Westinghouse design with 3 vertical U-tube steam generators. The loops are designated as loop A, B, and C. Each loop contains hot leg, steam generator, pump suction leg, reactor coolant pump, and cold leg. The pressurizer and presurizer surge line are attached to the hot leg in loop C. In level II/III of low power & shutdown source term analysis, the engineered safety system is assumed unavailable. The MELCOR input deck used in this calculation has been modified from the original deck which prepared by Sandia National Laboratory. The nodalization of the SURRY power plant results in total of 17 control volumes (6 for the RPV, 3 for the primary system, 8 for the containment, and the environment), 33 flow paths and 127 heat structures. The core itself, is nodalizedinto 39 cells with 3 radial rings, and 13 axial levels.The nodalizations are based on the following assumptions:

1. The primary reactor coolant systems are isolated. The steam generator,cold leg, and the secondary cooling system are not included. Also the residual heat removal system is not included.

2. The primary reactor coolant system is at low pressure; i.e., 1 atmosphere with the initial temperature of 333 K. The RCS is opened to the containment by the pressurizer safety relieve valve. The containment is also opened to the environment.

3. The calculation start after the reactor has been shutdown for 24 hrs with the coolant water level above the top of the reactor core. The decay heat power was calculated base on the ANS's standard for light water reactors (ANS-5.1-1979) with the 2 years reactor operation period and 80% capacity factor.

4. The release of the gap fission product inventories in the fuel-cladding gap occurs if a zircaloy cladding temperature reached 1173 K. The inventories for the entire ring is instantaneous released to the core channel control volume.

5. The radioactive and nonradioactive materials released from the core are calculated by an empirical CORSOR release rate model with the surface-to-volume ratio of the material correction. Also the release model allowed combination of Cs and I classes instantaneousely by the elemental molecular weights.

6. The engineered safety system are not available in the base case calculation. However, the containment spray and the low pressure injection are included in the sensitivity analysis.

MELCOR Base Case Results

The MELCOR base case calculation is based on the accident initiated at 24 hrs. after reactor has been shutdown. The RCS is opened to the containment via pressurizer safety valve. Also the containment is opened to the environment. The containment spray and the core recovery injection are not available. The hydrogen burn package is not active. The calculation resulted in core boiled off, radionuclides released to the RCS, the containment, and the environment respectively. The fuels were melt and relocated to the lower plenum of the reactor vessel. The reactor vessel breached, and the molten ejected to the containment cavity. The core-concrete inteaction caused more radionuclides released to the containment and the environment. The total in-core zircaloy oxidation is about 37% with 210 Kg of hydrogen generation. The ex-vessel hydrogen production is about 500 Kg at the end of calculation. The time sequences of the calculation are sumarized in Table 1.

Table 1. The MELCOR calculation time sequence of the base case

Event	Time (min)
Start calculation	0
Half core uncovered	60
Ring 1 gap released	100
1 Kg Hydrogen produced	104
Ring 2 gap released	105
Ring 1 fuel start to melt	117
Ring 3 gap released	118
Ring 2 fuel start to melt	123
Ring 3 fuel start to melt	136
30% Core melt	141
Core dry out	153
60% Core melt	219
Ring 3 lower head failed	249
Molten core ejected to cavity	249
Ring 2 lower head failed	270
Ring 1 lower head failed	282
End of calculation	2000

Sensitivity Analysis and Results.

The sensitivity analysis has been performed on the base case to evaluate the effect of the containment spray, the core recovery injection, the pressure effect to the containment due to hydrogen combustion, and the decay power when the accident occured. The results are summarizes as follows;

Case 1. The base case with the hydrogen combustion is active. The results are core damaged, vessel breached, and hydrogen start to burn at about 500 min. in the cavity due to the core-concrete interaction, and propagated to all the containment compartments. The calculation is ended at 2000 min.

Case 2. The base case with the containment spray is on at 220 min. and is on until the end of calculation; i.e. 2000 min. with the flow rate of 3200 gpm. The results are core damaged, vessel breached.

Case 3. The base case with the core recovery injection into the RPV down commer at 175 min. with the injection rate of 600 gpm. The results are core arrested, with no vessel breached. The calculation terminated at 215 min. when there are no further radionuclides released.

Case 4. The base case with both core recovery injection into the RPV down commer and the containment spray is turn on at the same time; i.e., 175 min. The spray flow rate and the core recovery injection rate are the same as case 2 and case 3 respectively. The results are no vessel breached. The calculation is terminated at 215 min.

Case 5. The base case with the containment closed and the hydrogen combustion is active. The containment withstands the pressure increase up to 120 psi. The results are core damaged, vessel breached, hydrogen is burn inside the reactor vessel, in the cavity, and all the containment compartments. The radionuclides released from the fuel and CCI reaction are retained in the containment, no released to the environment. The containment pressure incresed very slowly with the rate of 1 psi per hr.

Case 6 and 7. The base case with the accident occured at 72 hrs. and 240 hrs. after the reactor has been shut down respectively. Both cases have similar results as the base case; i.e., core damaged, fuel melt and relocated to the lower plenum, lower head failed and molten fuel ejected to the cavity. The vessel breached occured at 340 min. and 590 min. for 72 hrs and 240 hrs decay power respectively. The calculation of both cases are terminated at 2500 min.

The time sequences of accident resulting from the MELCOR calculation are used as an input to the SURSOR computer code to estimate the source terms released for the low power and shutdown condition are summarized in Table 2.

Table 2. MELCOR Time Sequence Calculations

MELCOR case	Accid. Int. After SD(Hr)	Time (Min.) Gap Release	Core Melt	Injection Recovery	Vessel Breach	Cont. Spray	End of Calc.	Event I.D.
BC	24	100	120	No	250	No	2000	7
Case 1	24	100	120	No	250	No	2000	-
Case 2	24	100	120	No	250	220	700	6
Case 3	24	100	120	140	No	No	200	5
Case 4	24	100	120	180	No	180	220	4
Case 5	24	100	120	No	215	No	1000	-
Case 6	72	140	160	No	340	No	2500	-
Case 7	240	230	260	No	590	No	2500	-

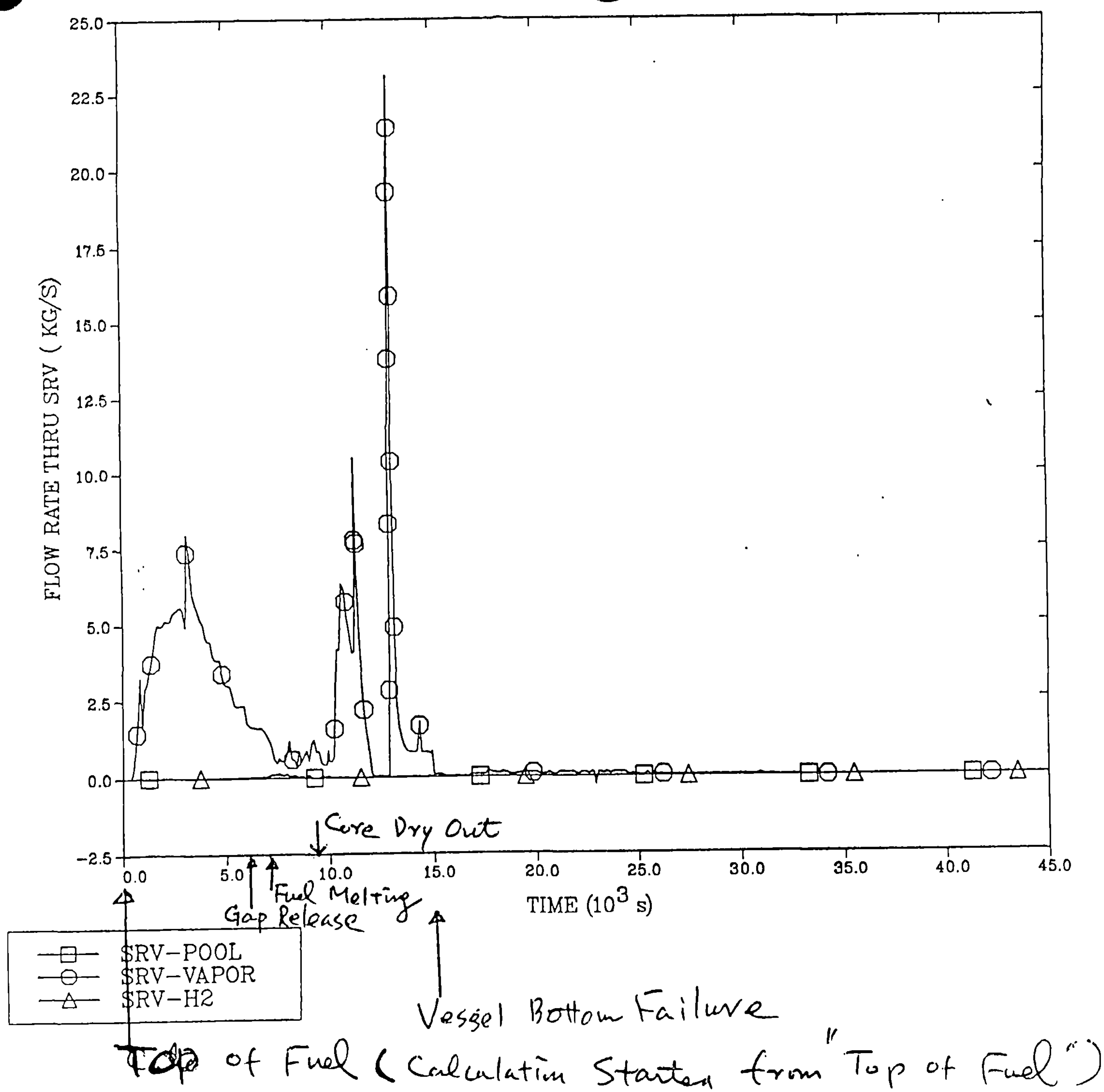
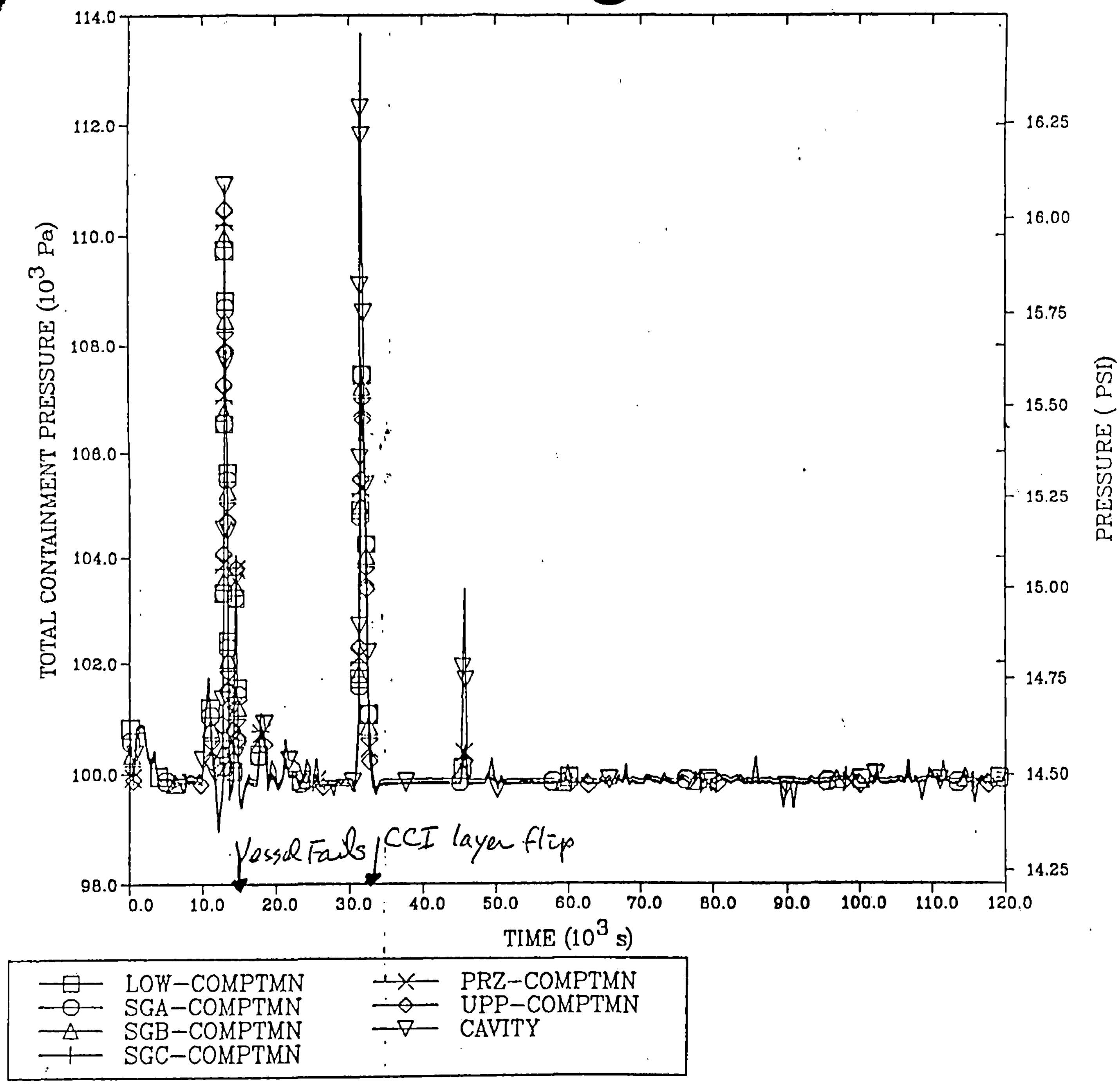


Figure 1. Mass Release from RCS

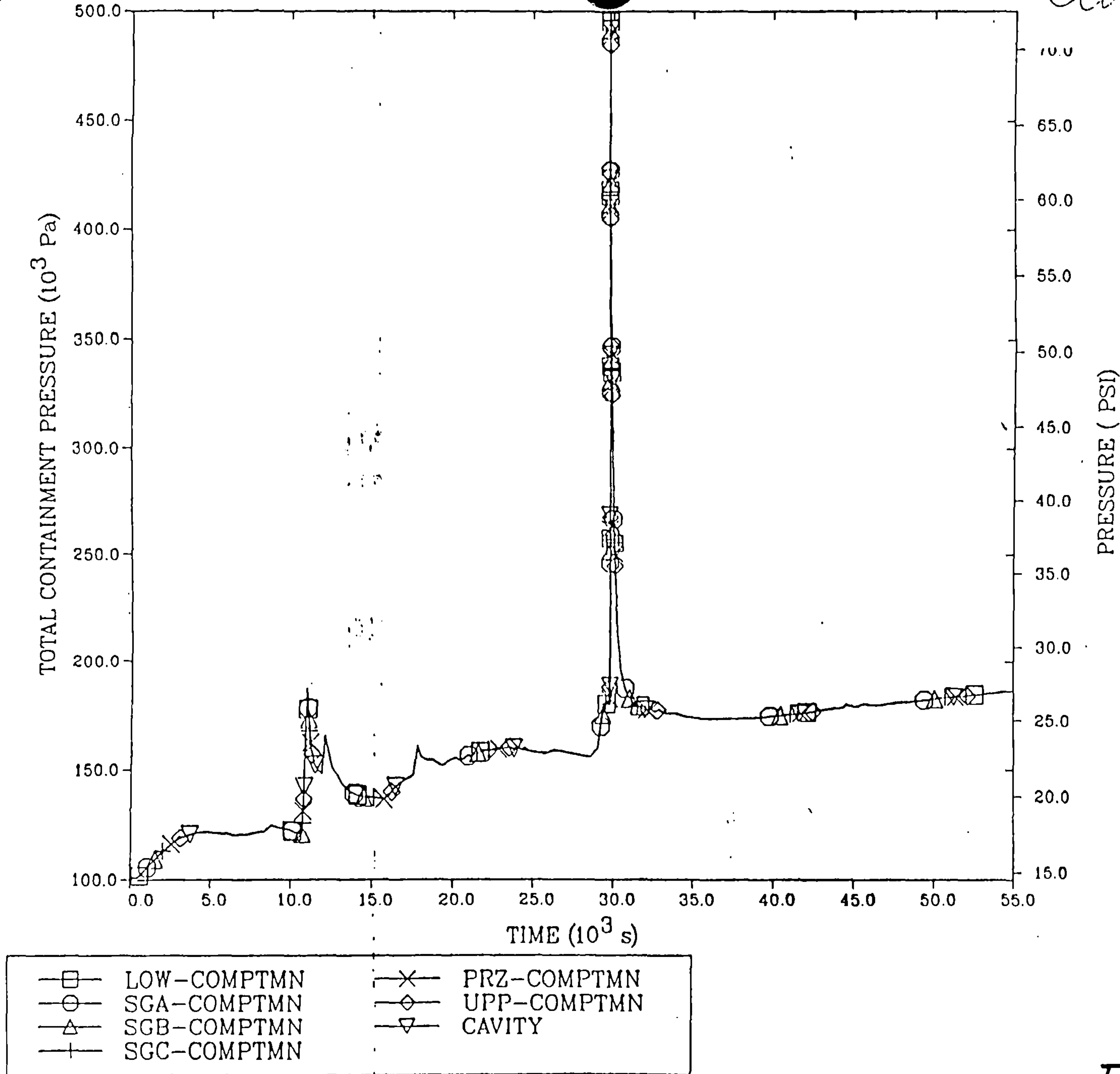
Item 1-5

Containment open

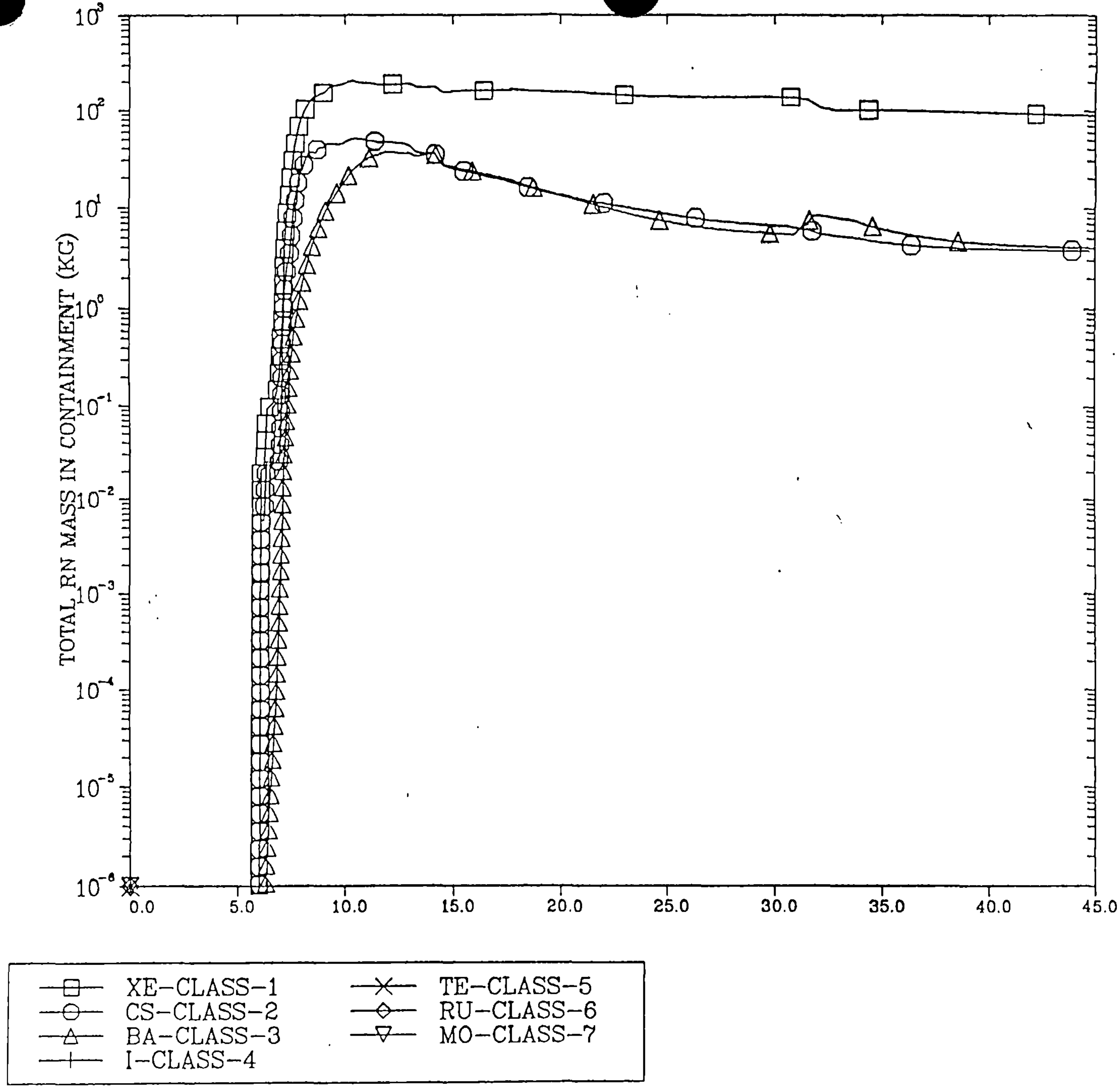


Item 2-5

Closed.

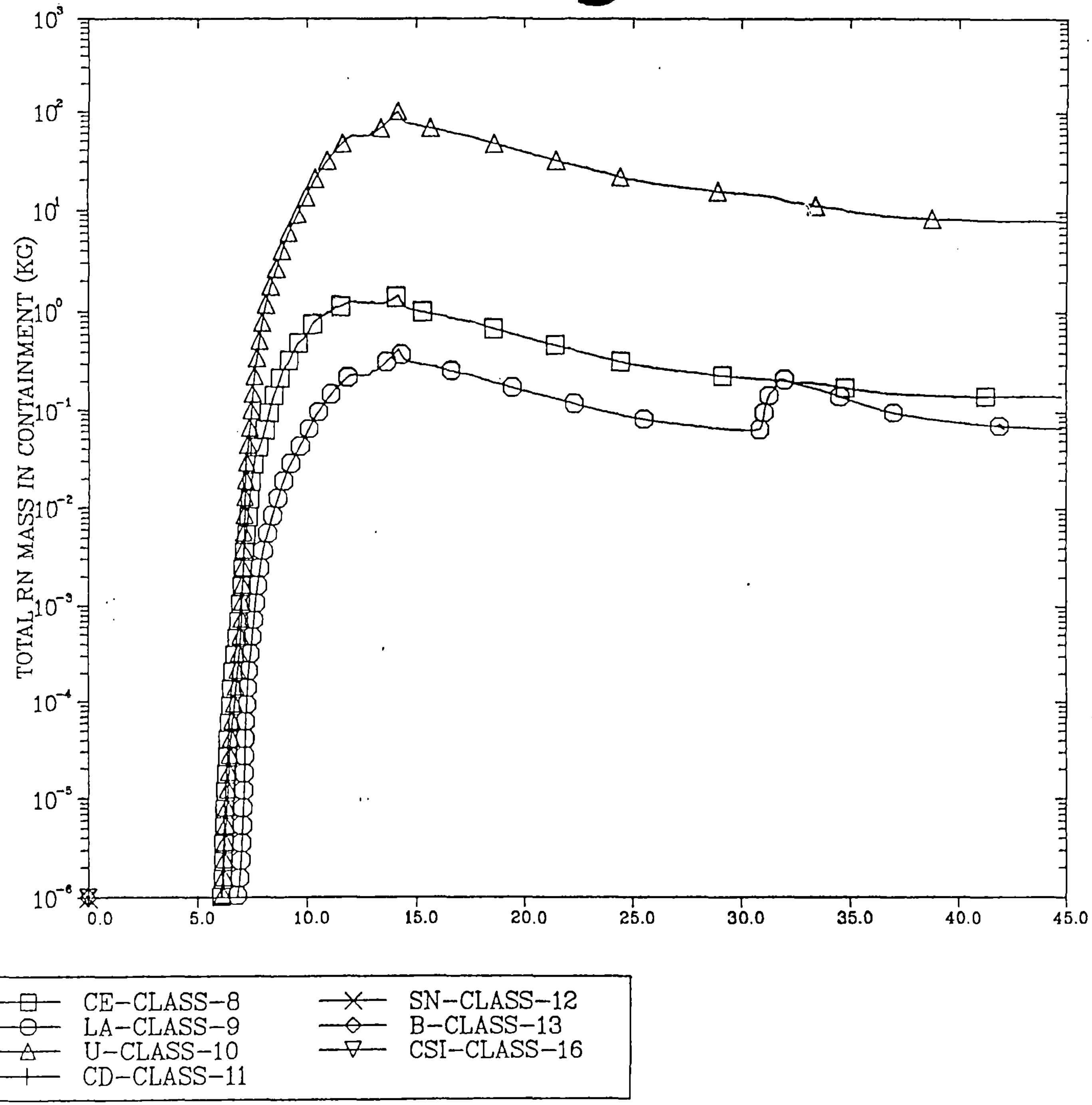


Item 3-5



TOTAL RN MASS IN CONTAINMENT (KG)
10^{3}
10^{2}
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
XE-CLASS-1
CS-CLASS-2
BA-CLASS-3
I-CLASS-4
TE-CLASS-5
RU-CLASS-6
MO-CLASS-7

Item 5a-5


TOTAL RN MASS IN CONTAINMENT (KG)
10^{3}
10^{2}
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
0.0
5.0
10.0
15.0
20.0
25.0
30.0
35.0
40.0
45.0
CE-CLASS-8
LA-CLASS-9
U-CLASS-10
CD-CLASS-11
SN-CLASS-12
B-CLASS-13
CSI-CLASS-16

Item 56-5

Table 1

Total mass of	UO_2	= 80711.00	kg
	Zircaloy	= 16071.20	kg
	Steel	= 700.50	kg
	Control Poison	= 2005.90	kg

from CORE package

Coolant	CV100 (Down comer)	= 10,748	kg
	CV110 (lower plenum)	= 20,547	kg
	CV120 (Core channel)	= 14,320	kg
	CV130 (Core Bypass)	= 5,297	kg
	total	= 50,912	kg.

Coolant at the top of Active fuel level

Item 7-S

CVφφ1

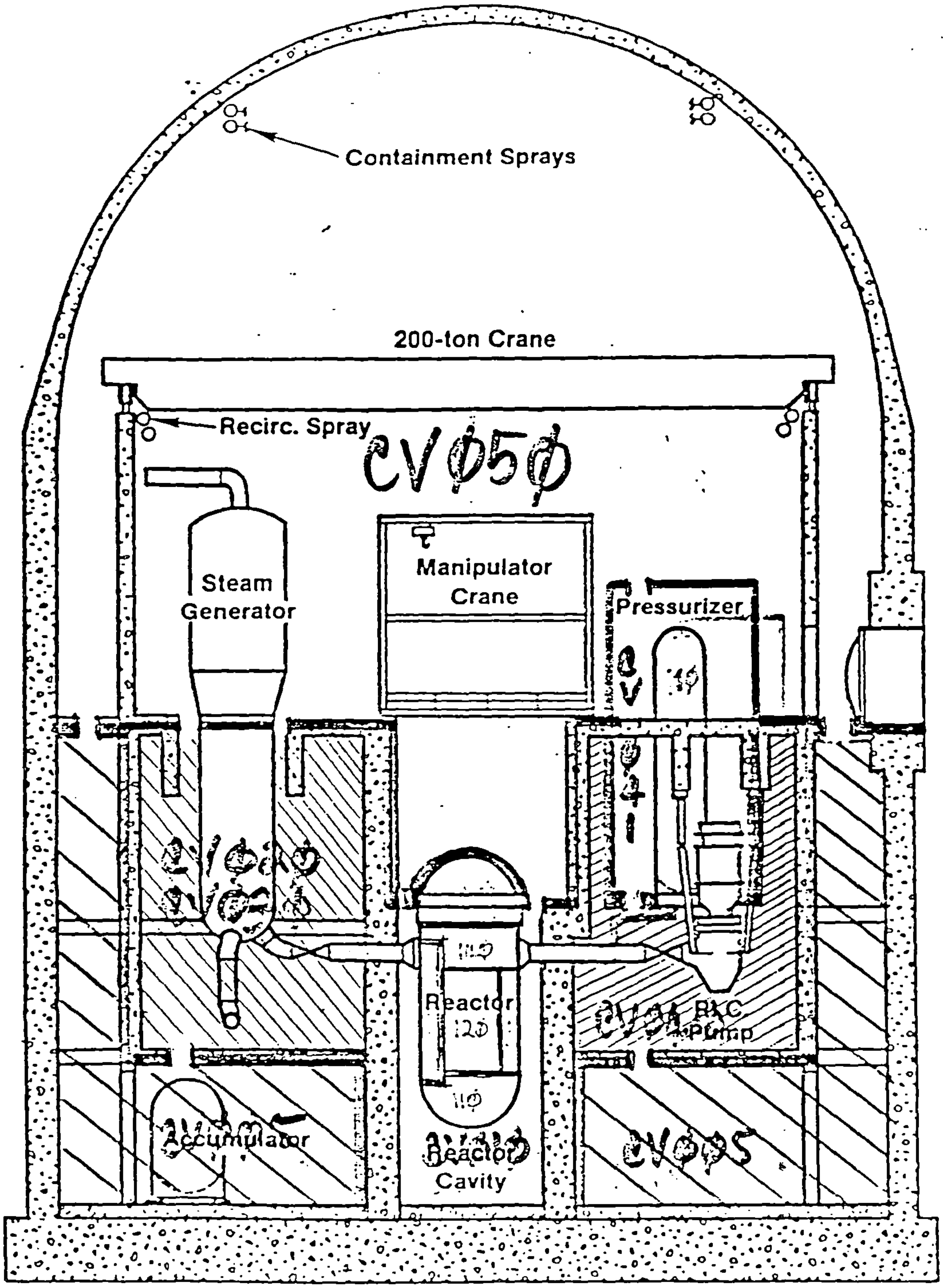


Figure Section of Surry Containment.

Item 9a-5

Containment control volume

9

CV Ø1Ø = Cavity
CV ØØ5 = Lower Compartment control volume
CV Ø2Ø = Steam generator A compartment
CV Ø3Ø = Steam generator B compartment
CV Ø4Ø = Steam generator C compartment
CV Ø41 = Pressurizer compartment
CV Ø5Ø = Upper compartment

Item 96-S