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Results and Insights of Internal Fire and Internal Flood Analyses of the Surry Unit 1 Nuclear Power Plant during Mid-Loop Operations*

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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 (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL). The objectives of the program are to assess the risks of severe accidents initiated during plant operational states (POSs) other than full power operation and to compare the estimated core damage frequencies (CDFs), important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The scope of the program includes that of a Level 3 PRA for internal events and a Level 1 PRA for seismically induced and internal fire and flood induced core damage sequences.

A phased approach was used in the Level 1 program. In Phase 1, which was completed in Fall 1991, a coarse screening analysis examining accidents initiated by internal events (including internal fire and flood) was performed for all plant operational states. The objective of the Phase 1 study was to identify potential vulnerable plant configurations, to characterize (on a high, medium, or low CDF basis) the potential core damage accident scenarios and risk , and to provide a foundation for a detailed Phase 2 analysis.

In Phase 2, mid-loop operation was selected as the plant configuration to be analyzed based on the results of the Phase 1 study. The objective of the Phase 2 study is to perform a detailed analysis of the potential accident scenarios that may occur during mid-loop operation, and compare the results with those of NUREG-1150. The scope of the Phase 2 study includes a Level 3 PRA.

The core damage frequencies (per year) of the Surry plant due to different initiating events that may take place during mid-loop operations are estimated as follows:

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plant has two Pressurized Water Reactors (PWRs), each rated at 788 megawatts (electrical) capacity, and is located near Surry in Virginia. Grand Gulf, a boiling water reactor, was selected as the plant for a parallel analysis performed by Sandia National Laboratories.

A phased approach was taken in this project. In Phase 1, a broadly scoped screening analysis,[3] which included internal fire, flood, and seismic events was completed in late 1991. The objective of the Phase 1 study was to identify potential vulnerable plant configurations, to characterize (on a high, medium, or low CDF basis) the potential core damage accident scenarios, and to provide a foundation for a detailed Phase 2 analysis. This analysis produced a preliminary Level 1 PRA for accidents initiated during low power and shutdown (LP&S) and also gave insights on potential accident scenarios and potentially vulnerable configurations during low power and shutdown conditions.

In Phase 1, plant outages were grouped into 4 outage types: refueling, drained maintenance, non-drained maintenance with use of the residual heat removal (RHR) system, and non-drained maintenance without the RHR system. Due to the continuously changing plant configuration in any outage, plant operational states (POSs) were defined and characterized within each outage type. Each POS represents a unique set of operating conditions (e.g., temperature, pressure, and configuration). For example, in a refueling outage, up to 15 POSs were used, representing the evolution of the plant throughout a refueling from low power down to cold shutdown and refueling, and back-up to low power. An extensive effort was made to collect Surry-specific data to characterize each POS, that included reviewing operating and abnormal procedures for shutdown operations, the shift supervisor's log books, and the monthly operating reports, and performing supporting thermal-hydraulic calculations.

To accurately address each of the conditions identified in the Phase 1 study in detail would represent a very large effort. Consequently, the NRC decided to perform a detailed Phase 2 analysis on mid-loop operation. This configuration was selected because many incidents have occurred during mid-loop operations throughout the world. In addition, recent studies,[4] including Phase 1 of this program, found that the core damage frequency during mid-loop operation is comparable to that of power operation.

This paper summarizes the results and highlights of the internal fire and flood analysis documented in Volumes 3 and 4 of NUREG/CR-6144[5,6] performed for the Surry plant during mid-loop operation.

Modifications made to the model developed for this study account for procedures and other plant information available as of April 30, 1993. Regarding the plant's policy of avoiding mid-loop operation, it was decided that this study would use the data collected from past outages before the Unit 1 refueling outage of 1992. Consequently, the core damage frequency reported in this study could be overestimated by making this assumption. However, it is emphasized that the core damage frequency calculated in the current study was reduced significantly by changes made before April 1, 1993.

Internal Fires

The internal fire analysis follows the approach of a typical fire PRA. Surry-specific fire frequencies for important plant areas and various types of fire: cable fires, transient fires, and equipment fires (e.g., switchgear panels, pumps, etc.) were assessed using the latest available information. The fire is considered in this study as a localized phenomenon (as it usually is in nuclear power plants). Hence, the analysis pinpoints the precise location of possible sources of fire and vulnerable equipment and/or electrical cables within a plant fire area. For that reason, the cables of the most important systems (for this study) have been traced with a high degree of confidence. Fire growth calculations (with COMPBRN-IIIe) have been performed for the most vulnerable locations uncovered in our analysis. A sophisticated, transition-diagram type of suppression model has been used, which, in conjunction with the fire growth model, gives the damage fraction (i.e., the probability of damage given a fire in that location). The damage fraction was used in the event tree and fault tree models to arrive at the core damage frequency (CDF) from a given fire scenario. The scenario-dependent human error probabilities (HEPs) have been estimated using the same human reliability analysis (HRA) method used in the internal event analysis.

Key Assumptions

(1) Multiple initiators were not considered (e.g., loss of offsite power subsequent to a fire) for probabilistic reasons. For example, loss of offsite power probability in any 24 hour period is about 7.E-4 for the Surry plant at mid-loop. Multiplying this by the mid-loop time fraction of 0.066 means that the scenario frequency would have to be a few times 10^{-3} in order for it to survive screening. No such scenarios were found.

(2) Barrier failure was not considered as it was determined to be probabilistically unimportant. For initiator-causing fires that occur near an important barrier (very few examples were found), the combination of fire frequency, probability of equipment failures causing an initiator, the mid-loop POS fraction, and barrier failure probability (on the order of 10^{-2}-10^{-3}) was such that the fire scenario frequency was below the screening value. Also, comparing

Flood Event Frequency Analysis

The initiating frequency data was developed based on actual flooding events and updated to include plant-specific features and data. The data was gathered from various sources including the IPE Surry flood analysis, industry sources, licensing event reports (LERs). In some cases, plant specific models were developed especially with regard to the most important circulating water (CW) and service water (SW) lines in the Turbine Building. These two sources of flood dominate the risk, and therefore required special treatment. The most dominant failure mechanism for large pipes is pipe rupture due to water hammer events. Generic pipe failure data and models[9] were used when plant-specific data was unavailable with the exception of the CW and SW lines. The effects of flood barriers and other mitigating mechanisms were also considered in the development of damage states.

The equipment damage due to internal flood was determined as a function of flood level. The flood level depends on various factors such as the leakage rate, area of interconnections, drainage pathways, and potential mitigating actions. In order to take these factors into account, simple flood damage categories or damage states were developed. The damage states effectively incorporate the time dependency of the flood event and any potential mitigating or recovery actions. The damage states reflect a particular flood-exceedance frequency at a particular location with a predefined equipment damage level. This allows the incorporation of any partial failures due to flooding or differentiation between equipment damages at a particular location.

The flood initiating frequencies were developed to incorporate simple recovery actions as expected from the existing experience or data base. In addition, the data base was examined for its applicability to the Surry plant and modified to take into account certain plant-specific features.

Location and Scenario Identification

The identification of potential accident scenarios and flood locations was accomplished by initially using the general arrangement diagrams and the RHR system fault tree developed for the internal event analysis. The fault tree identifies all equipment and support components required for the RHR system operation. These were located and identified on the plant drawings and correlated with the flood areas as identified during the analysis. The scenario development also included the identification of potential propagation pathways and their physical nature (doors, stairways, etc.) which may limit the flood rate. In addition, drainage potential and other mitigating features which could help in the recovery process were also identified. Once the flood scenarios were identified, the effect on other front line systems or their components was determined. This was limited to only those systems which were required to prevent core damage and were identified by examining the

Internal Event Insights

Operator Response

The dominant cause of core damage was the operator's failure to mitigate the accident. (There is very large uncertainty in the human error probabilities used in this study.) In general, it would be beneficial to have good training, procedures, and instrumentation to ensure that the utility's staff can respond to shutdown accidents.

Procedures for Shutdown Accidents

Very few procedures are available for accidents during shutdown; the procedure for loss of decay heat removal, AP 27.00, is the only one that was written specifically for the shutdown scenarios analyzed in this study. The procedure is conservative with regard to the equipment needed to establish reflux cooling and feed-and-bleed. In this study, the use of fewer than the number of steam generators specified in the procedure for reflux cooling was treated as a recovery action, and a more realistic success criteria was used for feed-and-bleed when the decay heat is high. In most cases, the information in the procedures for power operation is helpful, for shutdown accidents. For example, the procedure for station blackout, ECA-0.0, gives instructions for dumping steam to the condenser. Credit for this procedure was taken into account in this study. However, some procedures written for power operation would mislead the operator if followed during shutdown. For example, the procedure for loss of offsite power, AP 10.00, states that "When the EDG is the only source of power to an emergency bus, the Component Cooling Pump should NOT be in service". During shutdown, CCW flow to the RHR heat exchanger is necessary for decay heat removal. Therefore, following this procedure under these circumstances would not be the most appropriate operator response.

Instrumentation

The level instrumentation used during mid-loop operation (i.e., standpipe level instrumentation and ultra-sonic level instrumentation) has limited applicability during a shutdown accident. The standpipe system indicates the correct level only when there is no build-up of pressure in the system. The ultra-sonic level instrumentation only provides level indication when the level is within the reactor coolant loops, and therefore, may not be useful during a feed and bleed operation.

Isolation of Reactor Coolant Loops

Isolation of the RCS loops is an important contributor to the core damage frequency. Review of the plant shutdown experience indicated that the reactor coolant loops are isolated for extended periods in a refueling outage, making the steam generators unavailable for decay heat removal upon loss of RHR. In a cold shutdown condition, the steam generators are usually maintained in the wet lay-up condition with the secondary side filled with water. During mid-loop operation, the availability of the SGs makes reflux cooling a possible method of mitigating a loss of RHR; this might be the only mitigation function available in a station blackout.

Valve Arrangement of Auxiliary Feedwater System and Main Steam System During Shutdown

The auxiliary feedwater system has six motor operated valves in the flow path to the steam generators, that are normally closed during shutdown. They are difficult to locate during a station blackout. Similarly, the main steam non-return valves are normally closed during shutdown, and have to be opened to use steam dump to the condenser. These valves depend on offsite power and would be very difficult to open without it.

Potential for Plugging the Containment Sump When Recirculation is Needed

Because of various activities, transient material and equipment are brought into containment during shutdown. For example, large plastic Herculite sheets are often used to separate work areas from the rest of the containment. When an accident requiring recirculation from the containment sump occurs (as in time windows 1 and 2) the material would increase the potential for plugging the containment sump.

Internal Fire Insights

Fires during shutdown was not as prevalent as fires during power operation (after the construction events are taken out). However, a greater potential does exist for fires in certain categories (e.g., transient or welding igniting cables or other equipment fires) during shutdown operations, but the possibility of some types of fires is reduced (e.g., deenergized equipment, 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 such that deenergizing equipment extinguished the fire.) At Surry, a fire watch is in place during welding operations and the fire doors are kept closed.

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.

4 Conclusion

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. The following sections summarize the conclusions of the study.

Internal Events

This study shows that the core damage frequency due to internal events during mid-loop operation at the Surry plant is lower by an order to magnitude than that of power operation. This is mainly due to the much smaller fraction of time that the plant is at mid-loop. The core damage frequency conditional on the plant being at mid-loop is actually higher than that of power operation.

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.

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

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

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 at mid-loop, the core damage

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Table 1 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)		4.0E-05	6.8E-06	2.3E-05	1.3E-04	4.4
	Full Power Operation- IPE		7.4E-05*	-	-	-	-
	Mid-Loop Operation (per year while at mid-loop)		4.9E-06	4.8E-07	2.1E-06	1.5E-05	5.7
Internal Fires	Full Power Operation - NUREG 1150 (per year)		1.1E-05	-	-	-	-
	Full Power Operation- IPE		**	-	-	-	-
	Mid-Loop Operation (per year while at mid-loop)		2.2E-05	1.4E-06	9.1E-06	7.6E-05	7.2
Internal Flood	Full Power Operation - NUREG 1150 (per year)		***	-	-	-	-
	Full Power Operation- IPE		5.0E-05**	-	-	-	-
	Mid-Loop Operation (per year while at mid-loop)		4.8E-06	2.2E-07	1.7E-06	1.8E-05	9.0
Seismic Events	Full Power Operation - NUREG 1150 (per year)	LLNL	1.2E-04	-	-	-	33
		EPRI	4.0E-05	-	-	-	4.4
	Full Power Operation- IPE		**	-	-	-	-
	Mid-Loop Operation (per year while at mid-loop)****	LLNL	3.5E-07	1.3E-09	4.0E-08	1.4E-06	32
		EPRI	8.6E-07	2.5E-10	9.7E-09	3.7E-07	37

* point estimate
** not available
*** below truncation of 1.0E-08 per year
**** refueling outage only (no drained maintenance)