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ABRIDGED RISK STUDY DURING LOW POWER/SHUTDOWN OPERATION AT SURRY

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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 operations at a PWR (Surry) and a BWR (Grand Gulf). One such program is a risk study of accident progressions and consequences.

The objective of this study was 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 (NRC FIN L-1680). The study was designed to obtain results for regulatory decisions that were 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) (NRC FIN L-1678).

1.2 Scope of the Study

The ***abridged risk*** analysis took place from January through April 1992. 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* 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 because the frequencies had yet to be determined in companion Level 1

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and human reliability analysis (HRA) studies. Uncertainty was 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, mid-loop operation (referred to as POS 6), where the reactor vessel head is on and the containment is closed but has no pressure-retaining capability. The sister study at SNL investigated an early stage of a refueling operation. Figure 1.1. shows the scope of both studies.

1.3 Methodology

The abridged process of computing the conditional consequences is shown in Figure 1.2. In general, both this study and that done at SNL follow this scheme; some differences in the details of the procedures 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 began with the assumption that core damage has occurred, making the consequences conditional. Given core damage, the reasonable accident progressions were 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 progressions is reflected vis-a-vis branch point probabilities. In large-scale risk studies, the assignment of such probabilities can be made by a formal process of obtaining information from groups of experts (see Section 2.7 of the NUREG-1150 report, Vol. 1[1]). Here, because of resource limitations, most assignments were made by the contractor staff. Thus, the probabilities are not as rigorous as they could be; this is one of several limitations of the study. This lack of rigor was partially offset by repeating the calculations with other reasonable input values; together, these repeated calculations 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 were 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 is assigned to the branch points and multiplied through to the ends of the APET. The calculations were repeated using the sets of inputs to build 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 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 from the characteristics of the accident progression pathways through the APET 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). Although 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 input values for repeated calculations. The result is a distribution of source terms for each accident progression pathway.

An internal "Source Term Advisory Group," including W. T. Pratt and H. P. Nourbakhsh of BNL, and J. Kelly and D. Powers of SNL, was formed to support this study. The results of the accident progression and source term analysis were discussed with the group at two meetings during this analysis.

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 by Ramsdell[10] and a combination of the older Wilson[11] and Regulatory Guide 1.145 models.[12]
- Off-site consequences were computed using the MACCS code.[6]

To compute consequences, a few 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 for each accident progression pathway by multiplying the consequences by their associated accident probability determined with APET. The products of the pathways were summed. This process was repeated for each Monte Carlo sample of the source terms. Then, high, medium, and low results were reported. In the NUREG-1150 study,[1] high, medium and low results are represented by the 95 percentile, median and 5 percentile values of the distribution of the results. However, in this study, the number of samples taken was not sufficient enough to define them statistically. Therefore, they were referred to as high, medium and low results. They would approach the statistical 95, 50 or 5 percentiles, if sufficient numbers of samples were taken.

Differences

This study differs slightly from its sister program at SNL in three ways.

(1) Two hundred samples of uncertainty distributions were taken for source terms compared to one hundred at SNL.

(2) 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 the SNL study used twelve samples.

(3) Dose rates inside the containment or the reactor building were not calculated because the releases were assumed to take place through the equipment hatch directly to the outside. At Grand Gulf, the release path is through the reactor building, so the in-building dose rates were computed.

1.4 Limitations and Strengths of the Study

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

Limitations

- The subject of the study is one POS, mid-loop operation. This POS was selected because it was identified in a preliminary Level 1 study, known as a coarse screening analysis (NRC FIN L-1344),[7] as having the potential to occur at a relatively high frequency. Also, the POS had characteristics (e.g., reduced inventory) of interest to the staff in the NRC's Office of Nuclear Reactor Regulation.

- 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. Frequency was not incorporated into the Level 2 and 3 calculations to determine risk because the frequency estimate was only an approximation.

- The simple APET, having only nine top event questions, accounts for a limited number of factors. This compares to about seventy questions in a large-scale PRA such as the NUREG-1150 study.

- The on-site dose estimates stem from simple equations, yielding rough estimates.

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

- Because of gaps in knowledge of the plant configuration and operator actions, assumptions were necessary which are documented in the following sections. Some gaps will be filled later with more rigorous determinations with results from detailed Level 1 and HRA studies and 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 at least 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 were determined to at least a first approximation.
- Estimates of both on-site and off-site conditional consequences were made.

Figure 1.1 Scope of the Current Risk Study

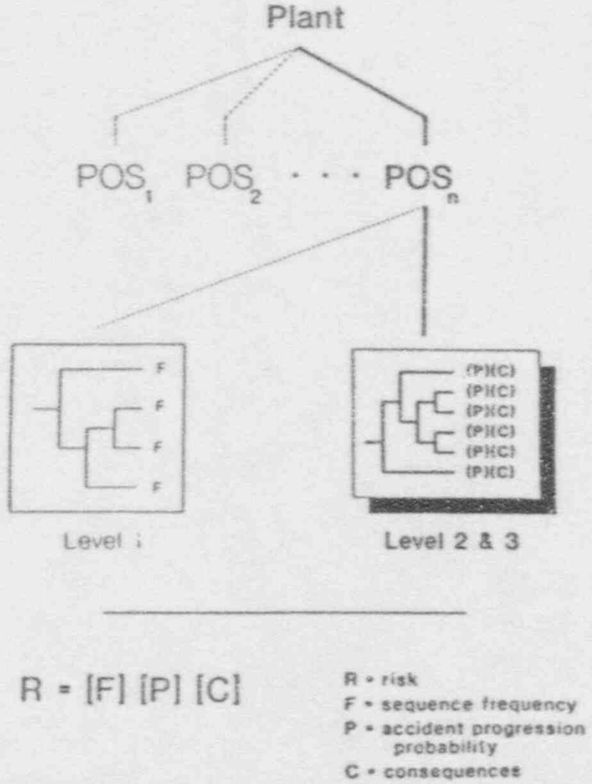
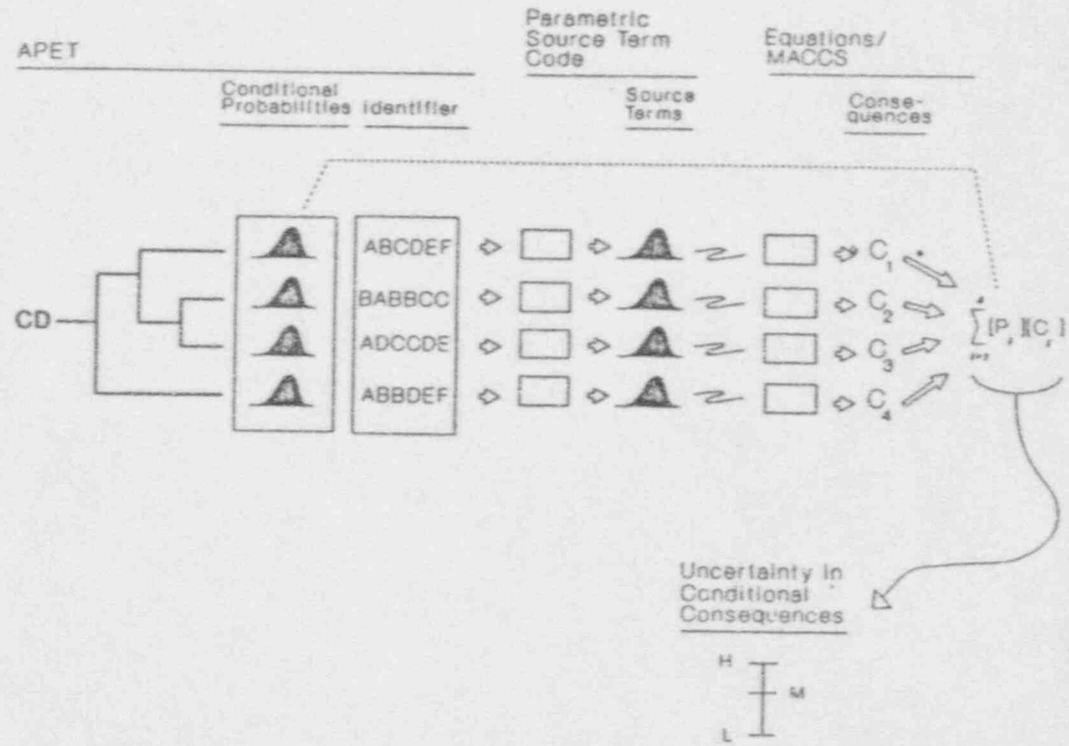


Figure 1.2 Abridged Process of Computing Conditional Consequences



2 Accident Progression Analysis

2.1 Approach

Following core damage in a severe accident, a risk analysis of the accident progression is done with an 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. The APET involves modelling of the physical processes occurring in the vessel and containment during the various accident sequences, and the availability and status of various safety equipment which could be used to mitigate the severity of the accident, and assessing the containment's capability to retain the fission products under severe accident loads. A series of questions are asked which represent these events and phenomena. Each path through the APET defines a unique accident progression path that potentially could give rise to the release of fission products. The number of questions in a APET can vary, depending on the details desired, and the number of relevant, important phenomena to be modelled.

To determine the extent of detail needed for this APET, extensive use was made 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. That study showed that the major cause of release was containment bypass, followed by basemat melt-through. The probability of early failure of the containment caused by various mechanisms and late failure resulting from gradual pressurization was either very small or negligible. Thus, once the containment boundary is closed, the containment retains the fission products most of the time (except by very late basemat melt-through). In other words, phenomena such as direct containment heating or steam explosions were not important contributors to the estimated probability of containment failure and the eventual release of fission products. For accidents during low power and shutdown 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 occurring at full power.

However, the containment cannot retain pressure during mid-loop operation at this plant, since some penetrations in the containment are closed with temporary barriers. Therefore it 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; because the integrity of the containment is already lost at accident initiation, many questions normally needed to assess the potential for containment failure are no longer relevant. However, several important questions must 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 level potentially increases the time available to take actions to recover core-cooling capability before the core is uncovered, and potentially reduces the inventory of fission products 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 LHS 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 shutdown to accident initiation.

2.2 Plant Configuration

The plant configuration during a low power and shutdown period can vary widely depending on the purpose of outage. Furthermore, there is a large degree of uncertainty for the operational state and availability of plant systems and components. Some examples are the number of loops isolated, the size of RCS venting, and the availability of containment sprays. For this abridged analysis, the BNL staff 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.

The two most important factors for determining the containment's response during an accident in POS 6 are the status of its integrity and the 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 technical specifications to have any of the containment sprays available once the plant enters the RHR entry condition at Surry.[8] Consequently, all of the containment sprays could be out of service during mid-loop operation. Therefore, spray availability was used as one of the uncertainty parameters in this study.

From several discussions with the Surry personnel, the BNL staff learned 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 a severe accident and prevent release of fission products.[8] This is due primarily to the presence of a temporary restraining plug, that has no overpressure capability, in place of the escape tunnel in the containment equipment hatch. Therefore, for this study, the containment was assumed to leak during POS 6.

2.3 Level 1 Sequence Description

Earlier, the BNL staff made a preliminary screening analysis of the systems reliability and characterized the accident sequences leading to core damage for the internally initiated events at the Surry Unit 1 plant.[7] The major objectives were to provide initial insights into any particularly vulnerable plant POSs during low power/shutdown operations, and to identify the set of major initiating events applicable to each POS. From this coarse screening analysis,[7] the BNL staff determined that POS 6, 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 a loss of RHR accident. 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 HRA. Since this HRA has a large band of uncertainty, it also was included as a uncertainty parameter. For accidents initiated by loss-of-power, its recovery determines the probability of recovering the core-cooling capability, and terminating an accident.

2.4 Event Tree Analysis

A relatively simple APET was used to describe events in the vessel and the containment responses after core damage. Figure 2.1 shows the containment event tree used. The first three questions refer to the status of containment. In this POS, the containment is assumed to be leaking from the start of the accident. Once the status of the containment is identified, the fourth 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 could significantly decrease the magnitude of release of fission products. Therefore, the timing of recovery of core-cooling capability was divided into five periods; Very early, Early, Intermediate, Late, and Never. 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 breaching the vessel. From 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, here, 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.

Table 2.1 shows the timing of core-melt progression as calculated by the MELCOR code, which was used to determine the time available for recovery, assuming that the accident was initiated 24 hours after shutdown. However, this time can vary widely. Therefore, the BNL staff determined the time of accident initiation by sampling from the joint distributions of the time to enter the mid-loop operation and the duration of POS 6 for each observation. Data from the Surry plant, which were collected for the screening level 1 analysis,[7] were used to determine 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 for the accident sequences whose times of accident initiation were different from those selected for the MELCOR calculations. The recovery probability was based on the HRA recovery curve for human error,[9] the off-site power recovery curve,[7] and hardware availability for each of the time periods (the latter from data used in the screening level 1 study). The next three questions (fifth, sixth and seventh) address spray availability and whether the cavity is dry or wet, which determines the extent of core-concrete interaction. Spray availability was included as an uncertainty parameter.

The outcome of the accident sequences in the APET were classified into eight bins (including a *No Release* bin) depending on the extent of core damage, vessel breach and spray availability (Fig. 2.1).

The basic structure of the APET (Figure 2.1) is sufficiently general to be applied to other POSs of low power and shutdown operation where the containment is closed and able to contain the elevated pressure. However, because 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 which core damage was defined to have occurred when the coolant level dropped to the top of active fuel.

As discussed above, an accident can still be terminated without core damage if core cooling is recovered during the *Very early* period. One possible exception to this is 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 its solubility limits, 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 effects on-site. Due to the limited time we had available, these releases were not quantified. In estimating the final risks conditional on core damage, only accident sequences which were actually predicted to result in core damage were included; accident sequences which were terminated in the *Very early* period were not included in the calculations of conditional risk. Figure 2.2 compares the conditional probability of arrest of core damage before vessel breach for the this analysis 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

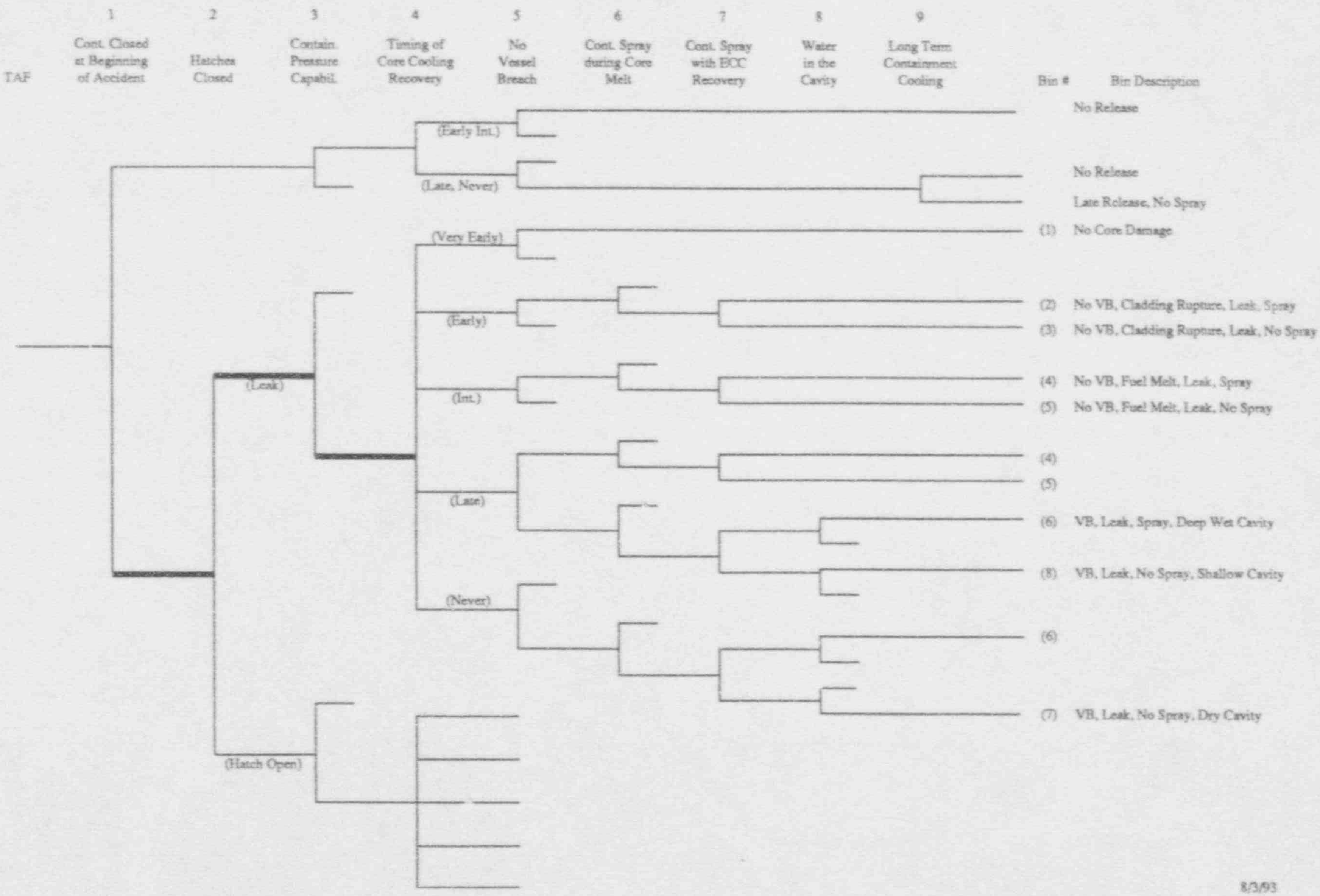


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Figure 2.2 Fraction of Core Damage Sequences Which are Terminated Without Vessel Breach

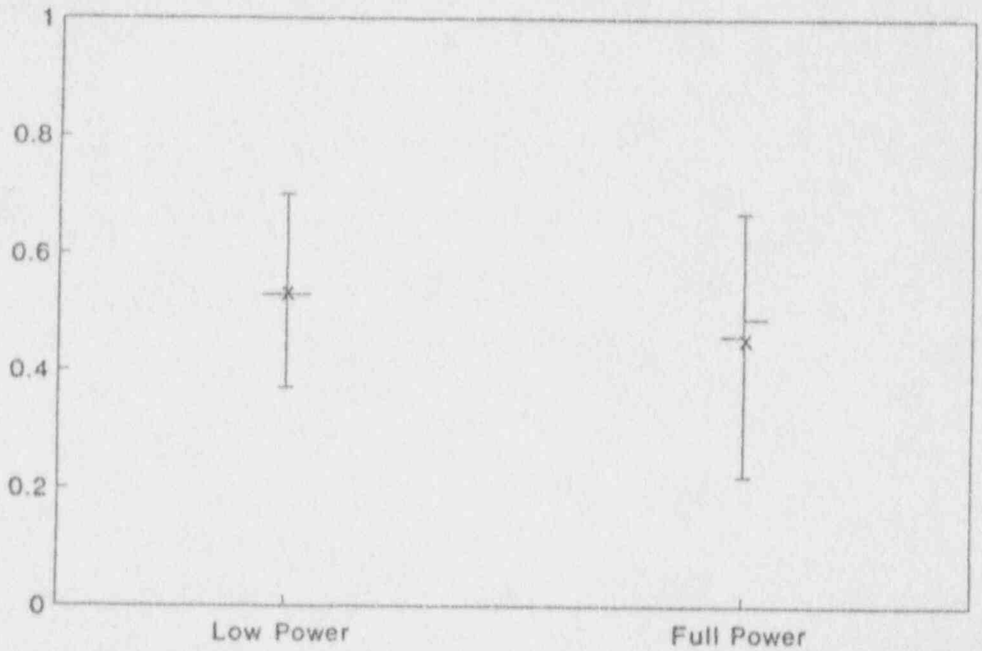

Table 2.1 Accident Timing (MELCOR Calculation with 24 Hours from Shutdown)

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 should address uncertainty and wherever possible, the NUREG-1150 distributions of source terms would be used to calculate the source terms from an accident during mid-loop operation. The parametric code, SURSOR,[4] that was developed in NUREG-1150 for Surry, was used to define source terms in the present study. Two measures were taken to assure the adequacy of the source terms: The first involved comparing the calculations from MELCOR with the data used in and results from SURSOR. Second, an advisory group called the Source Term Advisory Group was established to provide guidance, and any additional information on modifying the SURSOR code for the present study.

3.2 Description of Parametric Model

The SURSOR code, together with its associated distributions from NUREG-1150, was selected as the basis for source term definition. This section briefly discusses the SURSOR code, its evaluation (for modification, if required), and the final parametric model used.

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 formal expert elicitation.[5] A distribution, instead of a single value, was assigned to each parameter to address uncertainty. Considering the differences between full power and shutdown operations, the Source Term Advisory Group identified two parameters in SURSOR as important and possibly different than the values used in NUREG-1150. The first parameter is the fraction of the fission products in the core that are released to the vessel before vessel breach. The second parameter is the fraction of the fission products released to the vessel that are subsequently released to the containment. The distributions of these two parameters as defined in NUREG-1150 were compared with 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) using eleven characteristics. Table 3.2.1 shows the APBs and the attribute assigned to each of the 11 characteristics for the APBs derived in Section 2 that would cause significant off-site fission product releases. Because the containment cannot retain pressure during a mid-loop operation, even if it is closed, the two characteristics related to containment conditions, namely, the time and size of containment failure, are assigned attributes "early" and "leak," respectively. The mode of vessel breach is assigned either as "no vessel breach" or as a bottom-head failure, according to the definition of the APBs (Section 2). The containment spray condition also is assigned a value according to APB definition, and the mode of core-concrete interaction 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.

SURSOR was used to predict the release fractions of fission product for the five 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 source term 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. These figures also show the median (50 percentile) and mean values of the release fractions from the 200 observations, and the calculations from the MELCOR cases that are related to the individual APBs. Figures 3.2(a) and 3.2(b) give the distributions (the range and the median value) of FCOR (fraction of fuel inventory released to vessel) and FVES (material leaving the RCS as a fraction of those which are entering the RCS) used in SURSOR, and the calculated values from MELCOR for three cases. As shown in Figure 3.2(a), the difference between the MELCOR calculations for two cases with different accident-initiation times (24 and 72 hours after reactor shutdown) is not significant, and the values predicted for a core-recovery case 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.

These figures shows that generally, the MELCOR values fall within the ranges of SURSOR predictions. Although, for some radionuclide categories, the MELCOR values are closer to the upper ranges of the SURSOR predictions, they can be attributed to uncertainties; there are no apparent reasons to modify the SURSOR distributions. Consequently, the Source Term Advisory Group did not recommend any change to the SURSOR code for source term predictions.

3.3 Results

To limit the number of MACCS calculations, and 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 APBs for off-site consequence calculations.[*] When combined with the two time parameters defined in Section 2 (associated with drained maintenance and refueling), this gives 38 source terms for each APB.[†] Figure 3.3 shows the ranges, median, and mean values for the release fractions of all the selected source terms. For comparison, the figure shows the values calculated by MELCOR for a core-recovery case and a vessel breach case.

In addition to release fractions, a complete description of a source term requires 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 off-site emergency response (e.g., evacuation). Table 3.2 presents the mean values of the release fractions for the nine radionuclide categories, the release time (time when fission product release begins), and the duration of release. Both the release times and durations in Table 3.2 are obtained from MELCOR calculations. Since the release time is measured from accident initiation, it is not suitable for calculating radioactive decay after reactor shutdown, but can be used to determine the timing for emergency activities. The warning time for off-site emergency response is the time between the notification to the public, and fission product 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 fission product release is assumed to be at ground level for all bins.

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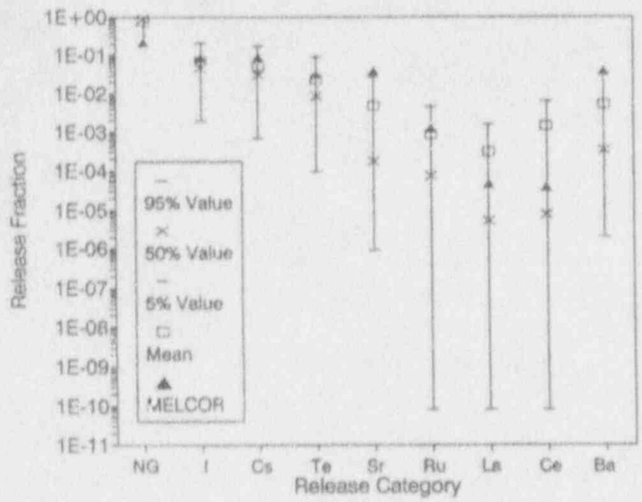
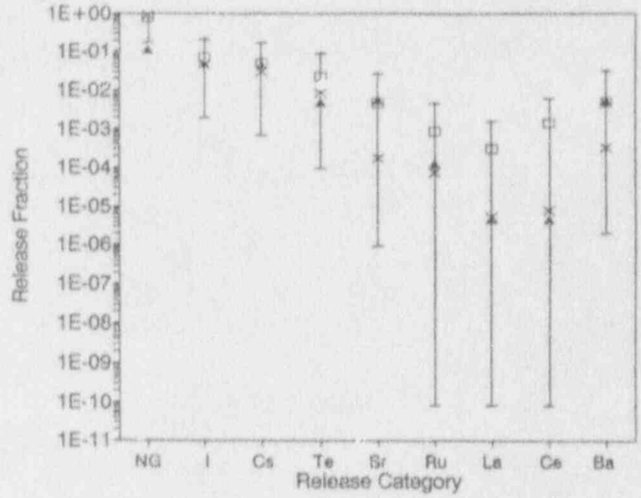
[*] This is the minimum number of STs to provide a 5% to 95% range. However, because of the low confidence level associated with such a small sample, they are simply referred to as the upper and lower limits of the calculations, with no percentiles associated with them.

[†] Source terms for APB-4 through APB-8 are given in this section for later MACCS calculations. Source terms for APB-1 through APB-3 are not given because they are not expected to cause significant off-site consequences (Due to early recovery, there is no core damage for APB-1 and only cladding damage for APB-2 and APB-3.). Also, the source terms for APB-4 and APB-5 obtained in this section are the same (thus, they are combined in the off-site consequence calculations). This is because the containment sprays, which are assumed to be recovered after the recovery of core injection, have a negligible effect on in-vessel fission product release, and, consequently, do not affect fission product release to the environment for this group of source terms.

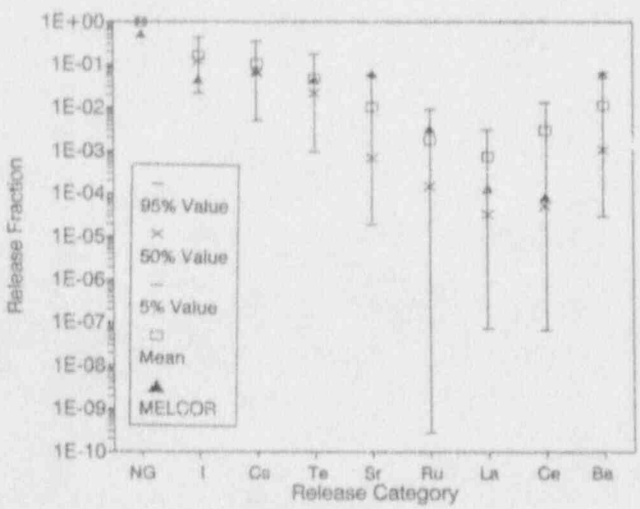
An important parameter in the source term definition, which is not considered in a full power analysis, is the time of accident initiation from reactor shutdown. This parameter determines the inventory available for release at accident initiation. The extended time period between accident initiation and reactor shutdown for an accident during the low power operation will result in significant radioactivity decay, and consequently, much reduced fission product inventory available for release. 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 3.4 shows the ranges, median, and mean values of this parameter used for both drained maintenance and refueling.



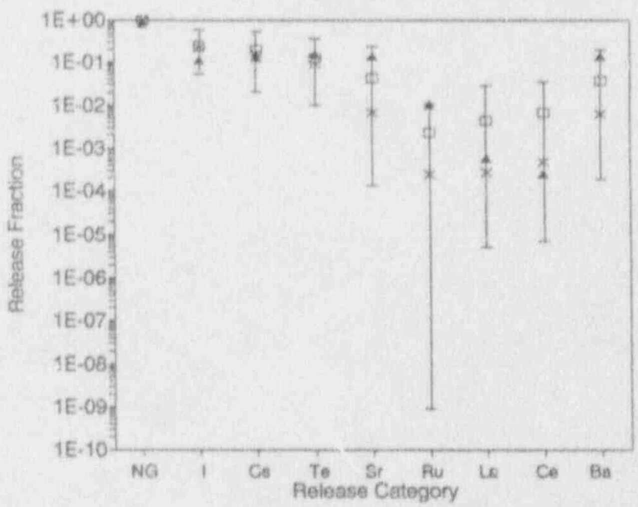
(a) APB 4: No VB, Early Containment Leak, Containment Spray



(b) APB 5: No VB, Early Containment Leak, No Containment Spray

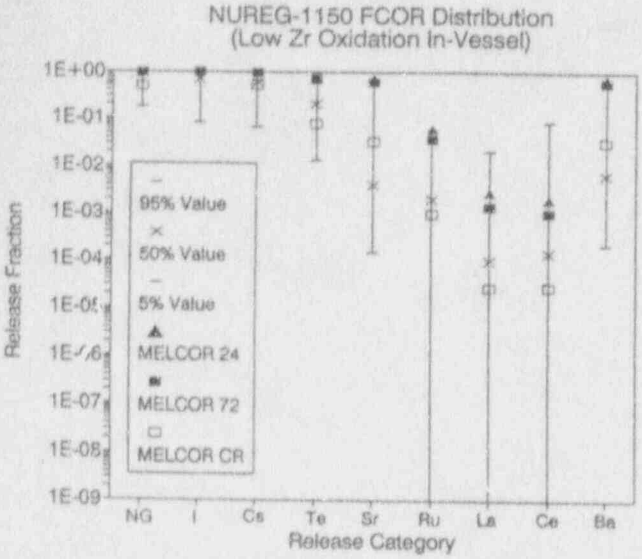


(c) APB 6: VB, Early Containment Leak, Containment Spray, and Deep Wet Cavity

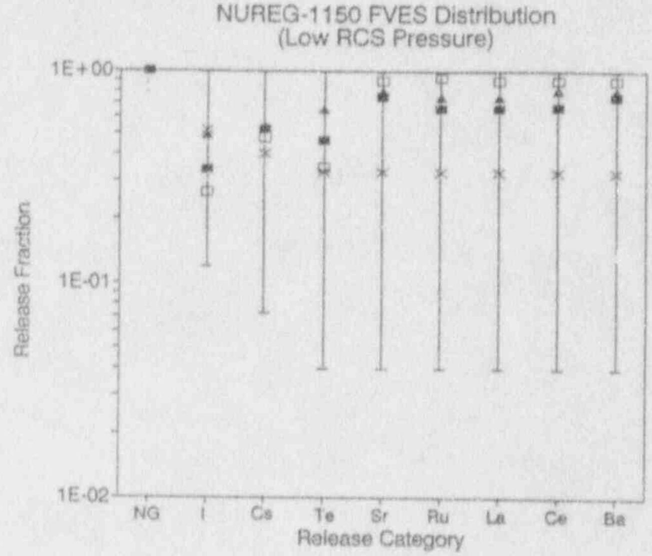


(d) APB 7: VB, Early Containment Leak, No Containment Spray, and Shallow Wet Cavity

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



(a) Fraction of fission products in the core that are released to the vesse'.



(b) Fraction of fission products in the vessel that are released into the containment.

Figure 3.2 Comparison of Important Values of Used in SURSOR and Calculated by MELCOR

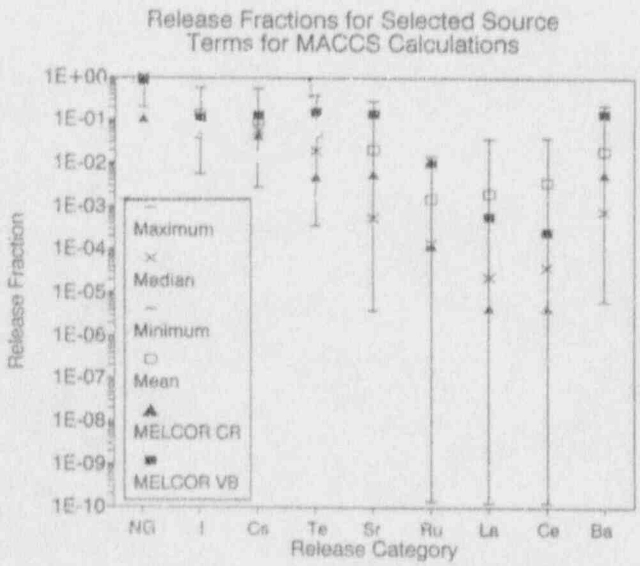


Figure 3.3 Comparison of the Release Fractions Used for MACCS Calculation (Maximum, Median, Minimum, and Mean) and Calculated by MELCOR

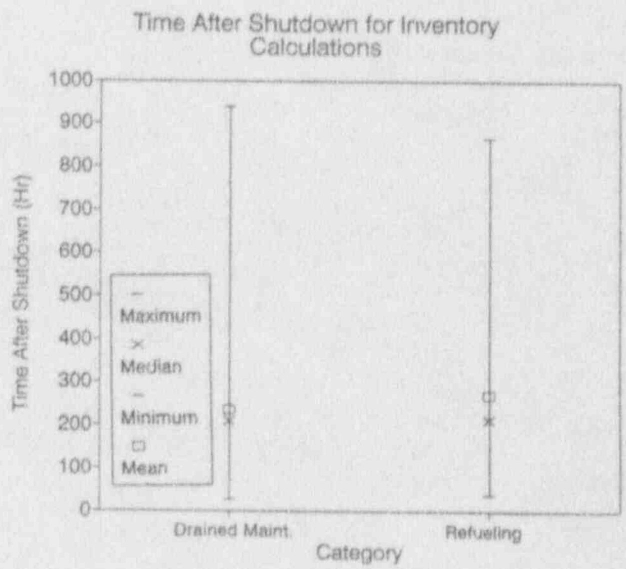


Figure 3.4 The Time of Acciden. Initiation After Reactor Shutdown for Radioactivity Inventory Calculation

Table 3.1 Low Power/Shutdown APBs for Source Term Calculations

APB #	APB ID[1]	1	2	3	4	5	6	7	8	9	10	11
		CF Time	Spray	CCI Mode	RCS Pres	VB Mode	SGTP	CCI Size	Zr Oxide	HPME	CF Size	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

The total on-site dose rate is a sum of the inhalation and cloud exposure dose rates based on the radionuclide concentration in the wake region of a building.

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,[10] Wilson,[11] Reg. Guide 1.145).[12] The on-site dose rate was estimated using the wake centerline concentration models: Ramsdell,[10] Wilson,[11] and Reg. Guide 1.145.[12] Brief descriptions of each model follow.

Ramsdell Model

The Ramsdell model[10] calculates the concentration in the far-region of the wake by including the effects of the lateral and vertical diffusion due to background turbulence:

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

assuming 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):

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

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, a multiplier of 5.0 was used to calculate the ground level release (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[12] 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 was calculated 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, $U = 1.2\ m/s$.

Wind speed 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 is based on the Reg. Guide 1.145 prediction, limited from above by the values predicted by the Wilson model. The results in Fig. 4 r the dose rate (Rem/h) indicate a variation of about two orders of magnitude as a function of the source term. The on-site 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 the containment suggest that a careful examination should be made of on-site evacuation schemes to limit the consequences.

4.2 Off-site Consequences

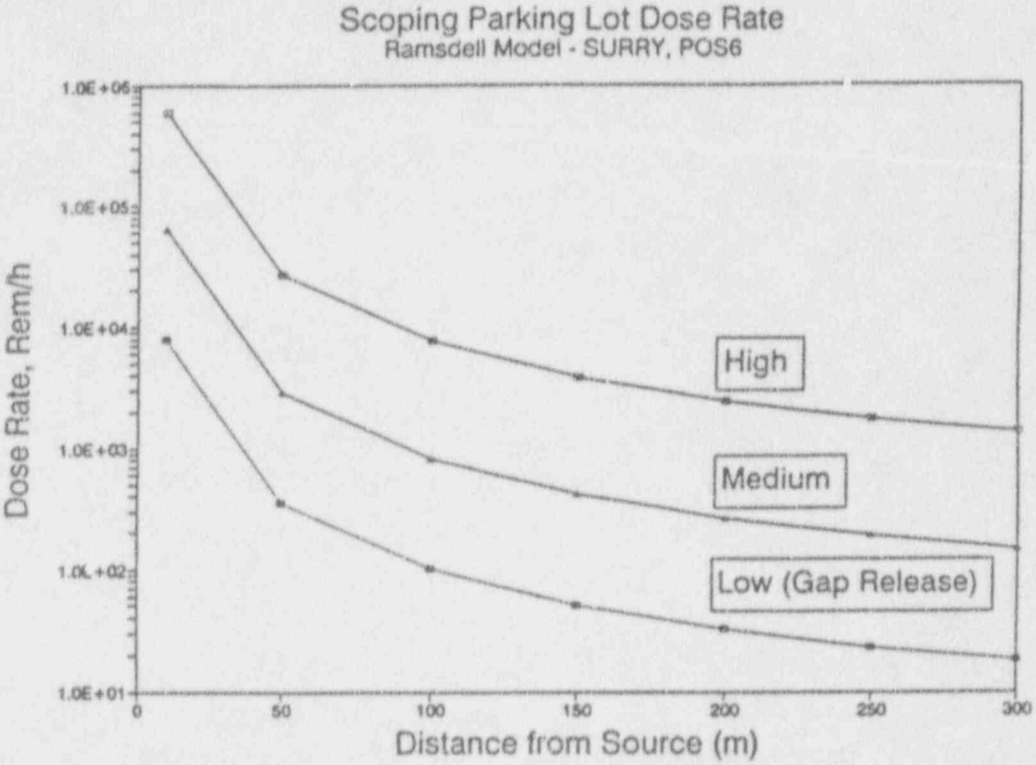
MACCS[5] calculations of the off-site consequences were made for all the source terms of POS6 generated by LHS sampling[3] of the SURSOR results. There were nineteen sample groups (one for Drained Maintenance and one for Refueling outages), each containing four distinct sets of release fractions for the nine radionuclide groups represented in the MACCS calculations. The time of release for each group was determined using the LHS technique. Then, the initial inventory for each source term was calculated using a logarithmic interpolation between the two closest data points. The inventories for various times after shutdown were taken from Denning et al.[13]

The following additional information/assumptions were used:

- Release power: 1.0 MW (sensitivity calculations with 0.0 MW).
- Release elevation: 28' (8.54 m).
- 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 calculations predict the highest number of early fatalities for Bin 7. The number of early fatalities is very small (less than 10^{-2}), even for the most severe accidents involving failure of the vessel's bottom head.

Figure 4.1 On-Site Parking-Lot Dose Rate



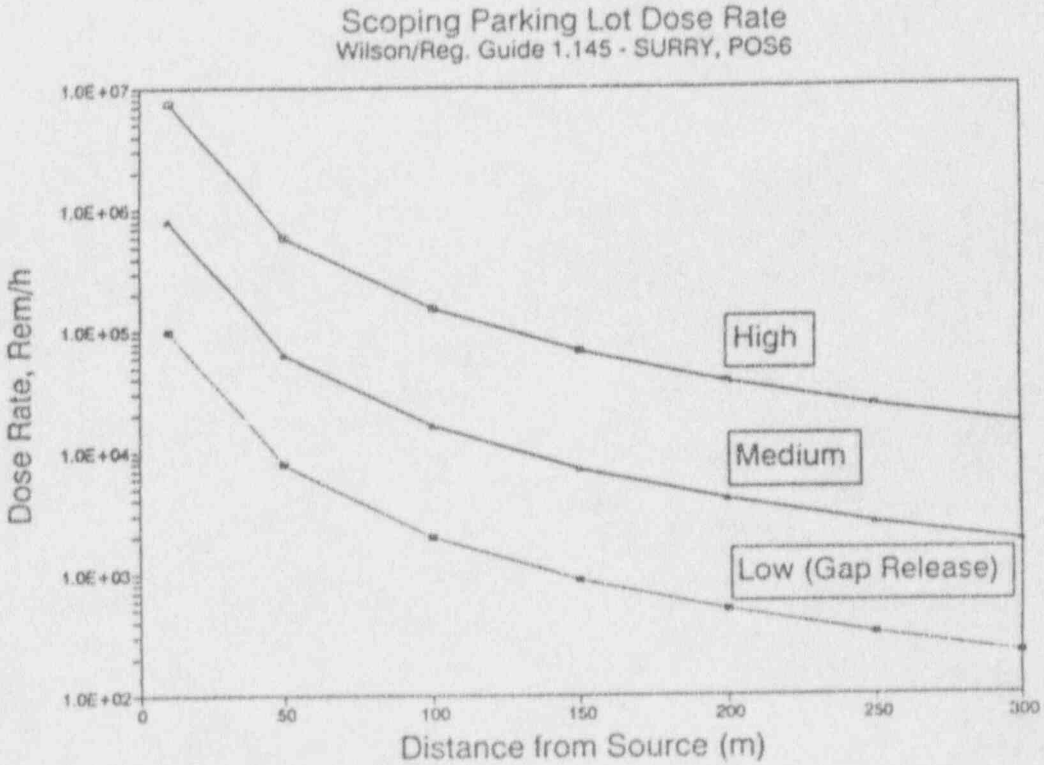
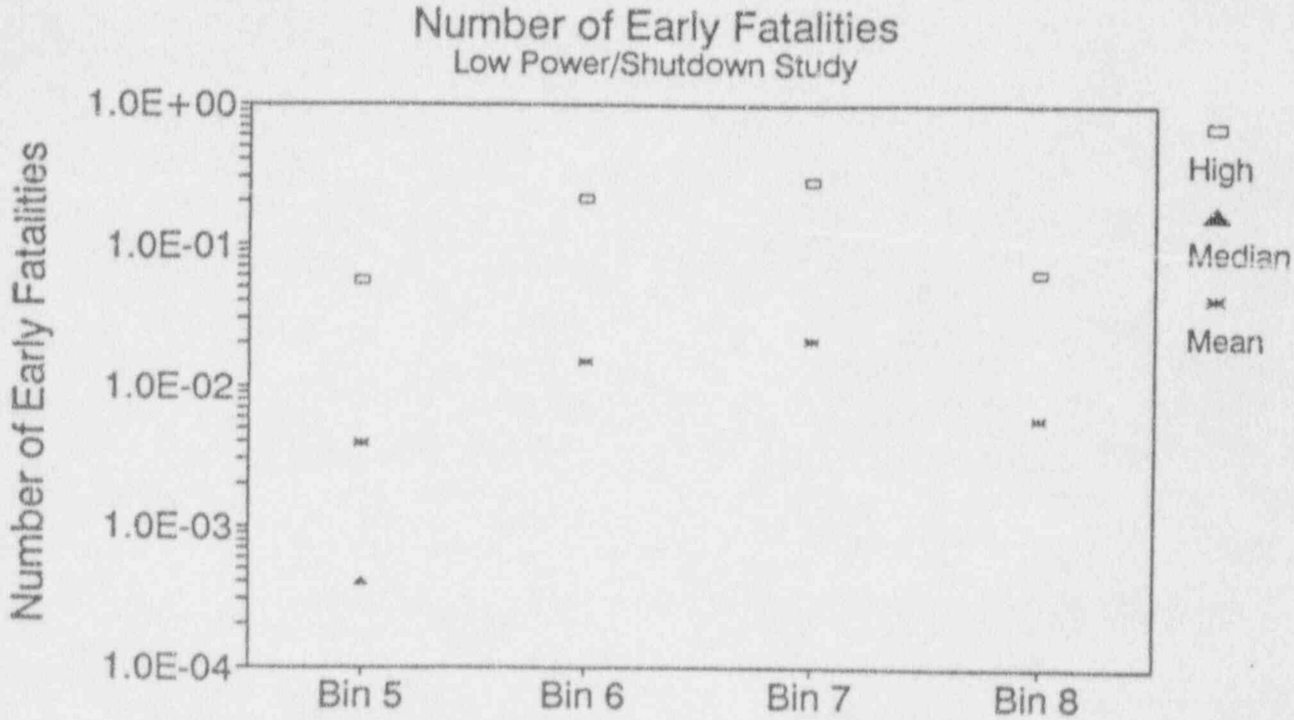
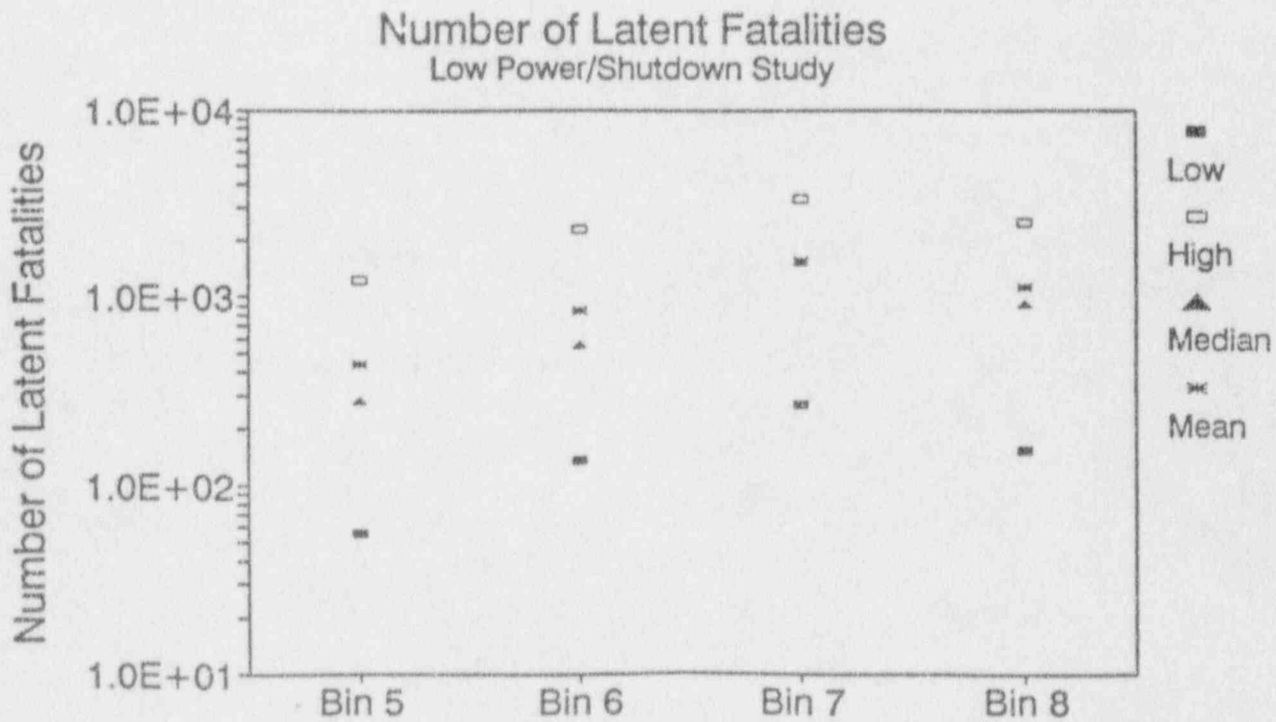


Figure 4.2 Off-Site Consequences Calculated by MACCS





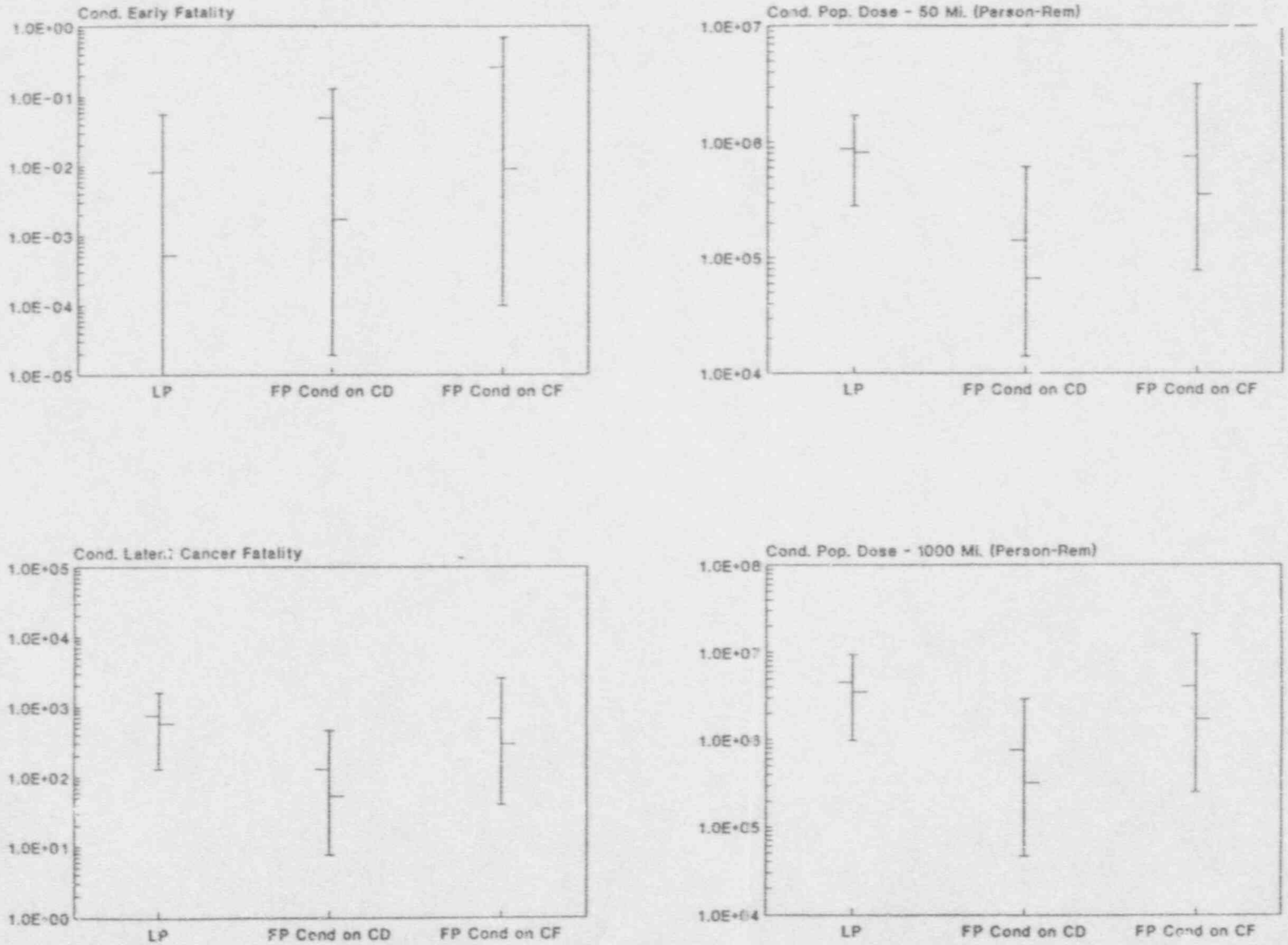
5 Integrated Risks Conditional on Core Damage

Once the consequences are calculated for each of the release bins, conditional risks can be evaluated by combining the accident progression analysis, source term analysis, and consequences. 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 they were unavailable, the risks 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 four risk measures (conditional on core damage), the early fatalities, late cancer fatalities, the population dose at 50 miles, and the dose at 1000 miles, calculated for the POS 6 at Surry. The upper and lower bounds shown in the figures do not represent any particular statistical measures because the number of samples was too small to attach any statistical significance to them. However, if sufficient samples are used, these bounds are expected to asymptotically approach the 5th and 95th percentiles. For comparison, the figure show the same risk measures for full power operation at Surry from the NUREG-1150 study; for this comparison, these results were converted to risks conditional on core damage and containment failure. The 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 will have had a longer time to decay and the species which have the greatest influence on the early fatalities generally have shorter half lives.

The figure also shows 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 because 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. These comparisons are conditional on core damage or containment failure, i.e., assuming the same frequencies of core damage or the same probability of containment failure. However, the 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 different at low 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 Conclusions

The abridged risk study has shown that during shutdown a severe release can occur with conditional long-term consequences approaching those of full power operation. In the mid-loop operation, POS 6, where the RCS inventory is less than half of the full inventory, the loss of RHR can proceed 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 several factors including 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 this time frame. Thus, the containment is expected to leak 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 cladding, the reactor vessel pressure boundary, and the containment. During shutdown operation and especially in the mid-loop condition based on currently available information, no credit could be assigned to the containment as a barrier once the fission products are released to the containment. 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 operational condition of the plant. Then, once the core damage occurs, the only possible mitigation (apart from containment sprays whose availability also is questionable) is 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 predictions of off-site consequences, which show almost no early fatalities, confirm this conclusion. However, these results also show that the conditional risk of long-term health effects due to long-lived isotopes, such as cesium, could be as severe, if not worse, as the corresponding risks at full power, mainly because the containment barrier is not present. The ultimate risk significance of the conditional results, however, depends on the frequencies of the accident sequences leading to core damage. Currently, these frequencies are being calculated in the Level 1 analysis. If the core damage frequency during low power/shutdown is about the same as at full power, then the result of this study show that probabilistic risk analysis of reactor accidents needs to be extended, in general, to cover the risk during shutdown operation.

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