BNL-NUREG--47564

DE93 008052

RISK ANALYSIS OF RELEASES FROM ACCIDENTS DURING MID-LOOP OPERATION AT SURRY*

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ABSTRACT

Studies and operating experience suggest that the risk of severe accidents during low power operation and/or shutdown (LP/S) conditions could be a significant fraction of the risk at full power operation. Two studies have begun at the Nuclear Regulatory Commission (NRC) to evaluate the severe accident progression from a risk perspective during these conditions: one at the Brookhaven National Laboratory for the Surry plant, a pressurized water reactor (PWR), and the other at the Sandia National Laboratories for the Grand Gulf plant, a boiling water reactor (BWR).

Each of the studies consists of three linked, but distinct, components: a Level 1 probabilistic risk analysis (PRA) of the initiating events, systems analysis, and accident sequences leading to core damage; a Level 2/3 analysis of accident progression, fuel damage, releases, containment performance, source term and consequences–off-site and on-site; and a detailed Human Reliability Analysis (HRA) of actions relevant to plant conditions during LP/S operations. This paper summarizes the approach taken for the Level 2/3 analysis at Surry and provides preliminary results on the risk of releases and consequences for one plant operating state, mid-loop operation, during shutdown.

I. Introduction

The objective of this study is an abridged risk analysis of the progressions (Level 2 analysis) and the consequences (Level 3 analysis) of accidents during low power and shutdown operation at the Surry plant. 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 the time of this study because the frequencies had yet to be determined in the companion Level 1 study. A limited level of uncertainty has been taken into account in a manner consistent with the detail of the abridged study.

The focus of the study was on a single plant operating state, POS 6, when the plant is in mid-loop operation. In the Phase 1 Level 1 screening analysis,[1] this POS was identified as having a special vulnerability due mainly to the reduced inventory.

II. Accident Progression Analysis

A. Approach

Following core damage in a severe accident, the accident progression is usually analyzed by using an Accident Progression Event Tree (APET). Quantification of the APET involves modeling of the physical processes occurring in the vessel and containment during the various accident sequences, the availability and status of safety equipment which could be used to mitigate the severity of the accident, and the assessment of the capability of the containment to retain the fission products when subjected to severe accident

*This work was performed under the auspices of the U. S. Nuclear Regulatory Commission.

loads. The number of questions in a APET can vary depending on the details desired, and the number of relevant and important phenomena to be modeled. The accident progression analysis for the Surry plant carried out for the NUREG-1150 program,[2] a PRA of the plant at full power, was used to identify the behavior of the Surry containment under accident conditions. The NUREG-1150 study showed 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 Surry containment succeeds in retaining the fission products most of the time (except by very late basemat melt-through) for accidents at full power. There is no reason to believe that the containment, if closed, would be more vulnerable during LP/S operation where the decay heat is significantly less and the reactor pressure is generally low.

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. The 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. Depending on the type of outage, at Surry, the time to enter POS 6 after shutdown ranges from one day to about 20 days and the duration of POS 6 varies from 10 hrs to more than one month. These times were selected as uncertainty parameters to be varied in the sampling process using the Latin Hypercube Sampling method.[3] To determine the timing of key events in the accident progression such as core melt and vessel breach, several MELCOR code[4] calculations were performed using different times of accident initiation.

B. Plant Configuration

The plant configuration during the LP/S period can vary widely depending on the purpose of the outage. Furthermore, a large degree of uncertainty exists for the operational state and availability of plant systems and components. 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.

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. There is no requirement under the existing plant technical specifications at Surry[5] to have any of the containment sprays available once the plant enters the residual heat removal (RHR) entry condition. It is possible that all of the containment sprays could be out of service and would not be 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[6] 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 retain the pressure which could be generated during the course of a severe accident and prevent release of fission products. This is due primarily to the presence of a temporary restraining plug in 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. This feature considerably simplified the APET; since the integrity of containment is assumed to be lost at accident initiation, many questions normally needed to assess the potential for containment failure are no longer relevant.

C. Phase 1 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 performed earlier for the Surry Unit 1 plant.[1] 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 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 the residual heat removal (RHR) system and loss of off-site power. Operating experience at nuclear power plants indicated a relatively high incidence of loss of RHR.[7] 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.

D. 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 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 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. 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. After 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 the vessel is assumed to be breached by the core debris. 'Never' indicates no core cooling recovery at all. Table 2 shows the timing of core melt progression as calculated by the MELCOR code for an accident initiated 24 hours after shutdown. MELCOR calculations were performed for several different times of accident initiation. Since this time can vary widely in POS 6, the time of accident initiation was treated as a random variable and 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. 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[9] and hardware availability for each of the time periods. The hardware availability was based on the data used in the screening Phase 1 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. 1.

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, 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 implications. Due to the limited time available for the abridged study, quantification of these releases was not carried out. 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. A comparison of the conditional probability of core damage arrest before vessel breach for the LP/S analysis with the full power analysis of NUREG-1150 at Surry showed that the vessel is not breached approximately half of the time given core damage for both low power and full power accidents.

III. Source Term Analysis

The parametric source term (ST) code, SURSOR,[10] that was developed in NUREG-1150 for Surry, was 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/S accidents with the data used in SURSOR (as well as the calculational results obtained from SURSOR). The second involved the establishment of a Source Term Advisory Group to provide guidance, and additional information if necessary, on possible modifications to SURSOR for LP/S conditions. The Source Term Advisory Group, based on a consideration of the differences between full power and LP/S operations, identified two parameters in SURSOR as possibly different (than the values used in NUREG-1150) for LP/S 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 the values to be used in the present study.

SURSOR was used to predict radionuclide release fractions for the five LP/S Accident Progression Bins (APBs) labelled as Bin #4 through Bin #8 in Fig. 1. Two hundred sets (or observations) of release fractions were produced for each of the five bins to address ST uncertainty. 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 off-site emergency response (e.g., evacuation). Table 2 presents the mean values of the release fractions for the nine radionuclide categories, the release time (i.e., the time when release begins), and the release duration. Both the release times and the release durations presented in Table 2 were obtained from MELCOR calculations.

The MELCOR calculated release fraction 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.

To limit the number of consequence calculations, and at the same time to provide a range of uncertainty, 19 source terms were (randomly) selected for each of the five APBs. This, when combined with the two time parameters defined in Section II (associated with drained maintenance and refueling), provides 38 source terms for each bin for the consequence calculations.

One of the most important parameters in the LP/S 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 radionuclide inventory available for release at accident initiation. Because of its importance, it is treated as one of the uncertainty parameters in the present study. The actual inventories for various times following shutdown were obtained from runs of the ORIGEN2 code for Surry.[11] A randomly selected value of time (and corresponding inventory) were assigned to each source term defined in this section.

IV. Consequence Analysis

Two sets of consequence calculations were performed for this study.

Offsite consequences, including early fatalities, population dose, and latent fatalities, were calculated using the MACCS code.[12] The input assumptions on meteorology, site data, emergency response, etc., required by MACCS, were the same as those used in the NUREG-1150 consequence analysis for Surry. The new data needed were the radionuclide release fractions and the initial inventories (as determined by the time of release) for each source term group. As outlined above, the time of release for each group was determined using the LHS technique, while the inventories for various times after shutdown were taken from ORIGEN2 code calculations for Surry.[11]

In addition to the offsite consequences, a scoping calculation of onsite dose rates (designated as the Parking Lot Dose Rate, PLDR) in the vicinity of the plant, following release, was performed in this study. The PLDR was calculated as a sum of the inhalation and cloud exposure dose rates based on the radionuclide concentration in the wake region of the containment building using three different models for the building wake centerline concentration, due to Ramsdell,[13] Wilson,[14] and Reg. Guide 1.145,[15] respectively. The scoping calculations were performed for three sets of source terms referred to as "High", "Medium", and "Low (Gap Release)", respectively, and used conservative values of weather stability and wind speeds at Surry.

V. 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 2 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, and late cancer fatalities, and the population dose out to 50 and 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. (In the NUREG-1150 study only about 20% of the core damage sequences result in containment failure).

The risk comparison shows that the early fatality risk of 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 LP/S core damage accidents are significantly different from those at full power, the integrated risk profiles will be dominated by those (Level 1) frequencies.

The results of the Parking Lot Dose Rates expressed in Rem/h, shown in Fig. 3 indicate a variation of about 2 orders of magnitude as a function of the source term. These 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-site evacuation schemes to limit consequences.

VI. 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 <u>conditional</u> long-term consequences approaching those of full power operation can occur. In the mid-loop operation, POS 6, 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 cladding, the reactor coolant pressure boundary, and the containment. During shutdown operation and especially in the mid-loop condition, little or 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 80 percent of the accidents, defense-in-depth at shutdown could be negated by the intrinsic operational condition of the plant. In this case, the most significant mitigation 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 in mid-loop operation the conditional long-term health effects due to the long-lived isotopes of cesium, etc. could in fact be as severe, as the corresponding results at full power, due mainly to the fact that the containment does not have a pressure retaining capability. The ultimate risk significance of the conditional results reported here, however, depends on the frequencies of the accident sequences leading to core damage. 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 probabilistic risk analysis of reactor accidents needs to be extended, in general, to cover the risk during LP/S operation.

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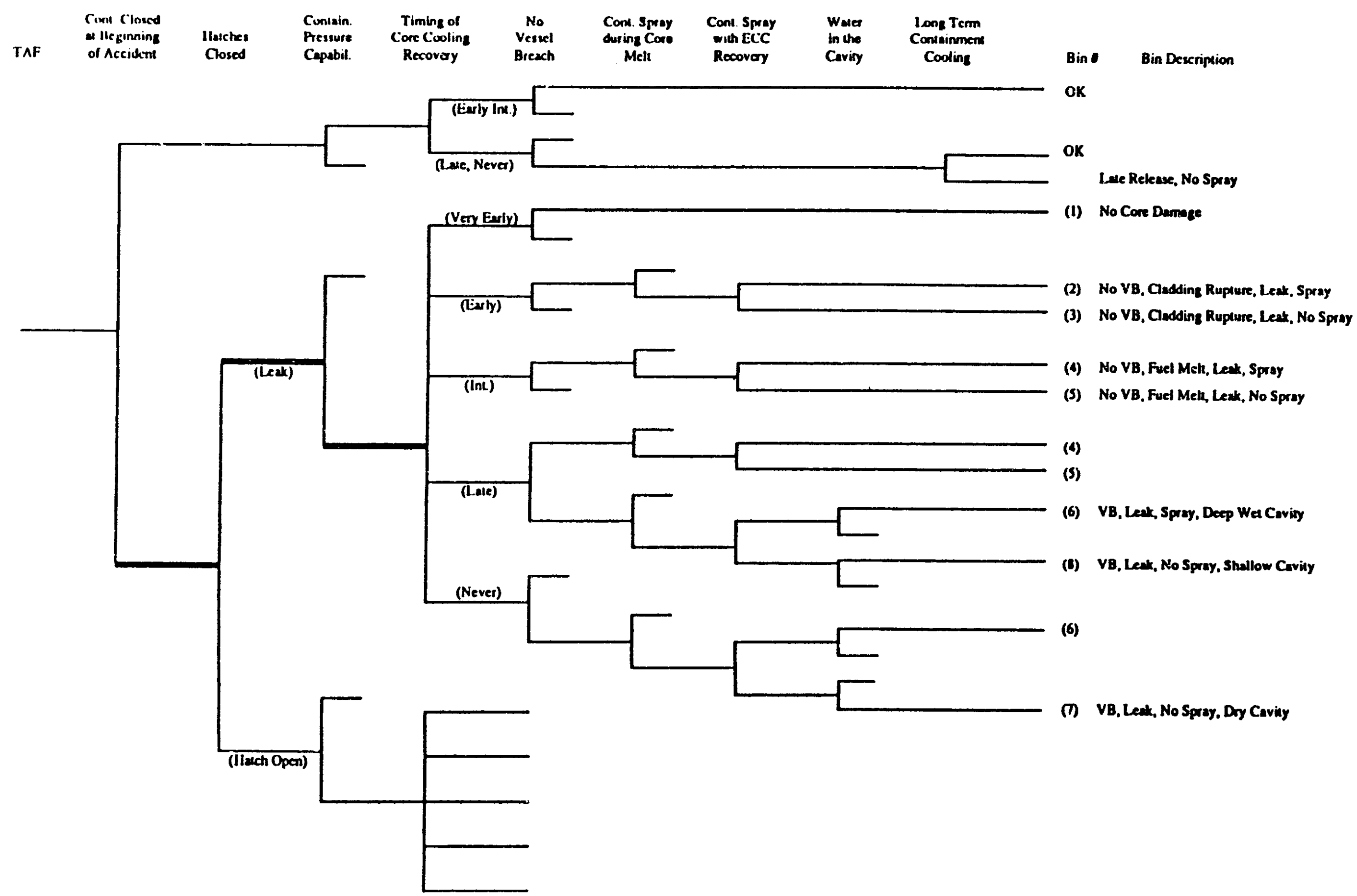
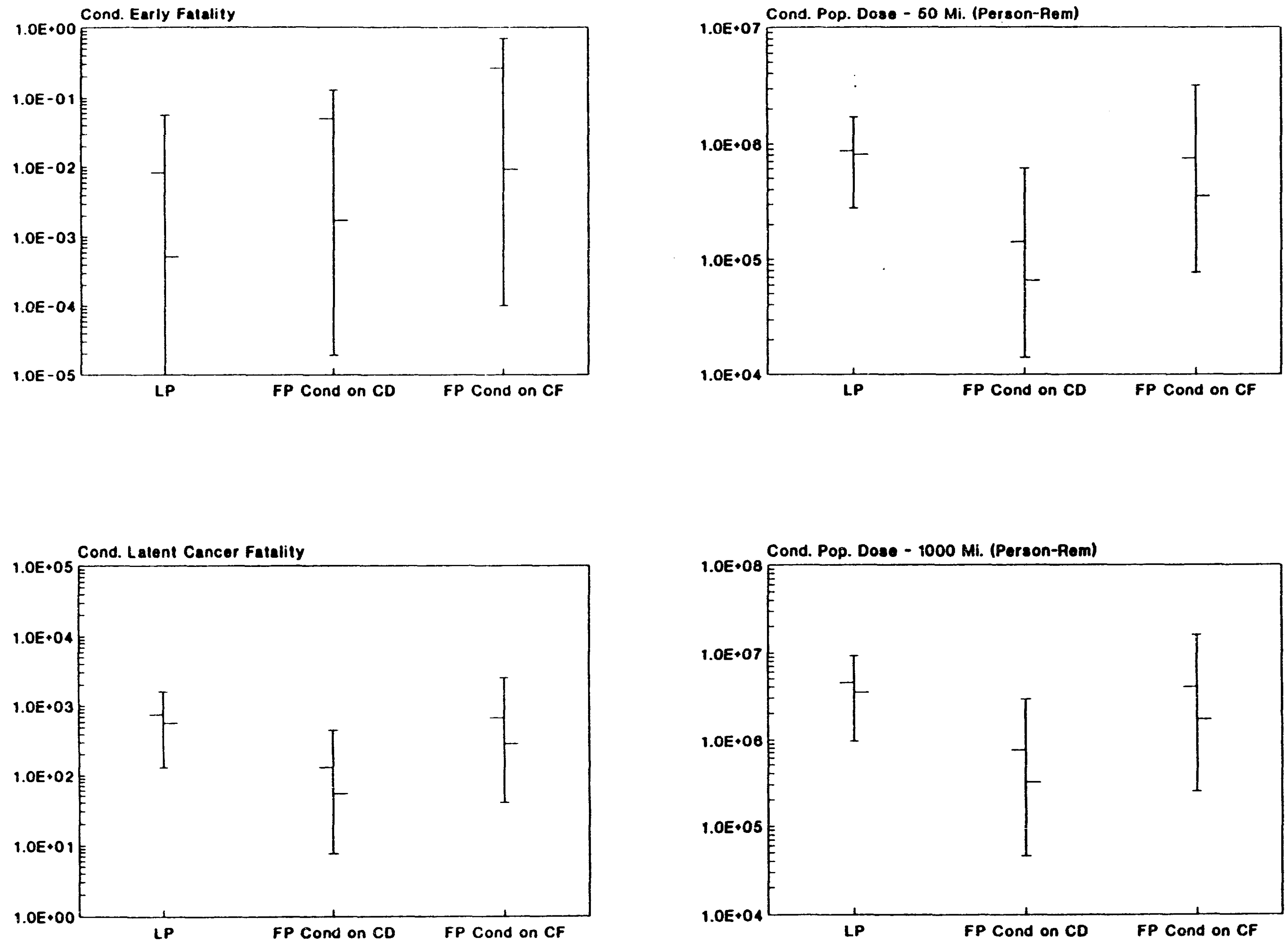


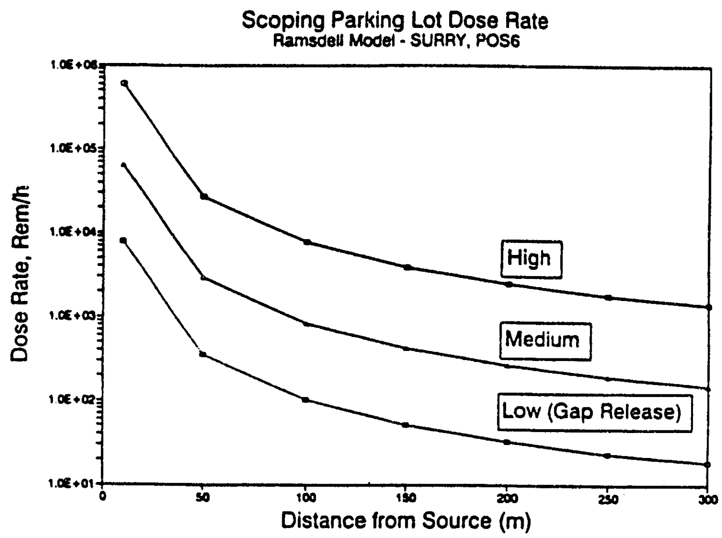
Fig. 1 Accident Progression Event Tree for the Abridged Low Power/Shutdown Risk Analysis



• FP results based on NUREG-1150.

Figure 2 Comparison of Risks Conditional on Core Damage

Figure 3 On-Site Parking Lot Dose Rate



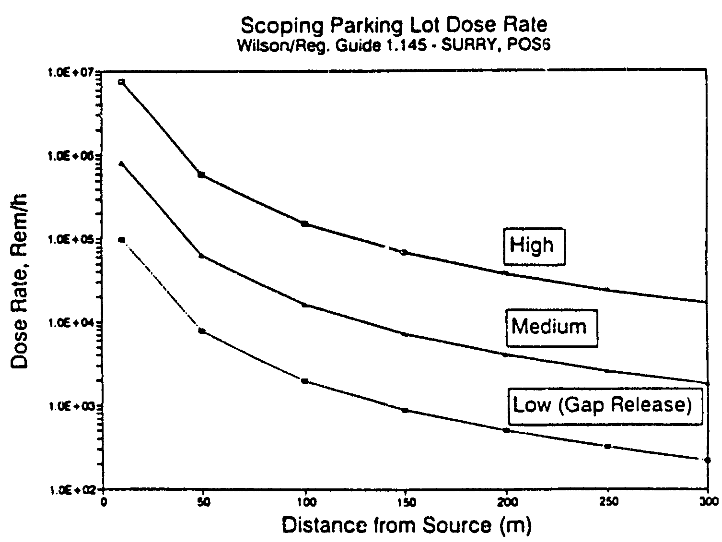


Table 1 Timing of Key Events in MELCOR Calculation
(Accident initiated 24 Hours after Shutdown)

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

APB 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

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

Table 2 Mean Release Fractions and Timing of Release

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