Fourth International Conference on Nuclear Engineering (ICONE-4), New Orleans, LA, March 10-14, 1996

BNL-NUREG-62012

CONF-960306--34

EVALUATION OF SEVERE ACCIDENT RISK DURING MID-LOOP OPERATION AT SURRY UNIT-1*

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1. ABSTRACT

In the past most probabilistic risk assessments (PRAs) of severe accidents in nuclear power plants have considered initiating events which could potentially lead to core damage and containment failure during normal full power operation. However, recent studies and operational experience during periods while plants were shutdown for maintenance or refueling indicated that potential accidents initiated during low power operation or shutdown conditions could also potentially become important contributors to risk. In 1989, the Nuclear Regulatory Commission (NRC) began an extensive program to assess the risk during low power and shutdown operation. Two plants, Surry (a pressurized water reactor, PWR) and Grand Gulf (a boiling water reactor, BWR) were selected as the plants to be studied.

This paper describes an analysis of accident progression and offsite consequences (level 3 PRA) carried out for the Surry plant. The focus of the level 3 PRA was on mid-loop operation, which is a plant operational state (POS) that can occur while the plant is shutdown for maintenance or refueling. Mid-loop refers to a configuration when the reactor coolant system is lowered to the mid-plane of the hot leg to allow essential maintenance to be performed. This operational state was selected after an initial coarse screening study indicated that reduced inventory during mid-loop operation could pose higher risk than other POSs.

The methodology used to perform the level 3 PRA was based on the methods developed for the NUREG-1150 full power study of Surry. However, the interfaces between the core damage frequency analysis and the plant damage states (PDSs) and the accident progression event tree (APET) had to be modified to reflect the plant configuration during mid-loop operation. In addition, due to the long times over which an accident can occur during shutdown, appropriate modifications were made to incorporate the effects of reduced decay heat levels and correspondingly reduced radionuclide inventories. Specifically, modifications were made to the accident progression event tree (APET) analyses, the source term, and consequence analyses. The consequence measures analyzed to obtain risk were early fatalities, latent cancer fatalities, and population dose (person-rem) to 50 miles from the plant. In addition, the quantitative health objectives defined in the NRCs Safety Goal Policy statement (i.e., individual risk of early fatality to 1 mile and individual risk of latent cancer to 10 miles from the plant) were also calculated. These consequence measures are the same as those evaluated in the NUREG-1150 study and were selected to facilitate comparison between full power and shutdown risk estimates.

The results indicate that the risk of latent cancers during mid-loop operation is approximately the same as the risk during full power operation. This is due to the potential lack of mitigative features for a significant fraction of the accidents that could be initiated during mid-loop. This in turn means that the releases to the environment could be large and the radionuclide species which contribute to long-term health effects have long half lives. The risk of early fatalities were estimated to be much lower at mid-loop operation compared to full power due mainly to the decay of the short-lived isotopes of iodine and tellurium, which contribute significantly to the early health effects. The status of containment isolation is the largest contributor to the uncertainty in the risk estimates during mid-loop operation. If the containment is initially open, it is not clear that it can be isolated prior to the start of core damage. Even if the containment is isolated, it is not clear that pressure-retaining capability can be achieved within the time frame of the accident progression. The availability of containment sprays also contributes to the uncertainty in risk estimates.

1. INTRODUCTION

Traditionally, probabilistic risk assessments (PRAs) of severe accidents in nuclear power plants considered initiating events which could potentially lead to core damage and containment failure only during normal full power operation. However, recent studies and operational experience during periods while plants were shutdown for maintenance or refueling indicated that potential accidents initiated during low power operation or shutdown conditions could

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

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also potentially become important contributors to risk. In 1989, the Nuclear Regulatory Commission (NRC) began an extensive program to assess the risk during low power and shutdown operation. Two plants, Surry (a pressurized water reactor, PWR) and Grand Gulf (a boiling water reactor, BWR) were selected as the plants to be studied.

The Level 1 PRA of Surry during low power and shutdown operation was initiated in 1990 and carried out in two phases. Phase 1 undertook a coarse qualitative screening analysis of the accident sequences leading to core damage for all plant operational states during low power and shutdown. In Phase 2, a detailed quantitative analysis of the core damage frequency was performed for a particular plant operational state (POS), mid-loop operation, which can occur while the plant is shutdown for maintenance or refueling. (Chu, 1994). Mid-loop operation refers to a condition when the reactor coolant system is lowered to the mid-plane of the hot leg to allow essential maintenance to be performed. This particular POS was selected after the initial screening study indicated its greater vulnerability compared to other POSs due to the reduced fluid inventory in the reactor coolant system (RCS).

The Level 2/3 PRA was begun in late 1991 and also carried out in two phases. In Phase 1 an abridged risk study was performed which focused on a limited analysis of accident progression and consequences conditional on the occurrence of core damage. In Phase 2 an integrated Level 2/3 PRA of mid-loop operation was conducted to combine the quantitative estimates of core damage of the Phase 2 Level 1 PRA with accident progression and consequence analysis to obtain risk. This paper reports the results of the integrated Level 2/3 PRA for Surry (Jo, 1995).

2. METHODOLOGY OF THE STUDY

The methodology of the integrated PRA was based on the NUREG-1150 full power study of Surry (NRC, 1990). However, due to the long times over which an accident can occur during shutdown, appropriate modifications were made to incorporate the effects of reduced decay heat levels and correspondingly reduced radionuclide inventories in the interfaces between the core damage frequency, plant damage state (PDS) and the accident progression event tree (APET) analyses and the APET outcomes, source term, and consequence analyses. The overall structure of the methodology is shown in Figure 1. The Level 1 analysis was carried out using the IRRAS (Russell, 1993) code; the minimal cut-sets were grouped into PDSs for entry to the APET which was analyzed using the EVNTRE (Griesmeyer, 1989) code. The APET was modified from the Surry full-power APET. It used 40 questions instead of the 71 used at full-power. The APET outcomes, grouped into accident progression bins, were fed to the SURSOR (Jow, 1989) code to determine source terms and the partitioned source terms were analyzed by the MACCS (Chanin) code to obtain consequences. Risk was evaluated by combining the results of each of the constituent analyses shown in Figure 1.

The consequence measures analyzed to obtain risk were early fatalities, latent cancer fatalities, and population dose (person-rem) to 50 miles from the plant. In addition, the quantitative health objectives of the Safety Goal Policy, individual risk of early fatality to 1 mile and individual risk of latent cancer to 10 miles from the plant were also calculated. These consequence measures are the same as those evaluated in the NUREG-1150 study and were selected in order to allow for a comparison between the full power and shutdown risk.

3. SURRY PLANT CONFIGURATION

Surry Unit 1 is a 2441 MWth pressurized water reactor (PWR) designed and constructed by Westinghouse. It is a three-loop plant; the reactor coolant system has three U-tube steam generators and three reactor coolant pumps. The containment and balance of plant were designed and constructed by Stone and Webster. Commercial operation of Unit 1 began in 1972.

Emergency ac power at the site is supplied by three diesel generators (DGs). Emergency dc power is supplied by separate battery banks at each unit. The DGs have their own separate set of batteries for starting power. The auxiliary feedwater (AFW) system has three trains. Two trains have electric pumps, the third train has a steam turbine driven pump. The condensate storage tank provides suction for the AFW system. The chemical volume and control system has three charging pumps which also serve as the high-pressure injection (HPI) pumps. There are two low pressure injection (LPI) pumps. Both the HPI and the LPI systems can function in the injection or recirculation mode. In the injection mode, they take suction from the refueling water storage tank (RWST) while in the recirculation mode they take suction from the sump. Surry also has three accumulators which provide a source of immediate, low-pressure, high flow injection. Overpressure protection for the reactor coolant system is provided by three code safety/relief valves (SRV) and two power operated relief valves (PORV).

The Surry containment is a reinforced concrete cylinder with a hemispherical dome. The free volume of the containment is 1.8 million cubic feet and the design pressure is 45 psig. Due to design conservatisms, realistic estimates of the loads needed to fail the containment are between two and three times the design pressure. The mean of the distribution for the failure pressure of the Surry containment provided by the expert panel in the NUREG-1150 study was 126 psig.

Cooling of the containment is normally provided by fan coolers which are not safety grade and will be partially submerged if the sump is filled with water. Emergency cooling of the containment is provided by the containment spray systems (CSS). Another feature of the Surry containment at a low elevation is that there is no connection between the sump and the reactor cavity. If a pipe break occurs, the water will flow to the sump. The cavity remains dry unless the containment sprays operate.

The general description given above indicates the main plant systems available during full power operation at Surry. However, during shutdown the plant is configured differently than during full power operation and some of the systems described above will not be available.

3.1 Plant and System Configuration During Mid-loop Operation

Three mid-loop operating states were identified and analyzed in the level 1 analysis; two mid-loop operating states during refueling outages (one early in the outage during cooldown using the residual heat removal (RHR) system and the other later after completion of refueling), and another mid-loop operating state during the cooldown period of a drained maintenance outage. A detailed analysis of plant systems, their response to various accident initiators and their status in accident sequences leading to core damage are described in Chu, et. al. (1994). In this study, the focus is on those plant systems and features which are important to the progression of the accident and to the possible releases to the containment and the environment following core damage. Accident progression can be influenced by the status of the reactor coolant system, recovery of coolant injection systems, containment integrity, containment spray systems and cavity flooding.

The reactor coolant system (RCS) is at low pressure during mid-loop operation as soon as the plant is placed in the RHR entry level condition. This implies that potential accidents during mid-loop operation will not involve any high pressure sequences such as those modeled in the full power PRA. Also during mid-loop operation the relief valves in the pressurizer are open connecting the pressurizer to the pressurizer relief tank which is vented to the process vent system. The vessel head vent is connected to the discharge side of the PORVs through piping that consists of a section of tygon tube which can withstand about 40 psia of pressure. Additionally, the safety valves could be removed for maintenance during mid-loop and a temporary partition placed on the opening. This creates the possibility of a direct vent path into containment for any released fission products in the event of an accident. These features of the RCS during mid-loop operation were incorporated in the accident progression event tree.

The ECCS at Surry consists of the High Pressure Injection/Recirculation (HPI/HPR) system and the Low Pressure Injection/Recirculation (LPI/LPR) system. ECCS is important to the accident progression because for some plant damage states it could be restored after the start of core damage. If the ECCS is restored while the damaged core is still in the reactor vessel it may be possible to terminate the accident prior to vessel breach. A relatively high probability of terminating the accidents in-vessel was estimated in the accident progression analysis for three out of the four plant damage states. If the core debris has melted through the vessel and is attacking the reactor cavity restoration of the ECCS will supply water to the cavity and flood the core debris. A flooded cavity could terminate the core-concrete interaction and considerably mitigate the associated source term. If core-concrete interactions continue, flooding of the cavity would lead to a scrubbing of the fission product release.

At the inception of the abridged risk study, the status of containment isolation during mid-loop operation was analyzed. At that time it was determined that while containment was considered "closed" during mid-loop operation, what closure meant was that all penetrations were isolated from the outside, some with temporary barriers, so that there is no air/vapor exchange with the environment. However, "closure" in the above sense did not mean that the containment was capable of achieving the design pressure.

Recognizing perhaps the potential problems regarding containment status during low power and shutdown operation, the Surry staff developed additional procedures to address the concerns about the closure of the containment during or shortly after the initiation of an accident in mid-loop operation. However, these concerns could not be fully resolved during the time-frame of the study. Since containment status during shutdown is, perhaps, the single most important feature of the plant which affects risk, it was deemed prudent to model the probability of pre-existing leakage (as assumed in the abridged risk study) and the containment failure pressure as uncertainty parameters and perform a sensitivity analysis to evaluate the impact of different assumptions regarding containment status on the risk.

Containment heat removal in an emergency at Surry is by means of the containment spray system (CSS) in the injection mode. The requirements on the availability of the CSS apply when the RCS temperature and pressure is in excess of 350°F and 450 psig, respectively. When the reactor is operating at power, both CSS trains must be operable. Considering the operating parameters of mid-loop operation, there are no Technical Specifications which require CSS to be available during this plant operational state. Discussions with Surry personnel indicated, informally, that the probability of at least one train of CSS being available was likely to be fairly high, on the order of 70 percent. Accordingly, spray availability was treated as an uncertainty parameter in the development of the APET. If CSS is available during mid-loop operation, it would have to be manually actuated since automatic actuation is not available at RCS temperature below 350°F.

The Inside Spray Recirculation (ISR) and the Outside Spray Recirculation (OSR) systems provide the long term containment cooling and pressure reduction following an accident. At Surry, these systems also provide long term core cooling after the accident. There are no Technical Specifications for ISR and OSR systems below the limits of 350°F and 450 psig. Thus it is possible that neither of the recirculation spray systems, ISR or OSR, would be available during mid-loop operation. In discussions with Surry plant personnel it was indicated, informally, that the likelihood of availability of at least one train of either ISR or OSR is high (about 70%) during shutdown.

The reactor cavity at Surry is normally dry as all water in the containment drains to the sump and there is no connection between the sump and the cavity. This feature of the Surry cavity has important implications for the progression of severe accidents and the source terms where the vessel is breached and core-concrete interactions occur. The only way for the cavity to have water is either if the containment sprays operate or if core injection is recovered after vessel breach. This feature was incorporated in the accident progression event tree.

4. ACCIDENT ANALYSIS

The study was limited to internal event initiators only. Specific initiators for mid-loop operation are: loss of residual heat removal (RHR), loss of offsite power, loss of 4 kV bus, loss of component cooling water, inadvertent safety feature actuation, and boron dilution events. Loss of RHR is the most important initiator in terms of frequency and impact on the accident analysis followed by loss of offsite power.

The time to enter mid-loop and the average duration of mid-loop operation are important parameters, which have a large impact on the probability of recovering from the accident. The criteria used for success of the safety systems to prevent core damage differ depending on the decay heat level, which is a function of the time that the accident occurs after shutdown. These times also have a significant impact on the progression of the accident and on possible releases and the consequences. In order to incorporate these times formally into the analysis, a "time window" approach was developed. A total of four time windows after shutdown were defined in the accident frequency analysis. Table 1 shows the definition of the time windows. Each window is characterized by a time interval (measured from the time of reactor shutdown), and a representative level of decay heat, which corresponds to the mid-point of the time interval. The decay heat then determines the timing of key events in the accident such as the time to boiling if the RHR system is lost, the time to reach various pressures which will challenge sub-systems such as the (temporary) tygon tubing, and the pressurizer relief tank (PRT), time to core uncovery and eventually core damage. These times are displayed for each window. The definition of time windows used in the accident progression analysis was also used in the definition of the plant damage states, and the accident progression event tree.

In constructing the APET for mid-loop operation, extensive use was made of the results of the accident progression analysis for the Surry plant carried out in the NUREG-1150 program (Breeding, 1993), which was a PRA of the plant at full power. The NUREG-1150 study showed that the major contributor to risk was from containment bypass followed by basemat melt-through (BMT) accidents. Phenomena leading to early containment failure such as direct containment heating (DCH) or steam explosions were not important contributors nor did hydrogen burning or gradual pressurization of the containment significantly contribute to containment failure. For accidents during low power and shutdown operation the decay heat is significantly less, the reactor pressure is generally low and the pressure generated in the containment is lower than for accidents occurring at full power. Therefore, early containment failure modes such as DCH and hydrogen burning could be excluded from the low power and shutdown risk study if the capability of the containment to hold pressure was the same as that at full power. However, the status of the containment during mid-loop operation is uncertain so these containment failure modes could not be eliminated based simply on the results of the full power analysis.

The APET for this study contains 40 questions; it was modified from the full power analysis (71 questions) to reflect the conditions during mid-loop operation. The APET was divided into five time periods: (1) Initial: the conditions at the beginning of the accident, (2) Early: the in-vessel accident progression period up to the time of vessel breach (VB), (3) Intermediate: the progression of the accident at and immediately after vessel breach (VB), including the possibility of containment failure at VB, (4) Late: the progression of the accident during core-concrete interaction (CCI), and (5) Very Late: the accident progression in the period following CCI, including the possibility of containment failure due to hydrogen combustion.

Several calculations were performed with the MELCOR code (Summers, 1991) to support the determination of the various time windows and associated success criteria, the time periods used in the APET and also to help in the quantification of the APET. Predictions of the MELCOR code were also used to compare the source term distributions calculated by the SURSOR code.

The major impact of the MELCOR calculations on the APET quantification related to two potential containment failure mechanisms, BMT and overpressurization of the containment by steam and noncondensible gases. BMT was found to be a significant cause of fission product releases for accidents during full power operation although the core debris was determined to penetrate the basemat very late in an accident sequence. However, the MELCOR calculations indicate much slower concrete erosion rates for accidents during mid-loop operation due to the relatively low decay heat. The erosion depth was calculated to be about 0.75 m (compared with a basemat thickness of 3 m) 30 hours after the start of an accident in time window 1 (which has the highest representative decay heat). Even in the full power analysis, it was calculated to take several days to breach the basemat. Since the probability of not recovering some safety injection system in this time period is extremely small, it was determined that basemat melt-through is not a credible failure mode for accidents during mid-loop operation.

Overpressurization of the containment by steam and noncondensible gases was also found to be not a credible failure mode during mid-loop operation based on MELCOR calculations. This is true even if the containment is assumed to leak at pressures above 45 psig. Again the low decay heat levels associated with accidents during mid-loop operation means that the driving force for containment pressurization is low and the rate of pressurization is very slow.

The outcomes of the evaluation of the APET were placed in the following categories: (1) No Vessel Breach, No Containment Failure (2) No Vessel Breach, Open Containment, (3) Vessel Breach, No Containment Failure, (4) Vessel Breach, Open Containment, and (5) Vessel Breach, Containment Failure (including steam explosions, DCH, & Hydrogen burn)

The "Containment Failure" group contains energetic events that cause structural failure of the containment. The "Open Containment" group includes leakage through the equipment hatch or other temporary barriers (which can occur even after "successful" isolation of containment) as well as failure to isolate containment before the onset of core damage. Generally, the containment failure probability is dominated by the probability of whether the containment is successfully isolated prior to core damage. Containment failure due to energetic events (DCH or hydrogen burning) is relatively small as in the full power study even if the containment is assumed to fail at pressures above 45 psig. This is partly because the fraction of accidents with high or intermediate vessel pressure is very small, and partly because the fraction of accidents where the containment was not isolated is high. Very late containment failure due to basemat melt-through and gradual pressurization due to loss of containment cooling was assumed not to happen based on the results of MELCOR calculations.

Source terms were calculated from the accident progression bins using the SURSOR code as done in the NUREG-1150 study. Partitioning of the source terms based on early and latent fatality weights was carried out for different time windows. The consequences were calculated using the latest version of MACCS.

5. RESULTS

The accident sequences from the integrated Level 1 and 2 analyses were binned into the following plant damage state groups:

PDS Group 1: Station Blackout (SBO). The SBO PDSs contribute approximately 10% to the mean total core damage frequency. The accidents belonging to this group are initiated by a loss of off-site power and coupled with other failures result in a SBO. The recirculation and containment cooling systems are not available due to the loss of power. In this PDS, an important factor in the accident progression is the recovery of the off-site AC power. The mean conditional probability of core damage arrest prior to vessel failure, averaged over all four time windows, is about 0.55. The mean conditional probability of loss of containment integrity for this PDS group averaged over all time windows is approximately 0.51. Energetic containment failure is significant only for this PDS group, with a mean conditional probability of about 0.15. This mostly comes from hydrogen burning late in the accident. This mode of failure is prominent in this PDS group, since hydrogen burning is more likely when the power is recovered after a substantial amount of hydrogen has accumulated in the containment.

PDS Group 2: Human Errors (HX). This PDS group is the largest contributor to the internal event core damage frequency for mid-loop operation. About two thirds of all core damage accidents belong to this group which are attributable to human errors. Following loss of core cooling due to some initiator, operators either fail to diagnose the accidents or to take correct actions. The progression of accidents is somewhat different depending on whether the human error is in diagnosis or action. For example, if it is a diagnostic error, then it is assumed that the same error results in failure to recognize the need for containment isolation. If the error was a failure to take the correct action, it was more likely that the containment was closed before core damage. In most cases, the electric power and some core cooling systems are available. The dominant factor in the accident progression is the recovery from human errors.

The mean conditional probability of core damage arrest without vessel failure is about 0.42 averaged over all windows. This probability is lower than that of PDS group 1 indicating that the recovery probability from human error is less likely than recovery of electric power once the accident progresses to core damage. The mean conditional probability of loss of containment integrity for this PDS group is very high, about 0.9. This result reflects the assumption that the containment would most likely remain unisolated in this PDS group. Energetic containment failure is insignificant for this PDS group. Since this PDS group is the largest contributor to the core damage frequency, it also significantly contributes to the overall probability of loss of containment integrity.

PDS Group 3: Recirculation Failure. The PDSs in this group contribute about 18% to the mean core damage frequency. The accidents in this group occur only in Windows 1 and 2. In this group, core cooling was successfully initiated and continued until the RWST is depleted; but the recirculation fails and the accident progresses to core damage. The conditional probability of core damage arrest before vessel failure in this PDS group is zero since it is assumed that core cooling is permanently lost once recirculation is lost. The mean conditional probability of loss of containment integrity for this PDS group is relatively low, about 0.13. The probability of isolating the containment in this PDS group is considered to be high because core cooling is established and the reactor has been in a stable condition for a relatively long time. Energetic containment failure is unimportant for this PDS group, contributing only about 3% to containment failure.

PDS Group 4: Loss of 4 kV Bus. This PDS group contributes about 5% to the mean core damage frequency. There are no occurrences of this PDS in Windows 3 and 4. The accidents in this group are similar to those of PDS group 1 (SBOs) except that accidents are initiated by loss of 4 kV bus. This group is separated from Group 1 since the recovery probabilities are different, however, the accident progression for this group is similar to that of Group 1. The mean conditional probability of core damage arrest without vessel failure was determined to be about 0.6. The mean conditional probability of loss of containment integrity for this PDS group is approximately 0.45. Hydrogen burning is a significant contributor to the conditional containment failure probability as in Group 1.

Table 2 shows distributions of the core damage frequency for mid-loop and full-power operation respectively. The mean CDF during mid-loop is approximately one order magnitude less than the full power CDF. Table 3 shows distributions of the risks for the selected measures of offsite consequences.

The results indicate that the mean risk of population dose during mid loop operation is approximately the same as the risk during full power operation. This is due to the potential lack of mitigative features for a significant fraction of the accidents initiated during mid-loop so the releases to the environment are relatively large and the radionuclide species which contribute to long-term health effects have long half lives. (The mean risk of latent cancers is three times higher due to the difference in the latent cancer risk coefficient between the later version of the MACCS code used in the shutdown study, which incorporates the higher BEIR V risk coefficient, and the earlier version of MACCS used in the NUREG-1150 study).

The risk of early fatalities is much lower at mid-loop compared to full power. This is due mainly to the long time after reactor shutdown that the accidents occur and the consequent decay of the short-lived isotopes of iodine and tellurium which contribute significantly to the early health effects.

6. DISCUSSION

Several issues were identified in the course of the study which potentially impact the risk of mid-loop operation and the uncertainty in the risk. A number of them relate to modeling of physical processes while others relate to lack of information. In some cases, if more information was made available then the uncertainty in the risk estimates could be reduced. In other cases, significant additional analysis would be required to reduce uncertainty.

The largest uncertainty in the risk estimates during mid-loop is contributed by the uncertainty in the status of containment isolation and achievement of a pressure-retaining capability, if the containment was initially unisolated, within the time frame of the accident progression. In the abridged risk study it was assumed that the containment could not be isolated in the time frame available before core damage and the start of the release of the core inventory. New procedures have been subsequently developed at Surry to address containment closure during mid-loop operation. However, questions still remained in the present study as to the adequacy of these procedures in ensuring the pressure retaining capability of the containment even if it is successfully isolated. This issue therefore remains an important contributor to the uncertainty associated with containment performance and determination of risk during mid-loop operation.

There are no procedures in place to ensure that the containment sump will be available as a source of water for recirculation cooling during an accident occurring in mid-loop operation. Plugging of the sump by temporarily stored materials required for performing plant maintenance during shutdown was found to be one of the contributors to core damage and risk due to failure of recirculation cooling.

A smaller contribution to uncertainty is made by the status of the availability of containment sprays. Containment sprays are an important system during accident conditions for condensing steam and removing heat. Sprays are also potentially effective as a mitigation system for scrubbing fission products released as an aerosol and reducing the source term to the environment. Spray availability was therefore treated as an uncertainty parameter in the analysis; its potential availability during mid-loop operation was based on discussions with Surry plant personnel. However, if the sprays are available they would have to be manually actuated during mid-loop operation as automatic actuation is disabled at RCS temperature below 350°F.

One issue relates to the effect of spray activation after core damage when a large amount of radioactive aerosols and gases could be present in the containment atmosphere. If the containment is unisolated water droplets from the sprays could displace the atmosphere inside containment and cause the aerosols and gases to be released through the opening in the containment boundary at a faster rate than if the sprays had not been activated. This effect could exacerbate the release to the environment; however, it was not modeled in the present study.

The impact of environmental conditions in the plant after the start of bulk boiling on the potential for successfully performing recovery actions is another important issue. It may be difficult to carry out recovery actions, which cannot be carried out from the control room, after bulk boiling of the reactor coolant inventory begins. There are several actions during mid-loop operation that can only be performed by entering the containment, for example, restoring RHR and, for station blackout sequences, opening valves to feed the steam generator. The HRA considered the impact of environment as part of the quantification of recovery actions. At temperatures around 140–150°F, the air is too hot for normal pulmonary function and self-breathing respirators may be required for emergency personnel which would also significantly decrease the possibility of success of recovery actions. The uncertainty in the status of containment, referred to above, cuts across this issue. If the containment was isolated, it is unlikely that it would be re-opened to undertake a recovery action once it was recognized that core uncovery was imminent or had occurred as indicated by the radiation monitors. On the other hand, if the containment were unisolated or had no pressure holding capability, the high radiation levels in its immediate vicinity as shown by the onsite dose rates would also make recovery actions inside it unlikely. The impact of environment on the ability of operators to perform recovery actions remains an important issue which contributes to the overall uncertainty.

The impact of recovering cooling water early in the accident progression after core uncovery but before vessel breach is also an open issue. If the clad becomes embrittled on heat up it could fracture on quenching releasing the gap inventory. Water could then enter the ruptured fuel rods and leach out iodine (and other volatile fission products) from the fuel matrix. Depending on temperature and solubility limits, the iodine would be partitioned between the water in the vessel and the containment atmosphere. While this accident scenario is not likely to have any significant offsite consequences, it could have important onsite implications particularly for recovery actions. This type of release was not modeled in the study.

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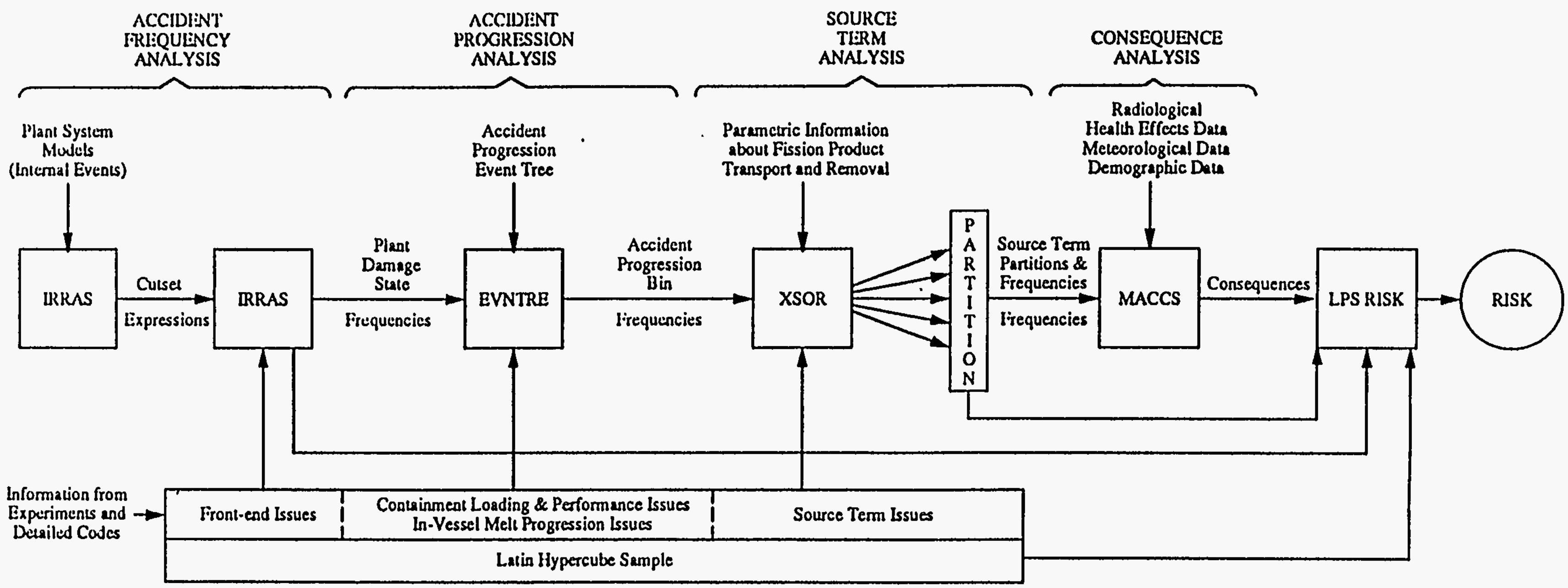

FIGURE 1 OVERVIEW OF PLANT ANALYSIS USED FOR MID-LOOP OPERATION

TABLE 1 DEFINITION AND CHARACTERIZATION OF TIME WINDOWS

	WINDOW 1	WINDOW 2	WINDOW 3	WINDOW 4
Definition	≤ 75 hours	> 75 hours and ≤ 240 hours	> 240 hours and ≤ 32 days	> 32 days
Representative Decay Heat	13 MW (2 days)	10 MW (5 days)	7 MW (12 days)	5 MW (32 days)
Time to Boiling	15 min.	20 min.	27 min.	37 min
Time to Tygon Tube Rupture (40 psia)	23 min.	31 min.	43 min.	59 min.
Time to PRT Rupture (100 psig)	51 min.	63 min.	78 min.	96 min.
Time to 165 psia	41 min. with 2 PORV 43 min. with 1 PORV	63 min. with 2 PORV 60 min. with 1 PORV	227 min. with 2 PORV 89 min. with 1 PORV	352 min. with 2 PORV 147 min. with 1 PORV
Time to 615 psig	145 min. with 1 PORV	—	—	—
Time to RWST Depletion	10 hrs	13.5 hrs	18.7 hrs	38.6 hrs
Time to AFW Initiation (with 25% SG inventory remaining)	743 min.	669 min.	925 min.	628 min.
Time to Core Uncovery	120 min.	157 min.	209 min.	273 min.
Time to Core Damage	219 min.	297 min.	411 min.	557 min.

*Reproduced from Table 5.4-2 in Volume 2 of this report.

TABLE 2 CORE DAMAGE FREQUENCY ESTIMATES

	Core Damage Frequency for Mid-Loop Operation (per reactor year)	Core Damage Frequency for Full-Power Operation (per reactor year)
95th Percentile	1.9E-5	1.0E-4
Mean	4.2E-6	4.1E-5
50th Percentile	2.0E-6	2.5E-5
5th Percentile	3.2E-7	9.8E-6

TABLE 3 COMPARISON OF DISTRIBUTIONS OF RISKS FOR MID-LOOP AND FULL-POWER OPERATION (ALL VALUES PER REACTOR YEAR; POPULATION DOSES IN P-SV PER YEAR)

	5th Percentile		Median		Mean		95th Percentile		Standard Deviation	
	Mid-Loop	Full-Power	Mid-Loop	Full-Power	Mid-Loop	Full-Power	Mid-Loop	Full-Power	Mid-Loop	Full-Power
Early Fatalities	1.26E-10	7.60E-10	3.57E-09	7.00E-08	4.90E-08	2.00E-06	1.59E-07	5.40E-06	1.69E-07	N.A.
Latent Fatalities within 50 mi	1.55E-04	N.A.	8.34E-04	N.A.	2.46E-03	N.A.	8.78E-03	N.A.	3.68E-03	N.A.
Latent Fatalities within 1000 mi	7.97E-04	3.10E-04	5.35E-03	2.20E-03	1.57E-02	5.20E-03	5.50E-02	1.90E-02	2.52E-02	N.A.
Population Dose within 50 mi	3.77E-03	5.90E-03	1.98E-02	2.70E-02	5.79E-02	5.80E-02	1.89E-01	2.50E-01	8.77E-02	N.A.
Population Dose within 1000 mi	1.87E-02	1.90E-02	1.25E-01	1.30E-01	3.66E-01	3.10E-01	1.29E+00	1.20E+00	5.90E-01	N.A.
Individual Early Fatalities Risk within 1 mi	6.00E-12	1.40E-11	1.27E-10	8.70E-10	1.74E-09	1.60E-08	6.94E-09	4.90E-08	5.52E-09	N.A.
Individual Latent Fatalities Risk within 10 mi	1.20E-10	1.60E-10	7.48E-10	4.90E-10	2.09E-09	1.70E-09	7.10E-09	8.10E-09	3.01E-09	N.A.

N.A. – Not Available