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ANALYSIS OF ACCIDENTS DURING THE MID-LOOP OPERATING STATE AT A PWR*

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ABSTRACT

Studies suggest that the risk of severe accidents during low power operation and/or shutdown conditions could be a significant fraction of the risk at full power operation. The Nuclear Regulatory Commission has begun two risk studies to evaluate the progression of severe accidents during these conditions: one for the Surry plant, a pressurized water reactor (PWR), and the other for the Grand Gulf plant, a boiling water reactor (BWR). This paper summarizes the approach taken for the Level 2/3 analysis at Surry for one plant operating state (POS) during shutdown. The current efforts are focussed on evaluating the risk when the reactor is at mid-loop; this particular POS was selected because of the reduced water inventory and the possible isolation of the loops.

The Level 2/3 analyses are conditional on core damage having occurred. Initial results indicate that the conditional consequences can indeed be significant; the defense-in-depth philosophy governing the safety of nuclear power plants is to some extent circumvented because the containment provides only a vapor barrier with no capability for pressure holding, during this POS at Surry. However, the natural decay of the radionuclide inventory provides some mitigation. There are essentially no predicted off-site prompt fatalities even for the most severe releases.

I. Introduction

The objective of this study is an abridged risk analysis of the progression (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 approximate estimates. In this abridged study risk refers to conditional consequences (defined as the probability of the various events which occur during the progression of the

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various accident sequences multiplied by the consequences) assuming that the core has been damaged. A limited level of uncertainty has also been taken into account in the level 2/3 analyses. An integrated risk estimate (calculated by multiplying the conditional consequences and the frequency of the core damage accident sequences) could not be made because the frequencies had not been determined in the companion Level 1 study.

Our focus 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 been potentially vulnerable due mainly to the reduced coolant inventory.

II. Accident Progression Analysis

A. Approach

Following core damage, the progression of the accident can be analyzed using an Accident Progression Event Tree (APET). Quantification of the APET involves modeling 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 also has to be determined. The assessment of the capability of the containment to retain the fission products when subjected to severe accident loads is also an important consideration. The number of questions in an 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 reported in NUREG-1150,[2] a probabilistic risk assessment (PRA) of the plant at full power, analyzed the behavior of the containment under severe accident conditions. The NUREG-1150 study showed that the major cause of release was due to containment bypass or by basemat melt-through. The probabilties of early containment failure (caused by various mechanisms) and late containment failure (resulting from gradual pressurization) were predicted to be very small. Thus, for severe accidents at full power, Surry's containment is expected to retain the fission products most of the time (except by very late basemat melt-through). There is no reason to believe that the containment, if closed, would be more vulnerable to severe accidents occuring 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 recover core cooling before core damage. The longer time from shutdown to release also potentially reduces the inventory of the fission products that are available for release. Therefore, it is very important to determine the time of the start of the accident relative to the time of shutdown. At Surry, depending on the type of outage, the time after shutodwn to enter POS 6 ranges from one day to about 20 days, and the duration of POS 6 varies from 10 hours to more than one month. These times were selected as parameters to be varied in the limited uncertainty study that was performed. The uncertainty study was performed by sampling from various uncertain parameters using the Latin Hypercube Sampling method (LHS).[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 done assuming various times for the start of the accident after shutdown.

B. Plant Configuration

The plant configuration during LP/S can vary widely depending on the purpose of the outage. Furthermore, there is a large degree of uncertainty related to the operational state and availability of plant systems and components. For this abridged analysis, we assumed that all the loops were isolated and the safety valves were removed for maintenance, which provides a vent path from the reactor coolant system (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 containment integrity and the 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, therefore, possible that all of the containment sprays could be out of service and would not be available during mid-loop operation. Therefore, the availability of sprays was used as one of the uncertainty parameters in this study.

The containment is closed during the mid-loop operation at Surry. However, after several discussions[6] with the Surry personnel, we determined that closure of the containment does not ensure that the containment can retain the pressure which could be generated during a severe accident and prevent the release of fission products. This is due primarily to the presence of a temporary restraining plug in the escape tunnel in the equipment hatch of the containment. This temporary plug has no overpressure capability. Therefore, for this study, the containment was assumed to leak during POS 6. This feature considerably simplified the APET; because the of containment is assumed to leak during the accident, 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 damage of the core for the internally initiated events were carried out earlier for the Surry Unit 1 plant.[1] The major objectives of this analysis were to provide initial insights into any particularly vulnerable plant operational states during low power/shutdown operations and to identify the set of major initiating events applicable to each POS. Based on this coarse screening analysis, we 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.[5] 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 an uncertainty parameter. For those accidents initiated by loss-of-power, recovery from the loss of power determines the probability of recovering the capability of the core cooling, and the termination of the accident.

D. Event Tree Analysis

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

Figure 1 shows the containment event tree used in this analysis. The first set of questions refer to the status of the containment. In this particular POS, the containment is assumed to be leaking from the start of the accident. Once the status of the containment is identified, the next question is on the timing of the recovery of the core cooling, which determines the extent of damage to the core. Arresting the degradation of the core before the vessel fails during a severe accident has the potential to significantly decrease the magnitude of the release of the fission products. The timing of the recovery of the capability of the core cooling was divided into five periods: Very early, Early, Intermediate, Late, and Never (no recovery). The timing of Very early extends to where core cooling is recovered without damaging the core. Early is recovery of cooling during the 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 stop the progress of the core melting without breaching the vessel. After consulting with the Source Term Expert Panel, we assummed this intermediate period to extend until 45% of the core melted. If core cooling is recovered during the Late period, the vessel is assumed to be breached by the core debris. Never indicates no recovery of core cooling at all. Table 1 shows the timing of the progression of the core melt as calculated by the MELCOR code for an accident occurring 24 hours after shutdown. MELCOR calculations were done for several different times of accident initiation. Since this time can vary widely in POS 6, the time of the start of an accident 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 an accident began, the timing of the progression of the core melt as calculated by the MELCOR code was adjusted by the decay heat to determine the time available for recovery of the core cooling. The probability of recovery was estimated based on the HRA recovery curve for human error,[7] the off-site power recovery curve[1], and the availability of hardware for each period. This availability was based on the data used in the screening Phase 1, Level 1 study.

The next questions in the APET address the availability of sprays and whether the cavity is dry or wet, which determines the extent of core-concrete interaction (CCI). The outcomes of the accident sequences in the APET were classified into eight bins, depending on the extent of damage to the core, breach of the vessel, and availability of the sprays as shown in Fig. 1.

This APET was applied to each of the major cutsets leading to core damage sequences identified in the level 1 study of the preliminary screening analysis. In this analysis, the damage of the core 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 damaging the core if the core cooling is recovered during the Very early period. However even during the Very Early period if the clad becomes embrittled during 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 have important consequences off-site, it could have significant on-site implications. Due to the limited time available for the abridged study, these releases were not quantified. In estimating the final risks conditional on damage of the core, only those accident sequences which were actually predicted to result in 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 arresting core damage before vessel breach for the LP/S analysis with the full power analysis showed that the vessel is not breached approximately half of the time given damage of the core for both low power and full power accidents.

III. Source Term Analysis

The parametric source term (ST) code, SURSOR,[8] that was developed in NUREG-1150[2] for Surry, was used as the basis for definition of the ST in this study.

Two additional efforts were undertaken to ensure 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 establishing 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 the definition of the LP/S source term. 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 used in this 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 were obtained from MELCOR calculations.

In general, the release fraction values calculated by MELCOR 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 for modifying 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. These, 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 is the time of the start of the accident from the reactor shutdown. This parameter determines the radionuclide inventory available for release at the start of the accident. Because of its importance, it is treated as one of the uncertainty parameters in this study. The actual inventories for various times following shutdown were obtained from runs of the ORIGEN2 code for Surry.[9] 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 carried out for this study.

Off-site consequences, including early fatalities, population dose, and latent fatalities, were calculated using the MACCS code.[10] The input assumptions on meteorology, site data, and emergency response, required by MACCS, were the same as those used for the consequence analysis for Surry in NUREG-1150.[2] 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.[9]

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 carried out in this study. The PLDR was calculated as a sum of the inhalation and cloud exposure dose rates based on the concentration of radionuclides in the wake region of the containment building using three different models for the centerline concentration of the building wake, due to Ramsdell,[11] Wilson,[12] and Regulatory Guide 1.145,[13] 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, 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 risks were calculated as conditional on core damage; i.e., the results are averaged over various accident progressions, given core damage.

Figure 2 shows the 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 latent cancer fatalities, and the population dose out to 50 and 1000 miles. (The upper and lower bounds do not represent any particular statistical measures, because the number of samples was not sufficiently large enough 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.) The results of the same risk measures for the full power operation at Surry from the NUREG-1150 study.[2] Also are shown in Figure 2 for comparison. These results were converted to risks conditional on damage of the core and conditional on containment failure for ease of comparison. (In the NUREG-1150 study only about 20% of the core damage sequences result in the failure of the containment[2]).

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 damage of the core or on failure of the containment). This result is expected because 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.

Figure 2 also shows that the latent cancer fatalities and population doses are higher than those predicted for the full power accidents conditional on damage of the core. However, these long term health effects are about the same for accidents conditional on the failure of the containment. This similarity is 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 the 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.

We emphasize again that these comparisons are conditional on damage of the core or failure of containment, 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 magnitudes 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 that on-site evacuation schemes be carefully examined to limit consequences.

VI. Insights and Conclusions

The abridged risk study, while preliminary and subject to confirmation in several 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 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 several factors, including the timing of the recovery of core cooling, and the availability of containment sprays. In POS 6, the isolation of the containment in the sense of achieving a pressure holding capability is 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, 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 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 results of the off-site consequences, 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 such as cesium, could, in fact, be as severe as the corresponding results at full power, due mainly because the containment does not have a capability for retaining pressure. The ultimate risk significance of the conditional results reported here, however, depends on the frequencies of the accident sequences leading to damage of the core. If the core damage frequency during low power/shutdown is the same magnitude as at full power, then the results 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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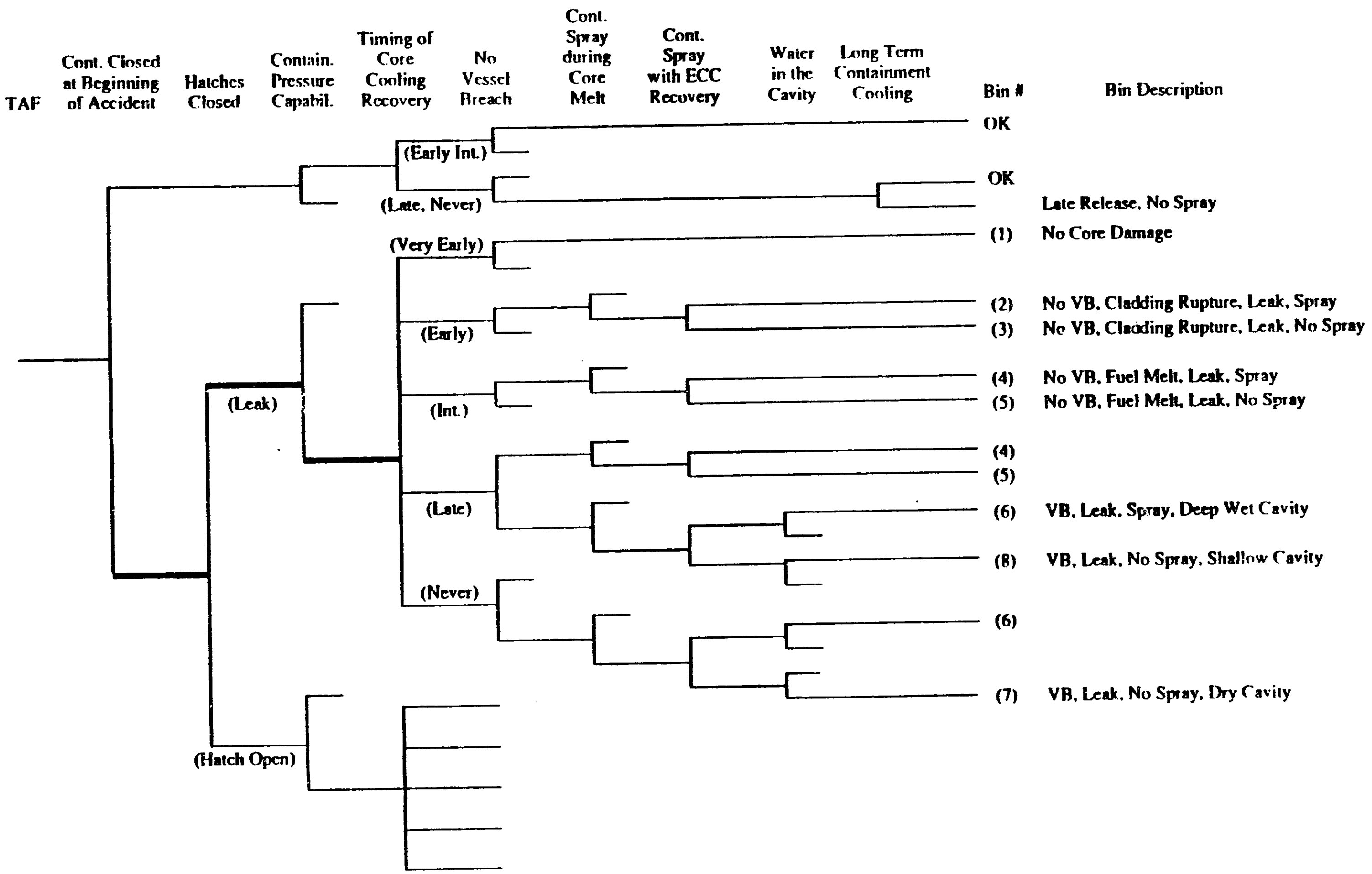
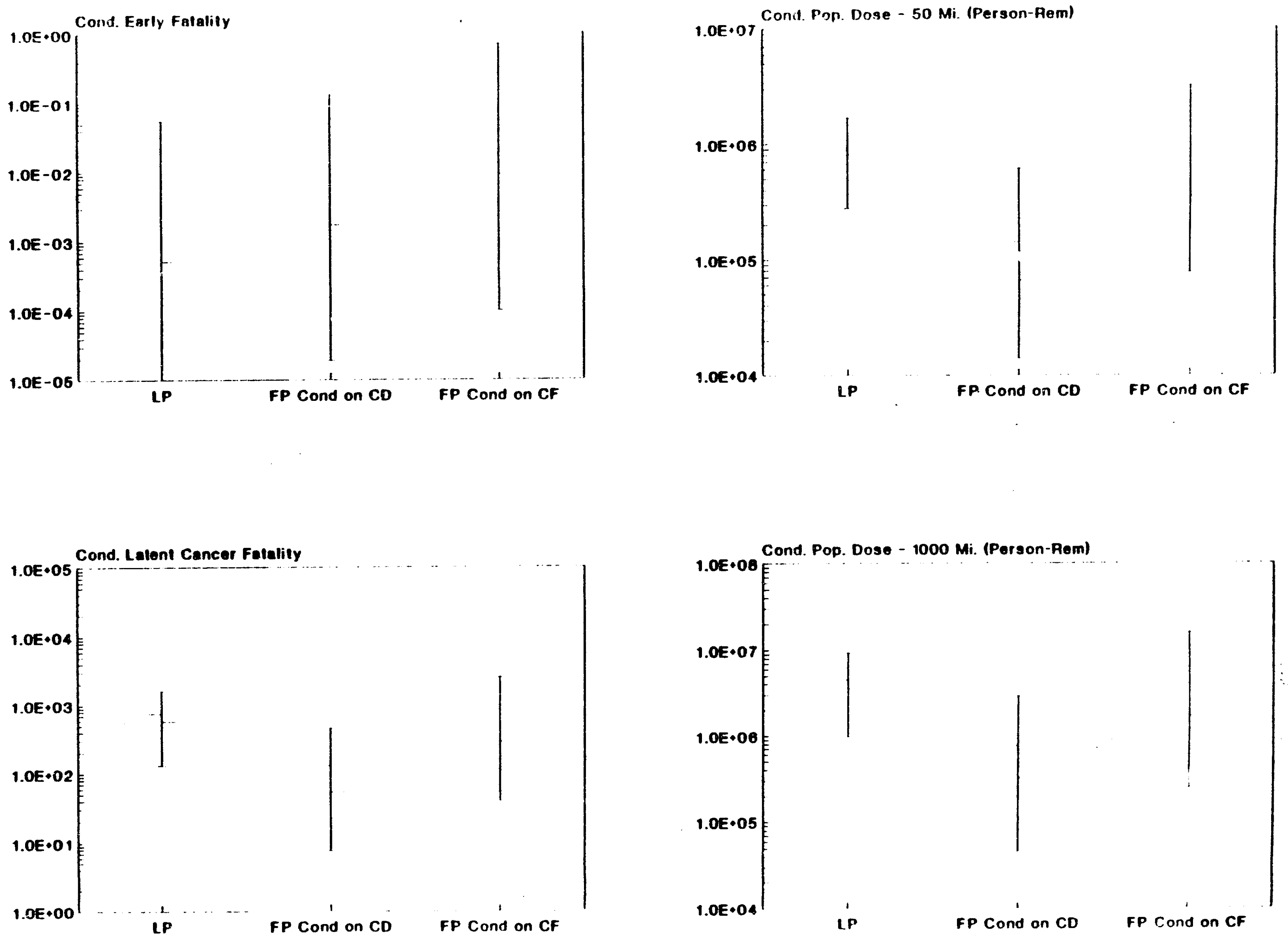


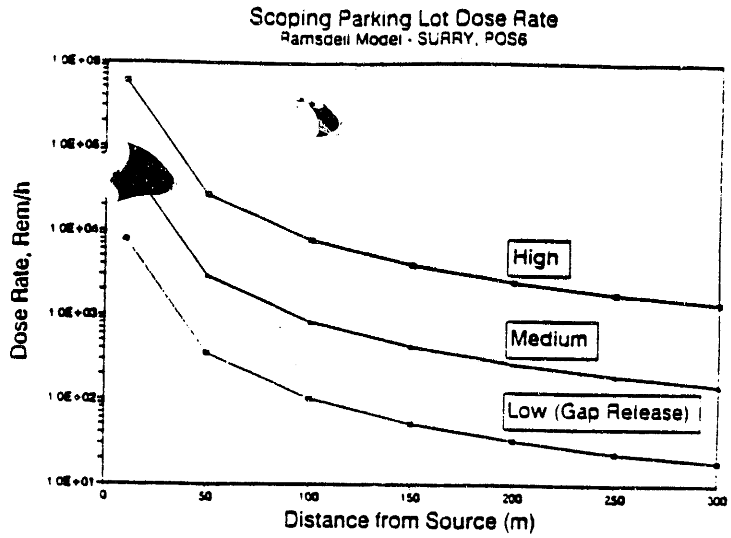
Figure 1 Accident Progression Event Tree for the Abridged Low Power/Shutdown Risk Analysis



- FP results based on NUREG-1150.

Figure 2 Comparison of Risks Conditional on Core Damage

Figure 3 On-Site Parking Lot Dose Rate



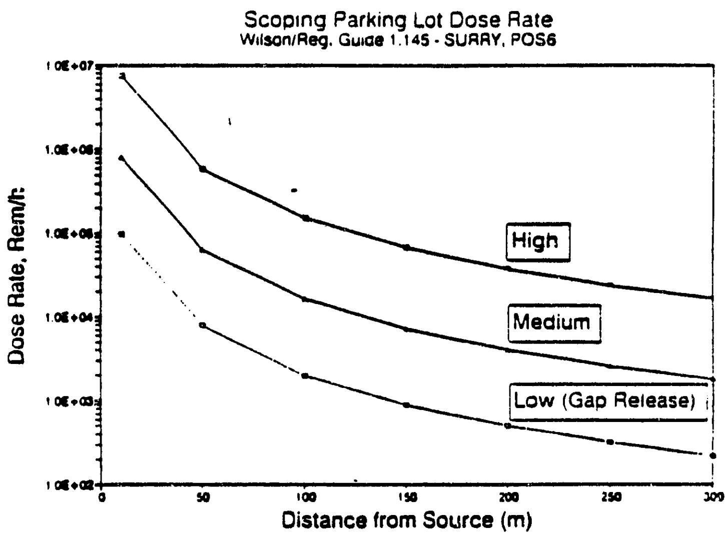


Table 1 Timing of Key Events in MELCOR Calculation (Accident Initiated 24 Hours After Shutdown)

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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