4.3.2.3 Quantification

The basic event that represents recovery of offsite power for plant-centered and grid-related LOSPs is REC-OSP-PC. The data in NUREG/CR-5496 indicates that one event in 102 plant centered events resulted in a loss for greater than 24 hours, and all 6 of the grid centered events were recovered in a relatively short time. Therefore a non-recovery probability of 1E-02 is assumed.

4.3.2.4 Basic Event Probability

Basic Event	Basic Event Probability
REC-OSP-PC	1E-02

4.3.3 Top Event OCS – Cooling System Restart and Run

4.3.3.1 Event Description and Timing

This top event represents restarting the SFP cooling system, given that offsite power has been recovered within 24 hours. There are two electrically operated pumps and the operator can start either one. If the operator starts the pump that was in operation, no valve alignment would be required. However, if operator starts the standby pump, some valve alignment may be required.

Fault tree LP1-OCS has several basic events: an operator action representing the failure to establish SFP cooling, and several hardware failures of the system. If power is recovered within 24 hours, the operator has 9 hours to start the system before boil-off starts.

4.3.3.2 Relevant Assumptions

- The operators have 9 hours to start the SFP cooling system.
- The SFP has at least one SFP water temperature monitor, with either direct indication or a trouble light in the control room (there could also be indications or alarms associated with pump flow and pressure) (NEI commitment no. 5).
- Procedures exist for response to and recovery from a loss of power, and the operators are trained in their use (NEI commitment no. 2).

4.3.3.3 Quantification

Human Error Probabilities

Event HEP-SFP-STR-LP1 represents operator failure to restart/realign the SFP cooling system in 9 hours. The operator can restart the previously running pump and may not have to make any valve alignment. If he decides to restart the standby pump he may have to make some valve alignment. The response part of the error was quantified using SPAR. The relevant performance shaping factors for this event included expansive time, high stress due to previous failures, moderately complex task due to potential valve lineups, highly trained staff, good ergonomics (well laid out and labeled matching procedures), and good work process.

A diagnosis error HEP-DIAG-SFPLP1, representing failure of the operators to recognize the loss of SFP cooling was also included. Success would most likely result from recognition that the electric pumps stop running once power is lost and require restart following recovery of power. If the operator fails to make an early diagnosis of loss of SFP cooling, then success could still be achieved during walkdowns following the loss of offsite power. Alternatively, if power is restored, the operator will have alarms available as well. Therefore this value consists of two errors. The diagnosis error was calculated using SPAR, and the walkdown error was calculated using THERP. The relevant performance shaping factors included greater than 24 hours for diagnosis, high stress, well-trained operators, diagnostic procedures, and good work processes. A low dependence for the walkdown error was applied.

Because it is assumed that at most 9 hours are available, no credit was given for repair of the SFP cooling system.

Non-HEP Probabilities

Fault tree LP1-OCS represents failure of the SFP cooling system to restart and run. Hardware failure rates have been taken from INEL-96/0334 (Ref. 13). It is assumed that SFPC system will be maintained since it is required to be running all the time.

4.3.3.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-DIAG-SFPLP1	1.0E-06
HEP-SFP-STR-LP1	5.0E-6
SPC-CKV-CCF-H	1.9E-5
SPC-CKV-CCF-M	3.2E-5
SPC-HTX-CCF	1.9E-5
SPC-HTX-FTR	2.4E-4
SPC-HTX-PLG	2.2E-5
SPC-PMP-CCF	5.9E-4
SPC-PMP-FTF-1	3.9E-3
SPC-PMP-FTF-2	3.9E-3

4.3.4 Top Event OMK – Operator Recovery Using Makeup Systems

4.3.4.1 Event Description and Timing

This top event represents the failure to provide makeup using the firewater pumps. If offsite power is recovered then the fault tree LP1-OMK-U represents this top event. In this case, the operator has both electric and diesel firewater pumps available. If offsite power is not recovered then fault tree LP1-OMK-L represents this top event. In this case, the operator has only the diesel firewater pump available.

4.3.4.2 Relevant Assumptions

- It is assumed that the procedures guide the operators to wait until it is clear that spent fuel pool cooling cannot be reestablished (e.g., using cues such as the level drops to below the suction of the cooling system or the pool begins boiling) before using alternate makeup sources. Therefore, they have 88 hours to start a firewater pump
- There is a means to remotely align a makeup source to the spent fuel pool without entry to the refuel floor, so that makeup can be provided even when the environment is uninhabitable due to steam and/or high radiation (NEI commitment no.8)
- Repair crew is different than onsite operators
- Repair crew will focus recovery efforts only on one pump
- On average, it takes 10 hours to repair a pump if it fails to start and run
- It takes 16 hours to contact maintenance personnel, make a diagnosis, and get new parts
- Both firewater pumps are located in a separate structure or protected from the potential harsh environment in case of pool bulk boiling.
- Maintenance is performed per schedule on diesel and electric firewater pumps to maintain operable status
- Operators have received formal training on relevant procedures

4.3.4.3 Quantification

Human Error Probabilities

The fault tree LPI-OMK-U includes five human failure events and LPI-OMK-L has three.

Two events are common. HEP-RECG-FWSTART represents the failure of the operator to recognize the need to initiate firewater as an inventory makeup system, given that a loss of fuel pool cooling has been recognized. This event was quantified using the SPAR HRA technique. The assumptions included expansive time (> 24 hours), a high level of stress, diagnostic type procedures, good ergonomic interface, and good quality of work process.

HEP-FW-START represents failure to start either the electric or diesel firewater pump (depending upon availability) within 88 hours after the onset of bulk boiling, given that the decision to start a firewater pump was made. No difficult valve alignment is required, but the operator may have to position a hose in the pool area. This event was quantified using the SPAR HRA technique. The PSFs included expansive time (> 50 times the required time), high stress, highly complex task because of the multiple steps, its non-routine nature, quality procedures available, as well as good ergonomics including equipment and tools matched to procedure, and finally a crew who had executed these tasks before, conversant with the procedures and one another.

HEP-FW-REP-NODEP represents the failure of the repair crew to repair a firewater pump for the scenario where power is not recovered. Note that since it has been assumed that since power is not recovered, the repair crew did not make any attempt to repair the SFPC system, and therefore no dependency was modeled in the failure to repair the firewater system. Assuming that it takes another 16 hours before technical help and parts arrive, then the operator has 72 hours (88 hours less 16 hours) to repair the pump. Assuming a 10-hour mean time to repair, the probability of failure to repair the pump would be Exp [-(1/10) (72] = 1.0E-3. This event is modeled in the fault tree, LP1-OMK-L.

HEP-FW-REP-DEPEN represents the failure of the repair crew to repair a firewater pump. Note that repair was not credited for top event OCS; however, it has been assumed that the repair crew would have made an attempt to restore the SFPC system, and so dependency was modeled in the failure to repair the firewater system. A probability of failure to repair a pump in 88 hrs is estimated to be 1.0E-3. For HEP-FW-REP-DEPEN a low level of dependence was applied modifying the failure rate of 1.0E-3 to 5.0E-2 using the THERP formulation for low dependence. This event is modeled in the fault tree, LP1-OMK-U.

In addition, in fault tree LP1-OMK-U, the possibility that no action is taken has been included by incorporating an AND gate with basic events HEP-DIAG-SFPLPI and HEP-RECG-DEPEN. The latter is quantified on the assumption of a low dependency.

Hardware Failure Probabilities

In the case of LP1-OMK-U, both firewater pumps are available. Failure of both firewater pumps is represented by basic event FP-2PUMPS-FTF. In the case of LP1-OMK-L, only the diesel-driven firewater pump is available, and its failure is represented by basic event FP-DGPUMP-FTF.

The pump may be required to run 8 to 10 hours at the most (250 gpm capacity), given that the water inventory drops by 20 ft (i.e., 3 ft above the top of the fuel). A failure probability of 3.7E-3 for failure to start and run for the electric pump and 0.18 for the diesel driven pump are used

from INEL-96/0334. These individual pump failures result in a value of 0.18 for event FP-DGPUMP-FTF and 6.7E-4 for event FP-2PUMPS-FTF.

4.3.4.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-RECG-DEPEN	5E-02
HEP-RECG-FWSTART	2.0E-5
HEP-FW-START	1.0E-5
HEP-FW-REP-DEPEN	5.0E-2
FP-2PUMPS-FTF	6.7E-4
FP-DGPUMP-FTF	1.8E-1

4.3.5 Top Event OFD – Operator Recovery Using Offsite Sources

4.3.5.1 Event Description and Timing

Given the failure of recovery actions using onsite sources, this event accounts for recovery of coolant makeup using offsite sources such as procurement of a fire engine. Adequate time is available for this action, provided that the operator recognizes that recovery of cooling using onsite sources will not be successful, and that offsite sources are the only viable alternatives. Fault tree LP1-OFD represents this top event for the lower branch, and LP1-OFD-U for the upper branch. These fault trees contains those basic events from the fault trees LP1-OMK-U and LP1-OMK-L that relate to recognition of the need to initiate the fire water system; if OMK fails because the operator failed to recognize the need for firewater makeup, then it is assumed that the operator will fail here for the same reason.

4.3.5.2 Relevant Assumptions

- The operators have 88 hours to provide makeup and inventory cooling
- Procedures and training are in place that ensure that offsite resources can be brought to bear (NEI commitment no. 2 and 4), and that preparation for this contingency is made when it is realized that it may be necessary to supplement the pool makeup
- Procedures explicitly states that if the water level drops below a certain level (e.g., 15 ft below normal level) operator must initiate recovery using offsite sources
- Operators have received formal training in the procedures
- Offsite resources are familiar with the facility

4.3.5.3 Quantification

Human Error Probabilities

The event HEP-INV-OFFSITE represents failure to recognize that it is necessary to take the extreme measure of using offsite sources, given that even though there has been ample time up to this point to attempt recovery of both the SFP cooling system and both firewater pumps it has not been successful. This top event should include failures of both the diagnosis of the need to provide inventory from offsite sources, and the action itself. The availability of offsite resources is assumed not to be limiting on the assumption of an expansive preparation time. However, rather than use a calculated HEP directly, a low level of dependence to account for the possible detrimental effects of the failure to complete prior tasks successfully.

4.3.5.4 Basic Event Probability

Basic Event	Basic Event Probability
HEP-INV-OFFSITE	5.0E-2

4.3.6 Summary

Table 4.3 presents a summary of basic event probabilities used in the quantification of the Plant-centered and Grid-related Loss of Offsite Power event tree.

As in the case of the loss of cooling, and fire initiating events, based on the assumptions made, the frequency of core uncovery can be seen to be very low. Again, a careful and thorough adherence to NEI commitments 2, 5, 8 and 10, the assumption that walkdowns are performed on a regular, (once per shift) basis is important to compensate for potential failures to the instrumentation monitoring the status of the pool, the assumption that the procedures and/or training are explicit in giving guidance on the capability of the fuel pool makeup system, and when it becomes essential to supplement with alternate higher volume sources, the assumption that the procedures and training are sufficiently clear in giving guidance on early preparation for using the alternate makeup sources, are crucial to establishing the low frequency.

Table 4.3 Basic Event Summary for Plant-centered and Grid-related Loss of Offsite Power

Basic Event Name	Description	Probability
IE-LP1	Loss of offsite power due to plant-centered or grid-related causes	8.0E-2
REC-OSP-PC	Recovery of offsite power within 24 hours	1.0E-2
HEP-DIAG-SFPLP1	Operators fail to diagnose loss of SFP cooling due to loss of offsite power	1.0E-6
HEP-SFP-STR-LP1	Operators fail to restart and align the SFP cooling system once power is recovered	5.0E-6
HEP-RECG-FWSTART	Operators fail to diagnose need to start the firewater system	2.0E-5
HEP-DIAG-DEPEN	Operators fail to recognize need to cool pool given prior failure	5E-02
HEP-FW-START	Operators fail to start firewater pump and provide alignment	1.0E-5
HEP-FW-REP-NODEP	Repair crew fails to repair firewater system	1E-3
SPC-PMP-CCF	SFP cooling pumps – common cause failure	5.9E-4
SPC-PMP-FTF-1	SFP cooling pump 1 fails to start and run	3.9E-3
SPC-PMP-FTF-2	SFP cooling pump 2 fails to start and run	3.9E-3
FP-2PUMPS-FTF	Failure of firewater pump system	6.7E-4
FP-DGPUMP-FTF	Failure of the diesel-driven firewater pump	1.8E-1

4.4 Severe Weather Loss of Offsite Power Event Tree

This event tree represents the loss of SFP cooling resulting from a loss of offsite power from severe-weather-related events. Until offsite power is recovered, the electrical pumps would be unavailable, and only the diesel fire pump would be available to provide makeup.

Figure 4.4 shows the Severe Weather Loss of Offsite Power (LOSP) event tree sequence progression.

4.4.1 Initiating Event LP2 – Severe Weather Loss of Offsite Power

4.4.1.1 Event Description

Initiating event IE-LP2 represents severe-weather-related losses of offsite power. Severe weather threatens the safe operation of a SFP facility by simultaneously causing loss of offsite power and potentially draining regional resources or limiting their access to the facility. This event tree also differs from the plant-centered and grid-related LOSP event tree in that the probability of offsite power recovery is reduced.

4.4.1.2 Quantification

The LOSP frequency from severe weather events is 1.1E-2/yr, taken from NUREG/CR-5496 (Ref. 16).

4.4.2 Top Event OPR – Offsite Power Recovery

4.4.2.1 Event Description and Timing

The fault tree for this top event (LP2-OPR) is a single basic event that represents the non-recovery probability of offsite power. It is assumed that if power is recovered before boil-off starts (33 hours), the operator has a chance to reestablish cooling using the SFP cooling system.

4.4.2.2 Relevant Assumptions

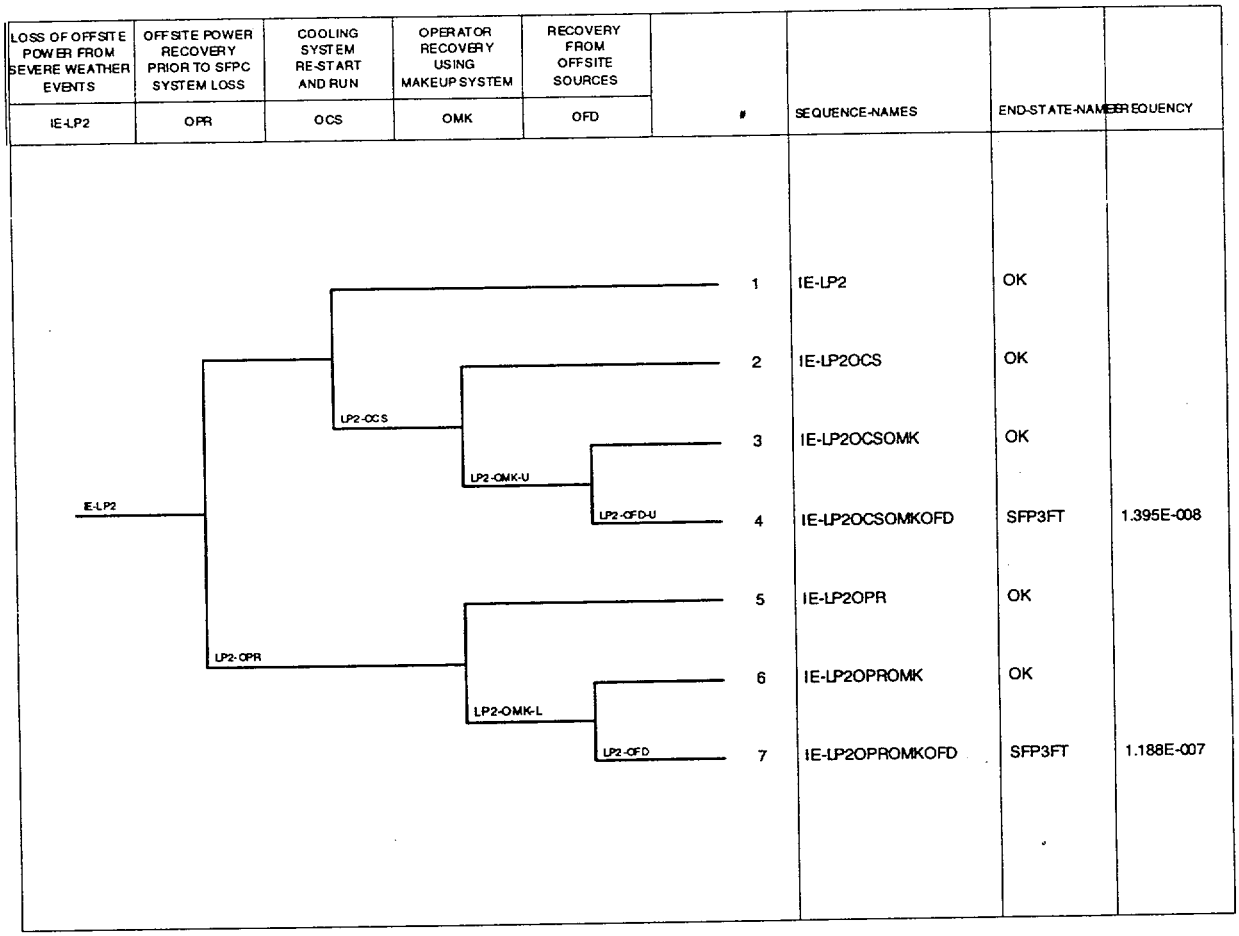
- See section 4.4.2.3 below.

4.4.2.3 Quantification

Non-HEP Probability

NUREG-1032 (Ref. 17) classified LOSP events into plant-centered, grid-related, and severe-weather-related categories, because these categories involved different mechanisms and also seemed to have different recovery times. Similarly, NUREG/CE-5496 divides LOSP events into three categories and estimates different values of non-recovery as functions of time. A non-recovery probability within 24 hrs for the offsite power from the severe weather event was estimated to be 2.0E-2 to <1.0E-4 depending on the location of the plant. In the operating plant, recovery of offsite power may be very efficient due to presence of skilled electricians. In the decommissioned plant, the skilled electricians may not be present at the site. Therefore, for the purpose of this analysis, a non-recovery probability for offsite power due to severe weather event (REC-OSP-SW) of 2.0E-2 is used.

Figure 4.4 Severe weather related loss of offsite power event tree



4.4.2.4 Basic Event Probability

Basic Event	Basic Event Probability
REC-OSP-SW	2.0E-2

4.4.3 Top Event OCS – Cooling System Restart and Run

4.4.3.1 Event Description and Timing

This top event represents restarting the SFP cooling system, given that offsite power has been recovered within 24 hours. There are two electrically operated pumps and the operator can start either one. If the operator starts the pump that was in operation, no valve alignment would be required. However, if operator starts the standby pump, some valve alignment may be required.

Fault tree LP2-OCS has several basic events: an event representing failure of the operators to realize they need to start the spent fuel pool cooling system, an operator action representing the failure to establish SFP cooling, and several hardware failures of the system. If power is recovered within 24 hours, the operator has 9 hours to start the system before boil-off starts. If he fails to initiate SFP cooling before boil-off begins, the operator must start a firewater pump to provide makeup.

4.4.3.2 Relevant Assumptions

- The operators have 9 hours to start the SFP cooling system before boil-off starts
- Operators have received formal training and there are procedures to guide them (NEI commitment no. 2)

4.4.3.3 Quantification

Human Error Probabilities

HEP-DIAG-SFPLP2 represents failure of the operator to recognize the loss of SFP cooling. Success could result from recognition that the electric pumps stop running once power is lost and require restart following recovery of power. If the operator fails to make an early diagnosis of loss of SFP cooling, then success could still be achieved during walkdowns following the loss of offsite power. Alternatively, if power is restored, the operator will have alarms available as well. Therefore this value consists of two errors. The diagnosis error was calculated using SPAR, and the walkdown error was calculated using THERP. The relevant performance shaping factors included greater than 24 hours for diagnosis, extreme stress, moderately complex task (due to potential complications from severe weather), diagnostic procedures, and

good work processes. A low dependence was applied to the walkdown error.

Event HEP-SFP-STR-LP2 represents operator failure to restart/realign the SFP cooling system in 9 hours. The operators can restart the previously running pump and may not have to make any valve alignment. If they decide to restart the standby pump they may have to make some valve alignment. This error was quantified using SPAR. The relevant performance shaping factors included expansive time, extreme stress due to severe weather, moderately complex task due to potential valve lineups and severe weather, poor ergonomics due to severe weather, and good work process.

If the system fails to start and run for a few hours then the operators would try to get the system repaired. Assuming that it takes another two shifts (16 hours) to contact maintenance personnel, make a diagnosis, and get new parts, and assuming an average repair time of 10 hours, there is not sufficient time to fix the system. Therefore, no credit was given for repair of the SFP cooling system.

Non-HEP Probabilities

Fault tree LP2-OCS represents failure of the SFP cooling system to restart and run. Hardware failure rates have been taken from INEL-96/0334. It is assumed that the SFPC system will be maintained since it is required to be running all the time.

4.4.3.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-DIAG-SFPLP2	2.0E-5
HEP-SFP-STR-LP2	5.0E-4
SPC-CKV-CCF-H	1.9E-5
SPC-CKV-CCF-M	3.2E-5
SPC-HTX-CCF	1.9E-5
SPC-HTX-FTR	2.4E-4
SPC-HTX-PLG	2.2E-5
SPC-PMP-CCF	5.9E-4
SPC-PMP-FTF-1	3.9E-3
SPC-PMP-FTF-2	3.9E-3

4.4.4 Top Event OMK – Operator Recovery Using Makeup Systems

4.4.4.1 Event Description and Timing

This top event represents the failure probability of the firewater pumps. If offsite power is recovered then the fault tree LP2-OMK-U represents this top event. In this case, the operators have both electric and diesel firewater pumps available. If offsite power is not recovered then fault tree LP2-OMK-L represents this top event. In this case, the operator has only the diesel firewater pump available.

4.4.4.2 Relevant Assumptions

- It is assumed that the procedures guide the operators to wait until it is clear that spent fuel pool cooling cannot be reestablished (e.g., using cues such as the level drops to below the suction of the cooling system or the pool begins boiling) before using alternate makeup sources. Therefore, they have 88 hours to start a firewater pump.
- Because of the severe weather, if one or both pumps fail to start or run, it is assumed that it takes another four to five shifts (48 hours) to contact maintenance personnel, perform the diagnosis, and get new parts. Therefore, the operator would have 40 hours (88 hours less 48 hours) to perform repairs.
- There is a means to remotely align a makeup source to the spent fuel pool without entry to the refuel floor, so that makeup can be provided even when the environment is uninhabitable due to steam and/or high radiation (NEI commitment no.8)
- Repair crew is different than onsite operators
- Repair crew will focus his recovery efforts on only one pump
- On average, it takes 10 hours to repair a pump if it fails to start and run
- It would take two days (48 hours) to contact maintenance personnel, make a diagnosis, and get new parts due to severe weather
- Both firewater pumps are located in a separate structure or protected from the potential harsh environment in case of pool bulk boiling
- Maintenance is performed per schedule on diesel and electric firewater pumps to maintain operable status
- Operators haves received formal training on relevant procedures

4.4.4.3 Quantification

Human Error Probabilities

The fault tree LP2-OMK-U has five operator actions, and LP2-OMK-I has three. Two of the events are common. HEP-RECG-FWST-SW represents the failure of the operator to recognize the need to initiate firewater as an inventory makeup system. This event was quantified using the SPAR HRA technique. The assumptions included expansive time (> 24 hours), extreme stress, highly trained staff, diagnostic type procedures, and good quality of work process. This diagnosis task provides the diagnosis for the subsequent actions taken to re-establish cooling to the pool.

HEP-FW-START-SW represents failure to start either the electric or diesel firewater pump (depending upon availability) within 88 hours after the onset of bulk boiling, given that the decision to start a firewater pump was made. No difficult valve alignment is required, but the operator may have to position a hose in the pool area. This event was quantified using the SPAR HRA technique. The PSFs chosen were; expansive time (> 50 times the required time), high stress, highly complex task because of the multiple steps and severe weather and its non-routine nature, quality procedures, poor ergonomics due to severe weather, and finally a crew who had executed these tasks before, conversant with the procedures and one another.

HEP-FW-REP-NODSW represents the failure of the repair crew to repair a firewater pump for the scenario where power is not recovered. Note that we have assumed that since power is not recovered, the repair crew did not make any attempt to repair the SFPC system, and therefore no dependency was modeled in the failure to repair the firewater system. We assume that the operator will focus his recovery efforts on only one pump. Assuming that it takes two days (48 hours) before technical help and parts arrive, then the operator has 40 hours (88 hours less 48 hours) to repair the pump. Assuming a 10-hour mean time to repair, the probability of failure to repair the pump would be Exp [-(1/10) (40)] = 2.5E-2. This event is modeled in the fault tree, LP2-OMK-L.

HEP-FW-REP-DEPSW represents the failure of the repair crew to repair a firewater pump for the scenario where power is recovered. Note that repair was not credited for top event OCS; however, we have assumed that the repair crew did make an attempt to restore the SFPC system, and so dependency was modeled in the failure to repair the firewater system. For HEP-FW-REP-DEPSW a low level of dependence was applied modifying the failure rate of 2.5E-2 to 7.0E-2 using the THERP formulation for low dependence.

In addition, in fault tree LP2-OMK-U, the possibility that no action is taken has been included by incorporating an OR gate with basic events HEP-DIAG-SFPLP2 and HEP-RECG-DEPEN. The latter is quantified on the assumption of a low dependency.

Non-HEP Probabilities

In the case of LP2-OMK-U, both firewater pumps are available. Failure of both firewater pumps is represented by basic event FP-2PUMPS-FTF.

In the case of LP2-OMK-L, only the diesel-driven firewater pump is available, and its failure is represented by basic event FP-DGPUMP-FTF.

The pump may be required to run 8 to 10 hours at the most (250 gpm capacity), given that the water inventory drops by 20 ft (i.e., 3 ft above the top of the fuel). A failure probability of 3.7E-3 for failure to start and run for the electric pump and 0.18 for the diesel driven pump are used from INEL-96/0334. These individual pump failures result in a value of 0.18 for event FP-DGPUMP-FTF and 6.7E-4 for event FP-2PUMPS-FTF.

4.4.4.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-RECG-FWST-SW	1.0E-4
HEP-RECG-DEPEN	5.0E-2
HEP-FW-START-SW	1.0E-3
HEP-FW-REP-DEPSW	7.0E-2
HEP-FW-REP-NODSW	2.5E-2
FP-2PUMPS-FTF	6.7E-4
FP-DGPUMP-FTF	1.8E-1
FP-DGPUMP-SW	5.0E-1

4.4.5 Top Event OFD – Operator Recovery Using Offsite Sources

4.4.5.1 Event Description and Timing

Given the failure of recovery actions using onsite sources, this event accounts for recovery of coolant makeup using offsite sources such as procurement of a fire engine. Adequate time is available for this action, provided that the operator recognizes that recovery of cooling using onsite sources will not be successful, and that offsite sources are the only viable alternatives. Fault tree LP2-OFD represents this top event for the lower branch (offsite power not recovered), and LP2-OFD-U for the upper branch. These fault trees contains those basic events from the fault trees LP2-OMK-U and LP2-OMK-L that relate to recognition of the need to initiate the firewater system; if OMK fails because the operator failed to recognize the need for firewater makeup, then it is assumed that the operator will fail here for the same reason.

4.4.5.2 Relevant Assumptions

- The operators have 88 hours to provide makeup and inventory cooling

- Procedures and training are in place that ensure that offsite resources can be brought to bear (NEI commitment no. 2, 3 and 4), and that preparation for this contingency is made when it is realized that it may be necessary to supplement the pool makeup

- Procedure explicitly states that if the water level drops below a certain level (e.g., 15 ft below normal level) operator must initiate recovery using offsite sources

- Offsite resources are familiar with the facility

4.4.5.3 Quantification

Human Error Probability

The event HEP-INV-OFFSITE represents failure to recognize that it is necessary to take the extreme measure of using offsite sources, given that even though there has been ample time up to this point to attempt recovery of both the SFP cooling system and both firewater pumps it has not been successful. This top event should include failures of both the diagnosis of the need to provide inventory from offsite sources, and the action itself. The availability of offsite resources is assumed not to be limiting on the assumption of an expansive preparation time. However, rather than use a calculated HEP directly, a low level of dependence to account for the possible detrimental effects of the failure to complete prior tasks successfully.

4.4.5.4 Basic Event Probability

Basic Event	Basic Event Probability
HEP-INV-OFFSITE	8.0E-2

4.4.6 Summary

Table 4.4 presents a summary of basic events used in the event tree for Loss of Offsite Power from severe weather events.

As in the case of the loss of offsite power from plant centered and grid related events, based on the assumptions made, the frequency of core uncovery can be seen to be very low. Again, a careful and thorough adherence to NEI commitments 2, 5, 8 and 10, the assumption that walkdowns are performed on a regular, (once per shift) basis is important to compensate for potential failures to the instrumentation monitoring the status of the pool, the assumption that the procedures and/or training are explicit in giving guidance on the capability of the fuel pool makeup system, and when it becomes essential to supplement with alternate higher volume sources, the assumption that the procedures and training are sufficiently clear in giving guidance on early preparation for using the alternate makeup sources, are crucial to establishing the low frequency. NEI commitment 3, related to establishing communication between on site and off site organizations during severe weather, is also important, though its importance is somewhat obscured by the assumption of dependence between the events OMK and OFD. However, if no such provision were made, the availability of offsite resources could

become more limiting.

Table 4.4 Basic Event Summary for Severe Weather Loss of Offsite Power

Basic Event Name	Description	Basic Event Probability
IE-LP2	LOSP event due to severe-weather-related causes	1.1E-02
HEP-DIAG-SFPLP2	Operators fail to diagnose loss of SFP cooling due to loss of offsite power	2.0E-5
HEP-RECG-DEPEN	Failure to recognize need to cool pool given prior failure	5.0E-2
HEP-SFP-STR-LP2	Operators fail to restart and align the SFP cooling system once power is recovered	5.0E-4
HEP-RECG-FWST-SW	Operators fail to diagnose need to start the firewater system	1.0E-4
HEP-FW-START-SW	Operators fail to start firewater pump and provide alignment	1.0E-3
HEP-FW-REP-DEPSW	Repair crew fails to repair firewater system	7.0E-2
HEP-FW-REP-NODSW	Repair crew fails to repair firewater system	2.5E-2
HEP-INV-OFFST-SW	Operators fail to provide alternate sources of cooling from offsite	8.0E-2
REC-OSP-SW	Recovery of offsite power within 24 hours	2.0E-2
SPC-CKV-CCF-H	Heat exchanger discharge check valves – CCF	1.9E-5
SPC-CKV-CCF-M	SFP cooling pump discharge check valves - CCF	3.2E-5
SPC-HTX-CCF	SFP heat exchangers – CCF	1.9E-5
SPC-HTX-FTR	SFP heat exchanger cooling system fails	2.4E-4
SPC-HTX-PLG	Heat exchanger plugs	2.2E-5

Basic Event Name	Description	Basic Event Probability
SPC-PMP-CCF	SFP cooling pumps – common cause failure	5.9E-4
SPC-PMP-FTF-1	SFP cooling pump 1 fails to start and run	3.9E-3
SPC-PMP-FTF-2	SFP cooling pump 2 fails to start and run	3.9E-3
FP-2PUMPS-FTF	Failure of firewater pump system	6.7E-4
FP-DGPUMP-FTF	Failure of the diesel-driven firewater pump	1.8E-1

4.5 Loss of Inventory Event Tree

This event tree (Figure 4.5) models general loss of inventory events, that are not the result of catastrophic failures that could result from dropped loads or seismic events. The following assumptions have been made in the development of the event tree.

- Maximum depth of siphon path is assumed to be 15 ft. below the normal pool water level (related to NEI commitments 6 and 7)
- Once the water level drops 15 ft below the normal pool water level, the losses would be only from the boiloff

4.5.1 Initiating Event LOI – Loss of Inventory

4.5.1.1 Event Description and Timing

This initiator (IE–LOI) includes loss of coolant inventory from events such as those resulting from configuration control errors, siphoning, piping failures, and gate and seal failures. Operational data provided in NUREG-1275 (Ref. 12), show that the frequency of loss of inventory events in which the level decreased more than one foot can be estimated to be less than one event per 100 reactor years. Most of these events were the result of operator error and were recoverable. NUREG-1275 shows that, except for one event that lasted for 72 hours, there were no events that lasted more than 24 hours. Eight events resulted in a level decrease of between one and five feet and another two events resulted in an inventory loss of between five and 10 feet.

4.5.1.2 Relevant Assumption

- NEI commitments 6 and 7 will reduce the likelihood of a significant initiating event

4.5.1.3 Quantification

The data reviewed during the development of NUREG-1275 (Ref. 12) indicated fewer than one event per 100 years in which level decreased over one foot. This would give a frequency of 1E-02. However, it is assumed that the NEI commitments 6 and 7 when implemented will reduce this frequency by an order of magnitude or more. Thus the frequency is estimated as 1E-03 per year.

4.5.2 Top Event NLL – Loss Exceeds Normal Makeup Capacity

4.5.2.1 Event Description and Timing

This phenomenological event divides the losses of inventory into two categories: those for which the leak size exceeds the capacity of the SFP makeup and therefore require isolation of the leak, and those for which the SFP makeup system's capacity is sufficient to prevent fuel uncovery without isolation of the leak.

4.5.2.2 Relevant Assumptions

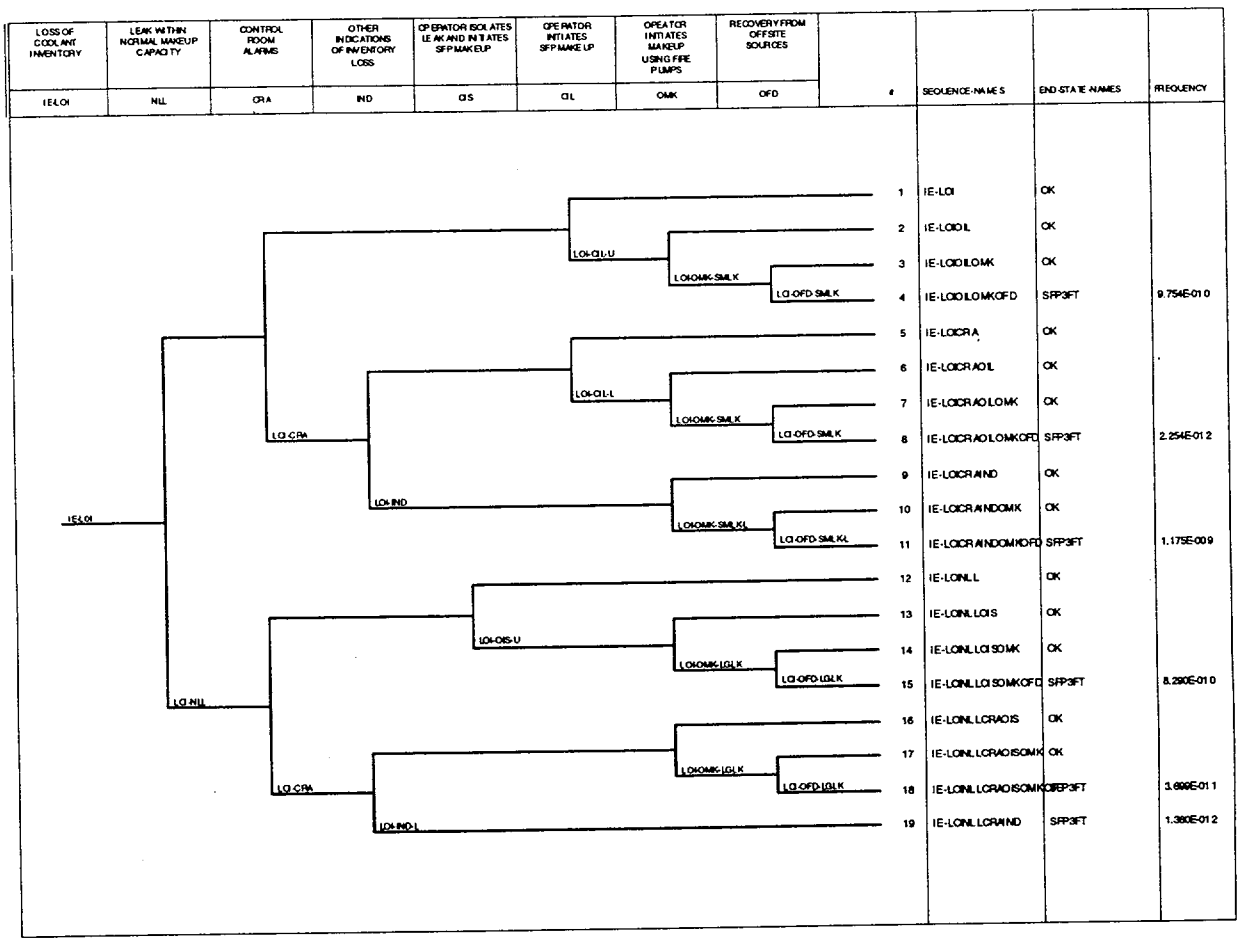
- In the case of a large leak, a leak rate is assumed to be twice the capacity of the SFP makeup system, i.e., 60 gpm
- The small leak is assumed for analysis purposes to be at the limit of the make-up system capacity, i.e., 30 gpm

4.5.2.3 Quantification

Non-HEP Probabilities

This top event is quantified by a single basic event, LOI-LGLK. From Table 3.2 of NUREG-1275, there were 38 events that lead to a loss of pool inventory. If we do not consider the load drop event (because this is treated separately), we have 37 events. Of these, 2 events involved level drops greater than 5 feet. Therefore, a probability of large leak event would be $2/37 \approx 0.06$ (6%). For the other 94% of the cases, operation of the makeup pump is sufficient to prevent fuel uncovery.

Figure 4.5 Loss of inventory event tree



4.5.3 Top Event CRA – Control Room Alarms

4.5.3.1 Event description and Timing

This top event represents the failure of the control room operators to respond to the initial loss of inventory from the spent fuel pool. This top event is represented by fault tree LOI-CRA. Depending on the leak size, the timings for the water level to drop below the level alarm set point (assumed 1 ft below the normal level) would vary. It is estimated that water level would drop below the low-level alarm set point in about 4 hours in the case of a small leak and in the case of a large leak, it would take 1 to 2 hours. Failure to respond could be due to operator failure to respond to an alarm, or loss of instrumentation system. Success for this event is defined as the operators recognizing the alarm as indicating a loss of inventory.

4.5.3.2 Relevant Assumptions

- Regular test and maintenance is performed on instrumentation (NEI commitment no. 10)
- Procedures are available to guide the operators on response to off-normal conditions, and the operators are trained on the use of these procedures (NEI commitment no. 2)
- System drawings are revised as needed to reflect current plant configuration
- SFP water level indicator is provided in the control room (NEI commitment no. 5)
- SFP low-water level alarm (narrow range) is provided in the control room (NEI commitment no. 5)
- Low level alarm set point is set to one foot below the normal level

4.5.3.3 Quantification

Human Error Probabilities

One operator error, HEP-DIAG-ALARM; is modeled under this top event. This event represents operator failure to respond after receiving a low-level alarm. Success is defined as the operator investigating the alarm and identifying the cause. This failure was quantified using The Technique for Human Error Prediction (THERP) Table 20-23. No distinction is made between the two leak sizes because this is treated as a simple annunciator response.

Non-HEP Probabilities

The value used for local faults leading to alarm channel failure, SPC-LVL-LOF (2.0E-3), was estimated based on information in NUREG-1275, Volume 12. This includes both local electrical faults and instrumentation faults.

4.5.3.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-DIAG-ALARM	3.0E-4
SPC-LVL-LOF	2.0E-3

4.5.4 Top Event IND – Other Indications of Inventory Loss

4.5.4.1 Event Description and Timing

This top event models operator failure to recognize the loss of inventory during walkdowns over subsequent shifts. Indications available to the operators include readouts in the control room, and a visibly decreasing water level. Eventually, when pool cooling is lost the environment would become noticeably hot and humid. Success for this event, in the context of the event tree, is treated differently for the small and large leaks.

For the small leak, it is defined as the operator recognizing the abnormal condition and understanding its cause in sufficient time to allow actions to prevent pool cooling from being lost. Failure of this top event does not lead to fuel uncovery. This top event is represented by the functional fault tree LOI-IND. Following an alarm, the operators would have in excess of 8 hrs before the water level would drop below the SFP cooling suction level. Therefore, for this event, only one shift is credited for recognition.

For the large leak, success is defined as recognizing there is a leak in sufficient time to allow make-up from alternate sources (fire water and offsite sources) before fuel uncovery. This top event is represented by the basic event LOI-IND-L. Based on the success criterion, there are many more opportunities for successive crews to recognize the need to take action. If the leakage is in the SFP cooling system, the leak would be isolated automatically once the water level drops below the SFP suction level. In this case, it would take more than 88 hrs (heatup plus boil-off) for the water level to reach 3 ft above the top fuel and the event would be similar to loss of spent fuel pool cooling. For the purpose of this analysis, it is assumed that leakage path is assumed to be below SFP cooling system suction level. It is assumed that once the water level drops 15 ft below normal pool level the leak is isolated automatically, and the inventory losses would be only due to boil-off. Time needed to boil-off to 3 ft above the top fuel is estimated to be 25 hours. Therefore, depending on the size of the leak and location and heatup rate, the total time available for operator actions after the first alarm before the water level drops below the SFP suction level to the 3 ft above the top of fuel would be more than 40 hrs. Furthermore, the indications become increasingly more compelling; with a large leak it would be expected that the water would be clearly visible, the level in the pool is obviously decreasing, and as the pool boils the environment in the pool area becomes increasingly hot and humid. Because of these very obvious physical changes, no dependence is assumed between the event IND and the event CRA. This lack of dependence is however, contingent on the fact that the operating crews performing walkdowns on a regular basis.

4.5.4.2 Relevant assumptions

- Operators have more than 40 hrs in the case of a large leak to take actions after the first alarm before the water level drops to the 3 ft above the top of fuel
- SFP water level indicator is provided in the control room e.g., camera or digital readout
- SFP low-water level alarm (narrow range) is provided in the control room
- System drawings are revised as needed to reflect current plant configuration
- Procedure/guidance exist for the operators to recognize and respond to indications of loss of inventory, and they are trained in the use of these procedures (NEI commitment no. 2)
- Water level measurement stick with clear marking is installed in the pool at a location that is easy to observe
- Operators are required to make a round per shift and document walkdowns in a log
- Training plans are revised as needed to reflect the changes in equipment configuration as they occur

4.5.4.3 Quantification

Human Error Probabilities

The top event LOI-IND, for small leaks, includes two HEPs, depending on whether the control room alarms have failed, or the operators failed to respond to the alarms. If the operators failed to respond to control room alarms, then event HEP-WLKDWN-DEPEN models the failure of the next shift to recognize the loss of cooling during a walkdown or during a control room review, taking into account a potential dependence on event HEP-DIAG-ALARM. A low dependence is assumed. If the alarms failed, then event HEP-WLKDWN-LOI models operator's failure to recognize the loss of inventory during walkdowns, with no dependence on previous HEPs. Because only one crew is credited, the HEP is estimated as 5E-03.

This failure probability is developed using THERP, and is based upon three individual failures: failure to carry out an inspection, missing a step in a written procedure, and misreading a measuring device.

The top event LOI-IND-L is modeled taking into account several opportunities for recovery by consecutive crews, and because the indications are so compelling no dependency is assumed between this HEP and the prior event.

4.5.4.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-WLKDWN-DEPEN	5.0E-2
HEP-WLKDWN-LOI-L	1.0E-5
HEP-WLKDWN-LOI	5.0E-3

4.5.5 Top Event OIS – Operator Isolates Leak and Initiates SFP Makeup

4.5.5.1 Event Description and Timing

This top event represents the operator's failure to isolate a large leak and initiate the SFP makeup system before the pool level drops below the SFP cooling system suction, and is represented by the fault tree LOI-OIS-U. Failure requires that the operators must provide the inventory using the firewater system or off-site resources.

The critical action here is the isolation of the leak. With the leak size assumed, and on the assumption that the low level alarm is set at 1 foot below the normal level, the operators have on the order of 4 hours to isolate the leak. Once the leak has been isolated, there would be considerable time available to initiate the normal make-up, since pool heat up to the point of initiation of boiling takes several hours.

If the loss of inventory is discovered through walkdowns, it is assumed that there is not enough time available to isolate the leak in time to provide for SFP makeup system success, and this event does not appear on the failure branch of event CRA.

4.5.5.2 Relevant Assumptions

- System drawings are kept up to date and training plans are revised as needed to reflect changes in plant configuration
- Operator has in excess of 4 hrs to isolate the leak and provide makeup
- There are procedures to guide the operators in how to deal with loss of inventory, and the operators are trained in their use (NEI commitment no. 2)
- Spent fuel pool operations that have the potential to rapidly drain the pool will be under strict administrative controls (NEI commitment no. 9). This increases the likelihood of the operators successfully terminating a leak should one occur.

4.5.5.3 Quantification

Human Error Probabilities

Two human failure events are included in the functional fault tree LOI-OIS, one for failure to

start the SFP makeup pump, HEP-MKUP-START, and one for failure to successfully isolate the leak, HEP-LEAK-ISO.

SPAR HRA worksheets were used to quantify each of these errors. For HEP-MKUP-START, it was assumed that the operator would be experiencing a high stress level, he is highly trained, the equipment associated with the task is well labeled and matched to a quality procedure, and the crew has effective interactions in a quality facility.

For HEP-LEAK-ISO it was assumed that the operators would be experiencing a high level of stress, the task is highly complex due to the fact that it is necessary to identify the source of the leak and it may be difficult to isolate, the operators are highly trained, have all the equipment available, and all components are well labeled and correspond to a procedure, and the crew has effective interactions in a quality facility.

Hardware Failure Probabilities

Unavailability of a SFP makeup system, SFP-REGMKUP-F, was assigned a value of 5.0E-2 from INEL-96/0334. It is assumed that SFP makeup system is maintained since it is required often to provide makeup.

4.5.5.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-LEAK-ISO	1.3E-3
HEP-MKUP-START	2.5E-4
SFP-REGMKUP-F	5.0E-2

4.5.6 Top Event OIL – Operator Initiates SFP Makeup System

4.5.6.1 Event Description and Timing

This top event represents the failure to initiate the SFP makeup system in time to prevent loss of spent fuel pool cooling, for a small leak. This top event is represented by the fault trees LOI-OIL-U and LOI-OIL-L, which include contributions from operator error and hardware failure. The leak is small enough that isolation is not required for success. If the operators respond to the initiator early (i.e., CRA is successful), they would have more than 8 hours to terminate the event using the SFP makeup system before the water level drops below the SFP suction level. If operators respond late (i.e., IND success), it is assumed that they would have on the order of 4 hours, based on the leak initiating at the start of one shift and the walkdown taking place at shift turnover.

4.5.6.2 Relevant Assumptions

- There are procedures to guide the operators in how to deal with loss of inventory, and the operators are trained in their use (NEI commitment no. 2).

- The manipulations required to start the make-up system can be achieved in less than 10 minutes.

4.5.6.3 Quantification

Human Error Probabilities

In the case of an early response operator would have more than 8 hours available to establish SFP makeup and the failure is represented by the basic event HEP-MKUP-START (see fault tree L OI-OIL-U). In the case of a late response, the operator is assumed to have 4 hours available to establish SFP makeup and is represented by the basic event HEP-MKUP-START-L (see fault tree L OI-OIL-L). Success is defined as the operator starting the makeup pump and performing valve manipulation as needed.

SPAR HRA worksheets were used to quantify each of these errors. For HEP-MKUP-START it was assumed that the 8 hour time window will allow more than 50 times the time required to complete this task, the operators are under high stress, are highly trained, have equipment that is well labeled and matched to a procedure, and the crew has effective interactions in a quality facility. For HEP-MKUP-START-L, the time available is not as extensive, and is considered nominal, all other PSFs being equal.

Hardware Failure Probabilities

Unavailability of a SFP makeup system, SFP-REGMKUP-F, was assigned a value of 5.0E-2 from INEL-96/0334. It is assumed that the SFP makeup system is maintained since it is required often to provide makeup.

4.5.6.4 Basic Event Proabablities

Basic Event	Basic Event Probability
HEP-MKUP-START-E	2.5E-4
HEP-MKUP-START	2.5E-6
SFP-REGMKUP-F	5.0E-2

4.5.7 Top Event OMK – Operator Initiates Makeup Using Fire Pumps

4.5.7.1 Event Description and Timing

This top event represents failure to provide make-up using the firewater pumps. The case of a large leak is represented by a fault tree LOI-OMK-LGLK. In this case the operators have 40 hours to start firewater system. The case of a small leak is represented by two functional fault trees, LOI-OMK-SMLK, and LOI-OMK-SMLK-L. The difference between the two trees is that in the first, the operators are aware of the problem and are attempting to solve it, whereas in the second, the operators will need to first recognize the problem. In both small leak cases, the

operator has more than 65 hrs to start firewater system. In all cases both the firewater pumps would be available.

4.5.7.2 Relevant Assumptions

- The operators have 40 to 65 hours to start a firewater pump depending on the leak size
- There is a means to remotely align a makeup source to the spent fuel pool without entry to the refuel floor, so that makeup can be provided even when the environment is uninhabitable due to steam and/or high radiation (NEI commitment no.8)
- Repair crew is different than onsite operators
- On average, it takes 10 hours to repair a pump if it fails to start and run
- It takes 16 hours to contact maintenance personnel, make a diagnosis, and get new parts
- Both firewater pumps are located in a separate structure and are protected from the potential harsh environment in the case of pool bulk boiling
- Maintenance and testing are performed on diesel and electric firewater pumps to maintain operable status (NEI commitment no. 10)
- There are procedures to guide the operators in how to deal with loss of inventory, and the operators are trained in their use. The guidance on when to begin addition of water from alternate sources is clear and related to a clearly identified condition, such as pool level or onset of boiling (NEI commitment no. 2).

4.5.7.3 Quantification

Human Error Probabilities

Each fault tree includes three human failure events. In the case of a functional fault tree LOI-OMK-SMLK, a basic event EP-RECG-FWSTART represents the failure of the operator to recognize the need to initiate firewater as an inventory makeup system; a basic event HEP-FW-START represents failure to start either the electric or diesel firewater pump; and a basic event HEP-FW-REP-NODSM represents the failure of the repair crew to repair a firewater pump.

For functional fault tree LOI-OMK-SMLK-L, the basic event EP-RECG-FWSTART is replaced by EP-RECG-FWSTART-L. This event requires that the operators recognize that the deteriorating conditions in the spent fuel pool are due to an inventory loss. The cues will include pool heat up due to the loss of spent fuel pool cooling which should be alarmed in the control room, as well as other physical indications such as increasing temperature and humidity, and a significant loss of level. Because of the nature of the sequence, the failure to

recognize the need for action will be modeled by assuming a low dependence between this event and the prior failures.

For functional fault tree LOI-OMK-LGLK, a basic event HEP-RECG-FW-LOI represents the failure of the operator to recognize the need to initiate firewater as an inventory makeup system; a basic event HEP-FW-START-LOI represents failure to start either the electric or diesel firewater pump; and a basic event HEP-FW-REP-NODLG represents the failure of the repair crew to repair a firewater pump.

SPAR HRA worksheets were also used to quantify the HEPs.

HEP-FW-START represents failure to start either the electric or diesel firewater pump (depending upon availability), given that the decision to start a firewater pump was made. No difficult valve alignment is required, but the operator may have to position a hose in the pool area, therefore, expansive time is assumed, with all other OSFs being the same as the other HEPs below.

For HEP-RECG-FWSTART it was assumed that extensive time is available to the operators for diagnosis, that the operators are under high stress, are highly trained, have a diagnostic procedure, have good instrumentation in the form of alarms, and are part of a crew that interacts well in a quality facility.

For HEP-RECG-FW-LOI it was assumed that extra time (>60 minutes) is available to the operators for diagnosis, that the operators are under high stress, are highly trained, have a diagnostic procedure, have good instrumentation in the form of alarms, and are part of a crew that interacts well in a quality facility.

For HEP-FW-START-LOI it was assumed that the operators are under high stress, are engaged in a highly complex task due to its non-routine nature, have a high level of training, have a diagnostic procedure, and are a part of a crew that interacts well in a quality facility.

Basic event HEP-FW-REP-NODS (see fault tree, OIL-OMK-SMLKL) represents the failure of the repair crew to repair a firewater pump for the small leak scenarios. Note that repairing the SFP regular makeup system is not modeled, as there would not be enough time to get help before the SFP makeup would be ineffectual and therefore no dependency was modeled in the failure to repair the firewater system. It is assumed that the operators will focus their recovery efforts on only one pump. Assuming that it takes another 16 hours before technical help and parts arrive, then the operators have about 50 hours (65 hours less 16 hours) to repair the pump. Therefore, assuming a 10-hour mean time to repair, the probability of failure to repair the pump would be Exp (-(1/10) * 49) = 7.5E-3 in the case of a small break scenario.

Basic event HEP-FW-REP-NODLG represents the failure of the repair crew to repair a firewater pump for the large leak scenarios. For this case there would only be 24 hours to repair the pump. Therefore, assuming a 10-hour mean time to repair, the probability of failure to repair the pump would be Exp (-(1/10) * 24) = 9.0E-2 in the case of a large break scenario.

Hardware Failure Probabilities

Failure of both firewater pumps is represented by basic event FP-2PUMPS-FTF. The pump may be required to run 8 to 10 hours at the most (250 gpm capacity), given that the water inventory drops by 20 ft (i.e., 3 ft from the top of the fuel). A failure probability of 3.7E-3 for failure to start and run for the electric pump and 0.18 for the diesel driven pump are used from INEL-96/0334. These individual pump failures result in a value 6.7E-4 for basic event FP-2PUMPS-FTF.

4.5.7.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-RECG-FWSTART	2.0E-5
HEP-RECG-FWSTART-L	5E-02
HEP-FW-START	1.0E-5
HEP-FW-REP-NODSM	7.5E-3
HEP-FW-REP-NODLG	9.0E-2
FP-2PUMPS-FTF	6.7E-4
HEP-RECG-FW-LOI	2.0E-4
HEP-FW-START-LOI	1.3E-3

4.5.8 Top Event OFD – Recovery From Offsite Sources

4.5.8.1 Event Description and Timing

Given the failure of recovery actions using onsite sources, this event accounts for recovery of coolant makeup using offsite sources such as procurement of a fire engine. This event is represented by the fault trees LOI-OFD-LGLK, LOI-OFD-SMLK and LOI-OFD-SMLK-L for the large break and two small break scenarios, respectively.

4.5.8.2 Relevant Assumptions

- The operator has 40 to 65 hours depending on the break size to provide makeup inventory and cooling
- Procedure explicitly states that if the water level drops below a certain level (e.g., 15 ft below normal level) operator must initiate recovery using offsite sources
- Operator has received formal training and there are procedures to guide him
- Offsite resources are familiar with the facility

4.5.8.3 Quantification

Human Error Probabilities

The only new basic events in these functional fault trees are HEP-INV-OFFST-LK and HEP-INV-OFFST. They were quantified using SPAR HRA worksheets. The diagnosis of the need to initiate the action is considered totally dependent on the recognition of the need to initiate inventory makeup with the fire water system. The PSFs are as follows: extreme stress (it's the last opportunity for success), high complexity because of the involvement of offsite personnel, highly trained staff with good procedures, good ergonomics (equipment is available to make offsite support straightforward) and good work processes. For both cases, a low level of dependence was assumed on the failure of prior tasks.

4.5.8.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-INV-OFFST-LK	5.0E-2
HEP-INV-OFFSITE	5.0E-2

4.5.9 Summary

Table 4.5 presents a summary of basic events.

As in the previous cases, the frequency of core uncovery can be seen to be very low. Again, a careful and thorough adherence to NEI commitments 2, 4, 5, 8 and 10, the assumption that walkdowns are performed on a regular, (once per shift) basis is important to compensate for potential failures to the instrumentation monitoring the status of the pool, the assumption that the procedures and/or training are explicit in giving guidance on the capability of the fuel pool makeup system, and when it becomes essential to supplement with alternate higher volume sources, the assumption that the procedures and training are sufficiently clear in giving guidance on early preparation for using the alternate makeup sources, are crucial to establishing the low frequency. NEI commitments 6, 7 and 9 have been credited with lowering the initiating event frequency.

Table 4.5 Basic Event Summary for the Loss of Inventory Event Tree

Basic Event Name	Description	Basic Event Probability
IE-LOI	Loss of inventory initiating event	1.0E-3
HEP-DIAG-LGLK	Operators fail to respond to a signal indication in the control room (large leak)	4.0E-4
HEP-DIAG-ALARM	Operators fail to respond to a signal indication in the control room	3.0E-4
HEP-WLKDWN-LOI	Operators fail to observe the LOI/loss of cooling in walkdowns, given failure to prevent loss of SFP cooling	5.0E-3
HEP-WLKDWN-LOI-L	Operators fail to observe the LOI/loss of cooling in walkdowns (independent case)	1.0E-5
HEP-WLKDWN-DEPEN	Operators fail to observe the LOI event walkdowns (dependent case)	5.0E-2
HEP-RECG-FW-LOI	Operators fail to diagnose need to start the firewater system	2.0E-4
HEP-RECG-FWSTART	Operators fail to diagnose need to start the firewater system	2.0E-5
HEP-RECG-FWSTART-L	Operators fail to diagnose need to start the firewater system given he failed to prevent loss of SFP cooling	5.0E-2
HEP-LEAK-ISO	Operators fail to isolate leak	1.3E-3
HEP-FW-START-LOI	Fails to start firewater pumps	1.3E-3
HEP-FW-START	Operators fail to start firewater pump and provide alignment	1.0E-5
HEP-FW-REP-NODLG	Fails to repair firewater pump (20 hrs)	9.0E-2
HEP-FW-REP-NODSM	Fails to repair firewater pump (49 hrs)	7.5E-3
HEP-INV-OFFST-LK	Operators fail to recover via offsite sources	5.0E-2
HEP-INV-OFFSITE	Operators fail to provide alternate sources of cooling from offsite	5.0E-2
FP-2PUMPS-FTF	Failure of firewater pump system	6.7E-4
LOI-LGLK	Loss exceeds normal makeup normal	6.0E-2
HEP-MKUP-START	Operators fail to start makeup(small leak)	2.5E-6
HEP-MKUP-START-E	Operators fail to start makeup(Early Respond)	2.5E-4
HEP-MKUP-START-L	Operators fail to start makeup(Late Respond)	1.0
SFP-REGMKUP-F	Regular SFP makeup system fails	5.0E-2
SPC-LVL-LOF	Failure of control room alarm channel	1.0E-5
SPC-LVL-LOP	Electrical faults leading to alarm channel failure	2.0E-3

5.0 SUMMARY OF RESULTS

The results of this analysis provide insight into the risks associated with storage of spent nuclear fuel in fuel pools at decommissioned nuclear power plants. The five accident initiators that were analyzed consist of: 1) Internal Fires, 2) Loss of Cooling, 3) Loss of Inventory, 4) Plant/Grid Centered Losses of Offsite Power, 5) Severe Weather Induced Losses of Offsite Power. The total frequency for the endstate is estimated to be 2.3E-7/year. Table 5.1 summarizes the core uncovery frequency for each accident sequence. The frequencies are point estimates, based on the use of point estimates for the input parameters. For the most part these input parameter values would be used as the mean values of the probability distributions that would be used in a calculation to propagate parameter uncertainty. Because the systems are essentially single train system, the point estimates therefore closely correlate to the mean values that would be obtained from a full propagation of parameter uncertainty.

The analysis has shown that, based on the assumptions made, the frequency of core uncovery from the loss of cooling, loss of inventory, loss of offsite power and fire initiating events is very low. The assumptions that have been made include that the licensee has adhered to NEI commitments 2, 4, 5, 8 and 10. In order to take full credit for these commitments, additional assumptions concerning how these commitments will be implemented have been made. These include: procedures and/or training are explicit in giving guidance on the capability of the fuel pool makeup system, and when it becomes essential to supplement with alternate higher volume sources; procedures and training are sufficiently clear in giving guidance on early preparation for using the alternate makeup sources; walkdowns are performed on a regular, (once per shift) basis. The latter is important to compensate for potential failures to the instrumentation monitoring the status of the pool.

NEI commitment 3, related to establishing communication between on site and off site organizations during severe weather, is also important, though its importance is somewhat obscured in the analysis by the assumption that there is some degree of dependence between the decision to implement supplemental makeup to the spent fuel pool from onsite sources such as fire water pumps, and that from offsite sources. However, if no such provision were made, the availability of offsite resources could become more limiting.

NEI commitments 6, 7 and 9 have been credited with lowering the initiating event frequency from its historical levels.

The worth of each individual commitment in achieving the low level of risk has not evaluated. The analysis has, however, demonstrated to the staff that, given an appropriate implementation of the commitments, the risk is indeed low, and would warrant consideration of granting exemptions. Check

Rich & John to review

Table 5.1 Summary of results

Sequence ID	*Core Uncovery Frequency (1/yr)*
IE-FIR-4	2.2E-008
IE-FIR-7	6.5E-010
IE-FIR-8	2.2E-008
IE-LOC-4	1.2E-008
IE-LOC-8	1.5E-010
IE-LOC-11	2.2E-009
IE-LOI-04	9.8E-010
IE-LOI-08	2.3E-012
IE-LOI-011	1.2E-009
IE-LOI-15	8.3E-010
IE-LOI-18	3.7E-011
IE-LOI-19	1.4E-012
IE-LP1-4	5.7E-009
IE-LP1-7	2.4E-008
IE-LP2-4	1.4E-008
IE-LP2-7	1.2E-007
TOTAL =	2.3E-007

1/23/00
11:00
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6.0 REFERENCES

1. Memorandum, G. M. Holahan (NRC) to J. A. Zwolinski (NRC), "Preliminary Draft Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants," June 16, 1999

2. Letter from L. Hendricks of the Nuclear Energy Institute (NEI) to R. Barrett of the USNRC, dated November 12, 1999

3. Letter, J. A. Lake (INEEL) to G. B. Kelly (NRC), "Details for the Spent Fuel Pool Operator Dose Calculations," CCN# 00-000479, October 20, 1999

4. K. D. Russell, et al., "Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 5.0: Technical Reference Manual," NUREG/CR-6116, July 1994

5. Williams, J. C., "A Data-Based Method for Assessing and Reducing Human Error to Improve Operational Performance", in Proceedings of the 1988 IEEE Conference on Human Factors and Poer PLants, Monterey, Ca., June 5-9, 1988, pp 436-450, Institiute of Electrical and Electronics Engineers, New York, NY, 1988

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7. Cooper, S. E., et al, "A Technique for Human Error Analysis (ATHEANA), NUREG/CR-6350, May 1996, USNRC

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11. U.S. Nuclear Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," NUREG-0612, July 1980

12. U.S. Nuclear Regulatory Commission, "Operating Experience Feedback Report - Assessment of Spent Fuel Cooling," NUREG-1275, Volume 12, February 1997

13. Idaho National Engineering and Environmental Laboratory, "Loss of Spent Fuel Pool Cooling PRA: Model and Results," INEL-96/0334, September 1996

14. Electric Power Research Institute, "Fire-Induced Vulnerability Evaluation (FIVE)," EPRI TR-100370s, April 1992

15. OREDA-92 Offshore Reliability Data Handbook, 2nd Edition, 1992.

6.0 REFERENCES

1. Memorandum, G. M. Holahan (NRC) to J. A. Zwolinski (NRC), "Preliminary Draft Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants," June 16, 1999

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   12

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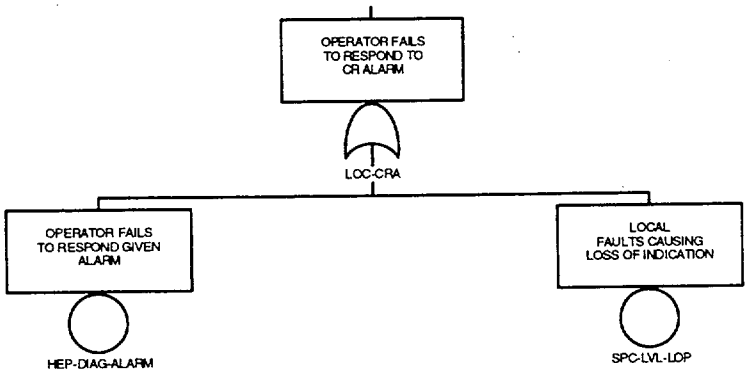
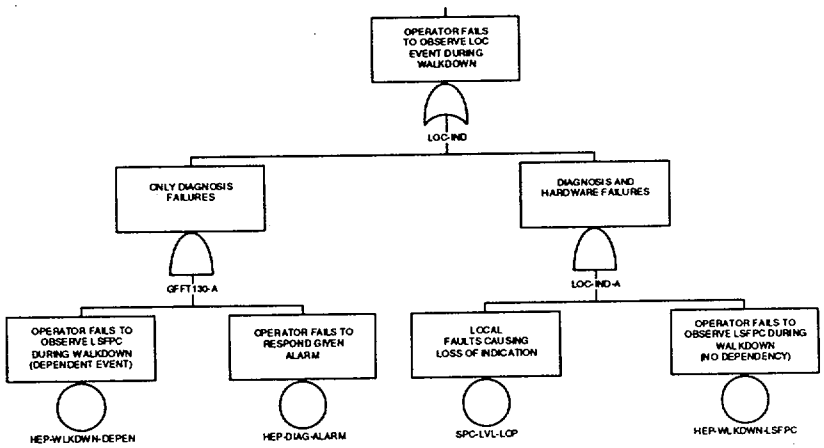
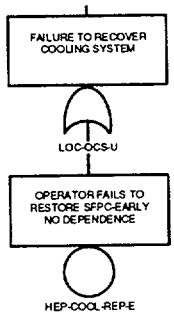
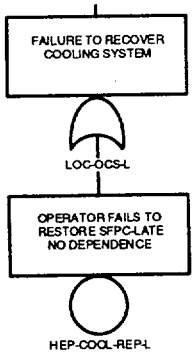
ATTACHMENT A

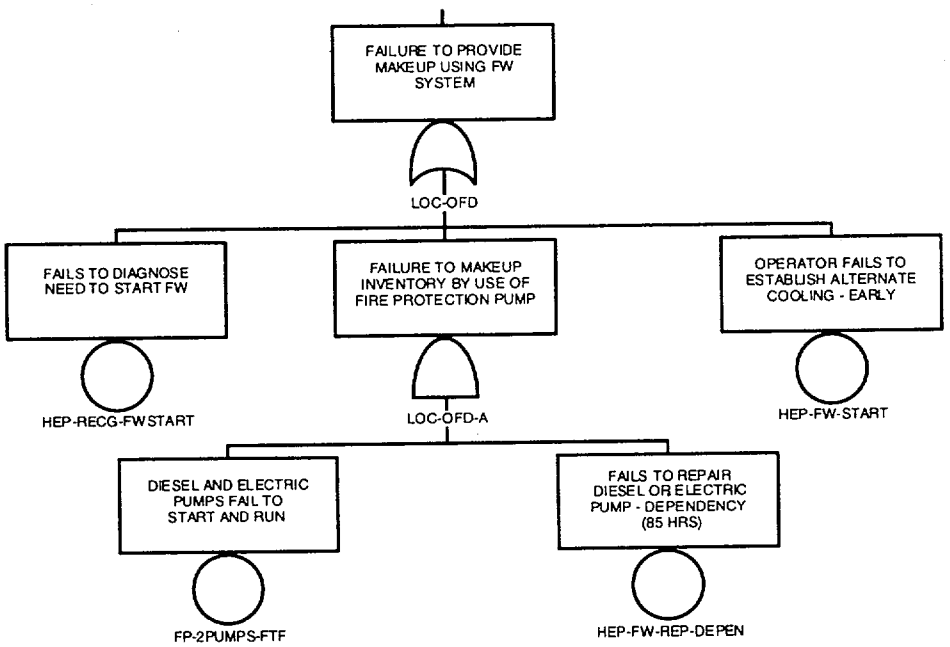
FAULT TREES USED IN THE RISK ANALYSIS

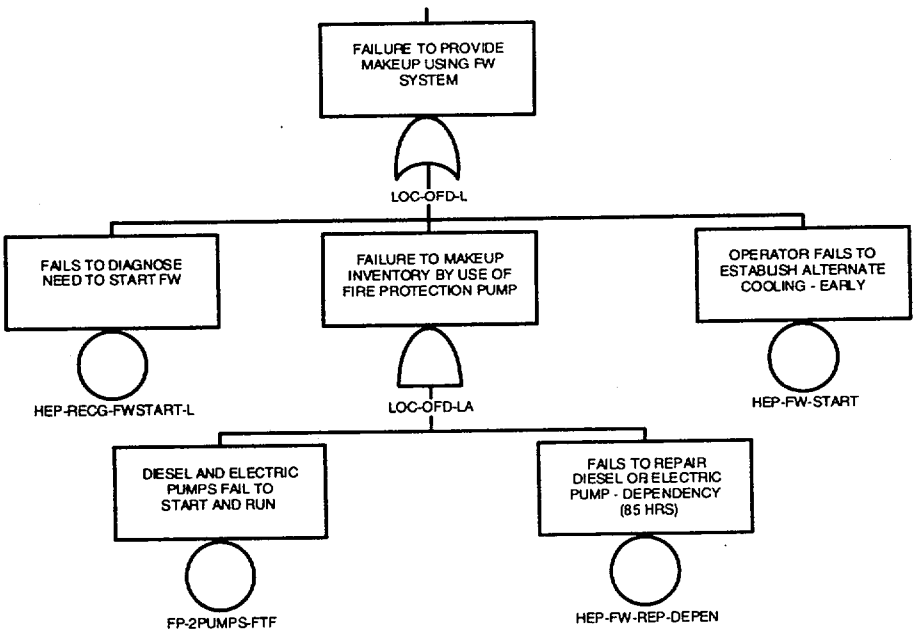

OPERATOR FAILS TO RESPOND TO CR ALARM
LOC-CRA
OPERATOR FAILS TO RESPOND GIVEN ALARM
LOCAL FAULTS CAUSING LOSS OF INDICATION
HEP-DIAG-ALARM
SPC-LVL-LOP


OPERATOR FAILS TO OBSERVE LOC EVENT DURING WALKDOWN
LOC-IND
ONLY DIAGNOSIS FAILURES
DIAGNOSIS AND HARDWARE FAILURES
GFFT130-A
LOC-IND-A
OPERATOR FAILS TO OBSERVE LSFPC DURING WALKDOWN (DEPENDENT EVENT)
OPERATOR FAILS TO RESPOND GIVEN ALARM
LOCAL FAULTS CAUSING LOSS OF INDICATION
OPERATOR FAILS TO OBSERVE LSFPC DURING WALKDOWN (NO DEPENDENCY)
HEP-WLKDWN-DEPEN
HEP-DIAG-ALARM
SPC-LVL-LOP
HEP-WLKDWN-LSFPC


FAILURE TO RECOVER COOLING SYSTEM
LOC-OCS-U
OPERATOR FAILS TO RESTORE SFPC-EARLY NO DEPENDENCE
HEP-COOL-REP-E


FAILURE TO RECOVER COOLING SYSTEM
LOC-OCS-L
OPERATOR FAILS TO RESTORE SFPC-LATE NO DEPENDENCE
HEP-COOL-REP-L


FAILURE TO PROVIDE MAKEUP USING FW SYSTEM
LOC-OFD
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPERATOR FAILS TO ESTABLISH ALTERNATE COOLING - EARLY
HEP-RECG-FWSTART
LOC-OFD-A
HEP-FW-START
DIESEL AND ELECTRIC PUMPS FAIL TO START AND RUN
FAILS TO REPAIR DIESEL OR ELECTRIC PUMP - DEPENDENCY (85 HRS)
FP-2PUMPS-FTF
HEP-FW-REP-DEPEN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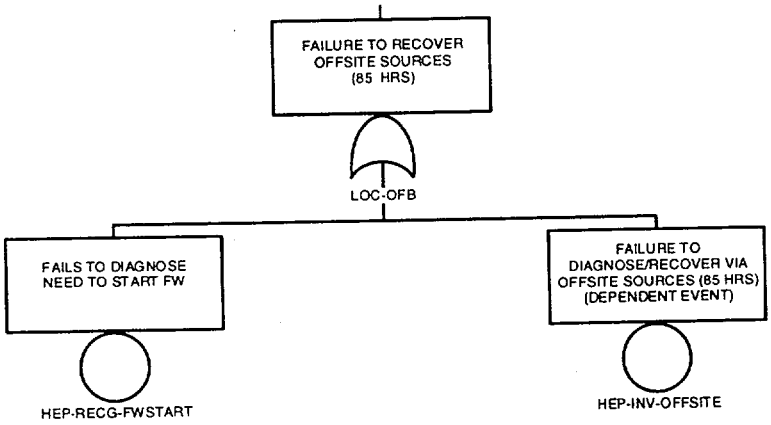
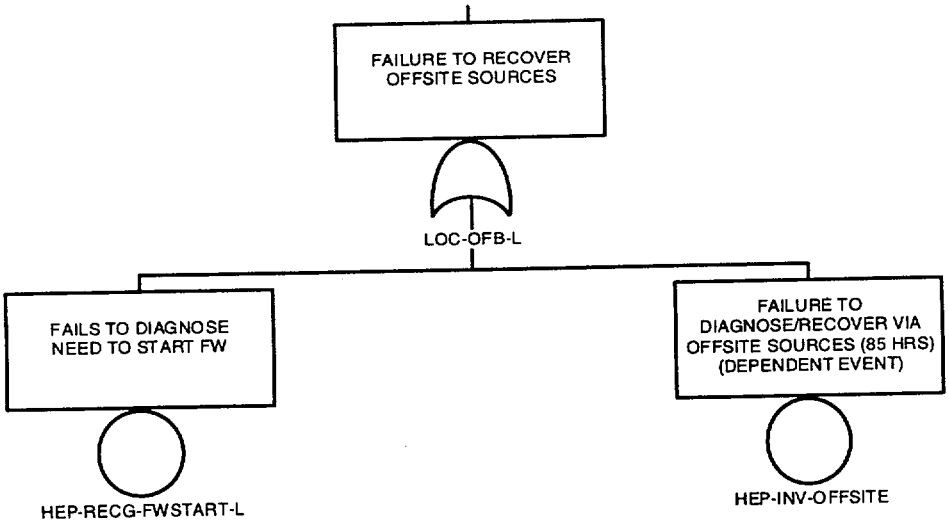
LOC-OFD-L
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPERATOR FAILS TO ESTABLISH ALTERNATE COOLING - EARLY
HEP-RECG-FWSTART-L
LOC-OFD-LA
HEP-FW-START
DIESEL AND ELECTRIC PUMPS FAIL TO START AND RUN
FAILS TO REPAIR DIESEL OR ELECTRIC PUMP - DEPENDENCY (85 HRS)
FP-2PUMPS-FTF
HEP-FW-REP-DEPEN


FAILURE TO RECOVER OFFSITE SOURCES (85 HRS)
LOC-OFB
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO DIAGNOSE/RECOVER VIA OFFSITE SOURCES (85 HRS) (DEPENDENT EVENT)
HEP-RECG-FWSTART
HEP-INV-OFFSITE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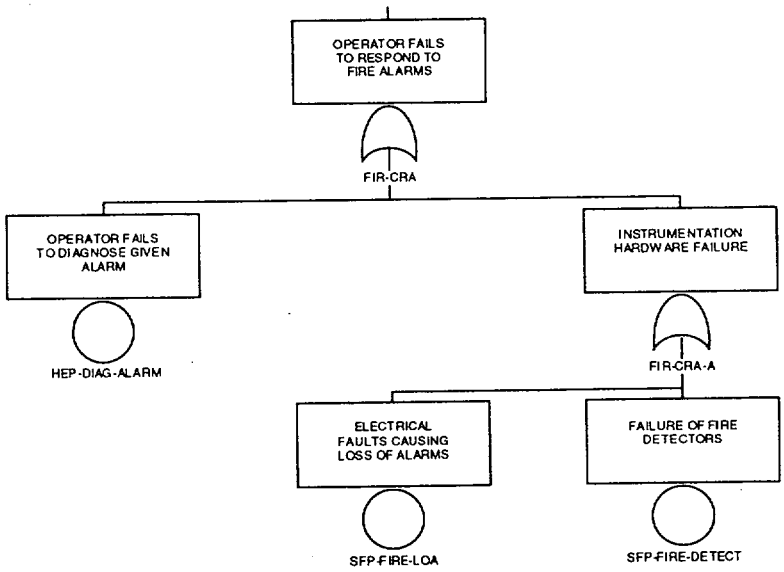
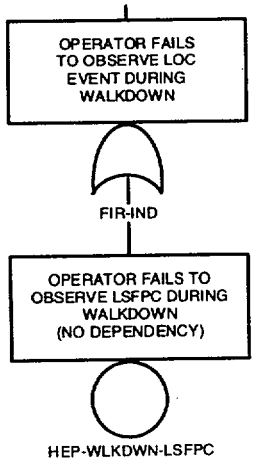
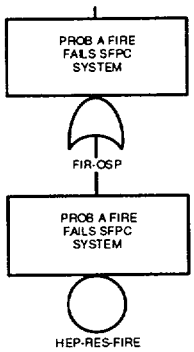
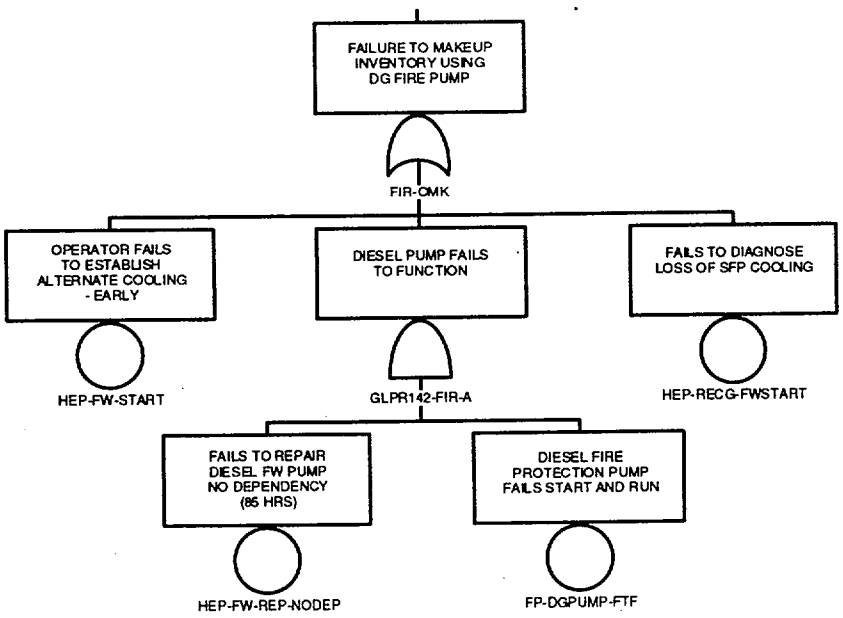
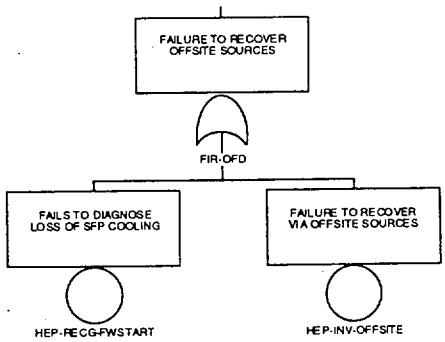
LOC-OFB-L
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO DIAGNOSE/RECOVER VIA OFFSITE SOURCES (85 HRS) (DEPENDENT EVENT)
HEP-RECG-FWSTART-L
HEP-INV-OFFSITE

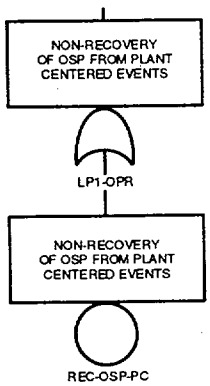

OPERATOR FAILS TO RESPOND TO FIRE ALARMS
FIR-CRA
OPERATOR FAILS TO DIAGNOSE GIVEN ALARM
HEP-DIAG-ALARM
INSTRUMENTATION HARDWARE FAILURE
FIR-CRA-A
ELECTRICAL FAULTS CAUSING LOSS OF ALARMS
SFP-FIRE-LOA
FAILURE OF FIRE DETECTORS
SFP-FIRE-DETECT

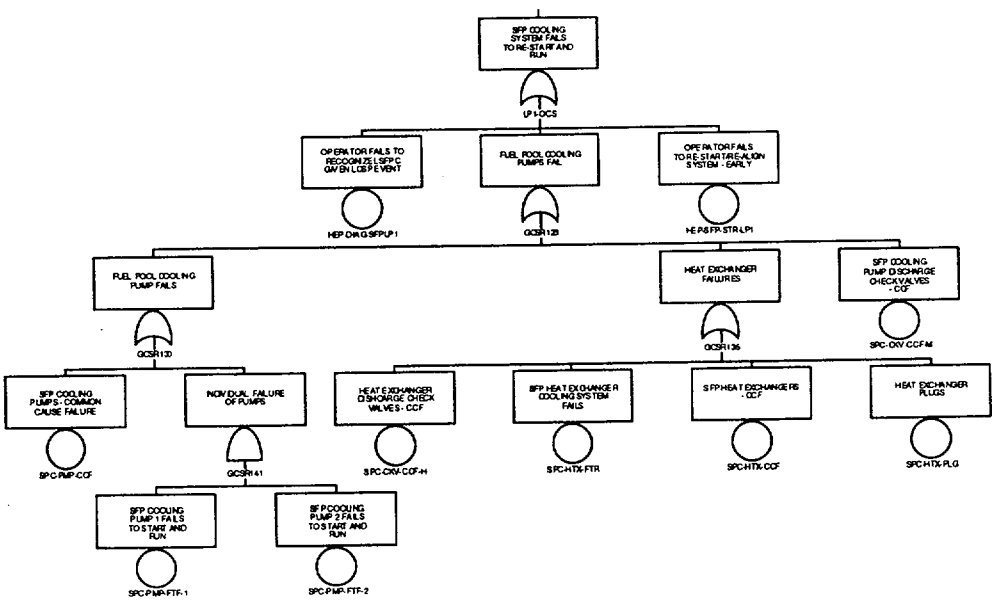

OPERATOR FAILS TO OBSERVE LOC EVENT DURING WALKDOWN
FIR-IND
OPERATOR FAILS TO OBSERVE LSFPC DURING WALKDOWN (NO DEPENDENCY)
HEP-WLKDWN-LSFPC

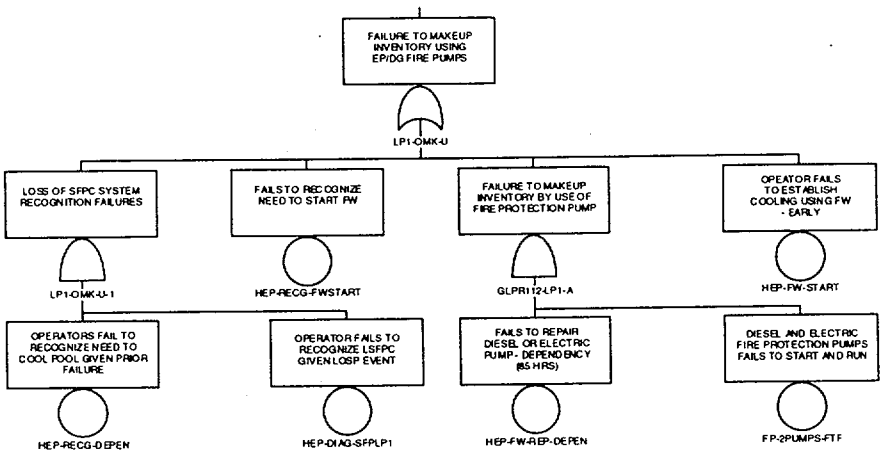

PROB A FIRE
FAILS SFPC
SYSTEM
FIR-OSP
PROB A FIRE
FAILS SFPC
SYSTEM
HEP-RES-FIRE

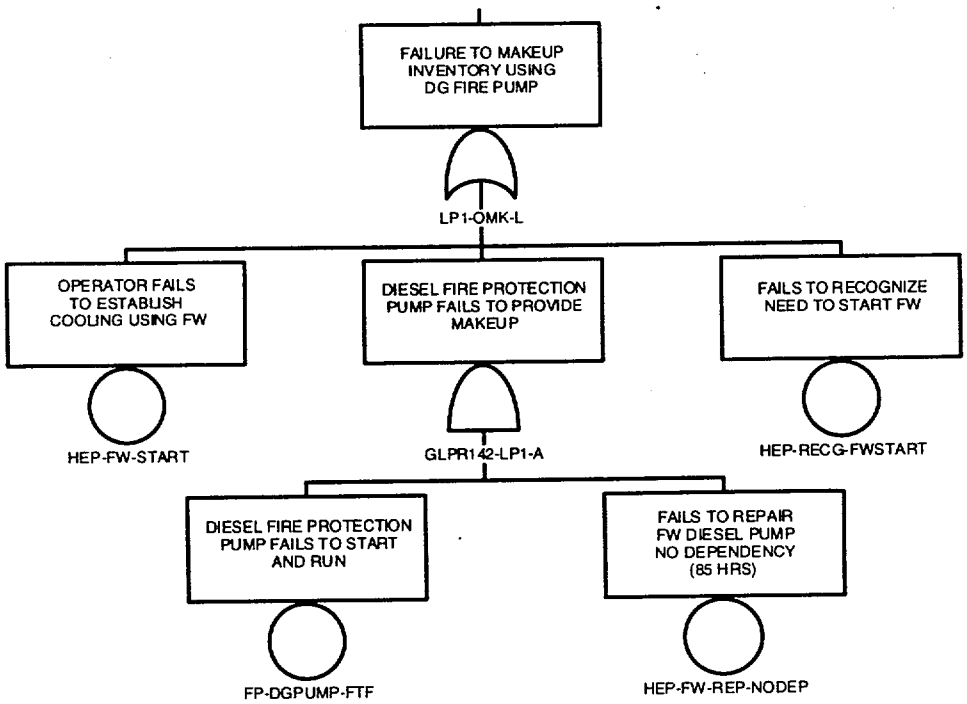

FAILURE TO MAKEUP
INVENTORY USING
DG FIRE PUMP
FIR-OMK
OPERATOR FAILS
TO ESTABLISH
ALTERNATE COOLING
- EARLY
DIESEL PUMP FAILS
TO FUNCTION
FAILS TO DIAGNOSE
LOSS OF SFP COOLING
HEP-FW-START
GLPR142-FIR-A
HEP-RECG-FWSTART
FAILS TO REPAIR
DIESEL FW PUMP
NO DEPENDENCY
(85 HRS)
DIESEL FIRE
PROTECTION PUMP
FAILS START AND RUN
HEP-FW-REP-NODEP
FP-DGPUMP-FTF

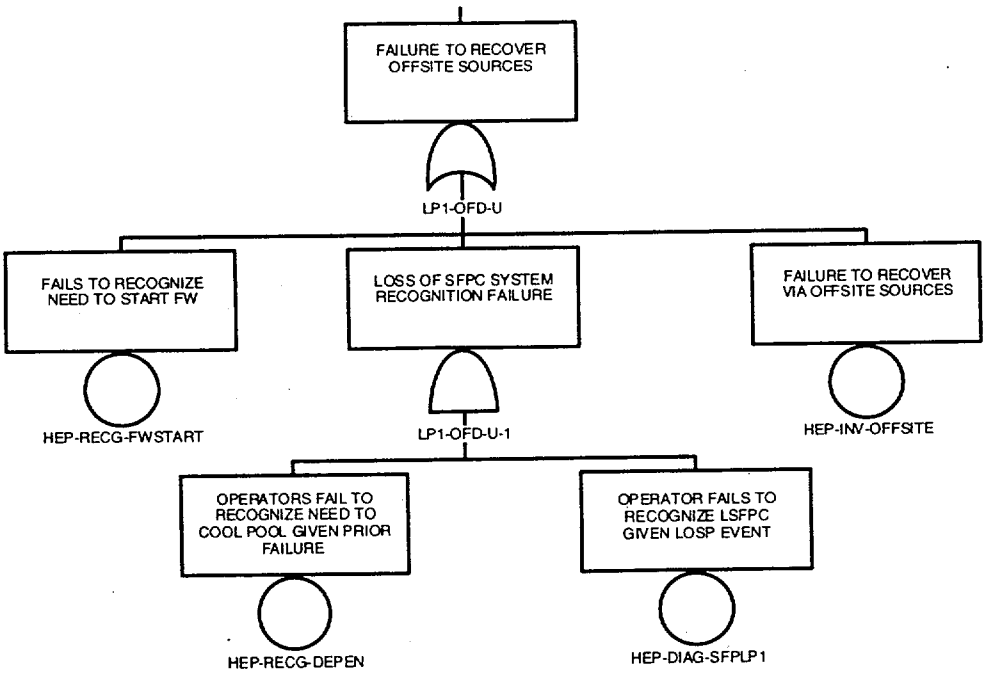

FAILURE TO RECOVER OFFSITE SOURCES
FIR-OFD
FAILS TO DIAGNOSE LOSS OF SFP COOLING
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FWSTART
HEP-INV-OFFSITE

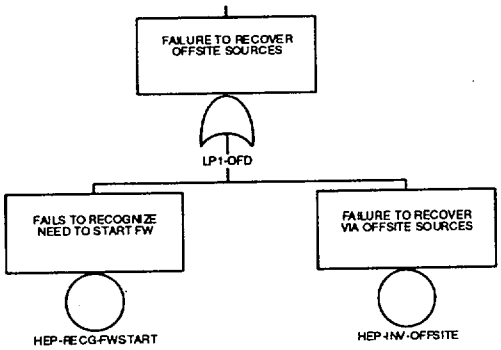

NON-RECOVERY
OF OSP FROM PLANT
CENTERED EVENTS
LP1-OPR
NON-RECOVERY
OF OSP FROM PLANT
CENTERED EVENTS
REC-OSP-PC

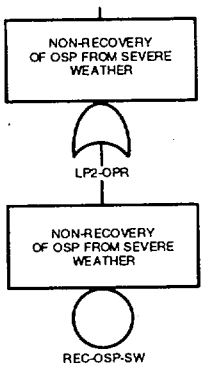

SFP COOLING
SYSTEM FAILS
TO RE-START AND
RUN
LP1-OCS
OPERATOR FAILS TO
RECOGNIZE LSFP C
GAIN LOSP EVENT
FUEL POOL COOLING
PUMPS FAIL
OPERATOR FAILS
TO RE-START/ALIGN
SYSTEM - EARLY
HEP-OIG-SFP-LP1
GCSR120
HEP-SFP-STR-LP1
FUEL POOL COOLING
PUMP FAILS
HEAT EXCHANGER
FAILURES
SFP COOLING
PUMP DISCHARGE
CHECK VALVES
- CCF
GCSR120
GCSR136
SFC-CKV-CCF-M
SFP COOLING
PUMPS - COMMON
CAUSE FAILURE
INDIVIDUAL FAILURE
OF PUMPS
HEAT EXCHANGER
DISCHARGE CHECK
VALVES - CCF
SFP HEAT EXCHANGER
COOLING SYSTEM
FAILS
SFP HEAT EXCHANGERS
- CCF
HEAT EXCHANGER
PLUGS
SFC-PMP-CCF
GCSR141
SFC-CKV-CCF-H
SFC-HTX-FTR
SFC-HTX-CCF
SFC-HTX-PLG
SFP COOLING
PUMP 1 FAILS
TO START AND
RUN
SFP COOLING
PUMP 2 FAILS
TO START AND
RUN
SFC-PMP-FTF-1
SFC-PMP-FTF-2

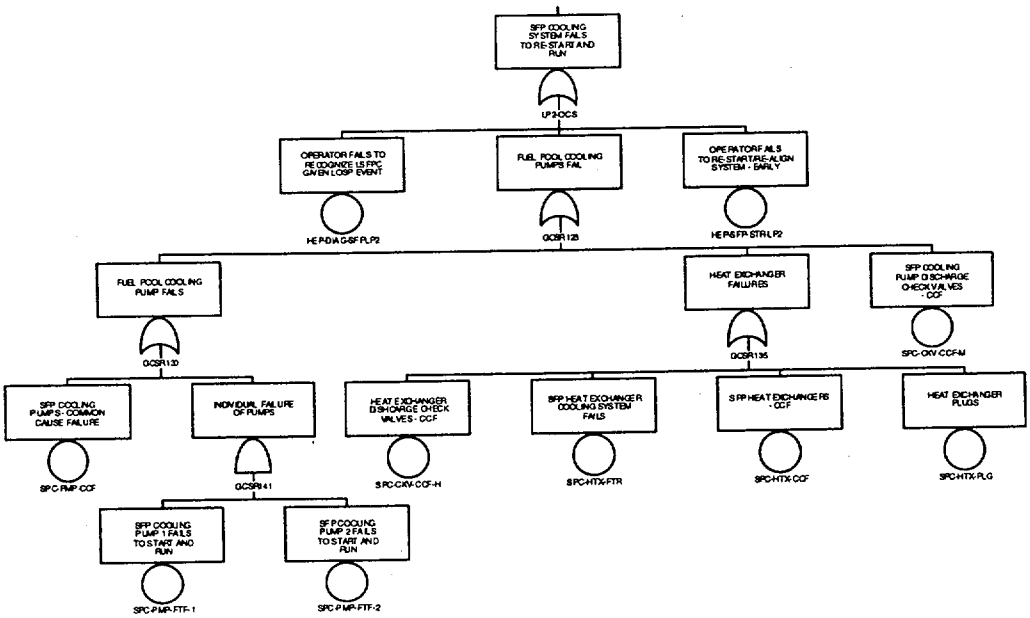

FAILURE TO MAKEUP INVENTORY USING EP/DG FIRE PUMPS
LP1-OMK-U
LOSS OF SFPC SYSTEM RECOGNITION FAILURES
FAILS TO RECOGNIZE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPEATOR FAILS TO ESTABLISH COOLING USING FW - EARLY
LP1-OMK-U-1
HEP-RECG-FWSTART
GLPR112-LP1-A
HEP-FW-START
OPERATORS FAIL TO RECOGNIZE NEED TO COOL POOL GIVEN PRIOR FAILURE
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP EVENT
FAILS TO REPAIR DIESEL OR ELECTRIC PUMP - DEPENDENCY (85 HRS)
DIESEL AND ELECTRIC FIRE PROTECTION PUMPS FAILS TO START AND RUN
HEP-RECG-DEPEN
HEP-DIAG-SFPLP1
HEP-FW-REP-DEPEN
FP-2PUMPS-FTF

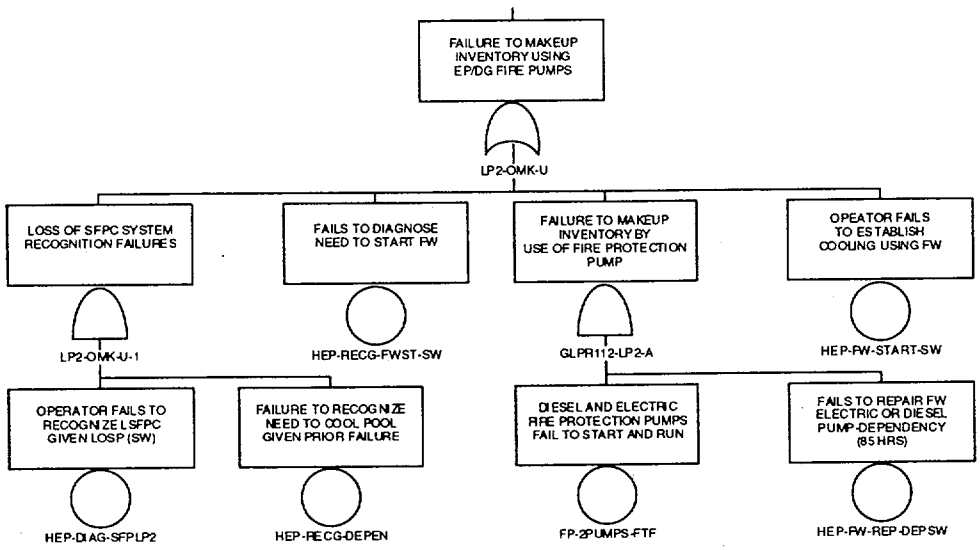

FAILURE TO MAKEUP INVENTORY USING DG FIRE PUMP
LP1-OMK-L
OPERATOR FAILS TO ESTABLISH COOLING USING FW
DIESEL FIRE PROTECTION PUMP FAILS TO PROVIDE MAKEUP
FAILS TO RECOGNIZE NEED TO START FW
HEP-FW-START
GLPR142-LP1-A
HEP-RECG-FWSTART
DIESEL FIRE PROTECTION PUMP FAILS TO START AND RUN
FAILS TO REPAIR FW DIESEL PUMP NO DEPENDENCY (85 HRS)
FP-DGPUMP-FTF
HEP-FW-REP-NODEP

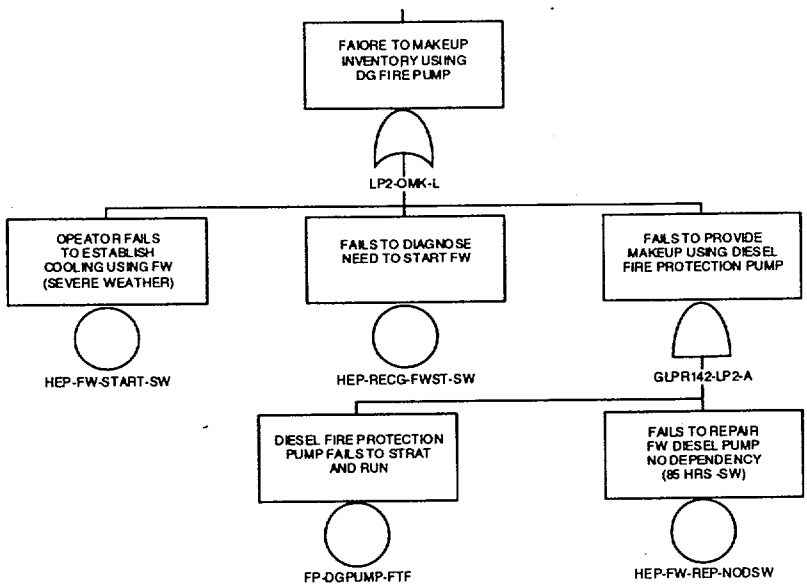

FAILURE TO RECOVER OFFSITE SOURCES
LP1-OFD-U
FAILS TO RECOGNIZE NEED TO START FW
LOSS OF SFPC SYSTEM RECOGNITION FAILURE
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FWSTART
LP1-OFD-U-1
HEP-INV-OFFSITE
OPERATORS FAIL TO RECOGNIZE NEED TO COOL POOL GIVEN PRIOR FAILURE
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP EVENT
HEP-RECG-DEPEN
HEP-DIAG-SFPLP1

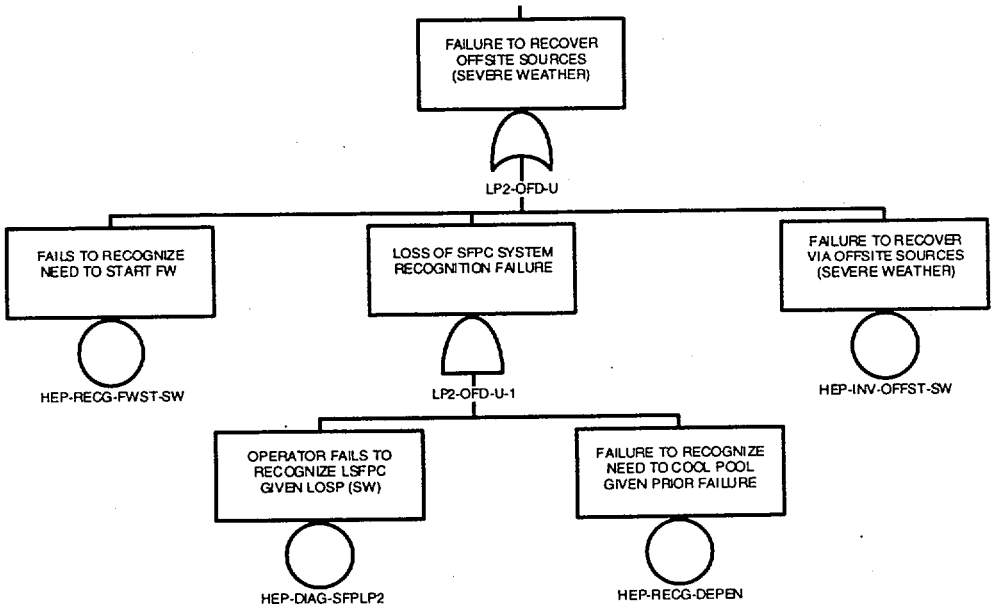

FAILURE TO RECOVER OFFSITE SOURCES
LP1-OFD
FAILS TO RECOGNIZE NEED TO START FW
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FWSTART
HEP-INV-OFFSITE

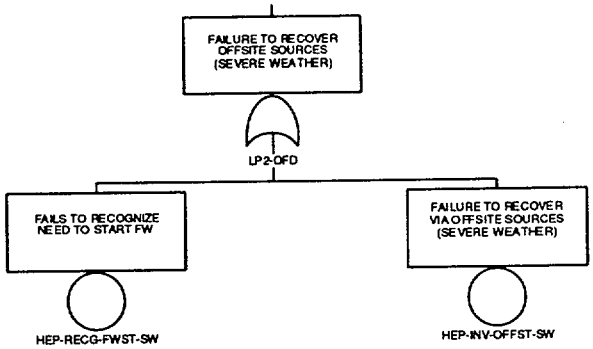

NON-RECOVERY OF OSP FROM SEVERE WEATHER
LP2-OPR
NON-RECOVERY OF OSP FROM SEVERE WEATHER
REC-OSP-SW

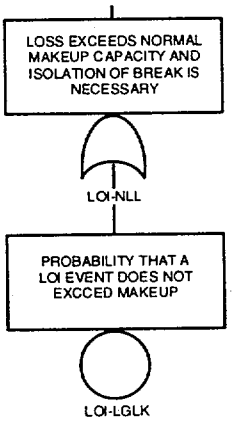

SFP COOLING SYSTEM FAILS TO RE-START AND RUN
LP2-OCS
OPERATOR FAILS TO RECOGNIZE LS PPC GIVEN LOSP EVENT
FUEL POOL COOLING PUMPS FAIL
OPERATOR FAILS TO RE-START/RE-ALIGN SYSTEM - EARLY
HE-P-DIAG-SF-PLP2
GCSR123
HE-P-SFP-STR-LP2
FUEL POOL COOLING PUMP FAILS
HEAT EXCHANGER FAILURES
SFP COOLING PUMP DISCHARGE CHECK VALVES - CCF
GCSR120
GCSR135
SFC-CKV-CCF-M
SFP COOLING PUMPS - COMMON CAUSE FAILURE
INDIVIDUAL FAILURE OF PUMPS
HEAT EXCHANGER DISHCARGE CHECK VALVES - CCF
SFP HEAT EXCHANGER COOLING SYSTEM FAILS
SFP HEAT EXCHANGERS - CCF
HEAT EXCHANGER PLUGS
SFC-PMP-CCF
GCSR141
SFC-CKV-CCF-H
SFC-HTX-FTR
SFC-HTX-CCF
SFC-HTX-PLG
SFP COOLING PUMP 1 FAILS TO START AND RUN
SFP COOLING PUMP 2 FAILS TO START AND RUN
SFC-PMP-FTF-1
SFC-PMP-FTF-2

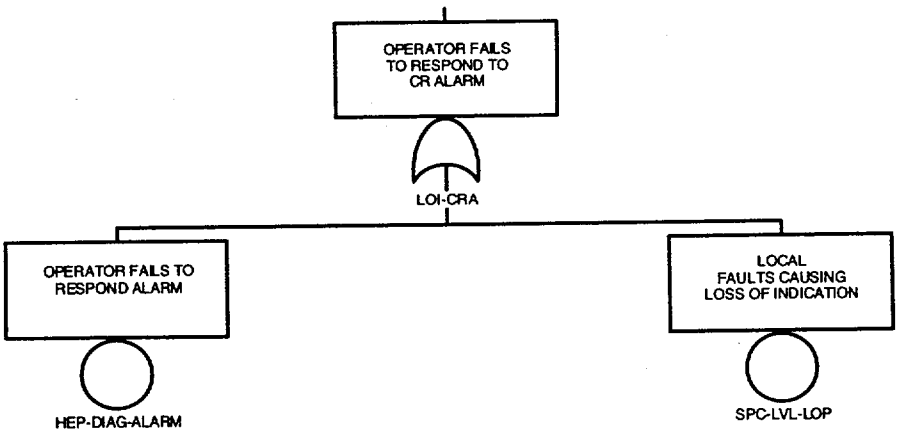

FAILURE TO MAKEUP INVENTORY USING EP/DG FIRE PUMPS
LP2-OMK-U
LOSS OF SFPC SYSTEM RECOGNITION FAILURES
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPEATOR FAILS TO ESTABLISH COOLING USING FW
LP2-OMK-U-1
HEP-RECG-FWST-SW
GLPR112-LP2-A
HEP-FW-START-SW
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP (SW)
FAILURE TO RECOGNIZE NEED TO COOL POOL GIVEN PRIOR FAILURE
DIESEL AND ELECTRIC FIRE PROTECTION PUMPS FAIL TO START AND RUN
FAILS TO REPAIR FW ELECTRIC OR DIESEL PUMP-DEPENDENCY (85 HRS)
HEP-DIAG-SFPLP2
HEP-RECG-DEPEN
FP-2PUMPS-FTF
HEP-FW-REP-DEPSW

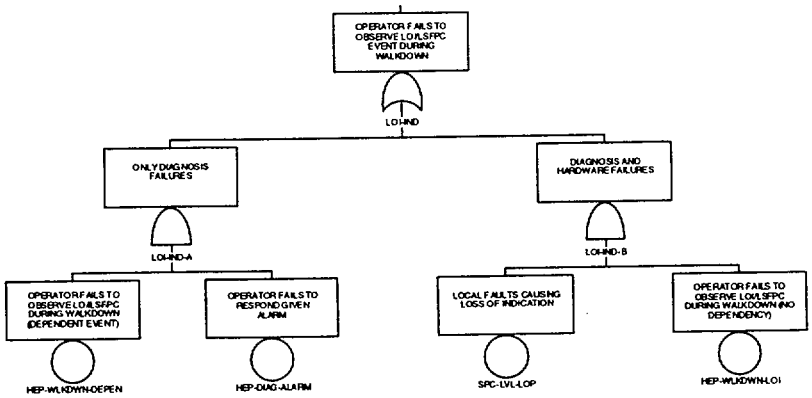

FAIORE TO MAKEUP INVENTORY USIING DG FIRE PUMP
LP2-OMK-L
OPEATOR FAILS TO ESTABLISH COOLING USING FW (SEVERE WEATHER)
FAILS TO DIAGNOSE NEED TO START FW
FAILS TO PROVIDE MAKEUP USING DIESEL FIRE PROTECTION PUMP
HEP-FW-START-SW
HEP-RECG-FWST-SW
GLPR142-LP2-A
DIESEL FIRE PROTECTION PUMP FAILS TO STRAT AND RUN
FAILS TO REPAIR FW DIESEL PUMP NO DEPENDENCY (85 HRS -SW)
FP-DGPUMP-FTF
HEP-FW-REP-NODSW

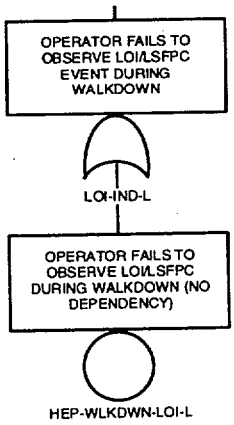

FAILURE TO RECOVER OFFSITE SOURCES (SEVERE WEATHER)
LP2-OFD-U
FAILS TO RECOGNIZE NEED TO START FW
LOSS OF SFPC SYSTEM RECOGNITION FAILURE
FAILURE TO RECOVER VIA OFFSITE SOURCES (SEVERE WEATHER)
HEP-RECG-FWST-SW
LP2-OFD-U-1
HEP-INV-OFFST-SW
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP (SW)
FAILURE TO RECOGNIZE NEED TO COOL POOL GIVEN PRIOR FAILURE
HEP-DIAG-SFPLP2
HEP-RECG-DEPEN

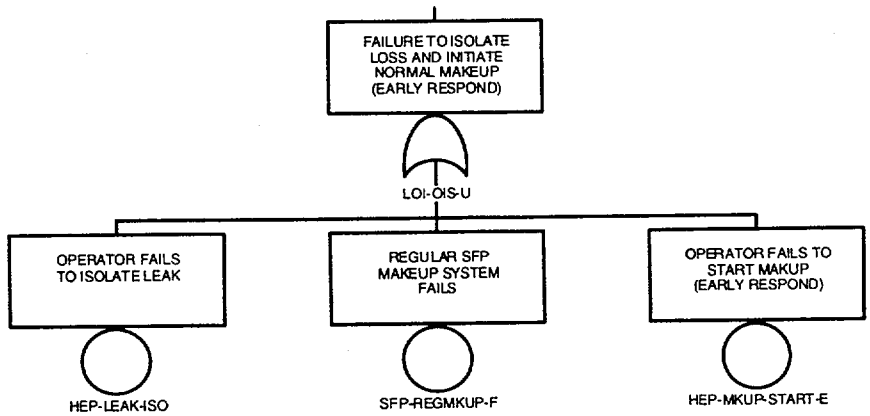

FAILURE TO RECOVER OFFSITE SOURCES (SEVERE WEATHER)
LP2-OFD
FAILS TO RECOGNIZE NEED TO START FW
FAILURE TO RECOVER VIA OFFSITE SOURCES (SEVERE WEATHER)
HEP-RECG-FWST-SW
HEP-INV-OFFST-SW

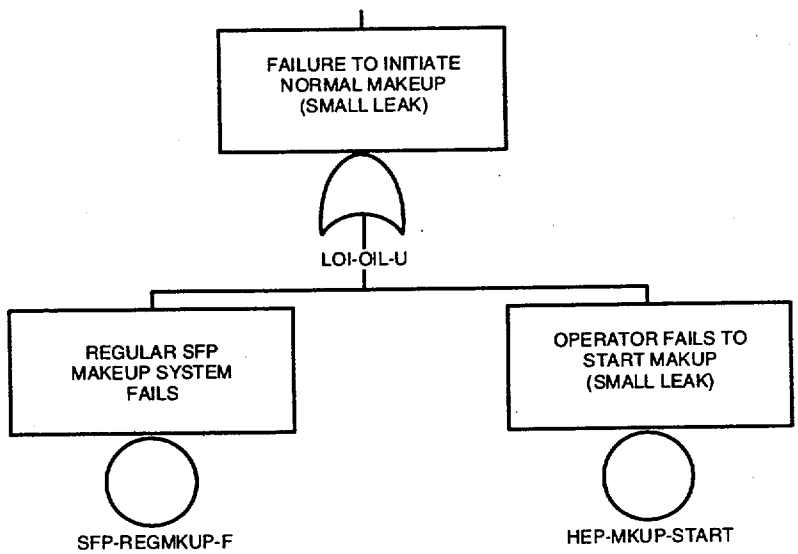

LOSS EXCEEDS NORMAL MAKEUP CAPACITY AND ISOLATION OF BREAK IS NECESSARY
LOI-NLL
PROBABILITY THAT A LOI EVENT DOES NOT EXCCED MAKEUP
LOI-LGLK

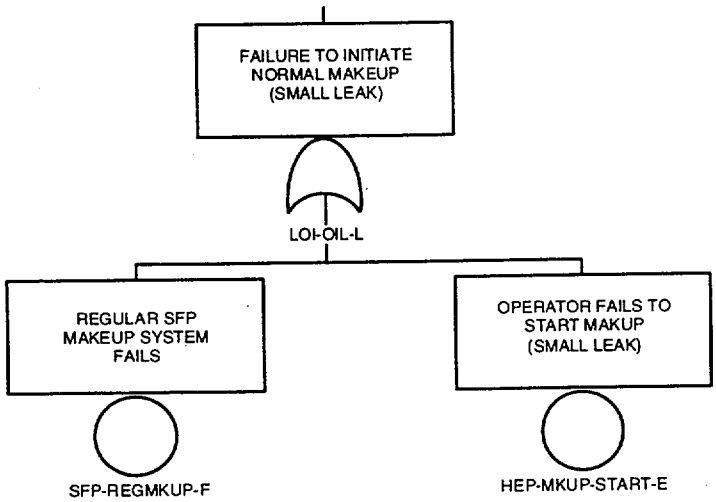

OPERATOR FAILS TO RESPOND TO CR ALARM
LOI-CRA
OPERATOR FAILS TO RESPOND ALARM
LOCAL FAULTS CAUSING LOSS OF INDICATION
HEP-DIAG-ALARM
SPC-LVL-LOP

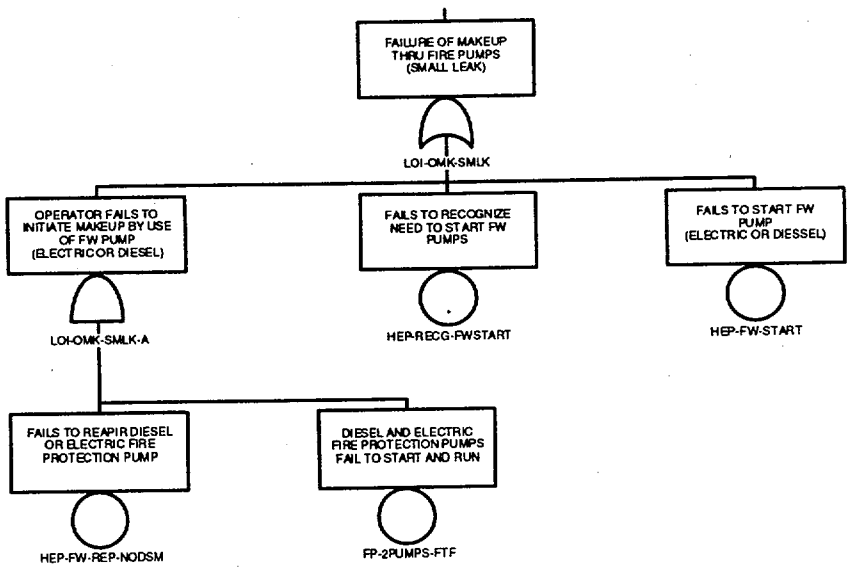

OPERATOR FAILS TO OBSERVE LOI/LSFPC EVENT DURING WALKDOWN
LOI-IND
ONLY DIAGNOSIS FAILURES
DIAGNOSIS AND HARDWARE FAILURES
LOI-IND-A
LOI-IND-B
OPERATOR FAILS TO OBSERVE LOI/LSFPC DURING WALKDOWN (DEPENDENT EVENT)
OPERATOR FAILS TO RESPOND GIVEN ALARM
LOCAL FAULTS CAUSING LOSS OF INDICATION
OPERATOR FAILS TO OBSERVE LOI/LSFPC DURING WALKDOWN (NO DEPENDENCY)
HEP-WLKDWN-DEPEN
HEP-DIAG-ALARM
SFC-LVL-LOP
HEP-WLKDWN-LOI

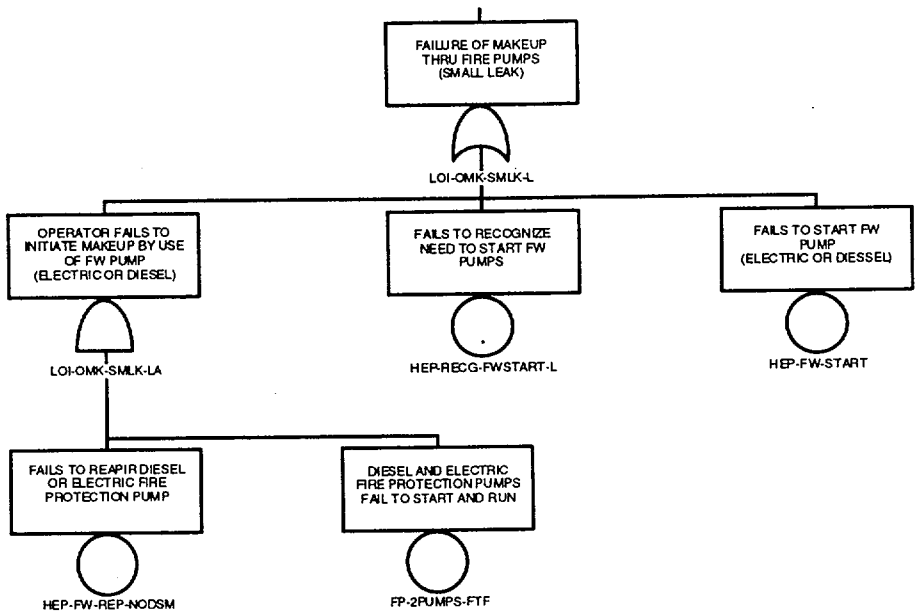

OPERATOR FAILS TO OBSERVE LOI/LSFPC EVENT DURING WALKDOWN
LOI-IND-L
OPERATOR FAILS TO OBSERVE LOI/LSFPC DURING WALKDOWN (NO DEPENDENCY)
HEP-WLKDWN-LOI-L

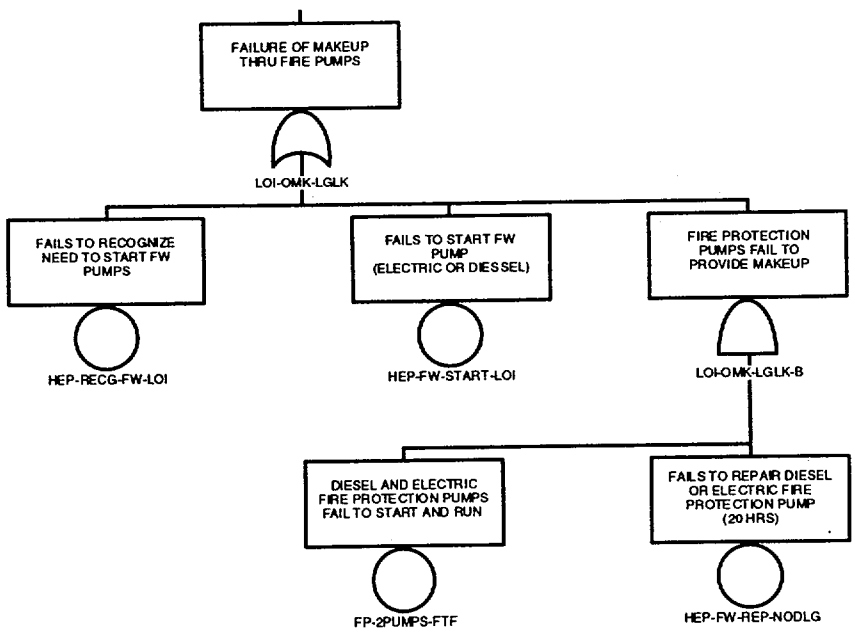

FAILURE TO ISOLATE LOSS AND INITIATE NORMAL MAKEUP (EARLY RESPOND)
LOI-OIS-U
OPERATOR FAILS TO ISOLATE LEAK
REGULAR SFP MAKEUP SYSTEM FAILS
OPERATOR FAILS TO START MAKUP (EARLY RESPOND)
HEP-LEAK-ISO
SFP-REGMKUP-F
HEP-MKUP-START-E

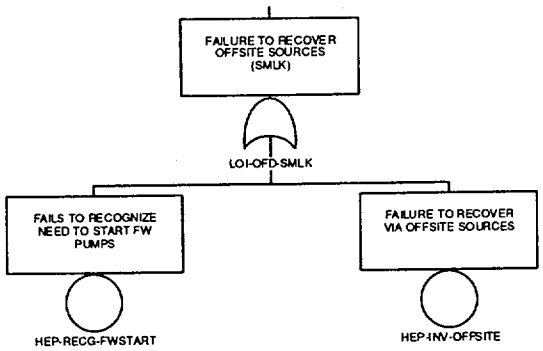

FAILURE TO INITIATE NORMAL MAKEUP (SMALL LEAK)
LOI-OIL-U
REGULAR SFP MAKEUP SYSTEM FAILS
OPERATOR FAILS TO START MAKUP (SMALL LEAK)
SFP-REGMKUP-F
HEP-MKUP-START

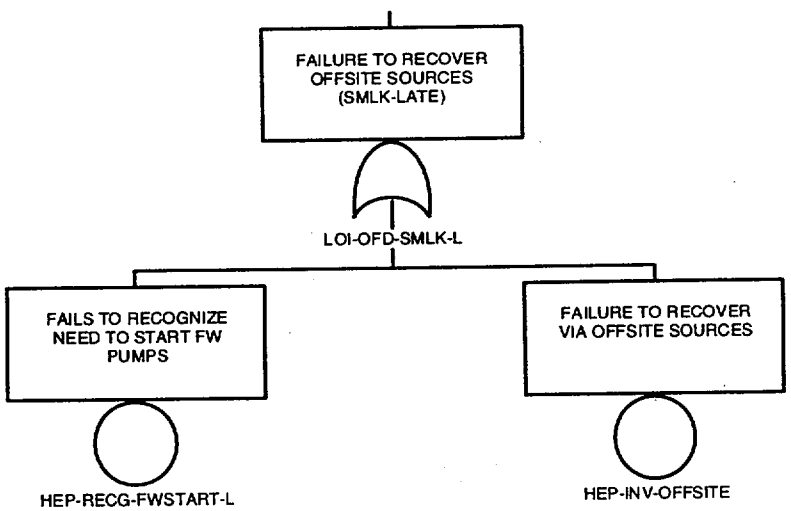

FAILURE TO INITIATE NORMAL MAKEUP (SMALL LEAK)
LOI-OIL-L
REGULAR SFP MAKEUP SYSTEM FAILS
OPERATOR FAILS TO START MAKUP (SMALL LEAK)
SFP-REGMKUP-F
HEP-MKUP-START-E


FAILURE OF MAKEUP THRU FIRE PUMPS (SMALL LEAK)
LOI-OMK-SMLK
OPERATOR FAILS TO INITIATE MAKEUP BY USE OF FW PUMP (ELECTRICOR DIESEL)
FAILS TO RECOGNIZE NEED TO START FW PUMPS
FAILS TO START FW PUMP (ELECTRIC OR DIESSEL)
LOI-OMK-SMLK-A
HEP-RECG-FWSTART
HEP-FW-START
FAILS TO REAPIR DIESEL OR ELECTRIC FIRE PROTECTION PUMP
DIESEL AND ELECTRIC FIRE PROTECTION PUMPS FAIL TO START AND RUN
HEP-FW-REP-NODSM
FP-2PUMPS-FTF


FAILURE OF MAKEUP THRU FIRE PUMPS (SMALL LEAK)
LOI-OMK-SMLK-L
OPERATOR FAILS TO INITIATE MAKEUP BY USE OF FW PUMP (ELECTRIC OR DIESEL)
FAILS TO RECOGNIZE NEED TO START FW PUMPS
FAILS TO START FW PUMP (ELECTRIC OR DIESSEL)
LOI-OMK-SMLK-LA
HEP-RECG-FWSTART-L
HEP-FW-START
FAILS TO REAPIR DIESEL OR ELECTRIC FIRE PROTECTION PUMP
DIESEL AND ELECTRIC FIRE PROTECTION PUMPS FAIL TO START AND RUN
HEP-FW-REP-NODSM
FP-2PUMPS-FTF


FAILURE OF MAKEUP THRU FIRE PUMPS
LOI-OMK-LGLK
FAILS TO RECOGNIZE NEED TO START FW PUMPS
FAILS TO START FW PUMP (ELECTRIC OR DIESSEL)
FIRE PROTECTION PUMPS FAIL TO PROVIDE MAKEUP
HEP-RECG-FW-LOI
HEP-FW-START-LOI
LOI-OMK-LGLK-B
DIESEL AND ELECTRIC FIRE PROTECTION PUMPS FAIL TO START AND RUN
FAILS TO REPAIR DIESEL OR ELECTRIC FIRE PROTECTION PUMP (20 HRS)
FP-2PUMPS-FTF
HEP-FW-REP-NODLG


FAILURE TO RECOVER OFFSITE SOURCES (SMLK)
LOI-OFD-SMLK
FAILS TO RECOGNIZE NEED TO START FW PUMPS
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FWSTART
HEP-INV-OFFSITE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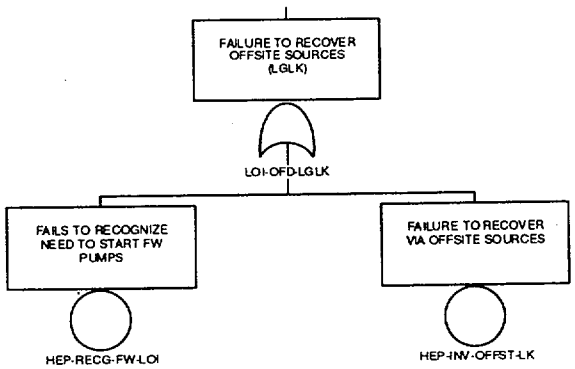
LOI-OFD-SMLK-L
FAILS TO RECOGNIZE NEED TO START FW PUMPS
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FWSTART-L
HEP-INV-OFFSITE


FAILURE TO RECOVER OFFSITE SOURCES (LGLK)
LOI-OFD-LGLK
FAILS TO RECOGNIZE NEED TO START FW PUMPS
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FW-LOI
HEP-INV-OFFST-LK

ATTACHMENT B

SPAR HRA Worksheet

SPAR HRA Human Error Worksheet (Page 1 of 3)

Plant:_____ **Initiating Event:**_____ **Sequence Number:**_____ **Basic Event Code:** _____

Basic Event Context: _____
Basic Event Description:_____

Does this task contain a significant amount of diagnosis activity? YES (start with Part I, p. 1) NO (skip Part I, p. 1; start with Part II, p. 2) Why?_____

Part I. DIAGNOSIS

A. Evaluate PSFs for the diagnosis portion of the task.

PSFs	PSF Levels	Multiplier for Diagnosis	If non-nominal PSF levels are selected, please note specific reasons in this column
Available Time	Inadequate time	P(failure) = 1.0	
	Barely adequate time <20 min	10	
	Nominal time . _30 min	1	
	Extra time >60 min	0.1	
	Expansive time >24 hrs	0.01	
Stress	Extreme	5	
	High	2	
	Nominal	1	
Complexity	Highly complex	5	
	Moderately complex	2	
	Nominal	1	
	Obvious diagnosis	0.1	
Experience/Training	Low	10	
	Nominal	1	
	High	0.5	
Procedures	Not available	50	
	Available, but poor	5	
	Nominal	1	
	Diagnostic/symptom oriented	0.5	
Ergonomics	Missing/Misleading	50	
	Poor	10	
	Nominal	1	
	Good	0.5	
Fitness for Duty	Unfit	P(failure) = 1.0	
	Degraded Fitness	5	
	Nominal	1	
Work Processes	Poor	2	
	Nominal	1	
	Good	0.8	

B. Calculate the Diagnosis Failure Probability

(1) If all PSF ratings are nominal, then the Diagnosis Failure Probability = 10E-2

) Otherwise,

	Time	Stress	Complexity	Experience/ Training	Procedures	Ergonomics	Fitness for Duty	Work Processes	
Diagnosis:	10E-2x___	x___	x___	x___	x___	x___	x___	x___	=___

SPAR HRA Human Error Worksheet (Page 2 of 3)

Plant:_____ **Initiating Event:**_____ **Sequence Number:**_____ **Basic Event Code:** _____

Basic Event Context: _____

Basic Event Description:_____

Part II. ACTION

A. Evaluate PSFs for the action portion of the task.

PSFs	PSF Levels	Multiplier for Action	If non-nominal PSF levels are selected, please note specific reasons in this column
Available Time	Inadequate time	P(failure) = 1.0	
	Time available . time required	10	
	Nominal time	1	
	Time available>50 x time required	0.01	
Stress	Extreme	5	
	High	2	
	Nominal	1	
Complexity	Highly complex	5	
	Moderately complex	2	
	Nominal	1	
Experience/Training	Low	3	
	Nominal	1	
	High	0.5	
Procedures	Not available	50	
	Available, but poor	5	
	Nominal	1	
Ergonomics	Missing/Misleading	50	
	Poor	10	
	Nominal	1	
	Good	0.5	
Fitness for Duty	Unfit	P(failure) = 1.0	
	Degraded Fitness	5	
	Nominal	1	
Work Processes	Poor	5	
	Nominal	1	
	Good	0.5	

B. Calculate the Action Failure Probability

(1) If all PSF ratings are nominal, then the Action Failure Probability = 10E-3

(2) Otherwise,	Time	Stress	Complexity	Experience/ Training	Procedures	Ergonomics	Fitness for Duty	Work Processes	
ction: 10E-3	x___	x___	x___	x___	x___	x___	x___	x___	=_____ Action

Plant:_____ Initiating Event:_____ Sequence Number:_____Basic Event Code: _____

PART III. CALCULATE THE TASK FAILURE PROBABILITY WITHOUT FORMAL DEPENDENCE ($P_{W/OD}$)

Calculate the Task Failure Probability Without Formal Dependence ($P_{w/od}$) by adding the Diagnosis Failure Probability (from Part I, p.1) and the Action Failure Probability (from Part II, p. 2).

Diagnosis Failure Probability: _____

Action Failure Probability: +_____

Task Failure Without Formal Dependence ($P_{w/od}$) =_____

If all PSFs are nominal, then

Diagnosis Failure Probability: 10E-2

Action Failure Probability: +10E-3

$P_{(w/od)}$ = 1.1x10E-2

Part IV. DEPENDENCY

For all tasks, except the first task in the sequence, use the table and formulae below to calculate the Task Failure Probability With Formal Dependence (P_{wd}).

If there is a reason why failure on previous tasks should not be considered, explain here: _____

Dependency Condition Table

Crew (same or different)	Time (close in time or not close in time	Location (same or different)	Cues (additional or not additional)	Dependency	Number of Human Action Failures Rule - Not Applicable. Why?_____
Same	Close	Same	-	complete	If this error is the **3rd error in the sequence**, then the dependency is at least **moderate**. If this error is the **4th error in the sequence**, then the dependency is at least **high.** This rule may be ignored only if there is compelling evidence for less dependence with the previous tasks. Explain above.
		Different	-	high	
	Not Close	Same	No Additional	high	
			Additional	moderate	
		Different	No Additional	moderate	
			Additional	low	
Different	Close	-	-	moderate	
	Not Close	-	-	low	

Using $P_{w/od}$ = Probability of Task Failure Without Formal Dependence (calculated in Part III, p. 3):

For Complete Dependence the probability of failure is 1.

For High Dependence the probability of failure is $(1+ P_{w/od})/2$

For Moderate Dependence the probability of failure is $(1+6 \times P_{w/od})/7$

For Low Dependence the probability of failure is $(1+19 \times P_{w/od})/20$

For Zero Dependence the probability of failure is $P_{w/od}$

Calculate $P_{w/d}$ using the appropriate values:

(1 + (*))/ = Task Failure Probability With Formal Dependence (P_{wd})

Appendix 2 b
seismic

Copy #6

Appendix 2b Structural Integrity of Spent Fuel Pools Subject to Seismic Loads

1. Introduction

As a part of the Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," NRC has studied the hypothetical event of an instantaneous loss of spent fuel pool water. The recommendation from a study in support of this generic issue indicates that a key part of a plant specific evaluation for the effect of such an event, is the need to obtain a realistic seismic fragility of the spent fuel pool. The failure or the end state of concern in the context of this generic issue is a catastrophic failure of the spent fuel pool which leads to an almost instantaneous loss of all pool water and the pool having no capacity to retain any water even if it were to be reflooded.

Spent fuel pool structures at nuclear power plants are constructed with thick reinforced concrete walls and slabs lined with stainless steel liners 1/8 to 1/4 inch thick. Dresden Unit 1 and Indian Point Unit 1 are exceptions to this in that these two plants do not have any liner plates. They were decommissioned more than 20 years ago and no safety significant degradation of the concrete pool structure has been reported. The spent fuel pool walls vary from 4.5 to 5 feet in thickness and the pool floor slabs are approximately 4 feet thick. The overall pool dimensions are typically about 50 feet long by 40 feet wide and 55 to 60 feet high. In boiling water reactor (BWR) plants, the pool structures are located in the reactor building at an elevation several stories above the ground. In pressurized water reactor (PWR) plants, the spent fuel pool structures are located outside the containment structure and are supported on the ground or partially embedded in the ground. The location and supporting arrangement of the pool structures help determine their capacity to withstand seismic ground motion beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than structural needs. Spent fuel structures at operating nuclear power plants are inherently rugged in terms of being able to withstand loads substantially beyond those for which they were designed. Consequently, they have significant seismic capacity.

2. Seismic Checklist

In the preliminary draft report published in June 1999, the staff assumed that the spent fuel pools were robust for seismic events less than about three times the safe shutdown earthquake (SSE). It was assumed that the high confidence, low probability of failure (HCLPF)[1] value for pool integrity is 3 times SSE. For most Central and Eastern U.S. (CEUS) sites, 3 X SSE is in the peak ground acceleration (PGA) range of 0.35 to 0.5 g (where g is the acceleration of gravity). Seismic hazard estimates developed by the Lawrence Livermore National Laboratory (NUREG-1488) show that, for most CEUS plants, the mean frequency for a PGA equal to 3 X SSE is less than 2E-5 per year. In the June 1999 report, the working group used the approximation that the frequency of a seismic event that will challenge the spent fuel pool integrity is 5% of 2E-5, or a value of 1E-6.

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[1]A HCLPF is the peak acceleration value at which there is 95% confidence that less than 5% of the time the structure, system or component will fail.

Several public meetings were held from April to July 1999 to discuss the staff's draft report. At the July public workshop, the NRC proposed, and the industry group agreed to develop a seismic checklist, which could be used to examine the seismic vulnerability of any given plant. In a letter dated August 18, 1999, the Nuclear Energy Institute (NEI) proposed a checklist which is based on assuring a robustness for a seismic ground motion with a PGA of approximately 0.5g. A copy of this submittal is included in Appendix 5.a.

The NRC contracted with Dr. Robert P. Kennedy to perform an independent review of the seismic portion of the June draft report, as well as the August 18, 1999, submittal from NEI. Dr. Kennedy's comments and recommendations were contained in an October 1999 report entitled "Comments Concerning Seismic Screening and Seismic Risk of Spent Fuel Pools for Decommissioning Plants," which is included as Appendix 5b of this report. Dr. Kennedy raised three significant concerns about the completeness of the NEI checklist.

The results of Dr. Kennedy's review, as well as staff comments on the seismic checklist, were forwarded to NEI and other stakeholders in a December 3, 1999, memorandum from Mr. William Huffman (Appendix 5c). In a letter from Mr. Alan Nelson, dated December 13, 1999 (Appendix 5d), NEI submitted a revised checklist, which addressed the comments from Dr. Kennedy and the NRC staff. Dr. Kennedy reviewed the revised checklist, and concluded in a letter dated December 28, 1999 (Appendix 5f), that the industry seismic screening criteria are adequate for the vast majority of CEUS sites.

The staff has considered the question of what criterion should be established for an acceptable HCLPF value; i.e., a HCLPF value which yields an acceptably low frequency of spent fuel pool failure. The design basis earthquake ground motion, or the SSE ground motion, for nuclear power plant sites were based on the assumption of the largest event geophysically ascribable to a tectonic province or a capable structure at the closest proximity of the province or fault to the site. In the case of the tectonic province in which the site is located, the event is assumed to occur at the site. For the eastern seaboard, the Charleston event is the largest magnitude earthquake and current research has established that such large events are confined to the Charleston region. The New Madrid zone is another zone in the central US where very large events have occurred. Recent research has identified the source structures of these large New Madrid earthquakes. Both of these earthquake sources are fully accounted for in the assessment of the SSE for currently licensed plants. The SSE ground motions for nuclear power plants are based on conservative estimates of the ground motion from the largest earthquake estimate to be generated under the current tectonic regime. If these SSE ground motions are amplified by a factor of three, the estimated ground motion borders on the limit of credibility for the particular site.

The seismic hazards at the west coast sites are generally governed by known active fault sources, consequently, the hazard curves, which are plots of ground acceleration versus frequency of occurrence, have a much steeper slope near the higher ground motion end. In other words, as the amplitude of the seismic acceleration increases, the probability of its occurrence decreases rapidly. Therefore, for west coast sites a seismic ground motion event greater than 2 times the SSE could be considered to be too large to be credible. Spent fuel pool structures at these sites would then need to have capacity against catastrophic failure at 2 times the SSE.

Therefore, it is reasonable to assume that a seismic ground motion greater than 3 times the SSE at a lower seismicity location (CEUS site) and 2 times the SSE at a higher seismicity location (west coast site) can be considered the maximum credible seismic ground motion for the site. Using these maximum credible seismic ground motions in conjunction with the seismic checklist simplifies the task of evaluating whether the seismic risk from the spent fuel pool is negligible. For those plants that can demonstrate that the maximum credible seismic ground motion, per the guidelines given above, are appropriate for the site and that they satisfy the seismic checklist, it can be concluded with reasonable assurance that they could be eliminated from any further seismic evaluation. For sites that fail the seismic checklist screening of the pool structure and cannot demonstrate a HCLPF value appropriate for the site, the NRC has proposed and the industry has agreed, that it would be necessary to conduct a detailed assessment of the seismically induced probability of failure of spent fuel pool structures.

In his letter of December 28, 1999, Dr. Kennedy concurred that this performance goal assures an adequately low seismic risk for the spent fuel pool.

3. Seismic Risk - Catastrophic Failure

As noted above, the preliminary risk assessment report published in June 1999 used an approximate method for estimating the risk of spent pool failure. It was assumed that the HCLPF value for the pool integrity is 3 times SSE. For most CEUS sites, 3 X SSE has a ground motion with a PGA range of 0.35 to 0.5 g. Seismic hazard curves from the Lawrence Livermore National Laboratory (NUREG-1488) show that, for most CEUS sites, the mean frequency for PGA equal to 3 X SSE is less than 2E-5. In the June report, the working group used the approximation that the frequency of a seismic event that will challenge the spent fuel pool integrity is 5% of 2E-5, or a value of 1E-6.

Dr. Kennedy, in his October 1999 report, pointed out that this approximation is nonconservative for CEUS hazard curves with shallow slopes; i.e., where an increase of more than a factor of two in ground motion is required to achieve a 10-fold reduction in annual frequency of exceedance. Dr. Kennedy proposed a calculation method, which had previously been shown to give risk estimates that were 5 to 20% conservative when compared to more rigorous methods, such as convolution of the hazard and fragility estimates. Using this approximation, Dr. Kennedy estimated the spent fuel pool failure frequency for a site with HCLPF of 1.2[2] peak spectral acceleration if sited at each of the 69 CEUS sites. A total of 35 sites had frequencies exceeding 1E-6 per year, and eight had frequencies in excess of 3E-6 per year. The remaining sites had frequencies below 1E-6[3]. Dr. Kennedy's report notes that spent fuel pools that pass

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[2]Damage to critical SSCs does not correlate very well to PGA of the ground motion. However, damage correlates much better with the spectral acceleration of the ground motion over the natural frequency range of interest, which is generally between 10 and 25 hertz for nuclear power plants SSCs. The spectral acceleration of 1.2g corresponds to the screening level recommended in the reference document cited in the NEI checklist, and this spectral ordinate is approximately equivalent to a ground motion with 0.5g PGA.

[3]These estimates are based on the Lawrence Livermore National Laboratory 1993 (LLNL 93) seismic hazard curves. Recently, the Senior Seismic Hazard Analysis Committee

the appropriately defined screening criteria are likely to have capacities higher than the screening level capacity. Thus, the frequencies quoted above are upper bounds.

The staff has no estimate of the seismic risk for sites west of the Rockies. However, based on considerations described above, the staff estimates that western plants which can demonstrate a HCLPF greater than 2 X SSE will have an acceptably low estimate of risk.

4. Seismic Risk - Support System Failure

In its preliminary draft report published in June 1999, the staff assumed that a ground motion three times the SSE was the HCLPF of the spent fuel pool. This meant that 95% of the time the pool would remain intact (i.e., would not leak significantly). The staff evaluated what would happen to the support systems to the spent fuel pool (i.e., the pool cooling and inventory makeup systems) in the event of an earthquake three times the SSE. We modeled some recovery as possible (although there would be considerable damage to the area's infrastructure at such earthquake accelerations). The estimate in the preliminary report for the contribution from this scenario was $1x10^{-6}$ per year. In this report, this estimate has been refined based on looking at a broader range of seismic accelerations and further evaluation of the conditional probability of recovery under such circumstances. The staff estimates that for an average site in the northeast United States the return period of an earthquake that would damage a decommissioning plant's spent fuel pool cooling system equipment (assuming it had at least minimal anchoring) is about once in 4,000 years. The staff quantified a human error probability of $1x10^{-4}$ that represents the failure of the fuel handlers to obtain off-site resources. The event was quantified using the SPAR HRA technique. The probability shaping factors chosen were as follows: expansive time (> 50 times the required time), high stress, complex task because of the earthquake and its non-routine nature, quality procedures, poor ergonomics due to the earthquake, and finally a crew who had executed these tasks before, conversant with the procedures and one another. In combination we now estimate the risk from support failure due to seismic events to be on the order of $1x10^{-8}$ per year. The risk from support system failure due to seismic events is bounded by other more likely initiators.

5. Conclusion

The staff concludes that the frequency of spent fuel pool failure for a CEUS plant is acceptably low if the seismic capacity of its spent fuel pool structure is at least equal to 3 times the plant's SSE value, and the plant satisfies the seismic checklist proposed in NEI's December 13, 1999 letter (See Appendix 5). Although the risk has not been rigorously calculated for these sites, deterministic considerations lead the staff to conclude that peak ground accelerations in excess of 3 times SSE are not credible. For these sites the frequency of failure is bounded by $3x10^{-6}$ per year, and other considerations indicate the frequency may be significantly lower.

(SSHAC) published NUREG-CR-6372, "Recommendation for Probabilistic Seismic Hazard Analysis: Guidance On Uncertainty and Use of Experts." The report gives guidance on future application of seismic hazards. However, site specific hazard estimates have not been performed for all sites with the new method.

For those CEUS plants with spent fuel pool structures that do not pass the seismic checklist, a detailed evaluation of HCLPF would be necessary. Similarly, a detailed HCLPF would be necessary for all western plants since seismic capacity at the high levels of ground motion associated with the western plants are well above the generic HCLPF value of 1.2g peak spectral acceleration. For all CEUS plants which can demonstrate a HCLPF equal to 3 times their SSE, the risk is judged to be bounded by $3x10^{-6}$ per year. Similarly, for western sites which can demonstrate a HCLPF equal to 2 times their SSE, the risk is judged to be bounded by $3x10^{-6}$ per year.

Appendix 2.b Structural Integrity of Spent Fuel Pools
Subject to Seismic Loads

Introduction

As a part of the Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," NRC has studied the hypothetical event of an instantaneous loss of spent fuel pool water. The recommendation from a study in support of this generic issue indicates that a key part of a plant specific evaluation for the effect of such an event is the need to obtain a realistic seismic fragility of the spent fuel pool. The failure or the end state of concern in the context of this generic issue is a catastrophic failure of the spent fuel pool which leads to an almost instantaneous loss of all pool water and the pool having no capacity to retain any water even if it were to be reflooded.

Spent fuel pool structures at nuclear power plants are constructed with thick reinforced concrete walls and slabs lined with thin stainless steel liners 1/8 to 1/4 inch thick. Dresden Unit 1 and Indian Point Unit 1 are exceptions to this in that these two plants do not have any liner plates. They were decommissioned more than 20 years ago and no safety significant degradation of the concrete pool structure has been reported. The spent fuel pool walls vary from 4.5 to 5 feet in thickness and the pool floor slabs are around 4 feet thick. The overall pool dimensions are typically about 50 feet long by 40 feet wide and 55 to 60 feet high. In boiling water reactor (BWR) plants, the pool structures are located in the reactor building at an elevation several stories above the ground. In pressurized water reactor (PWR) plants, the spent fuel pool structures are located outside the containment structure and are supported on the ground or partially embedded in the ground. The location and supporting arrangement of the pool structures help determine their capacity to withstand loads beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than structural needs. Spent fuel structures at operating nuclear power plants are inherently rugged in terms of being able to withstand loads substantially beyond those for which they were designed. Consequently, they have significant seismic capacity.

Seismic Checklist

In the preliminary draft report published in June 1999, the staff assumed that the spent fuel pools are robust for seismic events less than about three times the safe shutdown earthquake (SSE). It was assumed that the high confidence, low probability of failure (HCLPF)[1] value for pool integrity is 3 times SSE. For most Central and Eastern U.S. (CEUS) sites, 3 X SSE is in the peak ground acceleration (PGA) range of 0.35 to 0.5 g (where g is the acceleration of gravity). Seismic hazard estimates developed by the Lawrence Livermore National Laboratory (NUREG-1488) show that, for most CEUS plants, the mean frequency for a PGA equal to 3 X SSE is less than 2E-5 per year. In the June 1999 report, the working group used the approximation that the frequency of a seismic event that will challenge the spent fuel pool integrity is 5% of 2E-5, or a value of 1E-6.

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[1]A HCLPF is the peak acceleration value at which there is 95% confidence that less than 5% of the time the structure, system or component will fail.

Several public meetings were held from April to July 1999 to discuss the staff's draft report. At the July public workshop, the NRC proposed, and the industry group agreed to develop, a seismic checklist which could be used to examine the seismic vulnerability of any given plant. In a letter dated August 18, 1999, the Nuclear Energy Institute (NEI) proposed a checklist which would assure that any plant could show robustness for a seismic ground motion with a PGA of approximately 0.5g. A copy of this submittal is included in Appendix 5.a.

The NRC contracted with Dr. Robert P. Kennedy to perform an independent review of the seismic portion of the June draft report, as well as the August 18, 1999, submittal from NEI. Mr. Kennedy's comments and recommendations were contained in an October 1999 report entitled "Comments Concerning Seismic Screening and Seismic Risk of Spent Fuel Pools for Decommissioning Plants," which is included as Appendix 5.b of this report. Dr. Kennedy raised three significant concerns about the completeness of the NEI checklist.

The results of Mr. Kennedy's review, as well as staff comments on the seismic checklist, were forwarded to NEI and other stakeholders in a December 3, 1999, memorandum from Mr. William Huffman (Appendix 5.3). In a letter from Mr. Alan Nelson, dated December 13, 1999 (Appendix 5.4), NEI submitted a revised checklist, which addressed the comments from Mr. Kennedy and the NRC staff. Mr. Kennedy reviewed the revised checklist, and concluded in a letter dated December 28, 1999 (Appendix 5.6), that the industry seismic screening criteria are adequate for the vast majority of ~~Central and Eastern US~~ (CEUS) sites.

The staff has considered the question of what criterion should be established for an acceptable HCLPF value; i.e., a HCLPF value which yields an acceptably low frequency of spent fuel pool failure. The design basis earthquake ground motion, or the SSE ground motion, for nuclear power plant sites were based on the assumption of largest event geophysically ascribable to a tectonic province or a capable structure at the closest proximity of the province or fault to the site. In the case of the tectonic province in which the site is located, the event is assumed to occur at the site. For the eastern seaboard, the Charleston event is the largest magnitude earthquake and current research has established that such large events are confined to Charleston region. The New Madrid zone is another zone in the central US where very large events have occurred. Recent research has identified the source structures of these large New Madrid earthquakes. Both of these earthquake sources are fully accounted for in the assessment of the SSE for currently licensed plants. The SSE ground motions for nuclear power plants are based on conservative estimates of the ground motion from the largest earthquake estimate to be generated under the current tectonic regime. If we amplify these SSE ground motions by three, the estimated ground motion borders on the limit of credibility for the particular site.

The seismic hazards at the west coast sites are generally governed by known active fault sources, consequently, the hazard curves, which are plots of ground acceleration versus frequency of occurrence, have a much steeper slope near the higher ground motion end. Another way to say this, as the amplitude of the seismic acceleration increases, the probability of its occurrence goes down rapidly. Therefore, for west coast sites a seismic ground motion event greater than 2 times the SSE could be considered to be too large to be credible. Spent fuel pool structures at these sites would then need to have capacity against catastrophic failure at 2 times the SSE.

Therefore, it appears reasonable to assume that a seismic event greater than 3 times the SSE at a lower seismicity location (CEUS site) and 2 times the SSE at a higher seismicity location (west coast site) can be considered to be incredible. This proposed performance goal simplifies the task of evaluating whether the seismic risk from the spent fuel pool is negligible. Those plants that can demonstrate that they meet the proposed performance goal could be eliminated from any further seismic evaluation. For sites that fail the seismic checklist screening of the pool structure and cannot demonstrate a HCLPF equal to the performance goal, it would be necessary to conduct a detailed assessment of the seismically induced probability of failure of spent fuel pool structures.

In his letter of December 28, 1999, Dr. Kennedy concurred that this performance goal assures an adequately low seismic risk for the spent fuel pool. Therefore, those plants that do not meet these performance criteria, the NRC has proposed and the industry has agreed, that a more detailed assessment of seismic fragility is needed to establish the HCLPF capacity.

Seismic risk - Catastrophic Failure

As noted above, the preliminary risk assessment report published in June 1999 used an approximate method for estimating the risk of spent pool failure. It was assumed that the HCLPF value for the pool integrity is 3 times SSE. For most CEUS sites, 3 X SSE has a ground motion with a PGA range of 0.35 to 0.5 g (where g is the acceleration of gravity). Seismic hazard curves from the Lawrence Livermore National Laboratory (NUREG-1488) show that, for most CEUS sites, the mean frequency for PGA equal to 3 X SSE is less than 2E-5. In the June report, the working group used the approximation that the frequency of a seismic event that will challenge the spent fuel pool integrity is 5% of 2E-5, or a value of 1E-6.

Mr. Kennedy, in his October 1999 report, pointed out that this approximation is nonconservative for CEUS hazard curves with shallow slopes; i.e., where an increase of more than a factor of two in ground motion is required to achieve a 10-fold reduction in annual frequency of exceedance. Mr. Kennedy proposed a calculational method which had previously been shown to give risk estimates that were 5 to 20% conservative when compared to more rigorous methods, such as convolution of the hazard and fragility estimates. Using this approximation, Mr. Kennedy estimated the spent fuel failure frequency for a pool with HCLPF of 0.5 PGA for all 69 CEUS sites. A total of 35 sites had frequencies exceeding 1E-6 per year, and eight had frequencies in excess of 3E-6 per year.

Mr. Kennedy's report offers two additional considerations. First, spent fuel pools that pass the appropriately defined screening criteria are likely to have capacities higher than the screening level capacity. Thus the frequencies quoted above are upper bounds. Second, using the same approximations, Mr. Kennedy calculated frequencies approximately an order of magnitude lower, when using EPRI estimates of the seismic hazard rather than LLNL estimates.

The staff has no estimate of the seismic risk from western sites. However, based on considerations described above, the staff estimates that plants which can demonstrate a HCLPF greater than 2 X SSE will have an acceptably low estimate of risk.

Seismic Risk - Support System Failure

In its preliminary draft report published in June 1999, the staff assumed that a ground motion three times the SSE was the HCLPF of the spent fuel pool. This meant that 95% of the time the pool would remain intact (i.e., would not leak significantly.) We evaluated what would happen to the support systems to the spent fuel pool (i.e., the pool cooling and inventory makeup systems) in the event of an earthquake three times the SSE. We modeled some recovery as possible (although there would be considerable damage to the area's infrastructure at such earthquake accelerations.) Our estimate in the preliminary report for the contribution from this scenario was $1x10^{-6}$ per year. In this report, we have refined this estimate based on looking at a broader range of seismic accelerations and further evaluation of the conditional probability of recovery under such circumstances. We estimate that for an average site in the northeast United States the return period of an earthquake that would damage a decommissioning plant's spent fuel pool cooling system equipment (assuming it had at least minimal anchoring) is about once in 4,000 years. We quantified a human error probability of $1x10^{-4}$ that represents the failure of the fuel handlers to obtain offsite resources. The event was quantified using the SPAR HRA technique. The probability shaping factors chosen were as follows: expansive time (> 50 times the required time), high stress, complex task because of the earthquake and its non-routine nature, quality procedures, poor ergonomics due to the earthquake, and finally a crew who had executed these tasks before, conversant with the procedures and one another. In combination we now estimate the risk from support failure due to seismic events to be on the order of $1x10^{-8}$ per year. The risk from support system failure due to seismic events is bounded by other more likely initiators.

Conclusions

The staff concludes that the frequency of spent fuel pool failure for CEUS plants is acceptably low if they can demonstrate a HCLPF of 3 X SSE. The staff concludes that the vast majority of CEUS ~~plants~~ sites (61 of 69) can meet this criterion by showing compliance with the seismic checklist proposed by NEI in their December 13, 1999, letter. For those sites, the frequency is bounded by a value of 3E-6 per year calculated by Mr. Kennedy in his October 1999 report. Other considerations lead us to believe it may be significantly lower.

Those sites for which the ground motion at 3 X SSE exceeds 0.5g PGA, a detailed evaluation of HCLPF will be necessary. For plants which can demonstrate 3 X SSE, the risk has not been rigorously calculated. However, deterministic considerations lead the staff to believe that PGAs in excess of 3 X SSE are not credible, and the risk from such plants is acceptably low.

Western sites will have to perform a detailed HCLPF evaluation. For those that meet the performance criterion of 2 X SSE, the risk is judged to be acceptably low.

Appendix E
Heavy Loads

Wrong version

Appendix 2c

Structural Integrity of Spent Fuel Pool Structures Subject to Heavy Loads Drops

Summary

A heavy load drop into the spent fuel pool (SFP), or onto the spent fuel pool wall, can affect the structural integrity of the spent fuel pool. A loss-of-inventory from the spent fuel pool could occur as a result of a heavy load drop. For single failure proof systems where load drop analyses have not been performed at decommissioning plants, the mean frequency of a loss-of-inventory caused by a cask drop was estimated to be 2.2×10^{-7} per year (for 100 lifts). For a non-single failure proof handling system where load drop analyses have not been performed, the mean frequency of a loss-of-inventory event caused by a cask drop was estimated to be 2.3×10^{-5} per year. For decommissioning plants where load drop analyses have been performed, the frequency of a cask drop causing a loss-of-inventory event is less than 1×10^{-9} per year for single failure proof systems and less than 1×10^{-8} per year for non-single failure proof systems.

Analysis

The staff revisited NUREG-0612 to review the evaluation and the supporting data available at that time. Two additional sources of information were identified and used to reassess the heavy load drop risk:

1.01 1990s Navy crane experiences for the period 1996 through mid-1999, and

1.02 WIPP/WID-96-2196, "Waste Isolation Pilot Plant Trudock Crane System Analysis," October 1996 (WIPP).

The 1990s Navy data encompassed primarily bridge cranes with lift capacities of 20,000 lb. to 350,000 lb., at both shipyards and non-shipyard sites. The data are summarized in Table A2c-1 by incident type and incident cause. Improper operation caused 38% of the events, improper rigging 30%, poor procedures 20%, equipment failures 5%, and other causes 8%. Improper rigging was further divided into two parts: (a) 70% were identified as rigging errors and (b) 30% were rigging-related failures resulting from the crane operation. Reported load drops occurred in about 9% of the accidents, 3% related to the crane and its operation and 6% to improper rigging. The fault trees used to assess a heavy load drop leading to a loss-of-inventory are shown in Figure 1 (taken from NUREG-0612). Table A2c-1 includes the grouping of the incidents type for use in the fault tree quantification.

Based on the July 1999 SFP workshop, we assumed there will be a maximum of 100 cask lifts per year. Using the 1990s Navy database, for 100 lifts, about 3 lifts may lead to a load drop for the evaluation of the "failure of crane" event (CF). Using the new Navy database, for 100 lifts, about 6 lifts may lead to a load drop for the evaluation of the "failure of rigging" event (CR). In NUREG-0612, which was based on 200 lifts per year, the range of lifts leading to a load drop was estimated by the staff to be between 4 and 10 (2% to 5%).

The handling system failure rate was estimated in NUREG-0612 to be in the range of 1.0×10^{-5} to 1.5×10^{-4} incidents per year based on the 1970s Navy crane incident data and a staff estimate of the total number of lifts per year. The staff's evaluation included a factor of two reduction for

the range estimate based on improved procedures and conformance with the guidelines presented in Section 5.1.1 of NUREG-0612.

In the NUREG-0612 evaluation it was assumed that the number of reported incidents could have represented only about one-half of the actual number of incidents due to unknown reporting requirements. The 1990s Navy data identified about twice as many incidents over the same time span. This may support the earlier assumption since the Navy reporting requirements are now well defined in NAVFAC P-307, U.S. Navy, June 1998. For this evaluation we assumed that the handling system failure rate range was the same as used by the staff in NUREG-0612.

The base data used in this evaluation considered a range of values comprised of a high estimate (V_H) and a low estimate (V_L) to represent an initiator rate or a demand rate. The data were generally expressed in exponents of 10 and a log normal distribution for the variable V was used for the evaluation. Using the log normal distribution for V implies that the exponent has a normal distribution and that the exponent is viewed as the significant variable in the analysis.

We assumed the range of a value to be the 90% confidence interval to account for uncertainty. That is, there is a 5% chance that the high value may be higher than the estimate, and a 95% chance that the value is greater than the low estimate. This assumption provided a way to obtain the mean value for a range. A log normal distribution is, mathematically, a function of (μ,σ^2), where μ is the mean and σ^2 is the variance of the log normal distribution of V. μ and σ were calculated based on the 90% confidence interval consideration from the following two relationships:

$$V_H = \exp(\mu + 1.645\sigma) \qquad \text{and} \qquad V_L = \exp(\mu - 1.645\sigma)$$

The mean for the normal distribution of V was then calculated from the following relationship:

$$V_{mean} = \exp(\mu + \tfrac{1}{2}\sigma^2)$$

Heavy Load Drop

A heavy load drop could result from either the failure of the lifting equipment (mechanical or structural failures, or improper operation) or from failure to properly secure the load to the lifting device (human error). These two items are addressed separately.

Failure of the Lifting Equipment

The fault tree (Figure A2c-1) describing the failure of a crane comes from NUREG-0612. When heavy loads were evaluated in NUREG-0612, low density storage racks were in use and after 30 to 70 days (a period of about 0.1 to 0.2 per year) no release was expected if the pool were drained. It was assumed that after this period, the fuel gap noble gas inventory had decayed and no zircaloy fire would have occurred. To be consistent with the high density storage racks now in use, this evaluation presents the results for a period of 1.0 year, during which it is assumed a zirconium cladding fire may occur if the pool were drained.

Figure A2c-1 represents the "Releases exceed guidelines due to loads handled over spent fuel," the event 3.1(A) branch of Figure B-3 in NUREG-0612. The companion branch, "Releases exceed guidelines due to loads handled near spent fuel," the event 3.1(B) branch, was not considered in this evaluation for cask handling. Branch 3.1(B) considered movement of heavy loads near the spent fuel pool and the load drop would have resulted in damage to the spent fuel but not to the spent fuel pool.

The mean failure frequency of a component without a secondary device (for example, a crane cable/hook failure) was estimated in NUREG-0612 to be 1.2×10^{-6} per demand. This frequency estimate was further reduced by a factor of 10 in NUREG-0612 for the evaluation of a single failure proof system based on conformance with NUREG-0554 ("Single-Failure Proof Cranes for Nuclear Power Plants") and the expected increase in design safety factors.

Failure to Secure the Load

The improper rigging evaluation as presented in NUREG-0612 was based on an estimate of a common mode effect resulting in failure of the redundant rigging 5% to 25% of the time. The frequency of improper rigging incidents identified in the 1990s Navy data may not be representative of a single-failure proof load handling design that conforms to the guidelines in NUREG-0612. A literature search performed by the staff identified a study (WIPP report) which included a human error evaluation for improper rigging. This study was used to re-evaluate the contribution of rigging errors to the overall heavy load (cask) drop rate and to address both the common mode effect estimate and the 1990s Navy data.

Failure to secure a load was evaluated in the WIPP report for the Trudock crane. The WIPP report determined that failure to attach the load to the lifting mechanism, considering two trained personnel, numerous feedbacks, and verifications, was incredible. The more probable human error was for attaching the lifting legs to the lifting fixture using locking pins. In Appendix 4 of the WIPP report, the failure to secure the load (based on a 2-out-of-3 lifting device) was estimated (a mean point estimate) based on redundancy, procedures and a checker. The report assumed that the load could be lowered without damage if no more than one of the three connections were not properly made. Using NUREG/CR-1278 ("Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," August 1983) information, the mean failure rate due to improper rigging was estimated in the WIPP report to be 8.7×10^{-7} per lift. Our requantification of the fault tree using the WIPP improper rigging failure rate is summarized in Table A2c-2. The WIPP evaluation for the human error probabilities is summarized in Table A2c-3.

Heavy Load Drop Summary

The staff evaluation, based on the 1990s Navy crane data with the WIPP improper rigging evaluation as summarized in Table A2c-2, provides the basis for developing the estimate of a loss-of-inventory from a heavy load (cask) drop into a decommissioning plant's spent fuel pool.

The estimated mean value for a heavy load drop was 2.3×10^{-6} per year for 100 lifts (FHLS) for a single-failure proof handling system, with a range of 9.5×10^{-7} to 1.0×10^{-5} per year. The

contributors (mean values) included crane failure at 1.4×10^{-6} per year (CRANE), operator-related errors at 3.0×10^{-8} per year (CF1 + CF3) and improper rigging at 8.7×10^{-7} per year (RIGGING). For the non-single failure proof handling system, the estimated mean frequency for a heavy load drop was 1.0×10^{-3} per year for 100 lifts, with a range of 2.0×10^{-5} to 1.2×10^{-3} per year.

Evaluation of the Load Path

The path of the lift, and the portion of the path over which significant damage is likely to occur given a cask drop, needs to be factored into an overall estimate of a loss-of-inventory.

The load path assessment is plant-specific. In NUREG-0612 it was estimated that the heavy load was near or over the spent fuel pool for between 5% and 25% (event P in Table A2c-2) of the total path needed to lift, move, and set down the load. It was further estimated that if the load were dropped from 30 feet or higher (or from 36 feet and higher depending on the assumptions) and if a plant-specific load drop analysis had not been performed, then damage to the pool floor would result in loss-of-inventory. This works out that a (cask) drop between 0.5% and 6.25% of the path length could result in a loss-of-inventory. If the cask were dropped on the pool wall (from a height of 8 to 10 inches above the wall), it was assumed there is a 10% likelihood that damage to the wall would result in a loss-of-inventory based on Generic Safety Issue 82 studies (NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools').

Heavy Load Drop Leading to a Loss-of-Inventory

Our heavy load drop evaluation is based on the method and fault trees developed in NUREG-0612. New 1990s Navy data was used to quantify the failure of the lifting equipment. The WIPP human error evaluation was used to quantify the failure to secure the load. We estimated the mean frequency of a loss-of-inventory from a cask drop to be 2.0×10^{-7} per year for 100 lifts for a single-failure proof handling system (Table A2c-2, LOI-S). The range was estimated to be between 2.1×10^{-6} to 2.8×10^{-8} per year. Table A2c-2 presents the results for a heavy load drop on or near the spent fuel pool. If the cask were dropped on the spent fuel pool floor, the likelihood of a loss-of-inventory given the drop is 1.0. If the load were dropped on the spent fuel pool wall, the likelihood of a loss-of-inventory given the drop, is 0.1. Therefore the likelihood of a loss-of-inventory from a dropped spent fuel pool cask for a single-failure proof handling system was estimated to be 2.2×10^{-7} per year (for 100 lifts). The range was estimated to be between 2.3×10^{-6} to 3.1×10^{-8} per year.

For a non-single failure proof handling system, we based the mean frequency of a loss-of-inventory estimate on NUREG-0612. In NUREG-0612, an alternate fault tree (Figure B-2, page B-16 of NUREG-0612) was used to estimate the frequency of exceeding the release guidelines (loss-of-inventory) for a non-single failure proof system. The mean value was estimated to be about 2.1×10^{-5} per year (event 2.1.1) when corrected for the new Navy data and 100 lifts per year (Table A2c-2, LOI-N). The range was estimated to be between 7.5×10^{-5} to 1.0×10^{-7} per year. Table A2c-2 presents the results for a cask drop on or near the spent fuel pool. If the cask were dropped on the spent fuel pool floor, the likelihood of a loss-of-inventory given the drop is 1.0. If the cask were dropped on the spent fuel pool wall, the likelihood of a

loss-of-inventory given the drop is 0.1. Therefore we estimated the likelihood of a loss-of-inventory from a dropped spent fuel pool cask for a non-single failure proof handling system to be 2.3×10^{-5} per year (for 100 lifts). The range was estimated to be between 8.3×10^{-5} to 1.1×10^{-7} per year.

Comparison of results to other studies and data

Assessment of the Incident Rate

The incidents per year range was estimated to be on the order of 1.0×10^{-5} to 1.5×10^{-4} incidents per year. This range was based on Navy data and was used in the NUREG-0612 evaluation and in the current evaluation. The incident rate contains uncertainty because it is not well known how many crane operations occurred without a reportable incident. There is also some uncertainly in using the Navy data for nuclear power plant operations.

At nuclear power plants, dry cask storage has provided some additional information useful in assessing the incident rate. There have been about 150 casks loaded for dry storage at commercial reactor sites (LWRs) in the past 14 years. There have been about 250 cask loaded at the Fort St. Vrain gas-cooled reactor site (GCR). There have been no reportable incidents related to heavy loads per 10CFR 72.75, "Reporting requirements for special events and conditions."

Point estimates of the incident rate may be calculated with the following equations for those events not observed (zero occurrence — no drops or any other reportable event) in C number of components (lifts) for T years:

$$\lambda_{95\% \text{ confidence limit}} = 3.0/(C \times T) \text{ incidents per year}$$

$$\lambda_{50\% \text{ confidence limit}} = 0.69/(C \times T) \text{ incidents per year}$$

For the current experience base for LWRs, $\lambda_{95\%} = 7.1 \times 10^{-4}$ incidents per year (assuming each cask load requires two lifts). At the 50% confidence limit, $\lambda_{50\%} = 1.6 \times 10^{-4}$ incidents per year. If the GCR data is considered and added to the LWRs data, then $\lambda_{95\%} = 2.7 \times 10^{-4}$ incidents per year and $\lambda_{50\%} = 6.2 \times 10^{-5}$ incidents per year. The actual cask handling data does not call into question the incident rate range used in this assessment.

Summary of Other Heavy Load Drop Studies

Heavy load drops were evaluated as part of Generic Safety Issue 82. In NUREG/CR-4982 ("Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82) the total human error rate associated with cask movement was estimated to be 6.0×10^{-4} incidents per lift. It was further assumed that only 1-in-100 human errors would result in a cask drop. It was also estimated that the cask was above the pool edge (wall) about 25% of the lift time. Based on two shipment per week with two lifts per shipment (208 lifts), the estimate for a load drop on the spent fuel pool wall was 3.1×10^{-4} per year. Damage to the pool wall sufficient to cause a loss-of-inventory was further estimated to have a conditional probability of 0.1, based on the evaluation presented in NUREG/CR-5176, "Seismic Failure and Cask Drop Analyses of Spent

Fuel Pools at Two Representative Nuclear Power Plants," LLNL, P.G. Prassinos, et al., January 1989. The analysis assumes the height of the load above the pool wall is only about 8 to 10 inches. The estimate of a loss-of-inventory from a heavy load drop on the spent fuel pool wall was 3.1×10^{-5} per year (for a non-single failure proof handling system.) Damage resulting from a load drop onto the spent fuel pool floor was not addressed as part of Generic Safety Issue 82. We believe that if the load were dropped from a high enough elevation, e.g., 30 to 40 feet above the spent fuel pool floor, it is likely that significant damage would occur, resulting in a loss of inventory. Based on 100 lifts per year, the NUREG/CR-4982 evaluation would estimate the loss-of-inventory from a heavy load drop on the spent fuel pool wall to be about 1.5×10^{-5} per year (for a non-single-failure proof handling system).

In NUREG-1353, it was decided based on engineering judgement that conformance with NUREG-0612 guidelines would reduce the probability of a load drop as presented in NUREG/CR-4982 by a factor of 1,000. Based on Table A2c-2, the fault tree method indicates that the expected reduction was in the 10 to 100 range. For 100 lifts per year, the NUREG/CR-4982 evaluation would estimate the loss-of-inventory from a heavy load drop on the pool wall to be 1.5×10^{-8} per year. As a comparison to this current evaluation, for a load drop on the pool floor, this value should be increased by a factor of 10 to 1.5×10^{-7} per year to account for a load drop from 30 to 40 feet above the spent fuel pool floor (a drop onto the pool from this height likely will cause a loss of inventory.) Based on the fault tree quantification (Table A2c-2), the mean probability for the loss-of-inventory from a heavy load drop was estimated to be 2.0×10^{-7} per year for 100 lifts (for a single-failure proof handling system) for a drop on the spent fuel pool floor and 2.0×10^{-8} per year for a drop on the spent fuel pool wall.

Conclusion

This generic assessment of a heavy load (cask) drop that may result in significant damage to the spent fuel pool indicates that the likelihood of a loss-of-inventory from the spent fuel pool is in the range of 3.1×10^{-8} to 2.3×10^{-6} per year for 100 lifts with a mean value of 2.2×10^{-7} per year for a single-failure proof handling system. These values include the contribution from a heavy load drop on the spent fuel pool floor and a heavy load drop on the spent fuel pool wall.

Uncertainties

1. Incident rate.

   The range used in this evaluation (1.0×10^{-4} to 1.5×10^{-4} incidents per year) was based on the Navy data originally assessed by the staff in NUREG-0612. The 1999 Navy data, like the 1980 data, did not include the number of lifts made and only provided information about the number of incidents. The cask loading experience at LWRs and the GCR tends to support values used for the incident range.

2. Drop rate.

   The drop rate, about 1-in-10, was based on the 1999 Navy data. Previous studies used engineering judgement to estimate the drop rate to be as low as 1-in-100.

3. Load path.

   The fraction of the load path over which a load drop may cause sufficient damage to the spent fuel pool to result in a loss-of-inventory was estimated to be between 0.5% and 6.25% of the total path needed to lift, move, and set down the load. This range was developed by the staff for the NUREG-0612 evaluation.

4. Load handling design.

   The benefit of a single-failure proof load handing system to reduce the probability of a load drop was estimated to be about a factor of 10 to 100 improvement over a non-single failure proof load handling system, based on the fault tree quantifications in this evaluation. Previous studies have used engineering judgement to estimate the benefit to be as high as 1,000.

Table A2c-1 - Summary of the 1996-1999 Navy crane data

	Summary by Incident Type (fraction of events)	ID	Non-rigging Fraction	Rigging Fraction	Total Traction
	Crane collision	CC	0.17	0.00	0.17
	Damaged crane	DC	0.20	0.08	0.27
	Damaged load	DL	0.02	0.03	0.05
	Dropped load	DD	0.03	0.06	0.09
	Load collision	LC	0.11	0.03	0.14
	Other	OO	0.02	0.00	0.02
	Overload	OL	0.08	0.05	0.12
	Personnel injury	PI	0.03	0.05	0.08
	Shock	SK	0.00	0.02	0.02
	Two-blocking	TB	0.05	0.00	0.05
	Unidentified	UD	0.02	0.00	0.02
	Totals		0.70	0.30	1.00
	Summary by Incident Cause (fraction of total events)	ID	Fraction		
	Improper operation	IO	0.38		
	Procedures	PROC	0.20		
	Equipment failure	EQ	0.05		
	Improper rigging(1)	IR	0.30		
	Others	OTHER	0.08		
	Totals		1.00		
Fault Tree ID(2)	**Application of new Navy data to heavy load drop evaluation**	Fraction		NUREG-0612 Fraction	
F1	OL + 0.5*(DL+LC)	0.14		0.05	
F2	CC + DC + 0.5(DL+LC) + DD + OO + PI + SK + UD + 0.3*IR	0.61		0.53	
F3	TB	0.05		0.35	
F4	Assume next incident	(0.01)		(1/44)	
F5	Rigging 0.7*IR	0.21		0.07	
	Totals	1.00		1.00	

Notes:

1. Based on database description, 30% or "improper rigging" by incident cause were rigging failures during crane movement, and 70% of "improper rigging" by incident cause were rigging errors.

2. F1 - Load hangup resulting from operator error (assume 50% of "damaged load" and "load collision" lead to hangup)
   F2 - Failure of component with a backup component (assume 50% of "damaged load" and "load collision" lead to component failure)
   F3 - Two-blocking event
   F4 - Failure of component without a backup
   F5 - Failure from improper rigging

Table A2c-2 - Summary of NUREG-0612 heavy loads evaluation (for cask drop) with new 1990s Navy crane data values and WIPP rigging HEP method

Event	Description	Units	High	Low	Mean
N0	Base range of failure of handling system	/year	1.5e-04	1.0e-05	5.4e-05
	Crane Failure				
F1	Fraction of load hangup events (new 1990s Navy data)	---	0.14	0.14	0.14
CF11	Operator error leading to load hangup (N0*F1))	/year	2.0e-05	1.4e-06	7.4e-06
CF12	Failure of the overload device	/demand	1.0e-02	1.0e-03	4.0e-03
CF1	Load hangup event (CF11*CF12)	/year	**2.0e-07**	**1.4e-09**	**3.0e-08**
F2	Fraction of component failure events (new 1990s Navy data)	---	0.61	0.61	0.61
CF21	Failure of single component with a backup (N0*F2)	/year	9.1e-05	6.1e-06	3.3e-05
CF22	Failure of backup component given CF21	/demand	1.0e-01	1.0e-02	4.0e-02
CF2	Failure due to random component failure (CF21*CF22)	/year	**9.1e-06**	**6.1e-08**	**1.3e-06**
F3	Fraction of two-blocking events (new 1990s Navy data)	---	0.05	0.05	0.05
CF31	Operator error leading to Two-blocking (N0*F3)	/year	6.8e-06	4.5e-07	2.5e-06
CF32	Failure of lower limit switch	/demand	1.0e-02	1.0e-03	4.0e-03
CF33	Failure of upper limit switch	/demand	1.0e-01	1.0e-02	4.0e-02
CF3	Two-blocking event (CF31*CF32*CF33)	/year	**6.8e-09**	**4.5e-12**	**4.0e-10**
F4	Fraction of single component failure (new 1990s Navy data)	---	0.01	0.01	0.01
F4'	Credit for NUREG-0554	/demand	0.10	0.10	0.10
CF4	Failure of component that doesn't have backup (N0*F4*F4')	/year	**2.2e-07**	**1.5e-08**	**8.1e-08**
CRANE	**Failure of crane (CF1+CF2+CF3+CF4)**	**/year**	**9.5e-06**	**7.7e-08**	**1.4e-06**
D1	Lifts per year leading to drop (100 lifts per year, drops from non-rigging)	No.	3	3	3
CF	**Failure of crane leading to load drop (CRANE*D1)**	/year	**2.9e-05**	**2.3e-07**	**4.4e-06**
	Rigging failure - Based on WIPP method				
F5	Fraction of improper rigging events (new 1990s Navy data)	---	0.21	0.21	0.21
CR11	Failure due to improper rigging, mean from WIPP study	/year	8.7e-07	8.7e-07	8.7e-07
CR12	Failure of redundant/alternate rigging	N/A			
RIGGING	**Failure due to improper rigging (CR11)**	**/year**	**8.7e-07**	**8.7e-07**	**8.7e-07**
D2	Lifts per year leading to drop (100 lifts per year, drops from rigging)	No.	6	6	6
CR	**Failure of rigging leading to a load drop (RIGGING*D2)**	**/year**	**5.3e-06**	**5.3e-06**	**5.3e-06**
FHLS	**Failure of heavy load (crane and rigging) system (CRANE+RIGGING)**	**/year**	**1.0e-05**	**9.5e-07**	**2.3e-06**
CFCR	**Total failures (crane and rigging) leading to a load drop (CF+CR)**	**/year**	**3.4e-05**	**5.5e-06**	**9.6e-06**
	Loss-of-inventory for a single-failure proof crane				
RF	Fraction of year over which a release may occur	---	1.00	1.00	1.00
P	Fraction of path near/over pool	---	0.25	0.05	0.13
P'	Fraction of path critical for load drop	---	0.25	0.10	0.16
LOI-S	**(CFCR) * P * P' * RF**	**/year**	**2.1e-06**	**2.8e-08**	**2.0e-07**
	Loss-of-inventory for a non single-failure proof crane				
CFCRNON	Total failures leading to a dropped load (est. from NUREG-0612)	No.	7.5e-05	1.0e-07	2.1e-05
RF	Fraction of year over which a release may occur	---	1.00	1.00	1.00
LOI-N	**(CFCRNON) * P * P' * RF**	**/year**	**7.5e-05**	**1.0e-07**	**2.1e-05**
	Risk reduction for a single-failure proof crane (LOI-N /LOI-S)	**---**	**35**	**4**	**104**

Table A2c-3 - WIPP evaluation for failure to secure load (improper rigging estimate)

Symbol	HEP	Explanation of error	Source of HEP (NUREG/CR-1278)
A_1	3.75×10^{-3}	Improperly make a connection, including failure to test locking feature for engagement	Table 20-12 Item 13 Mean value (0.003, EF[1] = 3)
B_1	0.75	The operating repeating the actions is modeled to have a high dependency for making the same error again. It is not completely independent because the operator moves to the second lifting leg and must physically push the locking balls to insert the pins	Table 20-21 Item 4(a) High dependence for different pins. Two opportunities (the second and third pins) to repeat the error is modeled as 0.5+(1-0.5)*0.5 = 0.75
C_1	1.25×10^{-3}	Checker fails to verify proper insertion of the connector pins, and that the status affects safety when performing tasks	Table 20-22 Item 9 Mean value (0.001, EF = 3)
D_1	0.15	Checker fails to verify proper insertion of the connector pins at a later step, given the initial failure to recognize error. Sufficient separation in time and additional cues to warrant moderate rather than total or high dependency.	Table 20-21 Item 3(a) Moderate dependency for second check
F_1	5.2×10^{-7}	Failure rate if first pin improperly connected	$A_1 * B_1 * C_1 * D_1$
a_1	0.99625	Given first pin was improperly connected	
A_2	3.75×10^{-3}	Improperly make a connection, including failure to test locking feature for engagement	Table 20-12 Item 13 Mean value (0.003, EF = 3)
B_2	0.5	The operating repeating the actions is modeled to have a high dependency for making the same error again. It is not completely independent because the operator moves to the second lifting leg and must physically push the locking balls to insert the pins	Table 20-21 Item 4(a) High dependence for different pins. Only one opportunity for error (third pin)
C_2	1.25×10^{-3}	Checker fails to verify proper insertion of the connector pins, and that the status affects safety when performing tasks	Table 20-22 Item 9 Mean value (0.001, EF = 3)
D_2	0.15	Checker fails to verify proper insertion of the connector pins at a later step, given the initial failure to recognize error. Sufficient separation in time and additional cues to warrant moderate rather than total or high dependency.	Table 20-21 Item 3(a) Moderate dependency for second check
F_2	3.5×10^{-7}	Failure rate if first pin improperly connected	$a_1 * A_2 * B_2 * C_2 * D_2$
F_T	8.7×10^{-7}	Total failure due to human error	F1 + F2

(1) Note: The EF (error factor) is the 95th percentile/50th percentile (median). For an EF of 3, the mean-to-median multiplier is 0.8.

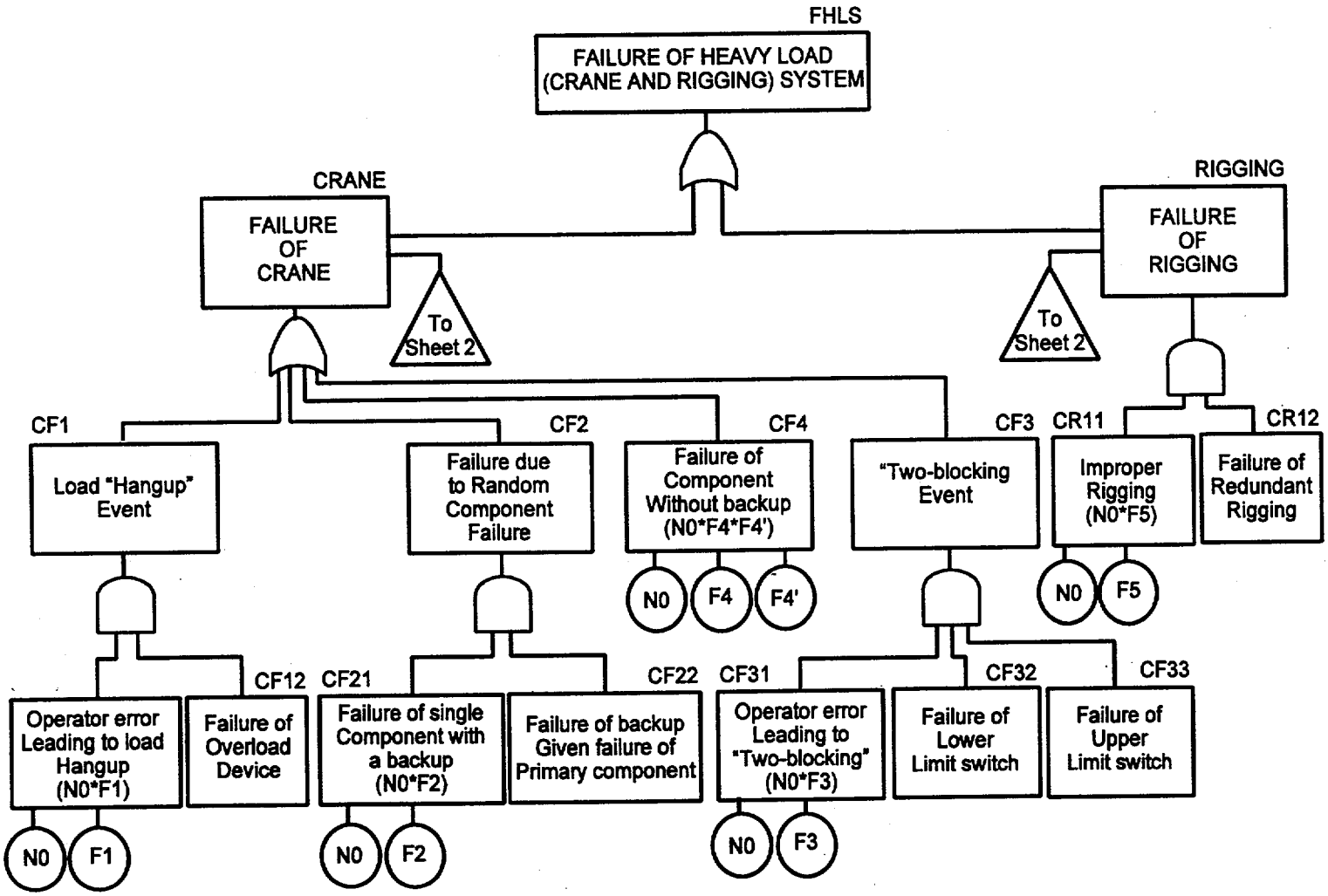

Figure A2c-1 (sheet 1 of 2) - Heavy load drop fault trees

RF

Fraction of year
Over which release
May occur

P'

Fraction of path
Critical for
Load drop

Figure A2c-1 (sheet 2 of 2) - Heavy load drop fault trees

App 2d Aircraft

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Appendix 2d Structural Integrity of Spent Fuel Pool Structures Subject to Aircraft Crashes

1. Introduction

The mean frequency for significant PWR or BWR spent fuel pool damage resulting from a direct hit from an aircraft was estimated based on the point target model for a 100 x 50-foot pool to be 2.9×10^{-9} per year. The estimated frequency of loss of support systems leading to spent fuel pool uncovery is bounded by other initiators.

2. Analysis

A detailed structural evaluation of how structures will respond to an aircraft crash is beyond the scope of this effort. The building or facility characteristics were chosen to cover a range typical of a spent fuel pool that is contained in a PWR auxiliary building or a BWR secondary containment structure. In general, PWR spent fuel pools are located on, or below grade, and BWR spent fuel pools, while generally elevated about 100 feet above grade, are located inside a secondary containment structure. The vulnerability of support systems (power supplies, heat exchangers and makeup water supplies) requires a knowledge of the size and location of these systems at decommissioning plants, information not readily available. However, we believe this analysis is adequately broad to provide a reasonable approximation of decommissioning plant vulnerability to aircraft crashes.

The staff used the generic data provided in DOE-STD-3014-96 [Ref. 1] to assess the likelihood of an aircraft crash into or near a decommissioned spent fuel pool. Aircraft damage can affect the structural integrity of the spent fuel pool or the availability of nearby support systems, such as power supplies, heat exchangers, and makeup water sources, and may also affect recovery actions.

The frequency of an aircraft crashing into a site, F, was obtained from the four-factor formula in DOE-STD-3014-96, and is referred to as the effective aircraft target area model:

$$F = \sum_{i,j,k} N_{ijk} \cdot P_{ijk} \cdot f_{ijk}(x,y) \cdot A_{ij} \qquad \text{Equation A2d-1}$$

where:

N_{ijk} =	estimated annual number of site-specific aircraft operations (no./yr)
P_{ijk} =	aircraft crash rate (per takeoff and landing for near-airport phases) and per flight for in-flight (nonairport) phase of operation
$f_{ijk}(x,y)$ =	aircraft crash location probability (per square mile)
A_{ij} =	site-specific effective area for the facility of interest, including skid and fly-in effective areas (square miles)
i =	(index for flight phase): i=1,2, and 3 (takeoff, in-flight, landing)
j =	(index for aircraft category, or subcategory)
k =	(index for flight source): there could be multiple runways and nonairport operations

The site-specific area is shown in Figure A2d-1 and is further defined as:

$$A_{eff} = A_f + A_s$$

where:

Equation A2d-2

$$A_f = (WS + R) \cdot (H \cdot \cot\theta) + \frac{2 \cdot L \cdot W \cdot WS}{R} + L \cdot W$$

$$A_s = (WS + R) \cdot S$$

and where:

A_{eff} = total effective target area
A_f = effective fly-in area
A_s = effective skid area
WS= wing span
cotθ= mean of cotangent of aircraft impact angle
H= height of facility
L= length of facility
W= width of facility
S= aircraft skid distance
R= length of facility diagonal

Alternatively, a point target area model was defined as the area (length times width) of the facility in question, which does not take into account the size of the aircraft.

Table A2d-1 summarizes the generic aircraft data and crash frequency values for five aircraft types (from Tables B-14 through B-18 of DOE-STD-3014-96). The data given in Table A2d-1 were used to determine the frequency of aircraft hits per year for various building sizes (length, width, and height) for the minimum, average, and maximum crash rates. The resulting frequencies are given in Table A2d-2. The product $N_{ijk}*P_{ijk}*f_{ijk}(x,y)$ for Equation A2d-1 was taken from the crashes per mi^2/yr and A_{ij} was obtained from Equation A2d-2 for aircraft characteristics. Two sets of data were generated: one included the wing and skid lengths, using the effective aircraft target area model, and the other considered only the area (length times width) of the site, using the point target area model.

The results from the DOE effective aircraft target area model, using the generic data in Table A2d-1, were compared to the results of two evaluations reported in Reference 2. The first evaluation of aircraft crash hits was summarized by C.T. Kimura et al. in Reference 3. The DWTF Building 696 was assessed in the Kimura report. It was a 1-story 254-feet-long 80-feet-wide, 39-feet-high structure. The results of Kimura's study are given in Table A2d-3.

Applying the DOE generic data to the DWTF resulted in a frequency range of 6.5×10^{-9} hits per year to 6.6×10^{-5} hits per year, with an average value of 4.4×10^{-6} per year, for the effective aircraft target area model. For the point target area model, the range was 4.4×10^{-10} to 2.2×10^{-6} per year, with an average value of 1.5×10^{-7} per year.

The second evaluation was presented in a paper by K. Jamali [Ref. 4] in which additional facility evaluations were summarized. For the Seabrook Nuclear Power Station, Jamali's application of the DOE effective aircraft target area model to the final safety analysis report (FSAR) data resulted in an impact frequency 2.4×10^{-5} per year. The Millstone Unit 3 plant area was reported as 9.5×10^{-3} square miles and the FSAR aircraft crash frequency as 1.6×10^{-6} per year. Jamali applied the DOE effective aircraft target area model to information in the Millstone Unit 3 FSAR. Jamali reported an impact frequency of 2.7×10^{-6} per year, using the areas published in the FSAR and 2.3×10^{-5} per year, and using the effective area calculated the effective aircraft target area model.

When the generic DOE data in Table A2d-1 were used (for a 514 x 514 x 100-foot site), the estimated impact frequency range was 6.3×10^{-9} to 2.9×10^{-5} per year, with an average of 1.9×10^{-6} per year, for the point target area model. The effective aircraft target area model gave an estimated range of 3.1×10^{-8} to 2.4×10^{-4} per year, with an average of 1.6×10^{-5} per year.

A site-specific evaluation for Three Mile Island Units 1 and 2 was documented in NUREG/CR-5042 [Ref. 5]. The NUREG estimated the aircraft crash frequency to be 2.3×10^{-4} accidents per year, about the same value as would be predicted with the DOE data set for the maximum crash rate for a site area of 0.01 square miles.

NUREG/CR-5042 summarized a study of a power plant response to aviation accidents. The results are given in Table A2d-4. The probability of the penetration of an aircraft through reinforced concrete was taken from that study.

Based on comparing these plant-specific aircraft crash evaluations with the staff's generic evaluation, there were no significant differences between the results from the DOE model whether generic data were used to provide a range of aircraft crash hit frequencies or whether plant-specific evaluations were performed.

3. Estimated Frequencies of Significant Spent Fuel Pool Damage

The frequency for significant PWR spent fuel pool damage resulting from a direct hit was estimated based on the point target model for a 100 x 50-foot pool with a conditional probability of 0.32 (large aircraft penetrating 6-ft of reinforced concrete) that the crash resulted in significant damage. If 1-of-2 aircraft are large and 1-of-2 crashes result in spent fuel uncovery, then the estimated range is 9.6×10^{-12} to 4.3×10^{-8} per year. The average frequency was estimated to be 2.9×10^{-9} per year.

The mean frequency for significant BWR spent fuel pool damage resulting from a direct hit was estimated to be the same as that for the PWR, 2.9×10^{-9} per year.

4. Support System Unavailability

The frequency for loss of a support system (e.g., power supply, heat exchanger, or makeup water supply) was estimated based on the DOE model, including wing and skid area, for a 400 x 200 x 30-foot area with a conditional probability of 0.01 that one of these systems is hit. The

estimated value range was 1.0×10^{-6} to 1.0×10^{-10} per year. The average value was estimated to be 7.0×10^{-8} per year. This value does not credit onsite or offsite recovery actions.

As a check, we calculated the frequency for loss of a support system supply based on the DOE model, including wing and skid area, for a 10 x 10 x 10-foot structure. The estimated frequency range was 1.1×10^{-9} to 1.1×10^{-5} per year with the wing and skid area modeled, with the average estimated to be 7.3×10^{-7} per year. Using the point model, the estimated value range was 2.4×10^{-12} to 1.1×10^{-8} per year, with the average estimated to be 7.4×10^{-10} per year. This value does not credit onsite or offsite recovery actions.

5. Uncertainties

Mark-I and Mark-II secondary containments do not appear to have any significant structures that would reduce the likelihood of penetration, although on one side there may be a reduced likelihood due to other structures. Mark-III secondary containments may reduce the likelihood of penetration, since the spent fuel pool may be considered to be protected by additional structures.

6. References

(1) DOE-STD-3014-96, "Accident Analysis for Aircraft Crash Into Hazardous Facilities," U.S. Department of Energy (DOE), October 1996

(2) A. Mosleh and R.A. Bari (eds),"Probabilistic Safety Assessment and Management," *Proceedings of the 4th International Conference on Probabilistic Safety Assessment and Management*, PSAM 4, Volume 3, 13-18 September 1998, New York City.

(3) C.T. Kimura et al.,"Aircraft Crash Hit Analysis of the Decontamination and Waste Treatment Facility (DWTF), Lawrence Livermore National Laboratory.

(4) K. Jamali, et al., "Application of Aircraft Crash Hazard Assessment Methods to Various Facilities in the Nuclear Industry."

(5) NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," Lawrence Livermore National Laboratory, December 1987.

Table A2d-1 Generic Aircraft Data

Aircraft	Wingspan (ft)	Skid distance (ft)	cotθ	Crashes per mi²/yr			Notes
				Min	Ave	Max	
General aviation	50	1440	10.2	$1x10^{-7}$	$2x10^{-4}$	$3x10^{-3}$	
Air carrier	98	60	8.2	$7x10^{-8}$	$4x10^{-7}$	$2x10^{-6}$	
Air taxi	58	60	8.2	$4x10^{-7}$	$1x10^{-6}$	$8x10^{-6}$	
Large military	223	780	7.4	$6x10^{-8}$	$2x10^{-7}$	$7x10^{-7}$	takeoff
Small military	100	447	10.4	$4x10^{-8}$	$4x10^{-6}$	$6x10^{-8}$	landing

Table A2d-2 Aircraft Hits Per Year

Building (L x W x H) (ft)	Average effective area (mi²)	Minimum hits (per year)	Average hits (per year)	Maximum hits (per year)
With the DOE effective aircraft target area model				
100 x 50 x 30	$6.9x10^{-3}$	$3.2x10^{-9}$	$2.1x10^{-6}$	$3.1x10^{-5}$
200 x 100 x 30	$1.1x10^{-2}$	$5.3x10^{-9}$	$3.7x10^{-6}$	$5.5x10^{-5}$
400 x 200 x 30	$2.1x10^{-2}$	$1.0x10^{-8}$	$7.0x10^{-6}$	$1.0x10^{-4}$
200 x 100 x 100	$1.8x10^{-2}$	$9.6x10^{-9}$	$5.1x10^{-6}$	$7.6x10^{-5}$
400 x 200 x 100	$3.3x10^{-2}$	$1.8x10^{-8}$	$9.6x10^{-6}$	$1.4x10^{-4}$
80 x 40 x 30	$6.1x10^{-3}$	$2.8x10^{-9}$	$1.8x10^{-6}$	$2.7x10^{-5}$
10 x 10 x 10	$2.9x10^{-3}$	$1.1x10^{-9}$	$7.3x10^{-7}$	$1.1x10^{-5}$
With the point target area model				
100 x 50 x 0	$1.8x10^{-4}$	$1.2x10^{-10}$	$3.7x10^{-8}$	$5.4x10^{-7}$
200 x 100 x 0	$7.2x10^{-4}$	$4.8x10^{-10}$	$1.5x10^{-7}$	$2.2x10^{-6}$
400 x 200 x 0	$2.9x10^{-3}$	$1.9x10^{-9}$	$5.9x10^{-7}$	$8.6x10^{-6}$
80 x 40 x 0	$1.1x10^{-4}$	$1.1x10^{-11}$	$2.4x10^{-8}$	$3.5x10^{-7}$
10 x 10	$3.6x10^{-6}$	$2.4x10^{-12}$	$7.4x10^{-10}$	$1.1x10^{-8}$

Table A2d-3 DWTF Aircraft Crash Hit Frequency (per year)

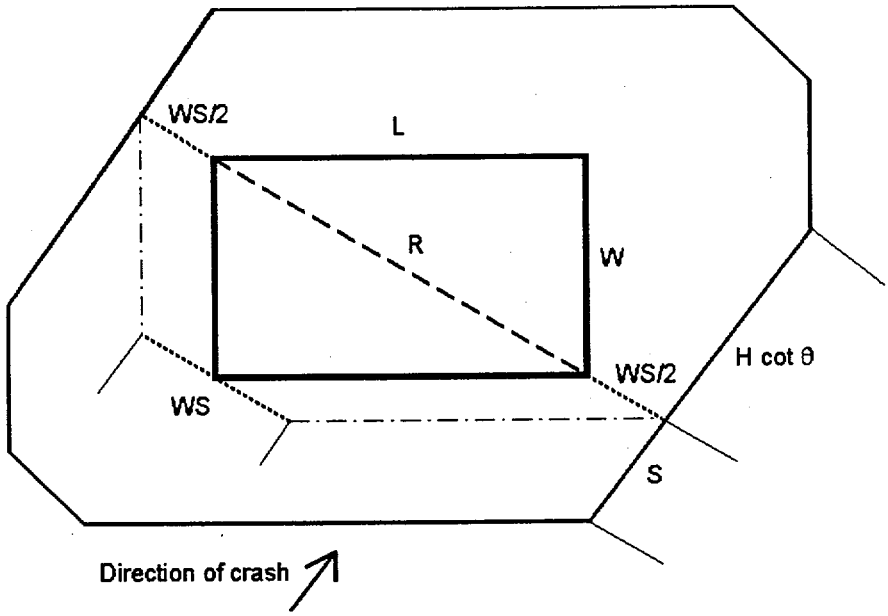
Period	Air Carriers	Air Taxes	General Aviation	Military Aviation	Total[1]
1995	1.72×10^{-7}	2.47×10^{-6}	2.45×10^{-5}	5.03×10^{-7}	2.76×10^{-5}
1993-1995	1.60×10^{-7}	2.64×10^{-6}	2.82×10^{-5}	6.47×10^{-7}	3.16×10^{-5}
1991-1995	1.57×10^{-7}	2.58×10^{-6}	2.89×19^{-5}	7.23×10^{-7}	3.23×10^{-5}
1986-1995	1.52×10^{-7}	2.41×10^{-6}	2.89×10^{-5}	8.96×10^{-7}	3.23×10^{-5}

Note (1): Various periods were studied to assess variations in air field operations.

Table A2d-4 Probability of Penetration as a Function of Location and Concrete Thickness

		Probability of penetration			
		Thickness of reinforced concrete			
Plant location	Aircraft type	1 foot	1.5 feet	2 feet	6 feet
≤ 5 miles from airport	Small ≤ 12,000 lbs	0.003	0	0	0
	Large > 12,000 lbs	0.96	0.52	0.28	0
> 5 miles from airport	Small ≤ 12,000 lbs	0.28	0.06	0.01	0
	Large > 12,000 lbs	1.0	1.0	0.83	0.32

Figure A2d-1 Rectangular Facility Effective Target Area Elements



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Appendix 2d Structural Integrity of Spent Fuel Pool Structures Subject to Aircraft Crashes

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Summary

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~~How are we numbering~~

The site-specific area is shown in Figure A2d-1 and is further defined as:

$$A_{eff} = A_f + A_s$$

where:

Equation A2d-2

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and where:

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H = height of facility
L = length of facility
W = width of facility
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Alternatively, a point target area model was defined as just the area (length times width) of the facility in question, which does not take into account the size of the aircraft.

Table A2d-1 summarizes the generic aircraft data and crash frequency values for five aircraft types (from Tables B-14 through B-18 of DOE-STD-3014-96). The data presented in Table A2d-1 were used to determine the frequency of aircraft hits per year for various building sizes (length, width, and height) for the minimum, average, and maximum crash rates. The resulting frequencies are presented in Table A2d-2. The product $N_{ijk}*P_{ijk}*f_{ijk}(x,y)$ for Equation A2d-1 was taken from the crashes per mi^2-yr and A_{ij} was obtained from Equation A2d-2 based on aircraft characteristics. Two sets of data were generated: one included the wing and skid lengths using the effective aircraft target area model and a second case which considered only the area (length times width) of the site using the point target area model.

The results from the DOE effective aircraft target area model, using the generic data in Table A2d-1, were compared to the results of two evaluations reported in "Probabilistic Safety Assessment and Management," A. Mosleh and R.A. Bari (Eds), PSAM 4, Volume 3, Proceedings of the 4th International Conference on Probabilistic Safety Assessment and Management, 13-18 September 1998, New Your City, USA. The first evaluation of aircraft crash hits was summarized by C.T. Kimura, et al., in "Aircraft Crash Hit Analysis of the Decontamination and Waste Treatment Facility (DWTF) at the Lawrence Livermore National Laboratory (LLNL)." The DWTF Building 696 was assessed in the Kimura report. It was a 254 feet long by 80 feet wide, 1-story, 39 feet high structure. The results of Kimura's study are shown in Table A2d-3.

Applying the DOE generic data to the DWTF resulted in a frequency range of 6.5×10^{-9} hits per year to 6.6×10^{-5} hits per year, with an average value of 4.4×10^{-6} per year, for the effective aircraft target area model. For the point target area model, the range was 4.4×10^{-10} to 2.2×10^{-6} per year, with an average value of 1.5×10^{-7} per year.

The second evaluation was presented in a paper by K. Jamali, et al., "Application of Aircraft Crash Hazard Assessment Methods to Various Facilities in the Nuclear Industry," in which additional facility evaluations were summarized. For the Seabrook Nuclear Power Station, Jamali's application of the DOE effective aircraft target area model to the Final Safety Analysis Report (FSAR) data resulted in an impact frequency 2.4×10^{-5} per year. The Millstone 3 plant area was reported as 9.5×10^{-3} square miles and the FSAR aircraft crash frequency was reported to be 1.6×10^{-6} per year. Jamali applied the DOE effective aircraft target area model to information found in the Millstone 3 FSAR. Jamali reported an impact frequency of 2.7×10^{-6} per year using the areas published in the FSAR and 2.3×10^{-5} per year using the effective area calculated the effective aircraft target area model.

When the generic DOE data in Table A2d-1 were used (for a 514x514x100 foot site), the estimated impact frequency range was 6.3×10^{-9} to 2.9×10^{-5} per year, with an average of 1.9×10^{-6} per year, for the point target area model. The effective aircraft target area model resulted in estimated range between 3.1×10^{-8} to 2.4×10^{-4} per year, with an average of 1.6×10^{-5} per year.

A site-specific evaluation for Three-Mile Island Units 1 and 2 was documented in NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," Lawrence Livermore National Laboratory, December 1987. The NUREG estimated the aircraft crash frequency to be 2.3×10^{-4} accidents per year, about the same value as would be predicted with the DOE data set for the maximum crash rate for a site area of 0.01 square miles.

NUREG/CR-5042 summarized a study of a power plant response to aviation accidents. The results are presented in Table A2d-4. The probability of the penetration of an aircraft through reinforced concrete was taken from that study.

Based on comparing these plant-specific aircraft crash evaluations with our generic evaluation, there were no significant differences between the results from the DOE model whether generic data were used to provide a range of aircraft crash hit frequencies or whether plant-specific evaluations were performed.

Estimated Frequencies of Significant Spent Fuel Pool Damage

The frequency for significant PWR spent fuel pool damage resulting from a direct hit was estimated based on the point target model for a 100x50 foot pool with a conditional probability of 0.32 (large aircraft penetrating 6-ft of reinforced concrete) that the crash resulted in significant damage. If 1-of-2 aircraft are large and 1-of-2 crashes result in spent fuel uncovery, then the estimated range is 9.6×10^{-12} to 4.3×10^{-8} per year. The average frequency was estimated to be 2.9×10^{-9} per year.

The mean frequency for significant BWR spent fuel pool damage resulting from a direct hit was estimated to be the same as that for the PWR, 2.9×10^{-9} per year.

4. Support System Unavailability

The frequency for loss of a support system (e.g., power supply, heat exchanger, or makeup water supply) was estimated based on the DOE model including wing and skid area for a 400x200x30 foot area with a conditional probability of 0.01 that one of these systems is hit. The estimated value range was 1.0×10^{-6} to 1.0×10^{-10} per year. The average value was estimated to be 7.0×10^{-8} per year. This value does not credit onsite or offsite recovery actions.

As a check, we calculated the frequency for loss of a support system supply based on the DOE model including wing and skid area for a 10x10x10 foot structure. The estimated frequency range was 1.1×10^{-9} to 1.1×10^{-5} per year with the wing and skid area modeled, with the average estimated to be 7.3×10^{-7} per year. Using the point model, the estimated value range was 2.4×10^{-12} to 1.1×10^{-8} per year, with the average estimated to be 7.4×10^{-10} per year. This value does not credit onsite or offsite recovery actions.

5. Uncertainties

Mark-I and Mark-II secondary containments do not appear to offer any significant structures to reduce the likelihood of penetration, although on one side there may be a reduced likelihood due to other structures. Mark-III secondary containments may reduce the likelihood of penetration as the spent fuel pool may be considered to be protected by additional structures.

Table A2d-1 Generic Aircraft Data

Aircraft	Wingspan (ft)	Skid distance (ft)	cotθ	Crashes per mi^2-yr			Notes:
				Min	Ave	Max	
General aviation	50	1440	10.2	$1x10^{-7}$	$2x10^{-4}$	$3x10^{-3}$	
Air carrier	98	60	8.2	$7x10^{-8}$	$4x10^{-7}$	$2x10^{-6}$	
Air taxi	58	60	8.2	$4x10^{-7}$	$1x10^{-6}$	$8x10^{-6}$	
Large military	223	780	7.4	$6x10^{-8}$	$2x10^{-7}$	$7x10^{-7}$	takeoff
Small military	100	447	10.4	$4x10^{-8}$	$4x10^{-6}$	$6x10^{-8}$	landing

Table A2d-2 Aircraft Hits Per Year

Building (L x W x H) (ft)	Average effective area (mi^2)	Minimum hits (per year)	Average hits (per year)	Maximum hits (per year)
With the DOE effective aircraft target area model				
100 x 50 x 30	$6.9x10^{-3}$	$3.2x10^{-9}$	$2.1x10^{-6}$	$3.1x10^{-5}$
200 x 100 x 30	$1.1x10^{-2}$	$5.3x10^{-9}$	$3.7x10^{-6}$	$5.5x10^{-5}$
400 x 200 x 30	$2.1x10^{-2}$	$1.0x10^{-8}$	$7.0x10^{-6}$	$1.0x10^{-4}$
200 x 100 x 100	$1.8x10^{-2}$	$9.6x10^{-9}$	$5.1x10^{-6}$	$7.6x10^{-5}$
400 x 200 x 100	$3.3x10^{-2}$	$1.8x10^{-8}$	$9.6x10^{-6}$	$1.4x10^{-4}$
80 x 40 x 30	$6.1x10^{-3}$	$2.8x10^{-9}$	$1.8x10^{-6}$	$2.7x10^{-5}$
10 x 10 x 10	$2.9x10^{-3}$	$1.1x10^{-9}$	$7.3x10^{-7}$	$1.1x10^{-5}$
With the point target area model				
100 x 50 x 0	$1.8x10^{-4}$	$1.2x10^{-10}$	$3.7x10^{-8}$	$5.4x10^{-7}$
200 x 100 x 0	$7.2x10^{-4}$	$4.8x10^{-10}$	$1.5x10^{-7}$	$2.2x10^{-6}$
400 x 200 x 0	$2.9x10^{-3}$	$1.9x10^{-9}$	$5.9x10^{-7}$	$8.6x10^{-6}$
80 x 40 x 0	$1.1x10^{-4}$	$1.1x10^{-11}$	$2.4x10^{-8}$	$3.5x10^{-7}$
10 x 10	$3.6x10^{-6}$	$2.4x10^{-12}$	$7.4x10^{-10}$	$1.1x10^{-8}$

Table A2d-3 DWTF Aircraft Crash Hit Frequency (Per Year)

Period	Air Carriers	Air Taxes	General Aviation	Military Aviation	Total(1)
1995	1.72×10^{-7}	2.47×10^{-6}	2.45×10^{-5}	5.03×10^{-7}	2.76×10^{-5}
1993-1995	1.60×10^{-7}	2.64×10^{-6}	2.82×10^{-5}	6.47×10^{-7}	3.16×10^{-5}
1991-1995	1.57×10^{-7}	2.58×10^{-6}	2.89×19^{-5}	7.23×10^{-7}	3.23×10^{-5}
1986-1995	1.52×10^{-7}	2.41×10^{-6}	2.89×10^{-5}	8.96×10^{-7}	3.23×10^{-5}

Note (1): Various periods were studied to assess variations in air field operations.

Table A2d-4 Probability Of Penetration As A Function Of Location And Concrete Thickness

		Probability of penetration			
		Thickness of reinforced concrete			
Plant location	Aircraft type	1 foot	1.5 feet	2 feet	6 feet
≤ 5 miles from airport	Small ≤ 12,000 lbs	0.003	0	0	0
	Large > 12,000 lbs	0.96	0.52	0.28	0
> 5 miles from airport	Small ≤ 12,000 lbs	0.28	0.06	0.01	0
	Large > 12,000 lbs	1.0	1.0	0.83	0.32

Figure A2d-1 - Rectangular Facility Effective Target Area Elements

Figure (I have)

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Appendix 2e Structural Integrity of Spent Fuel Pool Structures Subject to Tornados

1. Introduction

Tornado damage from missiles have the potential to affect the structural integrity of the spent fuel pool or the availability of nearby support systems, such as power supplies, cooling pumps, heat exchangers, and makeup water sources, and may also affect recovery actions. Department of Energy (DOE) studies indicate that the thickness of the spent fuel pool walls (greater than four feet of reinforced concrete) is more than sufficient protection from missiles that could be generated by the most powerful tornadoes ever recorded in the United States. In addition, the frequency of meeting or exceeding the wind speeds of F4 to F5 tornadoes (the most powerful tornadoes on the Fujita scale) is estimated to be on the order of $6x10^{-7}$ per year in the areas of the U.S. that are subject to the largest and most frequent tornadoes. The likelihood of meeting or exceeding the size tornado that could damage support systems is on the order of $2x10^{-5}$ per year. This is not the estimated frequency of fuel uncovery on a zirconium fire since the frequency estimate does not include credit for maintaining pool inventory from either on-site or off-site sources.

The probability of failing to maintain inventory was estimated for the case of loss of offsite power from severe weather, where it was assumed that the principal impact of the severe weather was to hamper recovery of offsite power and also to increase the probability of failing to bring off-site sources to bear because of damage to the infrastructure. The situation with tornados is different, because the damage caused by a tornado is relatively localized. Therefore, while a direct hit on the plant could also disable the diesel fire pump, it would be unlikely to also disable off-site resources to the same degree. Therefore, the probability of failing to bring in the off-site resources can be argued to be the same as for the seismic case, i.e., 1E-04, under the assumption that NEI commitments 3 and 4 are implemented.

2. Analysis

The methodology assessing tornado risk developed in NUREG/CR-2944, [Ref. 1] was used for this evaluation. The National Climatic Data Center (NCDC) in Asheville, N.C., keeps weather records for the U.S. for the period 1950 to 1995 [Ref. 2]. Tornado data are reported as the annual average number of (all) tornadoes per 10,000 square miles per state and the annual average number of strong-violent (F2 to F5) tornadoes per square mile per state, as shown in Figures A2e-1 and A2e-2.

The NCDC data were reviewed and a range of frequencies per square mile per year was developed based on the site location and neighboring state (regional) data. In general, the comparison of the NUREG/CR-5042 [Ref. 3] tornado frequencies for all tornadoes to the NCDC tornado frequencies for all reported tornadoes showed good agreement between the two sets of data.

Raw data from the Storm Prediction Center (SPC), for the period 1950 to 1995 was used to develop a database for this assessment. About 121 F5, and 924 F4, tornadoes have been recorded between 1950 and 1995 (an additional 4 in the 1996 to 1998 period). It was

estimated that about 30% of all reported tornadoes were in the F2 to F3 range and about 2.5% were in the F4 to F5 range.

The Department of Energy Report DOE-STD-1020-94, [Ref. 4] has some insights into wind-generated missiles:

(1) For sites where tornadoes are not considered a viable threat, to account for objects or debris a 2x4 inch timber plank weighing 15 lbs is considered as a missile for straight winds and hurricanes. With a recommended impact speed of 50 mph at a maximum height of 30 ft above ground, this missile would break annealed glass, perforate sheet metal siding and wood siding up to to 3/4-in thick. For weak tornadoes, the timber missile horizontal speed is 100 mph effective to a height of 100 ft above ground and a vertical speed of 70 mph. A second missile is considered: a 3-in diameter steel pipe weighing 75 lbs with an impact velocity of 50 mph, effective to a height of 75 ft above ground and a vertical velocity of 35 mph. For the straight wind missile, an 8-in concrete masonry unit (CMU) wall, single wythe (single layer) brick wall with stud wall, or a 4-inch concrete (reinforced) is considered adequate to prevent penetration. For the tornado missile, an 8-to-12-in CMU wall, single wythe brick wall with stud wall and metal ties, or a 4- to 8-inch concrete (reinforced) slab is considered adequate to prevent penetration (depending on the missile). (Refer to DOE-STD-1020-94 for additional details.)

(2) For sites where tornadoes are considered a viable threat, to account for objects or debris the same 2x4 inch timber is considered but for heights above ground to 50 ft. The tornado missiles are (1) the 15 lbs, 2x4 inch timber with a horizontal speed of 150 mph effective up to 200 ft above ground, and a vertical speed of 100 mph; (2) the 3-inch diameter, 75 lbs steel pipe with a horizontal speed of 75 mph and a vertical sped of 50 mph effective up to 100 ft above ground; and (3) a 3,000 lbs automobile with ground speed up to 25 mph. For the straight wind missile, an 8-in CMU wall, single wythe brick wall with stud wall, or a 4-inch concrete (reinforced) is considered adequate to prevent penetration. For the tornado missile, an 8 in CMU reinforced wall, or a 4-to-10-inch concrete (reinforced) slab is considered adequate to prevent penetration (depending on the missile). (Refer to DOE-STD-1020-94 for additional details.)

3. Recommended Values for Risk-informed Assessment of Spent Fuel Pools

The tornado strike probabilities for each F-scale interval were determined from the SPC raw data on a state-averaged basis. For each F-scale, the point strike probability was obtained from the following equation:

$$P_{fs} = \left(\frac{\Sigma_N <a>_T}{A_{ob}} \right) \times \frac{1}{Y_{int}} \qquad \text{Equation A2e-1}$$

where:

P_{fs} = strike probability for F-scale (fs)
$<a>_T$ = tornado area, mi^2
A_{ob} = area of observation, mi^2 (state land area)

Y_{int} = interval over which observations were made, years
Σ_N = sum of reported tornados in the area of observation

The tornado area, $<a>_T$, was evaluated at the midpoint of the path-length and path-width intervals shown in Table A2e-1, based on the SPC path classifications. For example, an F2 tornado with a path-length scale of 2 has an average path length of 6.55 miles and with a path-width scale of 3, an average width of 0.2 miles.

The tornado area, $<a>_T$, was then modified using the method described in NUREG/CR-2944 (based on Table 6b and 7b) to correct the area calculation by observations of the variations in a tornado's intensity along its path length and path width (see Figure A2e-3). Table A2e-2 gives the path-length correction data. Table A2e-3 gives the path-width correction data. The corrected effective area has a calculated $<a>_T$ of about 0.28 mi^2. The combined variation in intensity along the length and across the width of the tornado path is shown in Table A2e-4 (Table 15b from NUREG/CR-2944). For example, an F2 tornado with a path-length scale of 2 and a path-width scale of 3 has a calculated $<a>_T$ of about 0.28 mi^2. The total area is reapportioned using Table A2e-4 to assign 0.11 mi^2 to the F0 classification, 0.13 mi^2 to the F1 classification, and 0.04 mi^2 to the F2 classification.

The risk regionalization scheme from NUREG/CR-2944, as shown in Figure A2e-4, was used to determine the exceedance probability for each region identified. A continental U.S. average was also determined. Figure A2e-4 shows the approximate location of commercial LWRs and independent spent fuel storage facilities.

The SPC raw data for each state was used to determine the F-scale, path-length and path-width characteristics of the reported tornadoes. The effective tornado strike area was corrected using the data from NUREG/CR-2944. Equation A2e-1 was used for each state and the summation and averaging of the states within each region (A, B, C and D, as well as a continental USA average) performed. The results for the exceedance probability per year for each F-scale are given in Table A2e-5, and graphically presented in Figure A2e-5. The SPC data analysis is summarized in Table A2e-6.

4. Significant Pool Damage

An F4 to F5 tornado would be needed to consider the possibility of damage to the spent fuel pool by a tornado missile. The likelihood of having or exceeding this size tornado is estimated to be 5.6×10^{-7} per year (for Region A), or lower. In addition, the spent fuel pool is a multiple-foot thick concrete structure. Based on the DOE-DOE-STD-1020-94 information, it is very unlikely that a tornado missile would penetrate the spent fuel pool, even if it were hit by a missile generated by an F4 or F5 tornado.

5. Support System Availability

An F2 or larger tornado would be needed to consider damage to support systems (power supplies, cooling pumps, heat exchangers, and makeup water sources). The likelihood of the exceedance of this size tornado is estimated to be 1.5×10^{-5} per year (for Region A), or lower. This frequency is bounded by other more likely initiators that can cause loss of support systems.

6. References

1 NUREG/CR-2944, "Tornado Damage Risk Assessment," Brookhaven National Laboratory, September 1982

2 http://www.ncdc.noaa.gov/

3 NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," Lawrence Livermore National Laboratory, December 1987.

4 DOE-STD-1020-94, "Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities," January 1996, Department of Energy

Table A2e-1 Tornado Characteristics

F-scale	Damage and wind speed	Path-length scale		Path-width scale	
		Scale	Length (mi)	Scale	Width (yds)
0	Light Damage (40-72 mph)	0	< 1.0	0	< 18
1	Moderate Damage (73-112 mph)	1	1.0 - 3.1	1	18 - 55
2	Significant Damage (113-157 mph)	2	3.2 - 9.9	2	56 - 175
3	Severe Damage (158-206 mph)	3	10.0 - 31.9	3	176 - 527
4	Devastating Damage (207-260 mph)	4	32 - 99.9	4	528 - 1759
5	Incredible Damage (261-318 mph)	5	100 >	5	1760 >

Table A2e-2 Variation of Intensity Along Length Based on Fraction of Length per Tornado(*)

Local tornado state	Recorded tornado state					
	F0	F1	F2	F3	F4	F5
PL-F0	1	0.383	0.180	0.077	0.130	0.118
PL-F1		0.617	0.279	0.245	0.131	0.125
PL-F2			0.541	0.310	0.248	0.162
PL-F3				0.368	0.234	0.236
PL-F4					0.257	0.187
PL-F5						0.172

(*) - Table 6b from NUREG/CR-2944

Table A2e-3 Variation of Intensity Along Width Based on Fraction of Width Per Tornado(*)

Local tornado state	Recorded tornado state					
	F0	F1	F2	F3	F4	F5
PW-F0	1	0.418	0.154	0.153	0.152	0.152
PW-F1		0.582	0.570	0.310	0.264	0.262
PW-F2			0.276	0.363	0.216	0.143
PW-F3				0.174	0.246	0.168
PW-F4					0.122	0.183
PW-F5						0.092

(*) - Table 7b from NUREG/CR-2944

Table A2e-4 Combined Variation in Intensity Along Length and Across Width of Tornado Path(*)

Local tornado state	True maximum tornado state					
	F0	F1	F2	F3	F4	F5
CV-F0	1.0	0.641	0.380	0.283	0.298	0.286
CV-F1		0.359	0.471	0.433	0.358	0.333
CV-F2			0.149	0.220	0.209	0.195
CV-F3				0.064	0.104	0.116
CV-F4					0.031	0.054
CV-F5						0.016

(*) - Table 15b from NUREG/CR-2944

Table A2e-5 Exceedance Probability for Each F-scale

NUREG/CR-2944 Region	Exceedance probability (per year)					
	F0	F1	F2	F3	F4	F5
A	7.4E-05	4.4E-05	1.5E-05	3.5E-06	5.6E-07	3.1E-08
B	5.6E-05	3.3E-05	1.1E-05	2.5E-06	3.7E-07	2.1E-08
C	2.9E-05	1.5E-05	4.1E-06	8.9E-07	1.3E-07	4.7E-09
D	3.6E-06	1.6E-06	3.9E-07	8.7E-08	1.6E-08	---
USA	3.5E-05	2.0E-05	6.1E-06	1.4E-06	2.2E-07	1.0E-08

Table A2e-6 SPC Data Analysis Summary by State

	NUREG/CR -2944 Region					Tornado F-scale							Point Strike Probability (per year)						Land Area
State	A	B	C	D	Years	F0	F1	F2	F3	F4	F5	Total	F0	F1	F2	F3	F4	F5	(mi^2)
AL	X	X			46	165	364	323	129	36	14	1031	2.9e-05	3.2e-05	1.3e-05	3.7e-06	6.9e-07	4.3e-08	50750
AZ				X	44	90	57	11	2	0	0	160	6.7e-07	2.9e-07	3.6e-08	1.8e-09	0	0	113642
AR	X				46	198	298	331	149	31	0	1007	3.2e-05	3.5e-05	1.3e-05	2.4e-06	1.9e-07	0	52075
CA				X	45	142	58	21	2	0	0	223	5.1e-07	2.7e-07	6.0e-08	2.7e-09	0	0	155973
CO			X	X	46	616	441	99	15	1	0	1172	4.4e-06	2.0e-06	4.2e-07	3.9e-08	3.3e-11	0	103730
CT			X		46	9	29	20	5	2	0	65	1.1e-05	1.1e-05	3.6e-06	8.5e-07	2.2e-07	0	4845
DE			X		42	20	23	11	1	0	0	55	2.6e-05	1.5e-05	1.5e-06	6.4e-09	0	0	1955
DC*					1	1	0	0	0	0	0	1	1.3e-04	0	0	0	0	0	61
FL		X	X		46	1156	665	293	30	4	0	2148	1.5e-05	8.6e-06	2.2e-06	2.8e-07	2.0e-08	0	53997
GA		X			46	147	537	266	65	17	0	1032	2.9e-05	3.0e-05	1.2e-05	3.4e-06	4.3e-07	0	57919
ID				X	42	63	53	8	0	0	0	124	4.7e-07	1.9e-07	1.4e-08	0	0	0	82751
IN	X				46	246	336	263	108	77	8	1038	3.3e-05	3.5e-05	1.5e-05	5.2e-06	1.2e-06	6.7e-08	35870
IA	X				46	478	506	421	119	74	9	1607	3.7e-05	3.7e-05	1.4e-05	3.1e-06	6.1e-07	2.5e-08	55875
IL	X				46	431	440	316	113	39	3	1342	3.0e-05	2.7e-05	9.8e-06	2.5e-06	3.3e-07	2.1e-08	55875
KS	X	X			46	1111	610	404	168	54	16	2363	3.5e-05	3.0e-05	1.1e-05	3.0e-06	5.8e-07	1.1e-07	81823
KY	X				46	79	168	133	65	35	3	483	1.6e-05	1.7e-05	6.9e-06	1.8e-06	3.1e-07	1.4e-08	39732
LA		X			46	225	620	268	123	16	2	1254	2.4e-05	2.2e-05	6.9e-06	1.4e-06	1.2e-07	1.9e-08	43566
ME				X	42	21	44	17	0	0	0	82	1.8e-06	1.1e-06	1.7e-07	0	0	0	30865
MD			X		46	49	92	26	5	0	0	172	1.5e-05	9.2e-06	9.4e-07	8.2e-09	0	0	9775
MA			X		45	24	72	31	8	3	0	138	1.2e-05	1.1e-05	4.3e-06	1.6e-06	3.7e-07	0.0e+00	7838
MI		X	X		45	195	308	210	57	30	7	807	1.4e-05	1.4e-05	5.2e-06	1.4e-06	2.8e-07	1.4e-08	56809
MN		X	X		46	372	336	158	53	28	6	953	1.4e-05	1.2e-05	3.5e-06	7.2e-07	1.3e-07	6.6e-09	79617
MS	X	X			46	226	468	369	136	59	10	1268	4.4e-05	4.4e-05	1.7e-05	5.0e-06	1.0e-06	1.3e-08	46914
MO	X				46	298	577	334	109	48	1	1367	1.8e-05	1.6e-05	5.3e-06	1.3e-06	2.3e-07	2.6e-11	68898
MT				X	44	174	42	33	4	0	0	253	1.0e-06	7.0e-07	2.3e-07	2.2e-08	0	0	145556
NE		X	X		46	827	585	255	105	42	4	1818	2.9e-05	2.9e-05	1.2e-05	3.5e-06	3.5e-07	1.6e-08	76878

Table A2e-6 SPC Data Analysis Summary by State

	NUREG/CR -2944 Region					Tornado F-scale							Point Strike Probability (per year)						Land Area
State	A	B	C	D	Years	F0	F1	F2	F3	F4	F5	Total	F0	F1	F2	F3	F4	F5	(mi^2)
NV				X	34	41	8	0	0	0	0	49	2.9e-07	4.0e-08	0	0	0	0	109806
NH				X	45	24	34	15	2	0	0	75	4.7e-06	2.4e-06	4.7e-07	1.1e-08	0	0	8969
NJ			X		45	43	58	23	4	0	0	128	1.7e-05	6.6e-06	7.9e-07	7.1e-09	0	0	7419
NM			X		46	261	104	31	4	0	0	400	1.5e-06	5.2e-07	8.0e-08	1.1e-09	0	0	121365
NY				X	44	101	106	35	21	5	0	268	7.6e-06	6.1e-06	2.3e-06	8.8e-07	2.2e-07	0	47224
NC			X		46	153	321	143	44	26	0	687	1.5e-05	1.4e-05	4.9e-06	1.5e-06	2.5e-07	0	48718
ND			X		46	490	211	91	28	7	3	830	4.7e-06	3.2e-06	1.1e-06	3.6e-07	9.1e-08	1.1e-08	68994
OH	X				46	157	321	166	53	27	9	733	2.1e-05	1.8e-05	5.6e-06	1.3e-06	3.0e-07	2.8e-08	40953
OK	X				46	845	808	626	209	83	9	2580	4.1e-05	3.9e-05	1.4e-05	3.6e-06	7.0e-07	5.5e-08	68679
OR				X	45	31	15	3	0	0	0	49	2.9e-07	1.5e-07	3.1e-08	0	0	0	96003
PA			X		46	93	220	143	26	22	2	506	9.4e-06	9.0e-06	3.3e-06	9.3e-07	2.0e-07	5.4e-09	44820
RI			X		23	3	4	1	0	0	0	8	1.9e-05	1.3e-05	1.7e-06	0	0	0	1045
SC		X			46	136	234	100	31	15	0	516	1.9e-05	1.9e-05	6.8e-06	1.8e-06	3.0e-07	0	30111
SD		X	X		46	651	259	197	57	7	1	1172	9.7e-06	8.1e-06	3.0e-06	7.7e-07	1.5e-07	1.2e-08	75898
TN	X				46	107	241	139	76	29	4	596	2.2e-05	2.2e-05	8.3e-06	2.1e-06	2.0e-07	1.7e-10	41220
TX		X	X		46	2632	1837	1067	317	76	5	5934	1.6e-05	1.3e-05	4.3e-06	1.1e-06	1.8e-07	3.8e-09	261914
UT				X	43	53	19	6	1	0	0	79	5.1e-07	3.2e-07	1.0e-07	2.8e-08	0	0	82168
VT				X	41	7	14	12	0	0	0	33	3.3e-06	2.0e-06	3.4e-07	0	0	0	9249
VA			X		45	84	132	68	28	6	0	318	8.5e-06	7.0e-06	2.0e-06	4.4e-07	7.1e-08	0	39598
WA				X	41	24	17	12	3	0	0	56	4.9e-07	9.6e-08	2.3e-08	3.6e-09	0	0	66582
WV			X		45	27	36	16	8	0	0	87	2.2e-06	2.4e-06	9.7e-07	2.5e-07	0	0	24087
WI		X	X		46	204	378	276	62	24	5	949	2.6e-05	2.4e-05	7.9e-06	1.4e-06	2.5e-07	3.3e-08	54314
WY				X	46	247	145	43	8	1	0	444	2.5e-06	1.2e-06	3.1e-07	7.1e-08	1.9e-08	0	97105
Sum						13776	13251	7834	2553	924	121	38459							3536342

* DC was not included in the exceedance analysis.

Figure A2e-1

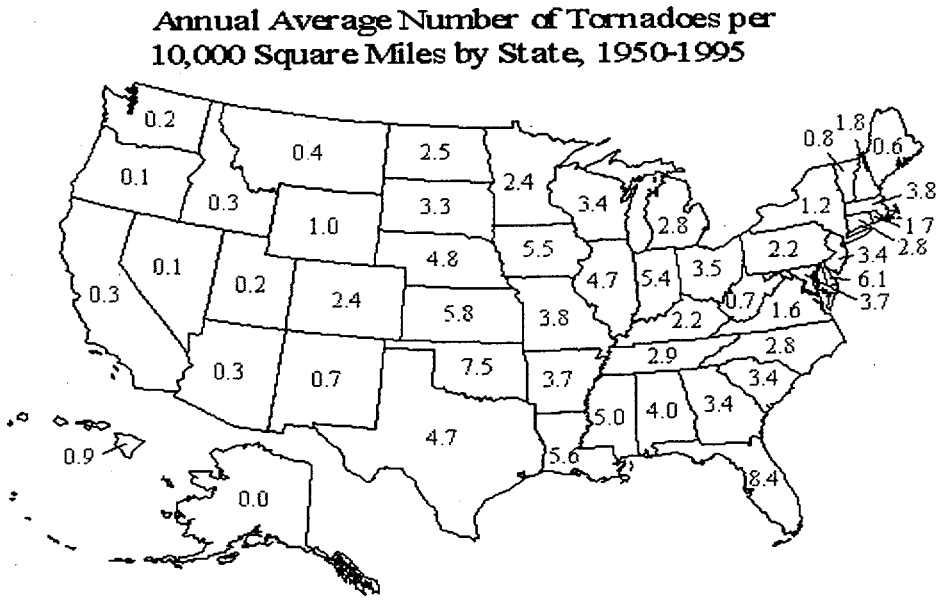


Figure A2e-2

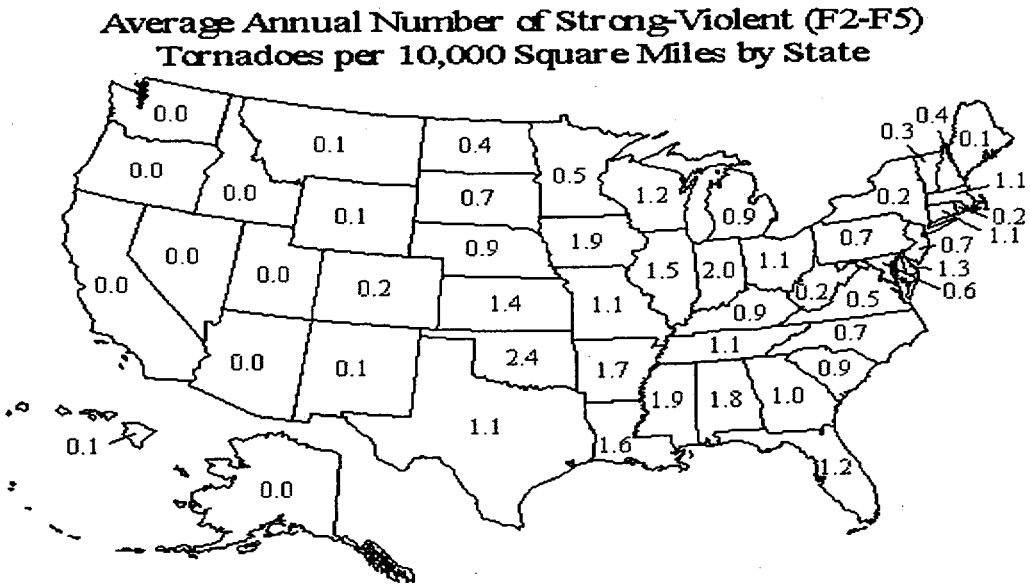


Figure A2e-3 Sketch of Hypothetical F2 Tornado Illustrating Variations

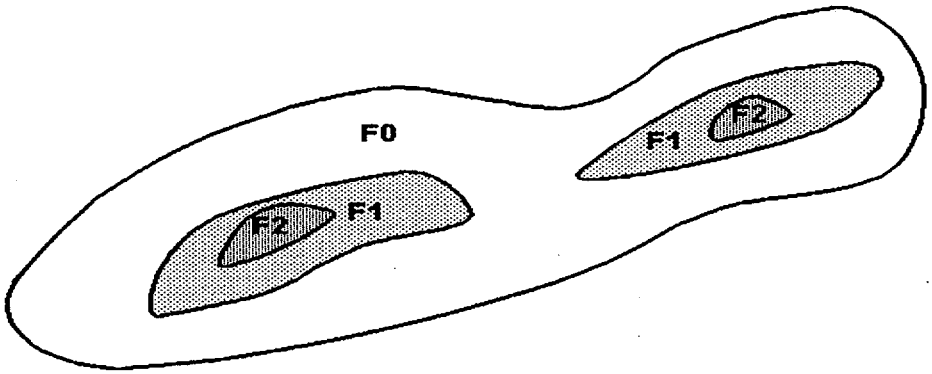


Figure A2e-4 Tornado Risk Regionalization Scheme (from NUREG/CR-2944)

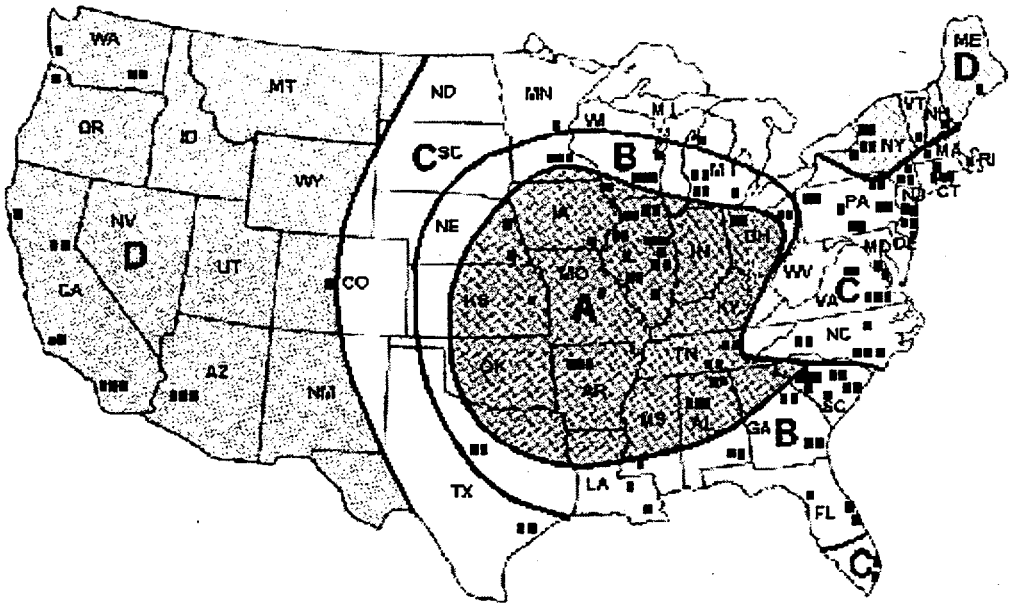
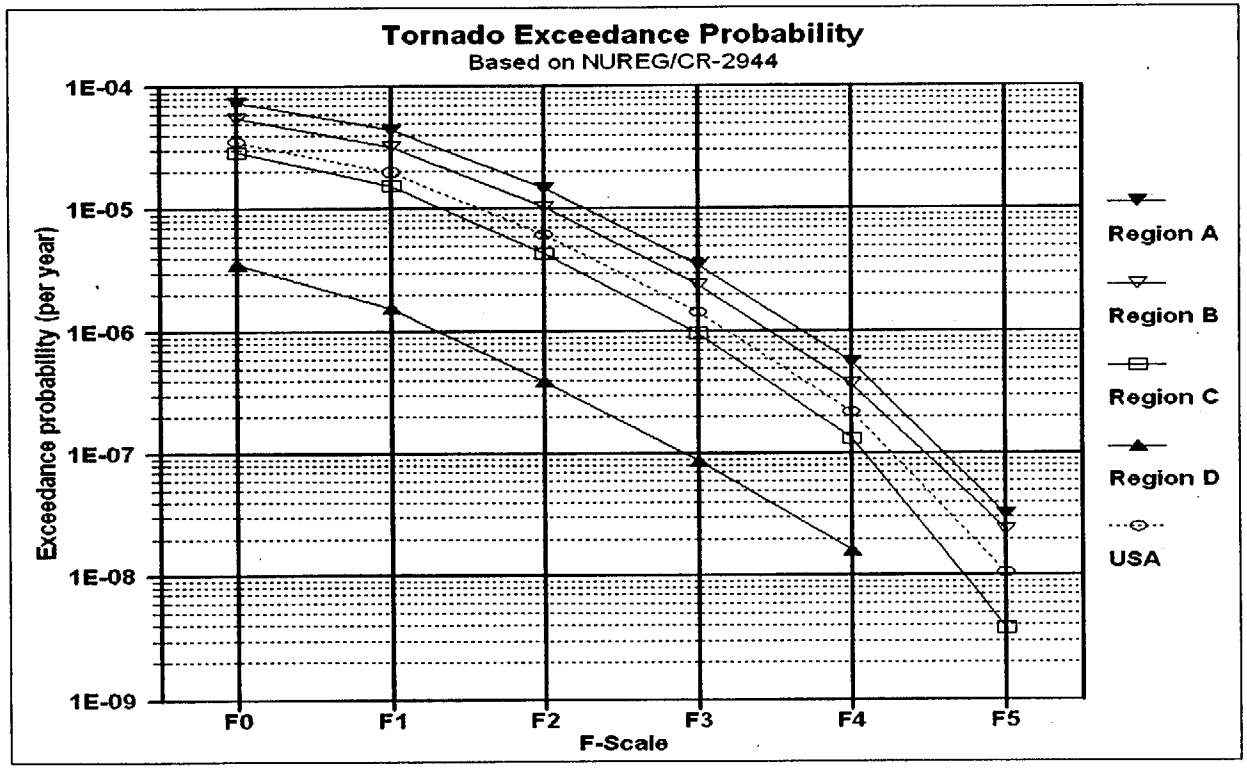


Figure A2e-5 Tornado Exceedance Probability For Each F-scale



Appendix 2e Structural Integrity of Spent Fuel Pool Structures Subject to Tornados and High Winds

1 Summary

Tornado or high winds damage, resulting from missile generation, have the potential to affect the structural integrity of the spent fuel pool or affect the availability of nearby support systems, such as power supplies, cooling pumps, heat exchangers, and water makeup sources, and may also affect recovery actions. Department of Energy studies indicate that the thickness of the spent fuel pool walls (greater than four feet of reinforced concrete) is more than sufficient protection from missiles that could be generated by the most powerful tornadoes ever recorded in the United States. In addition, the frequency of meeting or exceeding the wind speeds of an F5 tornado (the most powerful tornado on the Fujita scale) is estimated to be on the order of $6x10^{-7}$ per year in the areas of the U.S. that are subject to the largest and most frequent tornadoes. The likelihood of meeting or exceeding the size tornado that could damage support systems is on the order of $2x10^{-5}$ per year. The frequency of support system damage from tornadoes is bounded by other more likely events.

2 Analysis

A set of site-specific evaluations for tornados and high winds was documented in NUREG/CR-5042, [Ref. 1]. We note that the study was performed to assess core damage frequencies at operating plants. We used the methodology for the assessment of tornado risk developed In NUREG/CR-2944, [Ref. 2] for this evaluation.

The National Climatic Data Center (NCDC) in Asheville, N.C., keeps weather records for the U.S. for the period 1950 to 1995 [Ref. 3]. These data are reported as the annual average number of (all) tornadoes per 10,000 square mile per state, and the annual average number of strong-violent (F2 to F5) tornadoes per square mile per state, as shown in Figures A2e-1 and A2e-2.

A comparison of the site-specific evaluations (from NUREG/CR-5042) and general regional values from the NCDC database is presented in Table A2e-1. The NCDC data were reviewed and a range of frequencies per square mile per year was developed based on the site location and neighboring state (regional) data. In general, the comparison of the NUREG/CR-5042 tornado frequencies for all tornadoes to the NCDC tornado frequencies for all reported tornadoes showed good agreement between the two sets of data.

The Storm Prediction Center (SPC) raw data, for the period 1950 to 1995 was used to develop a data base for this assessment. About 121 F5, and 924 F4, tornadoes recorded between 1950 and 1995 (an additional 4 in the 1996 to 1998 period). It was estimated that about 30% of all reported tornadoes were in the F2 to F3 range and about 2.5% were in the F4 to F5 range.

The Department of Energy Report DOE-STD-1020-94, [Ref. 4] has some insights into wind generated missiles:

1 For sites where tornadoes are not considered a viable threat, to account for objects or debris a 2x4 inch timber plank weighing 15 lbs is considered as a missile for straight winds and hurricanes. With a recommended impact speed of 50 mph at a maximum height of 30 ft above ground, this missile would break annealed glass, perforate sheet metal siding and wood siding up to to 3/4-in thick. For weak tornadoes, the timber missile horizontal speed is 100 mph effective to a height of 100 ft above ground and a vertical speed of 70 mph. A second missile is considered: a 3-in diameter steel pipe weighing 75 lbs with an impact velocity of 50 mph, effective to a height of 75 ft above ground and a vertical velocity of 35 mph. For the straight wind missile, an 8-in concrete masonry unit (CMU) wall, single wythe (single layer) brick wall with stud wall, or a 4-inch concrete (reinforced) is considered adequate to prevent penetration. For the tornado missile, an 8-to-12-in CMU wall, single wythe brick wall with stud wall and metal ties, or a 4-to 8-inch concrete (reinforced) slab is considered adequate to prevent penetration (depending on the missile). (Refer to DOE-STD-1020-94 for additional details.)

2 For sites where tornadoes are considered a viable threat, to account for objects or debris the same 2x4 inch timber is considered but for heights above ground to 50 ft. The tornado missiles are (1) the 15 lbs, 2x4 inch timber with a horizontal speed of 150 mph effective up to 200 ft above ground, and a vertical speed of 100 mph; (2) the 3-inch diameter, 75 lbs steel pipe with a horizontal speed of 75 mph and a vertical sped of 50 mph effective up to 100 ft above ground; and (3) a 3,000 lbs automobile with ground speed up to 25 mph. For the straight wind missile, an 8-in CMU wall, single wythe brick wall with stud wall, or a 4-inch concrete (reinforced) is considered adequate to prevent penetration. For the tornado missile, an 8 in CMU reinforced wall, or a 4-to 10-inch concrete (reinforced) slab is considered adequate to prevent penetration (depending on the missile). (Refer to DOE-STD-1020-94 for additional details.)

The winds associated with hurricanes and other storms are generally less intense and lower in magnitude than those associated with tornadoes. Generally, high winds from wind storms and hurricanes are considered to be the controlling wind level at a higher frequency but at a lower magnitude.

Recommended Values for Risk-informed Assessment of Spent Fuel Pool

The tornado strike probabilities for each F-scale interval were determined from the SPC raw data on a state-averaged basis. For each F-scale, the point strike probability was obtained from the following equation:

$$P_{fs} = \left(\frac{\Sigma_N <a>_T}{A_{ob}} \right) \times \frac{1}{Y_{int}} \qquad \text{Equation A2e-1}$$

where:

P_{fs} = strike probability for F-scale (fs)
$<a>_T$ = tornado area, mi^2
A_{ob} = area of observation, mi^2 (state land area)

Y_{int} = interval over which observations were made, years
Σ_N = sum of reported tornados in the area of observation

The tornado area, $<a>_T$, was evaluated at the midpoint of the path-length and path-width intervals shown in Table A2e-2a, based on the SPC path classifications. For example, an F2 tornado with a path-length scale of 2 has an average path length of 6.55 miles and with a path-width scale of 3 has an average width of 0.2 miles.

The tornado area, $<a>_T$, was then modified using the method described in NUREG/CR-2944 (based on Table 6b, page 19 and Table 7b, page 21) to correct the area calculation based on observations of the variations in a tornado's intensity along its path length and path width, see Figure A2e-3. Table A2e-2b gives the path-length correction data. Table A2e-2c gives the path-width correction data. The corrected effective area has a calculated $<a>_T$ of about 0.28 mi^2. The combined variation in intensity along the length and across the width of the tornado path is shown in Table A2e-2d (Table 15b from NUREG/CR-2944). For example, an F2 tornado with a path-length scale of 2 and a path-width scale of 3 has a calculated $<a>_T$ of about 0.28 mi^2. The total area is reapportioned using Table A2e-2d to assign 0.11 mi^2 to the F0 classification, 0.13 mi^2 to the F1 classification, and 0.04 mi^2 to the F2 classification.

The risk regionalization scheme from NUREG/CR-2944, as shown in Figure A2e-4 was used to determine the exceedance probability for each region identified. A continental U.S. average was also determined. Included in Figure A2e-4 are the approximate location of commercial LWRs and independent spent fuel storage facilities.

The SPC raw data for each state was used to determine the F-scale, path-length and path-width characteristics of the reported tornadoes. The effective tornado strike area was corrected using the data from NUREG/CR-2944. Equation A2e-1 was used for each state and the summation and averaging of the states within each region (A, B, C and D, as well as a continental USA average) performed. The results for the exceedance probability per year for each F-scale are given in Table A2e-3, and graphically presented in Figure A2e-5. The SPC data analysis is summarized in Table A2e-4.

Significant Pool Damage

An F4 to F5 tornado would be needed to consider the possibility of damage to the spent fuel pool by a tornado missile. The likelihood of the exceedance of this size tornado is estimated to be 5.6×10^{-7} per year (for Region A), or lower. In addition, the spent fuel pool is a multiple-foot thick concrete structure. Based on the DOE-DOE-STD-1020-94 information, it is very unlikely that a tornado missile would penetrate the spent fuel pool, even if it were hit by a missile generated by an F4 or F5 tornado.

Support System Availability

An F2 or larger tornado would be needed to consider damage to support systems (power supplies, cooling pumps, heat exchangers, and water makeup sources). The likelihood of the exceedance of this size tornado is estimated to be 1.5×10^{-5} per year (for Region A), or lower.

This frequency is bounded by other more likely initiators that can cause loss of support systems.

References:

1 NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," Lawrence Livermore National Laboratory, December 1987.
2 NUREG/CR-2944, "Tornado Damage Risk Assessment," Brookhaven National Laboratory, September 1982
3 http://www.ncdc.noaa.gov/
4 DOE-STD-1020-94, "Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities," January 1996, Department of Energy

Table A2e-1 Tornado and High Wind Data Summary

	NUREG/CR-5042 Data				NCDC data	
Site	Tornado frequency (per mi^2-year)	Tornado strike frequency (per year)	High wind damage frequency (per year)	Tornado damage frequency (per year)	Frequency 1950-1995 average for F0-F5 (per mi^2-year)	Frequency 1950-1995 average for F2-F5 (per mi^2-year)
Indian Pt. 2	$1.00x10^{-4}$	$1.00x10^{-4}$	$2.50x10^{-5}$	$<1.0x10^{-7}$	$1.2\text{-}2.2x10^{-4}$	$0.2\text{-}0.7x10^{-4}$
Indian Pt. 3	$1.00x10^{-4}$	$1.00x10^{-4}$	$1.80x10^{-5}$	$<1.0x10^{-7}$	$1.2\text{-}2.2x10^{-4}$	$0.2\text{-}0.7x10^{-4}$
Limerick 1-2	$1.13x10^{-4}$	$2.30x10^{-4}$ (<F1)	$9.00x10^{-9}$	$<1.0x10^{-8}$	$2.2\text{-}3.4x10^{-4}$	$0.7\text{-}1.3x10^{-4}$
Millstone 3	$1.87x10^{-4}$	$1.87x10^{-4}$	Low	$<1.0x10^{-7}$	$2.8\text{-}3.4x10^{-4}$	$0.2\text{-}1.1x10^{-4}$
Oconee 3	$2.50x10^{-4}$	$3.50x10^{-3}$ 1 mi rad.	Low	$<1.0x10^{-9}$	$2.8\text{-}3.4x10^{-4}$	$0.7\text{-}0.9x10^{-4}$
Seabrook 1-2	$1.26x10^{-3}$	$7.75x10^{-5}$	$\leq 3.89x10^{-8}$	$2.06x10^{-9}$ LOSP & RWST	$1.8\text{-}3.8x10^{-4}$	$0.4\text{-}1.1x10^{-4}$
Zion ½	$1.00x10^{-3}$	$1.00x10^{-3}$	N.A.	$<1.0x10^{-8}$	$3.4\text{-}5.4x10^{-4}$	$1.2\text{-}2.0x10^{-4}$
GSI A-45 PRAs	Regional Local		w/o recovery of offsite power			
ANO 1	$5.18x10^{-4}$ $4.37x10^{-4}$	$1.53x10^{-3}$	$5.69x10^{-6}$	$2.53x10^{-4}$	$3.7\text{-}7.5x10^{-4}$	$1.7\text{-}2.4x10^{-4}$
Point Beach 1-2	$6.98x10^{-4}$ $4.11x10^{-4}$	$5.38x10^{-4}$	$1.00x10^{-5}$	$5.00x10^{-5}$	$3.4\text{-}4.7x10^{-4}$	$1.2\text{-}1.5x10^{-4}$
Quad Cities 1-2	$5.18x10^{-4}$ $5.44x10^{-4}$	$1.04x10^{-3}$	$\leq 1.0x10^{-8}$	$5.08x10^{-7}$	$3.4\text{-}5.4x10^{-4}$	$1.2\text{-}2.0x10^{-4}$
St. Lucie 1	$6.98x10^{-4}$ $1.20x10^{-3}$	$1.70x10^{-4}$	$\leq 1.0x10^{-8}$	$1.61x10^{-8}$	$8.4x10^{-4}$	$1.2x10^{-4}$
Turkey Pt. 3	$3.37x10^{-4}$ $5.83x10^{-3}$	$1.70x10^{-4}$	$3.30x10^{-5}$	$2.54x10^{-6}$	$8.4x10^{-4}$	$1.2x10^{-4}$

Table A2e-2a Tornado Characteristics

F-scale	Damage and wind speed	Path-length scale		Path-width scale	
		Scale	Length (mi)	Scale	Width (yds)
0	Light Damage (40-72 mph)	0	< 1.0	0	< 18
1	Moderate Damage (73-112 mph)	1	1.0 - 3.1	1	18 - 55
2	Significant Damage (113-157 mph)	2	3.2 - 9.9	2	56 - 175
3	Severe Damage (158-206 mph)	3	10.0 - 31.9	3	176 - 527
4	Devastating Damage (207-260 mph)	4	32 - 99.9	4	528 - 1759
5	Incredible Damage (261-318 mph)	5	100 >	5	1760 >

Table A2e-2b Variation of Intensity Along Length Based on Fraction of Length per Tornado(*)

Local tornado state	Recorded tornado state					
	F0	F1	F2	F3	F4	F5
PL-F0	1	0.383	0.180	0.077	0.130	0.118
PL-F1		0.617	0.279	0.245	0.131	0.125
PL-F2			0.541	0.310	0.248	0.162
PL-F3				0.368	0.234	0.236
PL-F4					0.257	0.187
PL-F5						0.172

(*) - Table 6b from NUREG/CR-2944

Table A2e-2c Variation of Intensity along Width Based on Fraction of Width per Tornado(*)

Local tornado state	Recorded tornado state					
	F0	F1	F2	F3	F4	F5
PW-F0	1	0.418	0.154	0.153	0.152	0.152
PW-F1		0.582	0.570	0.310	0.264	0.262
PW-F2			0.276	0.363	0.216	0.143
PW-F3				0.174	0.246	0.168
PW-F4					0.122	0.183
PW-F5						0.092

(*) - Table 7b from NUREG/CR-2944

Table A2e-2d Combined Variation in Intensity along Length And Across Width of Tornado Path(*)

Local tornado state	True maximum tornado state					
	F0	F1	F2	F3	F4	F5
CV-F0	1.0	0.641	0.380	0.283	0.298	0.286
CV-F1		0.359	0.471	0.433	0.358	0.333
CV-F2			0.149	0.220	0.209	0.195
CV-F3				0.064	0.104	0.116
CV-F4					0.031	0.054
CV-F5						0.016

(*) - Table 15b from NUREG/CR-2944

Table A2e-3 Exceedance Probability for Each F-scale

NUREG/CR-2944 Region	Exceedance probability (per year)					
	F0	F1	F2	F3	F4	F5
A	7.4E-05	4.4E-05	1.5E-05	3.5E-06	5.6E-07	3.1E-08
B	5.6E-05	3.3E-05	1.1E-05	2.5E-06	3.7E-07	2.1E-08
C	2.9E-05	1.5E-05	4.1E-06	8.9E-07	1.3E-07	4.7E-09
D	3.6E-06	1.6E-06	3.9E-07	8.7E-08	1.6E-08	---
USA	3.5E-05	2.0E-05	6.1E-06	1.4E-06	2.2E-07	1.0E-08

Table A2e-4 SPC Data Analysis Summary by State

	NUREG/CR -2944 Region					Tornado F-scale							Point strike probability (per year)						Land Area
State	A	B	C	D	Years	F0	F1	F2	F3	F4	F5	Total	F0	F1	F2	F3	F4	F5	(mi^2)
AL	X	X			46	165	364	323	129	36	14	1031	2.9e-05	3.2e-05	1.3e-05	3.7e-06	6.9e-07	4.3e-08	50750
AZ				X	44	90	57	11	2	0	0	160	6.7e-07	2.9e-07	3.6e-08	1.8e-09	0	0	113642
AR	X				46	198	298	331	149	31	0	1007	3.2e-05	3.5e-05	1.3e-05	2.4e-06	1.9e-07	0	52075
CA				X	45	142	58	21	2	0	0	223	5.1e-07	2.7e-07	6.0e-08	2.7e-09	0	0	155973
CO			X	X	46	616	441	99	15	1	0	1172	4.4e-06	2.0e-06	4.2e-07	3.9e-08	3.3e-11	0	103730
CT			X		46	9	29	20	5	2	0	65	1.1e-05	1.1e-05	3.6e-06	8.5e-07	2.2e-07	0	4845
DE			X		42	20	23	11	1	0	0	55	2.6e-05	1.5e-05	1.5e-06	6.4e-09	0	0	1955
DC*					1	1	0	0	0	0	0	1	1.3e-04	0	0	0	0	0	61
FL		X	X		46	1156	665	293	30	4	0	2148	1.5e-05	8.6e-06	2.2e-06	2.8e-07	2.0e-08	0	53997
GA		X			46	147	537	266	65	17	0	1032	2.9e-05	3.0e-05	1.2e-05	3.4e-06	4.3e-07	0	57919
ID				X	42	63	53	8	0	0	0	124	4.7e-07	1.9e-07	1.4e-08	0	0	0	82751
IN	X				46	246	336	263	108	77	8	1038	3.3e-05	3.5e-05	1.5e-05	5.2e-06	1.2e-06	6.7e-08	35870
IA	X				46	478	506	421	119	74	9	1607	3.7e-05	3.7e-05	1.4e-05	3.1e-06	6.1e-07	2.5e-08	55875
IL	X				46	431	440	316	113	39	3	1342	3.0e-05	2.7e-05	9.8e-06	2.5e-06	3.3e-07	2.1e-08	55875
KS	X	X			46	1111	610	404	168	54	16	2363	3.5e-05	3.0e-05	1.1e-05	3.0e-06	5.8e-07	1.1e-07	81823
KY	X				46	79	168	133	65	35	3	483	1.6e-05	1.7e-05	6.9e-06	1.8e-06	3.1e-07	1.4e-08	39732
LA		X			46	225	620	268	123	16	2	1254	2.4e-05	2.2e-05	6.9e-06	1.4e-06	1.2e-07	1.9e-08	43566
ME				X	42	21	44	17	0	0	0	82	1.8e-06	1.1e-06	1.7e-07	0	0	0	30865
MD			X		46	49	92	26	5	0	0	172	1.5e-05	9.2e-06	9.4e-07	8.2e-09	0	0	9775
MA			X		45	24	72	31	8	3	0	138	1.2e-05	1.1e-05	4.3e-06	1.6e-06	3.7e-07	0.0e+00	7838
MI		X	X		45	195	308	210	57	30	7	807	1.4e-05	1.4e-05	5.2e-06	1.4e-06	2.8e-07	1.4e-08	56809
MN		X	X		46	372	336	158	53	28	6	953	1.4e-05	1.2e-05	3.5e-06	7.2e-07	1.3e-07	6.6e-09	79617
MS	X	X			46	226	468	369	136	59	10	1268	4.4e-05	4.4e-05	1.7e-05	5.0e-06	1.0e-06	1.3e-08	46914

Table A2e-4 SPC Data Analysis Summary by State

	NUREG/CR -2944 Region					Tornado F-scale							Point strike probability (per year)						Land Area
State	A	B	C	D	Years	F0	F1	F2	F3	F4	F5	Total	F0	F1	F2	F3	F4	F5	(mi^2)
MO	X				46	298	577	334	109	48	1	1367	1.8e-05	1.6e-05	5.3e-06	1.3e-06	2.3e-07	2.6e-11	68898
MT				X	44	174	42	33	4	0	0	253	1.0e-06	7.0e-07	2.3e-07	2.2e-08	0	0	145556
NE		X	X		46	827	585	255	105	42	4	1818	2.9e-05	2.9e-05	1.2e-05	3.5e-06	3.5e-07	1.6e-08	76878
NV				X	34	41	8	0	0	0	0	49	2.9e-07	4.0e-08	0	0	0	0	109806
NH				X	45	24	34	15	2	0	0	75	4.7e-06	2.4e-06	4.7e-07	1.1e-08	0	0	8969
NJ			X		45	43	58	23	4	0	0	128	1.7e-05	6.6e-06	7.9e-07	7.1e-09	0	0	7419
NM			X		46	261	104	31	4	0	0	400	1.5e-06	5.2e-07	8.0e-08	1.1e-09	0	0	121365
NY				X	44	101	106	35	21	5	0	268	7.6e-06	6.1e-06	2.3e-06	8.8e-07	2.2e-07	0	47224
NC			X		46	153	321	143	44	26	0	687	1.5e-05	1.4e-05	4.9e-06	1.5e-06	2.5e-07	0	48718
ND			X		46	490	211	91	28	7	3	830	4.7e-06	3.2e-06	1.1e-06	3.6e-07	9.1e-08	1.1e-08	68994
OH	X				46	157	321	166	53	27	9	733	2.1e-05	1.8e-05	5.6e-06	1.3e-06	3.0e-07	2.8e-08	40953
OK	X				46	845	808	626	209	83	9	2580	4.1e-05	3.9e-05	1.4e-05	3.6e-06	7.0e-07	5.5e-08	68679
OR				X	45	31	15	3	0	0	0	49	2.9e-07	1.5e-07	3.1e-08	0	0	0	96003
PA			X		46	93	220	143	26	22	2	506	9.4e-06	9.0e-06	3.3e-06	9.3e-07	2.0e-07	5.4e-09	44820
RI			X		23	3	4	1	0	0	0	8	1.9e-05	1.3e-05	1.7e-06	0	0	0	1045
SC		X			46	136	234	100	31	15	0	516	1.9e-05	1.9e-05	6.8e-06	1.8e-06	3.0e-07	0	30111
SD		X	X		46	651	259	197	57	7	1	1172	9.7e-06	8.1e-06	3.0e-06	7.7e-07	1.5e-07	1.2e-08	75898
TN	X				46	107	241	139	76	29	4	596	2.2e-05	2.2e-05	8.3e-06	2.1e-06	2.0e-07	1.7e-10	41220
TX		X	X		46	2632	1837	1067	317	76	5	5934	1.6e-05	1.3e-05	4.3e-06	1.1e-06	1.8e-07	3.8e-09	261914
UT				X	43	53	19	6	1	0	0	79	5.1e-07	3.2e-07	1.0e-07	2.8e-08	0	0	82168
VT				X	41	7	14	12	0	0	0	33	3.3e-06	2.0e-06	3.4e-07	0	0	0	9249
VA			X		45	84	132	68	28	6	0	318	8.5e-06	7.0e-06	2.0e-06	4.4e-07	7.1e-08	0	39598
WA				X	41	24	17	12	3	0	0	56	4.9e-07	9.6e-08	2.3e-08	3.6e-09	0	0	66582
WV			X		45	27	36	16	8	0	0	87	2.2e-06	2.4e-06	9.7e-07	2.5e-07	0	0	24087
WI		X	X		46	204	378	276	62	24	5	949	2.6e-05	2.4e-05	7.9e-06	1.4e-06	2.5e-07	3.3e-08	54314

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	NUREG/CR -2944 Region					Tornado F-scale							Point strike probability (per year)						Land Area
State	A	B	C	D	Years	F0	F1	F2	F3	F4	F5	Total	F0	F1	F2	F3	F4	F5	(mi^2)
WY				X	46	247	145	43	8	1	0	444	2.5e-06	1.2e-06	3.1e-07	7.1e-08	1.9e-08	0	97105
Sum						13776	13251	7834	2553	924	121	38459							3536342

* - DC was not included in the exceedance analysis.

Figure **A2e-1**

Annual Average Number of Tornadoes per 10,000 Square Miles by State, 1950-1995

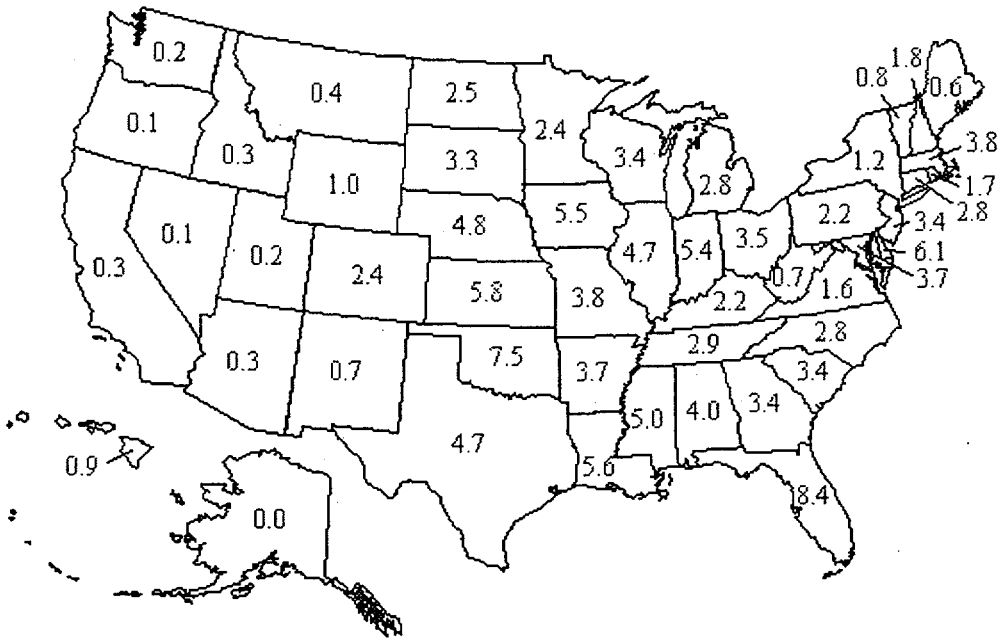


Figure A2 **e-2**

Average Annual Number of Strong-Violent (F2-F5) Tornadoes per 10,000 Square Miles by State

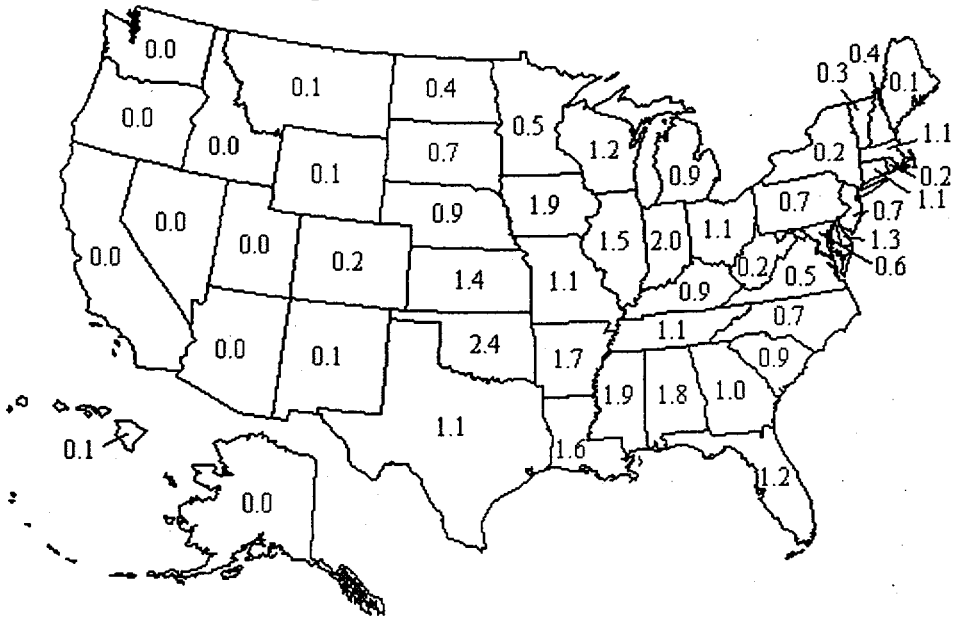
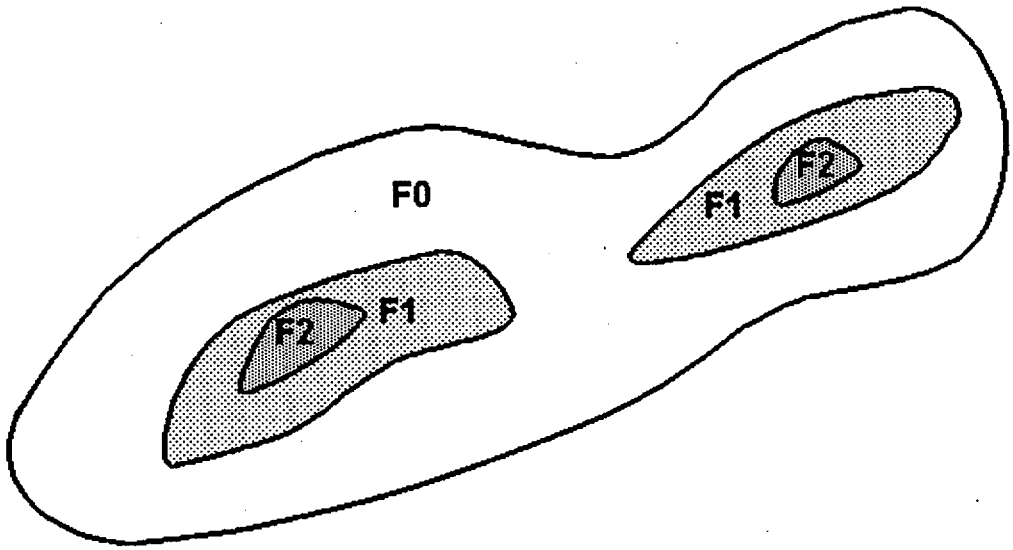


Figure A2e-3 Sketch of Hypothetical F2 Tornado Illustrating Variations



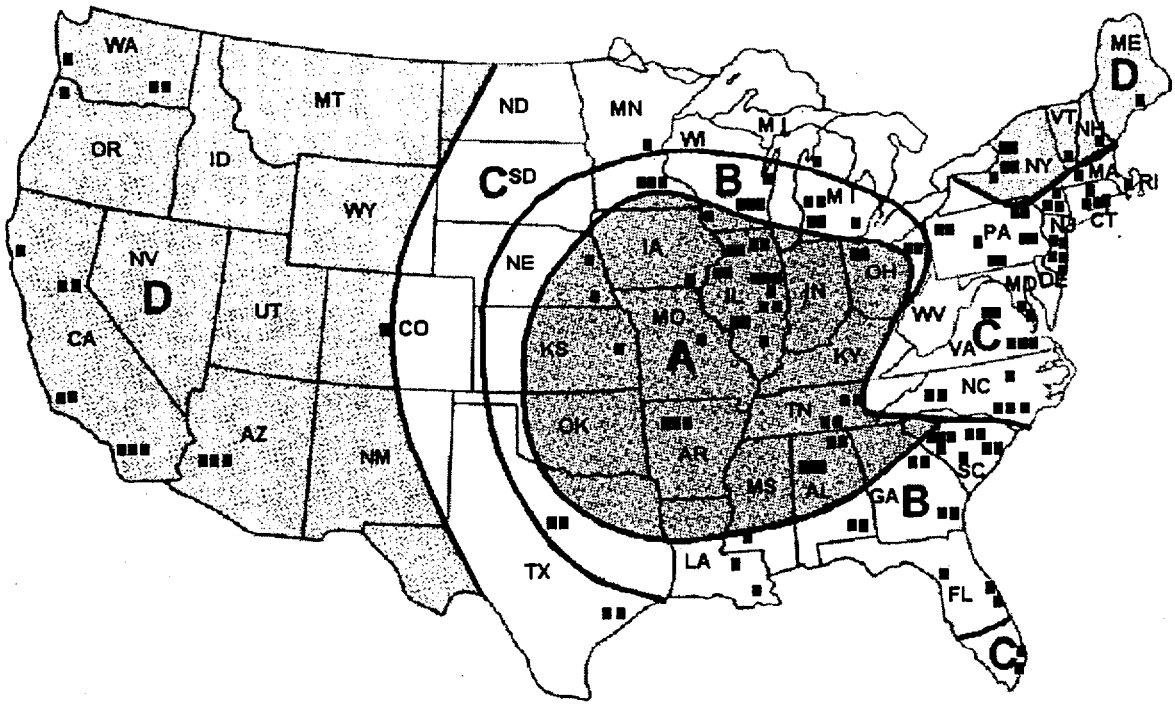
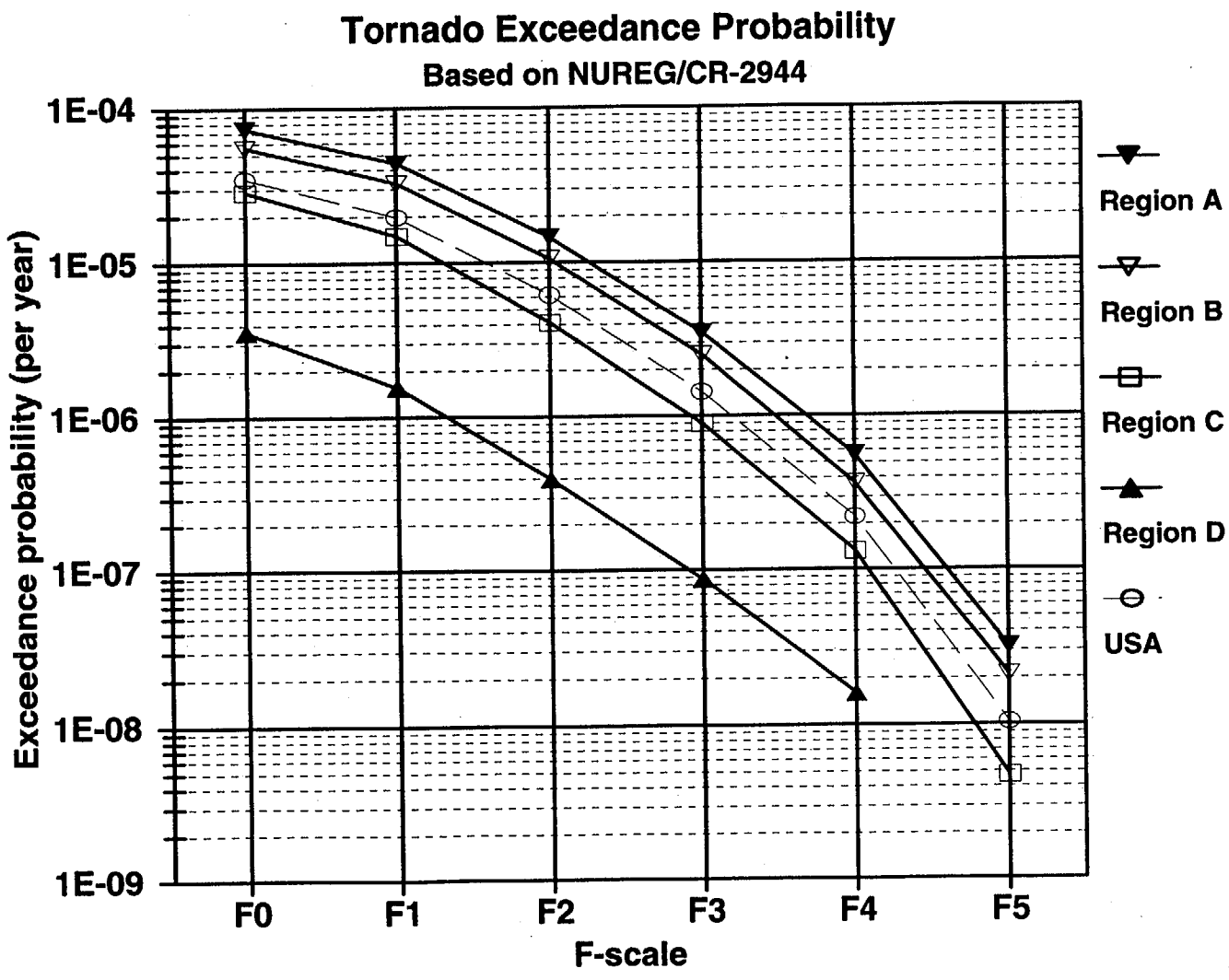


Figure A2e-4 Tornado Risk Regionalization Scheme (from NUREG/CR-2944)

Figure A2e-5 Tornado Exceedance Probability For Each F-scale



Appendix 3 Criticality

3.1 Introduction

The staff criticality assessment includes both a more classical deterministic study and a qualitative risk study. The conclusion in Chapter 3 of this report that criticality is not a risk significant event is based upon consideration of both of these studies. The deterministic study was used to define the possible precursor scenarios and any mitigative actions. The risk study considered whether the identified scenarios are credible and whether any of the identified compensatory measures are justified given the probability of the initiating scenario. This appendix combines both the risk study, the consequences, and the report on the deterministic criticality assessment into one location for easy reference.

3.2 Qualitative Risk Study

3.2.1 Criticality in Spent Fuel Pool

Due to the processes involved and lack of data, it was not possible to perform a quantitative risk assessment for criticality in the spent fuel pool. Enclosed as section 3.2.2 is a deterministic study in which the staff performs an evaluation of the potential scenarios that could lead to criticality and identified those that are credible. In this section the staff provides its qualitative assessment of risk due to criticality in the SFP, and its conclusions that the potential risk from SFP criticality is sufficiently small.

In the report enclosed in section 3.2.2, the staff assessed the various potential scenarios that could result in inadvertent criticality. This assessment identified two scenarios as credible, which are listed below.

(1) A compression or buckling of the stored assemblies could result in a more optimum geometry (closer spacing) and thus create the potential for criticality (see the NRC staff report "Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools," at the end of Appendix 3). Compression is not a problem for high-density PWR or BWR racks because they have sufficient fixed neutron absorber plates to mitigate any reactivity increase, nor is it a problem for low-density PWR racks if soluble boron is credited. But compression of a low-density BWR rack could lead to a criticality since BWR racks contain no soluble or solid neutron absorbing material. High-density racks are those that rely on both fixed neutron absorbers and geometry to control reactivity. Low-density racks rely solely upon geometry for reactivity control. In addition, all PWR pools are borated, whereas BWR pools contain no soluble absorbing material. If both PWR and BWR pools were borated, criticality would not be achievable for a compression event.

(2) If the stored assemblies are separated by neutron absorber plates (e.g., Boral or Boraflex), loss of these plates could result in a potential for criticality for BWR pools. For PWR pools, the soluble boron would be sufficient to maintain subcriticality. The absorber plates are generally enclosed by cover plates (stainless steel or aluminum alloy). The tolerances within a cover plate tend to prevent any appreciable

fragmentation and movement of the enclosed absorber material. The total loss of the welded cover plate is not considered feasible.

Boraflex has been found to degrade in spent fuel pools due to gamma radiation and exposure to the wet pool environment. For this reason, the NRC issued Generic Letter 96-04 to all holders of operating licenses, on Boraflex degradation in spent fuel storage racks. Each addressee that uses Boraflex was requested to assess the capability of the Boraflex to maintain a 5% subcriticality margin and to submit to the NRC proposed actions to monitor the margin or confirm that this 5% margin can be maintained for the lifetime of the storage racks. Many licensees subsequently replaced the Boraflex racks in their pools or reanalyzed the criticality aspects of their pools, assuming no reactivity credit for Boraflex.

Other potential criticality events, such as loose debris of pellets or the impact of water or firefighting foam (adding neutron moderation) during personnel actions in response to accidents was discounted due to the basic physics and neutronic properties of the racks and fuel, which would preclude criticality conditions being reached with any creditable likelihood. For example, without moderation, fuel at current enrichment limits (no greater than 5 wt% U-235) cannot achieve criticality, no matter what the configuration. If it is assumed that the pool water is lost, a reflooding of the storage racks with unborated water or fire-fighting foam may occur due to personnel actions. However, both PWR and BWR storage racks are designed to remain subcritical if moderated by unborated water in the normal configuration. The phenomenon of a peak in reactivity due to low-density (optimum) moderation (fire-fighting foam) is not of concern in spent fuel pools since the presence of relatively weak absorber materials such as stainless steel plates or angle brackets is sufficient to preclude neutronic coupling between assemblies. Therefore, personnel actions to refill a drained spent fuel pool containing undeformed fuel assemblies would not create the potential for a criticality. Thus, the only potential scenarios described above in 1 and 2 involve crushing of fuel assemblies in low density racks or degradation of Boraflex over long periods in time.

To gain qualitative insights on the criticality events that are credible, the staff considered the sequences of events that must occur. For scenario 1, above this would require a heavy load drop into the a low density racked BWR pool compressing assemblies. From appendix 2 on heavy load drop, the likelihood of a heavy load drop from a single failure proof crane is approximately 2E-6 per year, assuming 100 cask movements per year at the decommissioning facility. From the load path analysis done for that appendix it was estimated that the load could be over or near the pool between 25% and 5% of the movement path length, dependent on plant specific layout specifics. The additional frequency reduction in the appendix to account for the fraction of time that the heavy load is lifted high enough to damage the pool liner is not applicable here because the fuel assemblies could be crushed without the same impact velocity being required as for the pool liner. Therefore, if we assume 10% load path vulnerability, we observe a potential initiating frequency for crushing of approximately 2E-7 per year (based upon 100 lifts per year). Criticality calculations show that even if the low density BWR assemblies were crushed by a transfer cask, it is "highly unlikely" that a configuration would be reached that would result in a severe reactivity event, such as a steam explosion which could damage and drain the spent fuel pool. The staff judges the chances of such a criticality event to be well below 1 chance in 100 even given that the transfer cask drops

directly onto the assemblies. This would put the significant criticality likelihood well below 1E-8 per year, which justifies its exclusion from further consideration.

Deformation of the low density BWR racks by the dropped transfer cask was shown to most likely not result in any criticality events. However, if some mode of criticality was to be induced by the dropped transfer cask it would more likely be a small return to power for a very localized region, rather than the severe response discussed the above paragraph. This minor type of event would have essentially no offsite (or onsite) consequences since the reaction's heat would be removed by localized boiling in the pool and water would provide shielding to the site operating staff. The reaction could be terminated with relative ease by the addition of boron to the pool. Therefore, the staff believes that qualitative (as well as some quantitative) assessment of scenario 1 demonstrates that it poses no significant risk to the public from SFP operation during the period that the fuel remains stored in the pool.

With respect to scenario #2 from above, (the gradual degradation of the Boraflex absorber material in high density storage racks), there is currently not sufficient data to quantify the likelihood of criticality occurring due to its loss. However the current programs in place at operating plants to assess the condition of the Boraflex, and take remedial action if necessary provide sufficient confidence that pool reactivity requirements will be satisfied . In order to meet the RG 1.174 safety principle of maintaining sufficient safety margins, the staff judges that continuation of such programs into the decommissioning phase will required at all plants until all high density racks are removed from the SFP.

Additionally, to provide an element of defense in depth, the staff believes that inventories of boric acid be maintained on site, to respond to scenarios where loss of pool inventories have to be responded to by makeup of unborated water at PWR sites. The staff will also require that procedures be available to provide guidance to the operating staff as to when such boron addition may be beneficial.

Based upon the above conclusions and staff requirements, we believe that qualitative risk insights demonstrate conclusively that SFP criticality poses so meaningful risk to the public.

3.2.2 Deterministic Criticality Study

This section includes a copy of the report entitled "Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools" which is a deterministic study of the potential for spent fuel pool criticality.

Appendix 3
Criticality

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Appendix 3 Criticality

3.1 Introduction

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(2) If the stored assemblies are separated by neutron absorber plates (e.g., Boral or Boraflex), loss of these plates could result in a potential for criticality for BWR pools. For PWR pools, the soluble boron would be sufficient to maintain subcriticality. The absorber plates are generally enclosed by cover plates (stainless steel or aluminum alloy). The tolerances within a cover plate tend to prevent any appreciable fragmentation and movement of the enclosed absorber material. The total loss of the welded cover plate is not considered feasible.

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Other potential criticality events, such as loose debris of pellets or the impact of water or firefighting foam (adding neutron moderation) during personnel actions in response to accidents, were discounted due to the basic physics and neutronic properties of the racks and fuel, which would preclude criticality conditions being reached with any creditable likelihood. For example, without moderation, fuel at current enrichment limits (no greater than 5 wt% U-235) cannot achieve criticality, no matter what the configuration. If it is assumed that the pool water is lost, a reflooding of the storage racks with unborated water or fire-fighting foam may occur due to personnel actions. However, both PWR and BWR storage racks are designed to remain subcritical if moderated by unborated water in their normal configuration. The phenomenon of a peak in reactivity due to low-density (optimum) moderation (fire-fighting foam) is not of concern in spent fuel pools since the presence of relatively weak absorber materials, such as stainless steel plates or angle brackets, is sufficient to preclude neutronic coupling between assemblies. Therefore, personnel actions to refill a drained spent fuel pool containing undeformed fuel assemblies would not create the potential for a criticality. Thus, the only potential scenarios described above in 1 and 2 involve crushing of fuel assemblies in low-density racks or degradation of Boraflex over long periods in time.

To gain qualitative insights on the criticality events that are credible, the staff considered the sequences of events that must occur. For scenario 1 above, this would require a heavy load drop into a low-density racked BWR pool compressing assemblies. From Appendix 2c on heavy load drops, the likelihood of a heavy load drop from a single failure proof crane is approximately 2E-6 per year, assuming 100 cask movements per year at the decommissioning facility. From the load path analysis done for that appendix, it was estimated that the load could be over or near the pool between 5% and 25% of the movement path length, dependent on plant-specific layout specifics. The additional frequency reduction in the appendix, to account for the fraction of time that the heavy load is lifted high enough to damage the pool liner, is not applicable here because the fuel assemblies could be crushed without the same impact velocity being required as for the pool floor or wall. Therefore, if we assume 10% load path vulnerability, we observe a potential initiating frequency for crushing of approximately 1.2E-6 per year (based upon 100 lifts per year). Criticality calculations in this appendix show that even if the low-density BWR assemblies were crushed by a transfer cask, it is "highly unlikely" that a configuration would be reached that would result in a severe reactivity event, such as a steam explosion which could damage and drain the spent fuel pool. The staff judges the chances of such a criticality event to be well below 1 chance in 100, even given that the transfer cask drops directly onto the assemblies. This would put the significant criticality likelihood well below 1E-8 per year, which justifies its exclusion from further consideration.

Deformation of the low-density BWR racks by the dropped transfer cask was shown to most likely not result in any criticality events. However, if some mode of criticality were to be induced by the dropped transfer cask, it would more likely be a small return to power for a very localized region, rather than the severe response discussed in the above paragraph. This minor type of event would have essentially no off-site (or on-site) consequences since the reaction's heat would be removed by localized boiling in the pool and water would provide shielding to the site operating staff. The reaction could be terminated with relative ease by the addition of boron to the pool. Therefore, the staff believes that qualitative, as well as some quantitative assessment of scenario 1 demonstrates that it poses no significant risk to the public from SFP operation during the period that the fuel remains stored in the pool.

With respect to scenario 2 from above (i.e. the gradual degradation of the Boraflex absorber material in high-density storage racks), there is currently not sufficient data to quantify the likelihood of criticality occurring due to its loss. However, the current programs in place at operating plants to assess the condition of the Boraflex, and take remedial action if necessary provide sufficient confidence that pool reactivity requirements will be satisfied. In order to meet the RG 1.174 safety principle of maintaining sufficient safety margins, the staff judges that continuation of such programs into the decommissioning phase will be required at all plants until all high-density racks are removed from the SFP.

Additionally, to accommodate the potential for a loss in safety margin, the staff believes that inventories of boric acid should be maintained on-site, to assist in scenarios where loss of pool inventories have to be responded to with makeup of unborated water at PWR sites. The staff will also require that procedures be available to provide guidance to the operating staff as to when boron addition may be beneficial.

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