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RISK CONTRIBUTION FROM LOW POWER AND SHUTDOWN OF A PRESSURIZED WATER REACTOR*

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ABSTRACT

During 1989 the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. Two plants, Surry (a pressurized water reactor) and Grand Gulf (a boiling water reactor), were selected for study by Brookhaven National Laboratory and Sandia National Laboratories, respectively.

The program objectives included assessing the risks of severe accidents initiated during plant operational states other than full power operation and comparing estimated core damage frequencies, important accident sequences, and other qualitative and quantitative results with full power accidents as assessed in NUREG-1150. The scope included a Level 3 PRA for traditional internal events and a Level 1 PRA on fire, flooding, and seismically induced core damage sequences.

A phased approach was used in Level 1. In Phase 1 the concept of plant operational states (POSs) was developed to provide a better representation of the plant as it transitions from power to nonpower operation. This included a coarse screening analysis of all POSs to identify vulnerable plant configurations, to characterize (on a high, medium, or low basis) potential frequencies of core damage accidents, and to provide a foundation for a detailed Phase 2 analysis.

In Phase 2, selected POSs from both Grand Gulf and Surry were chosen for detailed analysis. For Grand Gulf, POS 5 (approximately Cold Shutdown as defined by Grand Gulf Technical Specifications) during a refueling outage was selected. For Surry, three POSs representing the time the plant spends in midloop operation were chosen for analysis.

Level 1 and Level 2/3 results from the Surry analyses are presented.

1. INTRODUCTION

During 1989 the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. Two plants, Surry (a pressurized water reactor) and Grand Gulf (a boiling water reactor), were selected as the plants to be studied by Brookhaven National Laboratory and Sandia National Laboratories, respectively.

The program objectives included assessing the risks of severe accidents initiated during plant operational states other than full power operation and comparing the estimated core damage frequencies, risks, important accident sequences, and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150.

A phased approach was used in the Level 1 program. In Phase 1 the concept of plant operational states (POSs) was developed to allow the analysts to obtain a better representation of the plant as it transitions from power operation to nonpower operation. This phase consisted of a coarse screening analysis for all POSs to identify potential vulnerable plant configurations, to characterize (on a high, medium, or low basis) the potential frequencies of core damage accidents, and to provide a foundation for a detailed Phase 2 analysis.

In Phase 2 selected POSs from both Grand Gulf and Surry were chosen for detailed analysis. For Grand Gulf, POS 5 (approximately Cold Shutdown as defined by Grand Gulf Technical Specifications) during a refueling outage was selected. For Surry, three POSs representing the time the plant spends in midloop operation

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were chosen for analysis. These included POS 6 and POS 10 of a refueling outage and POS 6 of a drained maintenance outage.

During the preliminary quantification of the accident sequences in Phase 2, it was found that the decay heat at which the accident initiating event occurs is an important parameter that determines both the success criteria for the mitigating functions and the time available for operator actions. In order to better account for the decay heat, a "time window" approach was developed. In this approach, time windows after shutdown were defined based on the success criteria established for the various methods that can be used to mitigate the accident.

In this paper, the results of the Surry [1-6] analysis are presented. Section 2 documents the results from the level 1 part of the analysis, Section 3 documents the results from the level 2/3 analysis. Section 4 is conclusions.

2. LEVEL 1 RESULTS

2.1 Results from Traditional Internal Events

Table 1 summarizes the results of the event tree quantification, showing the core-damage frequency as a function of the initiating events and POSs. The core damage frequency is the frequency that core-damage occurs while the reactor is at mid-loop, and includes the fraction of a year that the reactor is at mid-loop. POS 6 of a drained maintenance outage (D6), and POS 6 of a refueling outage are the most dominant POSs. Their characteristics are high decay-heat level and a relatively short time available for operator action. In contrast, POS 10 of a refueling outage has a very low decay heat, and its core-damage frequency is approximately one order of magnitude lower.

Table 2 compares the results of this study with those of NUREG-1150[7] and the individual plant examination (IPE)[8] performed by the utility for Surry. The results are displayed in two ways. The core-damage frequency, shown in the first row, is the frequency that core-damage occurs when the plant is at mid-loop (the core-damage frequencies in the parentheses are the contributions due to over-draining events), and the conditional core-damage frequency, shown in the third row, is the core-damage frequency (minus the contribution of over-draining events) divided by the fraction of time the plant is at mid-loop. The former accounts for the fact that the plant is at mid-loop only a small fraction of the time, while the latter is the conditional frequency at which core-damage occurs given the plant is at mid-loop. The core-damage frequency of mid-loop operations is approximately one eighth of that of power operation as estimated in NUREG-1150, while the plant is in mid-loop operation approximately 7% of a year. The numbers in the parentheses of the third row of the table are the conditional probability of core-damage due to over-draining events, given that the plant enters mid-loop operation in the POS.

The core-damage frequencies shown in the first row of Table 2 are additive. That is, the sum of the core-damage frequencies of the 3 POSs is the total core-damage frequency of mid-loop operation. This total, 5.06 E-06 per year, can be added to the core-damage frequency of power operation, e.g., 4.01 E-05 per year for NUREG-1150. Therefore, the sum of 4.51 E-05 per year is the frequency per year that core-damage occurs while the plant is at full power or mid-loop operation.

The conditional core-damage frequency shown in the third row of Table 2 is a measure of how susceptible a plant configuration is with respect to core-damage. For example, the fact that the conditional core-damage frequency of mid-loop operation, 7.62 E-05 per year, is higher than that of full power operation, 4.01 E-05 per year, shows that mid-loop operation is more susceptible to core-damage than full power operation, although the plant is at mid-loop only a small fraction of the time.

Table 3 lists the conditional core damage frequency as a function of the time windows and POSs. The conditional core damage frequency is the rate at which core damage occurs given that the plant is in the time window of the POS. It is obtained by dividing the core damage frequency by the fraction of time the plant is in the time window of the POS. The conditional core damage frequency is a measure that can be used to compare the vulnerability of the time windows and POSs with respect to core damage. It can be seen, from Table 3, that for each POS the conditional core damage frequency decreases with time window. This is due to the relaxed success criteria and more time available for operator actions. The conditional core damage frequency for R6 or R10 is higher than that of D6 mainly due to that the RCS loops have a high probability of being isolated in a refueling outage; that makes reflux cooling impossible. For example, in window 1, the probability that the loops are isolated in a refueling outage is 0.3, and the probability that reflux cooling fails in a drained maintenance outage is 0.1

Table 1 Summary of Results-Core-Damage Frequency by Initiating Event and Plant Operational States

	Initiating Event	Core-Damage Frequency (per year)			
1.	Loss of RHR	R6	R10	D6	Total
	RHR2A-Over Draining	1.8E-7	5.3E-8	2.6E-7	4.9E-7
	RHR2B-Failure to Maintain Level	2.1E-8	2.0E-8	2.9E-8	7.0E-8
	RHR3-Non-Recoverable Loss of RHR	1.5E-7	8.4E-9	3.0E-7	4.6E-7
	RHR4-Non-Recoverable Loss of Operating Train of RHR	7.6E-9	1.2E-9	2.3E-8	3.2E-8
	RHR5-Recoverable Loss of RHR	4.0E-8	4.1E-9	9.3E-8	1.4E-7
2.	LOOP-Loss of Offsite Power				
	L1-Both 1H and 1J Energized	3.3E-7	7.0E-8	7.6E-7	1.2E-6
	L2-1H and 2H energized, not 1J	1.0E-7	1.3E-8	1.7E-7	2.9E-7
	L3-1H energized, not 1J, unit 2 blackout	4.2E-8	1.3E-8	9.9E-8	1.5E-7
	B1-Unit 1 Black Out	4.8E-8	1.1E-8	1.7E-7	2.3E-7
	B2-2 Unit Blackout	3.8E-8	4.2E-8	1.1E-7	1.9E-7
3.	4KV-Loss of 4kv Bus	1.4E-7	1.9E-8	2.4E-7	4.0E-7
4.	VITAL-Loss of Vital Bus	2.8E-8	5.1E-9	7.3E-8	1.1E-7
5.	AIR-Loss of Outside Instrument Air	7.9E-10	-	3.2-9	4.0E-9
6.	CCW-Loss of CCW	6.3E-8	1.1E-10	2.1E-7	2.7E-7
7.	SWGR-Loss of Emergency Switchgear Room Cooling	3.6E-8	1.2E-8	7.4E-8	1.2E-7
8.	ESFAS-Inadvertent Safety Feature Actuation	2.7E-7	2.7E-8	6.8E-7	9.8E-7
9.	Dilute-Boron Dilution (CDF)				6.8E-8
TOTAL		1.5E-6	3.0E-7	3.3E-6	5.1E-6*

* Not including boron dilution

Table 2 Comparison of Total Core-Damage Frequency with NUREG-1150 and IPE

Study	Results				
PWR Low Power and Shutdown Study (Mid-Loop POSs, Internal Event Only)		R6	R10	D6	TOTAL
	CDF* per year	1.49E-6 (1.82E-7)**	3.06E-7 (5.47E-8)**	3.25E-6 (2.67E-7)**	5.06E-6 (5.04E-7)**
	Fraction of a year the plant is in mid-loop	1.63E-2	1.52E-2	3.49E-2	6.64E-2
	Conditional CDF*** per year (CDP)	8.09E-5 (3.03E-7)	1.65E-5 (1.82E-7)	8.55E-5 (2.23E-7)	7.62E-5 (2.40E-7)
NUREG-1150 (Internal Event Only)	4.01E-5				
IPE(Internal Event Only)	7.40E-5				

* CDF reflects the fraction of time the plant is at mid-loop
** Contribution of over-draining events
*** Frequency of core-damage given that the plant is at mid-loop
CDP probability of core-damage due to over-draining to the POS

Table 3 Conditional Core Damage Frequency As a Function of the Time Windows and POSs (per year)

	R6	R10	D6	Average
Window 1 (13hr-75hr)	9.96E-4	-	3.37E-4	3.77E-4
Window 2 (75hr-240hr)	7.55E-5	-	5.90E-5	7.25E-5
Window 3 (240-768hr)	5.49E-5	6.54E-5	5.18E-5	5.60E-5
Window 4 (> 768hr)	1.87E-5	1.57E-5	1.05E-5	1.80E-5
AVERAGE	8.09E-5	1.65E-5	8.55E-5	7.62E-5

(modeled as a recovery action). The difference between R6 and R10 in windows 3 and 4 is due to the difference in maintenance unavailabilities.

The averages in Table 3 represent the averaged conditional core damage frequency. For example, the averaged conditional core damage frequency for R6 is 8.09E-05 per year, while that for D6 is 8.55E-05 per year. This means that the plant is better off if in R6, given it is at mid-loop. This does not contradict the comparison made earlier for a given time window of the POSs, because given that plant is in D6 the plant is more likely to be in the earlier time windows that have higher conditional core damage frequency. The averaged conditional core damage frequency over the POSs, shown in the right most column of Table 3, does show the trend of decreasing with decay heat.

Table 4 lists the key uncertainty characteristics of the core-damage frequencies for mid-loop operation and power operation, and shows that the core-damage for mid-loop operation has a wider spread than that of power operation. Note also that the mean total CDF in Table 4 is slightly different for the total CDF in Tables 1 and 2. This is because the numbers in Tables 1 and 2 are point estimates whereas the information in Table 4 reflects an uncertainty analysis.

2.2 Results from Internal Fire Analysis

Table 5 summarizes the point estimate results of the fire analysis. Note that the CDF is the frequency at which core damage occurs when the plant is at mid-loop. It accounts for the fact that the plant is at mid-loop only a small fraction of the time. The quantification indicates that certain scenarios in the H and J compartments of the emergency switchgear room, one scenario in the cable vault and tunnel, and one containment scenario dominate the CDF. The most dominant scenarios occur in the cable vault and tunnel (due to proximity of many emergency cables from both divisions in a closed, constrained space) and in the J room of the ESGR, where many emergency cables from both the H and the J divisions come together in close proximity (before entering the control room). In the containment, the relatively high CDF is due to a relatively high scenario frequency combined with non-separation of RHR trains over significant distances. Other scenarios are also important, due to a moderate damage from the fire combined with a relatively high scenario frequency.

POSs D6 and R6 are much more important than R10 (as R10 occurs in later time windows). D6 is more important than R6 owing to constraints imposed by a drained maintenance outage and its tendency to occur in earlier time windows.

The earlier time windows are more important than the later ones, with window 4 being relatively unimportant. Windows 1 and 2 are of the highest importance, with window 2 being significantly more important than window 1. While the decay heat is higher and the success criteria are more stringent in window 1, this window doesn't last as long and the outages tend to occur in the later time windows. The most risk significant fire initiator occurs in the cable vault tunnel area, in window 2 and POS D6, followed by a few scenarios in the J room of the ESGR, in the same window and POS.

Table 4 summarizes the result of the uncertainty analysis for core damage accidents initiated by fires. Also shown in the table are the uncertainty analysis results of the internal event analysis, as well as the mean value of the internal fire analysis of NUREG-1150.

No prevalence of fires at shutdown was noticed in the data, compared with power operation fires (after the construction events are taken out). It is true that there is greater potential for fires in certain categories (e.g., transient fires, fires caused by welding igniting cables, or other equipment fires). It is also true that the possibility of some types of fires is reduced (e.g., deenergized equipment,

Table 4 Result of the Level -1 Uncertainty Analysis and Comparison with Full Power Operation(per year)

	Study		Mean	5th Percentile	50th Percentile	95th Percentile	Error Factor
Internal Events	Full Power Operation - NUREG 1150 (per year)		4.01E-5	6.75E-6	2.31E-5	1.31E-4	4.41
	Full Power Operation- IPE		7.40E-5*	-	-	-	-
	Mid-Loop Operation (per year while at mid-loop)		4.86E-6	4.76E-7	2.14E-6	1.54E-5	5.69
Internal Fires	Full Power Operation - NUREG 1150 (per year)		1.13E-5	-	-	-	-
	Full Power Operation- IPE		**	-	-	-	-
	Mid-Loop Operation (per year while at mid-loop)		2.2E-5	1.4E-6	9.1E-6	7.6E-5	7.2
Internal Flood	Full Power Operation - NUREG 1150 (per year)		***	-	-	-	-
	Full Power Operation- IPE		5.0E05**	-	-	-	-
	Mid-Loop Operation (per year while at mid-loop)		4.8E-6	2.2E-7	1.7E-6	1.8E-5	9.0
Seismic Events	Full Power Operation - NUREG 1150 (per year)	LLNL	1.2E-4	-	-	-	33
		EPRI	4.01E-5	-	-	-	4.41
	Full Power Operation- IPE		**	-	-	-	-
	Mid-Loop Operation (per year while at mid-loop)****	LLNL	3.5E-7	1.3E-9	4.0E-8	1.4E-6	32
		EPRI	8.6E-7	2.5E-10	9.7E-9	3.7E-7	37

* point estimate
** not available
*** below truncation of 1.0E-8 per year
**** refueling outage only (no drained maintenance)

Table 5 Summary of Point Estimate Core Damage Frequencies for Fire Events (per year)

Fire Area	R6	D6	R10	Total
Emergency Switchgear Room	4.1E-6	8.2E-6	2.1E-7	1.3E-5
Containment	7.0E-8	5.5E-7	5.0E-9	6.3E-7
Cable Vault and Tunnel	1.3E-6	2.7E-6	7.4E-8	4.0E-6
Normal Switchgear Room	1.5E-8	3.5E-8	1.4E-9	5.1E-8
Main Control Room	7.0E-8	5.3E-7	4.4E-9	6.0E-7
Total	5.5E-6	1.2E-5	2.9E-7	1.8E-5

oil dripping on hot piping). A fire at shutdown is liable to be detected much sooner and extinguished in its early phases because of increased floor traffic. (Credit is taken for this by disallowing events that were discovered in the smoking stage (without flames) or early enough so that deenergizing equipment extinguished the fire.) Increased vigilance by licensees may play a part in this also. At Surry, a fire watch is in place during welding operations; fire doors are kept closed.

Human error events are not prominent contributors individually in terms of the Fussell-Vesely importance range (a few percent). Part of the reason is that there are many human error probabilities (HEPs), each applicable in a small fraction of sequences; another reason is in the values assigned to the HEPs; the third is that in many important scenarios hardware failures dominate because of heavy damage by fire.

Table 4 provides a comparison of the fire induced core damage frequency during mid-loop operation with that of power operation. Although the plant spends much less time in mid-loop, the core damage frequency is comparable to that of power operation. The main reason is that the routing of the cables of the equipment needed to support RHR operation or mitigate an accident during mid-loop operation is such that a single fire at a few critical locations can damage almost all the equipment need to mitigate the accident, while during power operation there are fewer critical locations.

2.3 Results from Internal Flood Analysis

The main results of the flood analysis are presented in Table 6, listing the point estimate core damage frequencies of the analyzed operating states. It was found that the most dominant contributors to core damage due to internal floods are accident scenarios initiated in the turbine building leading to the draining of the intake canal. This potentially could result in a flood encompassing the plant Emergency Switchgear Rooms (ESGR) leading to a two unit loss of all emergency power (F1 and F2 scenarios).The scenarios account for approximately 85% of the total core damage frequency (CDF) due to internal floods. This result is mainly due to the specific features of the Surry circulating water system and may not be applicable to other plants. The second most dominating flood scenario involves flooding of the Safeguard/Auxiliary Building in combination with the unavailability of the Refueling Water Storage Tank (RWST). The contribution of these scenarios (F4 and F5) is approximately 13% of the total internal CDF. Again, these specific findings may not be generalized to other plants due to the plant specific nature of the actual evolvement of these accident scenarios.

The main results of the uncertainty analysis are shown in Table 4, indicating the uncertainty bounds of the core damage frequency due to internal floods.

The internal flood CDF is dominated by Turbine Building flood events. These events are primarily initiated by either valve or expansion joint failures in the main inlet lines of the circulating water system. These failures may lead to pipe ruptures upstream of the condenser water box and inlet valves. At Surry the circulating water system is gravity fed from a very large capacity intake canal and its isolation may not be accomplished in a timely manner. This is in contrast with other common design arrangement where dedicated pumps provide the required cooling water flow through the system. In these designs, stopping the pumps would effectively isolate the system limiting potential water outflow.

The potential draining of the intake canal inventory in the Turbine Building is dominant due to a plant-specific spatial interdependence. For both units the Emergency Switchgear Rooms are located in the Service Building on the same elevation as the Turbine Building basement. These areas are separated by a fire door with 2 foot high flood dikes in front of them. A large scale flood could potentially overflow the dikes and enter into the two unit ESGR, leading to the potential loss of emergency power in both units, including the loss of Residual Heat Removal (RHR) stub busses. The normal off-site power supply to the plant would not be affected since the normal SGR is located at higher elevation in the Service Building.

Another important contributor to the internal flood CDF is due to flood events originated or entering into the Auxiliary Building. These flood scenarios, mainly from supply pipe ruptures from the RWST, result in the loss of all Component Cooling Water (CCW) and consequently the RHR function at the plant. This coupled with the unavailability of the RWST inventory to be injected into the reactor core leads to core damage. Again, the plant-specific spatial arrangement of piping and equipment is the main reason for the development of the accident scenario and its risk significance.

2.4 Results from Seismic Analysis

Table 4 shows the base case results. The base case consists of the Surry plant (systems and fragilities) at the Surry site with EPRI[9] and LLNL[10] seismic hazard curves. In this table, the mean, median, 5 percentile and 95 percentile frequencies of the two plant operating states are shown. It is seen from the table that mean annual frequency of the two plant operating states is less than 10^{-6} per year using either the LLNL or the EPRI seismic hazard curves. Therefore, we conclude that the seismic contribution to mean annual core damage frequency during both POS 6 and POS 10 is very small at Surry Unit 1

The comparison of core-damage-frequency results is also shown in Table 4. From examining the table, several important observations emerge:

- During shutdown conditions, the total annual mean CDF arising from earthquakes is small compared to the CDF arising from internal initiators: a factor of about 15 smaller for the LLNL seismic hazard curves and a factor of about 60 smaller using the EPRI hazard curves.

- . The seismic mean CDF during shutdown is small compared to the mean CDF at full power from seismic initiators from NUREG-1150: a factor of about 350 times smaller for the LLNL hazard curves and about 300 times smaller for the EPRI hazard curves.

- The Error Factor (EF) in this seismic study is significantly greater than the EF in the CDF from internal initiators during shutdown This is primarily due to the large uncertainty in the seismic hazard curves but another contribution arises from the uncertainty in the seismic fragilities.

A number of important insights emerge from this Surry analysis, including:

Core-damage frequency - The core-damage frequency for earthquake-initiated accidents during refueling outages in POS 6 and POS 10 is found to be low in absolute terms, below 10^{-6}/year. The reasons for this are (i) Surry's seismic capacity in responding to earthquakes during shutdown is excellent, well above its design basis and similar to its ability to respond to earthquakes during full-power conditions; (ii) the Surry site enjoys one of the least seismically active locations in the United States; (iii) the Surry plant is only in POS 6 and POS 10 (combined) for an average (mean) of 6.6% of the time. The core-damage frequencies are also low relative to the frequencies during POS 6 and POS 10 for internal initiators. This can be seen in Table 4.

The results are plant-specific - We believe that the results for Surry are highly plant-specific, in the sense that the seismic capacities, the specific sequences that are found to be most important, and the seismicity of the site are all difficult to generalize to other reactors elsewhere.

Shutdown seismic sequences are similar to full-power seismic sequences - Nevertheless, it is important to observe that all of the sequence types, components, and human errors that emerge in the key sequences in this analysis are similar or identical to sequences, components, and human errors that appear in typical full-power seismic PRAs. That is, nothing that has arisen as important in this study appears to be unique to earthquakes occurring during shutdown conditions. Whether this observation is generalizable to other reactors at other sites is unknown to us.

Sensitivities - Sensitivity studies reveal that if the Surry reactor were moved to the Zion site in Illinois (a typical midwestern site) or the Pilgrim site in Massachusetts (one of the most seismically active sites among all of the reactor sites in the eastern U.S.), the mean annual CDF from this study would increase by factors of about 1.8 and 10, respectively.

Uncertainties - While there are significant uncertainties in the numerical values of core-damage frequencies found in this study (see Table 4), the above conclusions are relatively robust --- they do not depend on the detailed numerical values found.

3. LEVEL 2/3 RESULTS

Table 7 presents statistical measures of the distributions for seven consequence measures for accidents during mid-loop operation obtained from this study. Similar statistical measures for full power operation obtained from the NUREG-1150 study of Surry are also included in the table. Table 7 indicates that the mean risk of offsite early health effects is over two orders of magnitude lower for accidents during mid-loop operation than for full power. This is due to the natural decay of the radionuclide inventory (because the accidents occur a long time after shutdown) particularly the short-lived isotopes of iodine and tellurium, which are primarily associated with early health effects. The distributions obtained for population dose (50 miles and 1000 miles) for mid-loop and full power operation are very similar. However the distributions for latent cancer fatalities differ by a factor of about three. The mid-loop study used the latest version of the MACCS code,[11] which incorporates the BEIR V[12] update to the latent cancer versus dose relationship, whereas NUREG-1150 used an older version of MACCS. The latest BEIR V update gives approximately a factor of three higher latent cancers for the same value of population dose.

In addition, scoping estimates of onsite doses were performed which indicate that the parking lot dose rates for accidents involving unisolated containment were high. This would limit the ability to take corrective actions, which cannot be performed from the control room, for this class of accidents.

The main finding of the study is that during mid-loop operation the risk of consequence measures related to long-term health effects, latent cancer fatalities and population dose, are high, comparable to those at full power, despite the much lower level of the decay heat and the radionuclide inventory. The reason for this is that containment is likely to be unisolated for a

Table 6 Summary of Point Estimate Core Damage Frequencies for Flood Events (/yr)

Scenario	Core Damage Frequency w/ Recovery			
	POS 6 Refueling	POS 6 Drained	POS 10 Refueling	Total
Turbine Building (F1)	1.9E-6	9.3E-8	1.5E-6	3.5E-6
Turbine Building (F2)	4.5E-7	2.2E-8	3.6E-7	8.3E-7
Auxiliary Building (F3)	4.7E-8	4.3E-8	1.2E-8	1.0E-7
Auxiliary Building (F4)	1.6E-7	5.7E-8	6.7E-8	2.8E-7
Safeguard Area (F5)	2.0E-7	8.9E-8	9.4E-8	3.8E-7
Spray in Containment (F6)	-	-	-	-
Mechanical Equipment Room No. #3 (F7)	1.0E-8	1.5E-8	7.8E-9	3.3E-8
Total-Flood	2.8E-6	3.2E-7	2.0E-6	5.1E-6

significant fraction of the accidents initiated during mid-loop operation so the releases to the environment are potentially large and the radionuclide species which mostly contribute to long-term health effects (such as cesium) have long half-lives. Accident sequences involving failure to correctly diagnose the situation or take proper actions are the largest contributors to the integrated risk. Another finding of the study is that the risk of early fatalities is low despite the unisolated containment due to the decay of the short-lived radionuclide species such as iodine and tellurium which contribute to early fatality risk. The integrated risk estimates have a range of uncertainty extending over approximately two orders of magnitude from the 5th to the 95th percentile of the distribution.

The accident sequences in which operators did not achieve containment isolation were the largest contributors to the core damage frequency during mid-loop operation and even larger contributors to the offsite risk estimates. Therefore, during mid-loop operation the probability of loss of containment integrity conditional on core damage was assessed to be high.

In comparison, in the full power study accident sequences that lead to station blackout were the largest contributors to core damage frequency but not to the offsite risk estimates. This is because containment performance at Surry was found to be very good for this class of accidents even if the molten core penetrates the lower head of the reactor vessel. Therefore accidents with lower frequencies but higher source terms which bypassed the containment, such as interfacing system loss of coolant accidents (ISLOCAs) and steam generator tube ruptures (SGTRs) were found to be the largest contributors to mean risk estimates in the full power study. Thus the loss of containment integrity conditional on core damage was determined to be small for severe accidents at full power. Therefore, although the core damage frequency distributions are an order of magnitude lower for mid-loop operation than for full-power operation, the frequencies of relatively large source terms are similar in both studies and hence the distributions for population dose are also similar.

4. CONCLUSIONS

This study was successful in developing a methodology to estimate the risk associated with the operation of a PWR during mid-loop operation. The methodology developed and the lessons learned from its application provide the NRC with new tools that could be used in subsequent analyses. The study concentrated the effort on mid-loop operation only. The core damage frequency contributions of other low power and shutdown POSs were analyzed in the coarse screening analysis of the phase 1 study, and remain to be analyzed in the future.

The following sections summarize the conclusions of the study.

4.1 Level 1 Conclusions

Internal Events - This study shows that the core-damage frequency due to internal events during mid-loop

Table 7 **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		Sigma	
	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-9	7.00E-8	4.90E-8	2.00E-6	1.59E-7	5.40E-6	1.69E-7	N.A.
Latent Fatalities within 50 mi	1.55E-4	N.A.	8.34E-4	N.A.	2.46E-3	N.A.	8.78E-3	N.A.	3.68E-3	N.A.
Latent Fatalities within 1000 mi	7.97E-4	3.10E-4	5.35E-3	2.20E-3	1.57E-2	5.20E-3	5.50E-2	1.90E-2	2.52E-2	N.A.
Population Dose within 50 mi	3.77E-3	5.90E-3	1.98E-2	2.70E-2	5.79E-2	5.80E-2	1.89E-1	2.50E-1	8.77E-2	N.A.
Population Dose within 1000 mi	1.87E-2	1.90E-2	1.25E-1	1.30E-1	3.66E-1	3.10E-1	1.29E+00	1.20E+00	5.90E-1	N.A.
Individual Early Fatalities Risk within 1 mi*	6.00E-12	1.40E-11	1.27E-10	8.70E-10	1.74E-9	1.60E-8	6.94E-9	4.90E-8	5.52E-9	N.A.
Individual Latent Fatalities Risk within 10 mi*	1.20E-10	1.60E-10	7.48E-10	4.90E-10	2.09E-9	1.70E-9	7.10E-9	8.10E-9	3.01E-9	N.A.

N.A. – Not Available
*NRC quantitative health objectives:
- Individual early fatality risk within one mile to be less than 5×10^{-7} per reactor year.
- Individual latent cancer fatality risk within 10 miles to be less than 2×10^{-6} per reactor year.

operation at the Surry plant is lower than that of power operation. This is mainly due to the much smaller fraction of time that the plant is at mid-loop. The conditional core damage frequency, that provides a measure of the vulnerability of the plant configuration with respect to core damage, is actually higher than that of power operation.

The time window approach developed in this study provides a more realistic approach to accounting for the changing decay heat during shutdown. Without it, the core damage frequency estimates could be an order of magnitude higher.

This study identified that only a few procedures are available for mitigating accidents that may occur during shutdown. Procedures written specifically for shutdown accidents would be useful. Realistic thermal hydraulic analysis should be used as the basis of the procedures.

We assumed that reduced-inventory check list was followed, and found that the maintenance unavailability of equipment not on the list were dominant contributors to system unavailability. However, the check list is believe to be sufficient for ensuring the availability of essential equipment. The dominant cause of cause damage is due to operator errors. We recognize that there is very large uncertainty in the human error probabilities used in this study.

Internal Fires - A comparison of the fire induced core damage frequency during mid-loop operation with that of power operation shows that, although the plant spends much less time in mid-loop, the core damage frequencies are comparable. The main reason is that the routing of the cables of the equipment needed to support RHR operation or mitigate an accident during mid-loop operation is such that a single fire at a few critical locations can damage almost all the equipment need to mitigate the accident, while during power operation much fewer critical locations exist.

Risk significant scenarios are found mainly in the emergency switchgear room (ESGR), the cable vault and tunnel (CVT). In the ESGR, several important scenarios (which are also the most risk significant ESGR scenarios) occur in locations where many cables for the H and the J emergency divisions come together in a close proximity. In the CVT, the tunnel part is a constrained space, where damage would quickly propagate to both divisions (serving many different emergency equipment). In the containment (CT), the risk significance stems from the relatively high fire frequency and non-separation of the two RHR divisions.

Internal Floods - The internal flood CDF is dominated by Turbine Building flood events. These events are primarily initiated by either valve or expansion joint failures in the main inlet lines of the circulating water system. These failures may lead to pipe ruptures upstream of the condenser water box and inlet valves. At Surry the circulating water system is gravity fed from a very large capacity intake canal and its isolation may not be accomplished in a timely manner. This is in contrast with other common design arrangement where dedicated pumps provide the required cooling water flow through the system.

The potential draining of the intake canal inventory in the Turbine Building is dominant due to a plant-specific spatial interdependence. For both units the Emergency Switchgear Rooms are located in the Service Building on the same elevation as the Turbine Building basement. These areas are separated by a fire door with 2 foot high flood dikes in front of them. A large scale flood could potentially overflow the dikes and enter into the two unit ESGR, leading to the potential loss of emergency power in both units, including the loss of stub busses that support the RHR pumps. The normal off-site power supply to the plant would not be affected since the normal SGR is located at higher elevation in the Service Building.

The flood initiating event analysis indicated that the shutdown and specifically the mid-loop operational period does not pose a unique flood risk with the exception of flood events coupled with loop isolation in Time Windows 2, 3 and 4. In general, the risk contribution from flood events is relatively significant and is dominated by potential flood events into the ESGR coupled with loop isolation.

Seismic Events - The core-damage frequency for earthquake-initiated accidents during refueling outages in POS 6 and POS 10 is found to be low in absolute terms, below 10^{-6}/year. The reasons for this are (I) Surry's seismic capacity in responding to earthquakes during shutdown is excellent, well above its design basis and similar to its ability to respond to earthquakes during full-power conditions; (ii) the Surry site enjoys one of the least seismically active locations in the United States; (iii) the Surry plant is only in POS 6 and POS 10 (combined) for an average (mean) of 6.6% of the time.

The seismic mean CDF during shutdown is small compared to the mean CDF at full power from seismic initiators from NUREG-1150: a factor of about 350 times smaller for the LLNL hazard curves and about 300 times smaller for the EPRI hazard curves.

4.2 Level 2/3 Conclusions

Comparison with Full Power Study - The mean core damage frequency for accidents during mid-loop operation is about an order of magnitude lower than the mean frequency of accidents caused by internal events at full power. However, the risk distributions obtained for comparable long term health consequences are very similar in the two studies. What this finding implies is that the lower decay heat and lower radionuclide inventory of the mid-loop operating state, compared with full power, is offset by the likelihood of containment being unisolated. Finally, the mean risk of early health effects is over two orders of magnitude lower for accidents during mid-loop operation than for accidents during full power operation. This is due to the natural decay of those radionuclide species which have the greatest impact on early fatality risk because accidents during mid-loop operation occur a long time after shutdown.

Comparison With the Safety Goals -Comparison of the results of this study against the NRC safety goals is done only for the two quantitative health objectives identified in the Commission's policy statement of August 1986. These objectives deal with individual early fatality and latent cancer fatality risks within 1 mile and 10 miles of the site, respectively. The numerical value of these objectives are given in Table 4. The 95th percentile of the distribution for individual latent cancer fatality risk falls more than an order of magnitude below the objective. The 95th percentile of the distribution for individual early fatality risk falls over two orders of magnitude below the corresponding health objective. The health objectives, however, apply to the total risk of the Surry plant. The risk estimates of this study are for accidents initited by internal events during mid-loop operation and therefore reflect only a fraction of the total risk at Surry.

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